

SMiRT 20 – Book of abstracts Vol. 2

20th International Conference on
STRUCTURAL MECHANICS
IN REACTOR TECHNOLOGY



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Preface

The two biggest global challenges facing the energy world today are climate change and the huge increase in energy consumption. Global electricity consumption is expected to double by the year 2030. There will be increasing competition in the world for energy resources. On the other hand there is a need to lower energy intensity of the economy and turn to CO₂-free forms of energy production. Inevitably, the era of cheap energy is over. These are driving forces favourable for increasing nuclear energy production capacity.

In Europe, the EU has put forward a very concrete target. The Heads of State set very ambitious EU mandatory targets for the new Energy and Climate Change policy for 2020. One of those targets requires at least 20% less CO₂ emissions compared to 1990. Targets will not be realistic without considerable investments on new nuclear power plants and their development. Consequently, the European Commission with the support of nuclear industry have launched a Sustainable Nuclear Energy Technology Platform (SNE-TP) in September 2007. In Finland, a decision to build a new (fifth) nuclear power plant (EPR) was already made in 2002, and the licence for the construction was granted in early 2005. Recently new initiatives have been made by the industry for three more nuclear power plants in Finland. The applications for the Decision in Principle for these initiatives are being handled by the Government. Some other countries in Europe have also made initiatives for new builds.

The international conferences on Structural Mechanics in Reactor Technology (SMiRT) have traditionally provided innovative and practical mechanics-based solutions to the planning, design, construction, operation, and regulation of NPPs and related facilities. SMiRT 20 will continue this tradition, bringing together experts and practitioners from around the world to share their knowledge of technology that is most relevant at this time in the nuclear energy industry for both currently operating facilities and future development. Around 400 papers will give

answers to **Challenges Facing Nuclear Renaissance**. Besides technical papers the SMiRT20 Programme consists of Leadership Forum, Technical Plenary Sessions, Panel Workshops and Tutorial Workshops having topics of major interest.

This book of abstracts gives an outline of all technical papers presented in each Technical Divisions. Full papers are published in the DVD Proceedings.

We wish all the SMiRT20 participants and authors a successful conference and pleasant stay in Finland 2009.

Rauno Rintamaa
Chairman
SMiRT 20 Conference

Seppo Vuori
Chairman
Local Organizing Committee

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5. Modeling, Testing and Response Analysis of Structures, Systems and Components

Modeling and response analysis of structures (including foundations), systems, lifelines and components subjected to extreme loads. Validation of analytical methods based on experimental results.

(Part 2. 1844–3193)

Part 1 of the abstracts of Division 5 is included in Volume 1 of the SMiRT20 abstracts

Numerical studies on shear reinforced impact loaded concrete walls (5-1844)

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Protective concrete barrier walls of nuclear power plants are required to withstand the effects of impacts by projectiles. Numerical methods are developed and taken in use for predicting the response of reinforced concrete structures subjected to impacts of deformable projectiles. Structural behaviour, in terms of collapse mechanism type and the damage grade, are predicted both by simple analytical methods and by extensive non-linear FE-models. Experimental data is needed in order to verify the accuracy of numerical models. Numerical results obtained using different kinds of methods are compared with experimental data and observations on impact loaded reinforced concrete walls with shear reinforcement.

An experimental set-up has been constructed at Technical Research Centre of Finland VTT for medium scale impact tests. The main objective of this effort is to provide data for the calibration and verification of numerical models. Reinforced concrete walls have been tested in the IMPACT project using deformable aluminium missiles. The IMPACT test facility is described in Lastunen et al. [1].

Reference

1. A. Lastunen et al. Impact Test Facility in Transaction SMiRT-19 August 2007, Toronto, Canada.

Qualification against seismic and other external vibration, experiences from the oversight of Olkiluoto 3 (5-1851)

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Introduction

This paper presents experiences which the Finnish Radiation and Nuclear Safety Authority (STUK) has gained during the inspection and supervision of qualification of structures and equipment against seismic and other external vibration for Olkiluoto 3. Feedback from the Olkiluoto 3 is presented in order to bring some useful information to be considered in future projects for nuclear power plants. Experiences presented in this paper are from a nuclear safety authority's point of view.

Background

Qualification of nuclear safety related structures and equipment against external vibrations is a combination of studies in different technical domains. The first issue is to define the external source from which the vibrations are induced. Such sources are seismic earthquake, aircraft crash and explosion pressure wave. The vibration behaviour of a building framework must be analysed and, depending on selected equipment, corresponding design response spectra's must be defined for further qualification of nuclear safety related equipment.

The common procedure is to qualify different components separately against induced external vibrations. Beside that the equipment are studied as larger units, formed by, e.g., pump, motor, switchgear, foundation and connection to piping. Qualification against vibrations can be made by analysis, shake table tests and/or the combination of these analyses and vibration tests.

STUK is overseeing the construction and component manufacturing of nuclear power plants by inspections and supervision of design, manufacture and construction at the site. Also the installation and commissioning of nuclear safety related equipment will be followed as soon as these phases begin. If quality control and assurance from these later phases bring information in the form of non-conformance reports and audit findings, they will also be described in this paper.

Nuclear safety related vibration resistance questions are dealt with correspondence between STUK and the licensee of the Olkiluoto 3 in order to ensure, that all solutions fulfil nuclear safety requirements and from licensor point of view that these solutions are accepted by STUK. Important part for efficient handling of different, mostly complicated technical questions has been the close cooperation between the licensee and supplier of the nuclear power plant so that also specialists from STUK have participated in technical discussions.

Essential results

Feedback from STUK's studies and decisions and from non-conformance reports of Olkiluoto 3 qualification against external vibrations has been collected so that it can serve in future projects as well as in the further development of regulatory guides on nuclear safety [1...3].

Experiences are presented from design process point of view so that is possible to understand how design criteria have been formed from the initial design assumptions to final equipment qualification against external vibrations. Such design phases are sources of external vibration, behaviour of building framework and equipment qualification.

An important issue is cooperation between different technical domain areas, like civil engineering and equipment design.

Summary

Experiences from Olkiluoto 3 nuclear safety related equipment qualification against external vibrations will be described. In order to see the total picture, brief description of how STUK has learnt its lessons will be presented. Feedback is collected by design process so that is possible to understand the development of criteria for nuclear safety related equipment qualification against external vibrations. The Finnish seismic conditions and regulatory requirements will also be described.

Selective references from standards, regulatory guides and research studies, which are directly related to the reported feedback/experiences:

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5. Modeling, Testing and Response Analysis of Structures, Systems and Components

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Seismic motion incoherency effects on SSI response of nuclear islands with significant mass eccentricities and different embedment levels (5-1853)

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The paper shows the effects of seismic motion incoherency on the soil-structure interaction (SSI) response of a nuclear structures founded on different site conditions. The paper presents results obtained from a sequence of parametric SSI studies using the AP1000-based stick SSI model that was also employed in recent EPRI studies (Short, Hardy, Merz and Johnson, 2006 and 2007). The paper focuses on the effects of foundation embedment on incoherent SSI response. Different stochastic and deterministic incoherent SSI approaches are employed. These incoherent SSI approaches are a part of those used in the EPRI studies called the SASSI-based approaches (deterministic AS and SRSS approaches, and stochastic simulation approach). In addition, an alternate version of the SRSS approach is included. The 2005 and 2007 Abrahamson incoherency models for all sites, hard-rock sites and soil sites are applied. No wave passage effects are considered. The computed SSI results show that incoherent SSI effects are significant for both non-embedded and embedded structures. Conclusions and recommendations are stated at the end of the paper.

We considered both stochastic and deterministic incoherent SSI approaches. In addition to the stochastic simulation approach, three deterministic approaches were considered: i) linear superposition, or algebraic sum, of the scaled incoherent spatial modes (AS in EPRI studies), ii) quadratic superposition of the incoherent modal SSI complex response amplitudes (transfer function amplitudes) assuming a zero-phase for the incoherent SSI complex response phase (SRSS in EPRI studies), and iii) quadratic superposition of the incoherent modal SSI complex response amplitudes (transfer function amplitudes) assuming a non-zero phase for the incoherent SSI complex response that is equal to coherent SSI complex response phase (not used in EPRI studies). The last implementation is an alternate version of SRSS approach that does not neglect the complex response phase.

Two case studies are considered: i) A typical PWR Reactor Building (RB) with three different embedment levels, and ii) the AP1000-based stick model used in the EPRI studies with different embedment levels, and two different foundation mat sizes. The AP1000-based stick model that was surface founded in the EPRI studies was embedded in the analyses of this paper. The embedded foundation walls are modeled by shell elements. Seismic input and soil layering were assumed: i) identical with those used in EPRI studies for AP1000-based stick model, ii) typical hard-rock site-specific GRS with 2007 Abrahamson hard-rock coherency model and hard-rock soil profile with V_s of 9000 fps, iii) RG 1.60 GRS with 2007 Abrahamson soil coherency model and soil layering with V_s of 1000 fps. It should be noted that the 2007 Abrahamson soil coherence function is currently not accepted by US NRC. Only the 2007 Abrahamson hard-rock coherence function is permitted by US NRC at this time.

Based on the investigated case studies shown in this paper, the following conclusions are drawn:

- 1) The effects of motion incoherency are similar for non-embedded and embedded nuclear structures. The SSI results shown herein indicate that motion incoherency effects are significant for both rock and soil sites. Typically, motion incoherency effects are larger for rock sites in high frequency range.
- 2) Combined effects of embedment and motion incoherency are much more complex for soil sites than for rock sites. For rock sites, it appears that motion incoherency effects are to reduce the SSI response at all frequencies, but more drastically in the high frequency range, above 10-12 Hz. For soil sites, the motion incoherency effects manifest visible at much lower frequencies, below 10Hz, where global, dominant structural vibration modes exist.
- 3) For structures with significant mass eccentricities, motion incoherency effects could amplify the torsional SSI responses, as shown herein for the AP1000-based stick model on a soil site, in Y direction.

As practical recommendations, we believe that for soil sites, the combined effects of motion incoherency and embedment have to be considered. For hard-rock sites, since motion incoherency effects have similar trends for non-embedded and embedded SSI models, the use of simple reduction factors might be acceptable.

We believe that more study is worthwhile to propose and gain acceptance for the use of soil coherence function by the US NRC.

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Finite element analysis of the primary shield structure and evaluation for postulated reactor pressure vessel head drop (5-1857)

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Reactor pressure vessel head (RPVH) lifts are required for head replacement operations and refueling outages at nuclear power plants. To ensure that the reactor core remains covered with coolant and sufficient cooling is available following a postulated RPVH drop, it is required to evaluate the consequences of impact loading from a postulated concentric flat head drop (approximately 400 kips free-falling through 40 ft. or more) onto the vessel flange (Ref. 1). Available literature for this non-design basis event is limited and structural response acceptance criteria for it are currently under development jointly by the US Nuclear Regulatory Commission (NRC) and the industry group Nuclear Energy Institute (NEI) (Ref. 2). Determination of system responses for implementation of these criteria entails transient linear or nonlinear dynamic analysis of the nuclear steam supply system, internal concrete structure and containment system by Finite Element (FE) software such as ANSYS (Ref. 3). As it is convenient and necessary, due to contractual and division of responsibility requirements, to separate the design/analysis considerations for the nuclear steam supply system (NSSS) from those for the supporting internal concrete structure (ICS), a decoupled approximate approach must be taken. Also, there are unique and special response and design evaluation considerations for this impact loading condition which must be addressed. This investigation provides general guidance and recommendations for addressing the NSSS interface with boundary and supporting structures, i.e., soil, containment building/basemat, and ICS in the performance of FE analysis and design evaluations for this beyond design basis RPVH postulated drop event.

The important factors that need to be considered in development of an analysis methodology are discussed by addressing the various types of RPVH support interfaces. The structural response to the postulated impact loading is characterized by inelastic deformation of the NSSS system components and its structural steel supports, accompanied by linear or nonlinear response of the concrete in compression and a rebound response in tension. Tension response due to the rebound may yield or rupture the concrete anchorages (rupture of the anchorages per se does not constitute a “failure” condition provided that the deformation is limited and the system pressure boundary integrity is

maintained). The actual supporting concrete behavior varies in accord with the supporting concrete configuration which can be a shelf, ledge, a corbel, or the base slab itself. Other factors considered are the estimation of inelastic response utilizing a linearly elastic model, the effects of decoupling, soil-structure interaction effects, model boundaries and extent of model needed to capture response, the various concrete inelastic response modes, problem size and solution time/computational resources, anchorage stiffness and strength uncertainty, dynamic load factors and equivalent static load approaches, shear friction utilization as an energy absorber, and concrete local bearing response.

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Study on radiation shielding performance of reinforced concrete wall (2): shielding analysis (5-1865)

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Introduction

It is necessary to estimate the decrease in the radiation shielding ability of a cracked concrete wall in a nuclear facility for safety reasons. Since the crack width is very small, it is difficult to measure the change in the penetration rate of radiation through the thick concrete shield. In this study, the shielding abilities of cracked and uncracked walls were assessed by simulation using the three-dimensional Monte Carlo code MCNP⁵¹⁾ and its nuclear data library MCPLIB04²⁾.

Calculation method

The calculation models were two models, simple and practical building model of the repository for low level radioactive wastes. Only gamma rays were used as a source of radiation, and their spectrum was conservatively obtained from the

calculated spectrum of vitrified radioactive wastes. A cubic room model with dimensions of $3\text{ m} \times 3\text{ m} \times 3\text{ m}$ was used as the simple model, and the thickness of the wall was 1200 mm. The source was set at the center of the room, and its volume was 1 m^3 . It was assumed that a straight crack extending from the floor to the ceiling would be formed in one of the walls. The crack width was varied from 0 to 10 mm for the assessment. The cell tally was set outside the center of the crack, and had two types. One was rectangular type that was $100 \times 100 \times 10$ mm in size for simple model, and the other was cylindrical type that was 39 mm in diameter and 60 mm in thickness for practical model. The dimensions of the cylindrical one were the same as those of an ionization chamber. The number of gamma rays generated during the simulation was up to 2.0×10^{11} . The calculation was performed for about 2 months using 16 CPUs, each of which had a frequency of 4 GHz or more. Results obtained by calculation were normalized with the strength data of practical radioactive wastes. The calculated gamma-ray fluxes were converted into the dose rate by the flux-to-dose rate conversion factors based on ICRP Pub.74.

Calculated results

The ratio of the penetration rate of radiation through a cracked concrete shield to the penetration rate of radiation through an uncracked concrete shield was obtained from the analysis. The thickness of the concrete shield was 1200 mm. Ratios of 5, 50, and 1000 were obtained at crack widths of 1, 2, and 10 mm, respectively. Since it was experimentally verified that the maximum crack width should be less than 1 mm it was expected that the decrease of shielding ability would become about 1/10 in the minimum. Parametric estimates of the ratio were obtained at different wall thicknesses and a constant crack width of 1 mm. Up to a wall thickness of 500 mm, the penetration rate of radiation through the cracked wall was almost the same as that of radiation through the uncracked wall. This was because the number of gamma rays that penetrated the cracked wall was negligibly smaller than the number of gamma rays that penetrated the uncracked wall. The ratio of the two penetration rates, as defined above, increased when the wall thickness exceeded 500 mm. However, the ratio was at the most 10 when the wall thickness was 1200 mm. In the case of the practical model, the wall thickness was 1000 mm, and cracks were formed at intervals of 200 mm; further, the ratio was less than 3.

Conclusion

The penetration rate of radiation through a cracked concrete shield was compared with that of radiation through an uncracked concrete shield in order to estimate the decrease in shielding ability by using the Monte Carlo code MCNP5. When the crack width was 1 mm, which is a practical value, and the

wall thickness was less than 500 mm, there was almost no change in the two penetration rates. It was found that the penetration rate increased with the wall thickness. The ratio of the two penetration rates was at the most 10 when the wall thickness was 1200 mm. It was concluded that the increase in the penetration rate of radiation through the concrete shield due to crack formation is not a serious problem since the surface of the crack practically generated in concrete walls is irregular and never smooth as calculated above.

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Study on radiation shielding performance of reinforced concrete wall. (1) Loading test on concrete walls and modeling of concrete cracks (5-1866)

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Introduction

Reinforced concrete structures that store radioactive materials are required to maintain shielding capability as well as aseismic capability even after suffering damage, mainly cracks, due to earthquakes. Many researches have been performed on the aseismic capability of walls cracked during an earthquake, but few have been conducted on their shielding capability.

In order to evaluate the shielding capability of concrete walls, horizontal loading tests were conducted on RC and SC (steel plate reinforced concrete) wall specimens to confirm their aseismic capability, and to obtain a better understanding of their crack patterns and roughness of fracture surfaces. Cyclic horizontal loads were applied to the specimens. Images of crack patterns were taken with a high-resolution digital camera and crack width and length were obtained by analysing the images. Core samples were taken through cracked portions of wall specimens to investigate the roughness of the fracture surfaces.

Loading tests

Two RC wall specimens and one SC wall specimen were tested in this research. The rebar ratio and rebar spacing of the RC wall specimens and the plate thickness ratio and stud spacing of the SC wall specimen were made the same as those of real walls so that the crack patterns obtained from loading tests would be similar to real walls.

The RC wall specimens were 250 mm thick. One, called RC-1, had a rebar ratio of 0.508%, which corresponds to the minimum rebar ratio of real walls. The other, called RC-2, had a rebar ratio of 0.796%, which corresponds to the average rebar ratio of real walls. The rebar spacing was 200 mm, as for real

walls. The SC wall specimen, called SC, was 300 mm thick. The SC steel plate was 3 mm thick, based on the proportion for real walls. The stud spacing 200 mm, was the same as for real walls.

For RC specimen, cyclic loads were applied step by step up to a shear strain of 4×10^{-3} , and then increased until shear failure occurred. For the SC specimen, cyclic loads were applied step by step up to an ultimate shear strain of 6×10^{-3} , and it was then unloaded in order to observe the crack patterns of an unbroken SC specimen.

Investigation of cracks

A digital camera having 4500×3000 pixels was employed to obtain the crack patterns of the wall specimens. The resolution was 0.1 mm in crack width. The images were digitally analyzed to obtain crack lengths and widths.

The maximum crack width was 2.6 mm for RC-1 and 2.0 mm for RC-2 when subjected to 4×10^{-3} shear strain. After unloading, the maximum width was reduced to 1.1 mm and 0.7 mm respectively. The maximum crack width was 0.2 mm for SC in the unloaded condition from 6×10^{-3} shear strain.

In order to investigate the fracture surface roughness of the wall specimens, core samples were taken through cracked portions. Each core was easily broken into two pieces and the fracture surfaces were digitally scanned to create 3D images.

There were 6 core samples obtained from RC specimens. The maximum depth of the fracture surface was 25.6 mm and the minimum depth is 9.6 mm. There were 3 core samples obtained from SC specimen. The maximum depth of the fracture surface was 29.7 mm and the minimum depth was 9.8 mm.

Modeling of cracks

In order to perform analyses of shielding capability of walls, the crack model was established. A through crack was modeled by a straight slit having a constant width in order to make the model conservative against all through cracks. Then, the slit width was determined 1.0 mm considering the maximum crack width observed in RC specimens. The crack patterns were simplified to equally spaced parallel slits crossing at 45 degrees to the horizontal.

Conclusions

- It can readily be said that the loading tests of the RC and SC wall specimens simulated the shear failure behaviour of real RC and SC walls and that the crack patterns and the roughness of fracture surfaces observed were appropriate for investigating shielding capability of the real walls.

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- Cracks corresponding to 95% of total crack length had widths of less than 0.25 mm for both RC specimens. Cracks corresponding to almost 100% of total crack length had widths of less than 0.1 mm.
- The crack spacing was 200 mm, which corresponded to the rebar spacing of the RC specimens and the stud spacing of the SC specimen.
- Since depths of fracture surfaces are much greater than crack width, those are too rough for radiation to go through the walls. It can safely be said that radiation rays would hardly penetrate walls directly via these through cracks.
- The through cracks in the RC and SC wall specimens were modelled by a straight slit having a constant width of 1 mm. The crack patterns on the walls were simplified to parallel slits of equal spacing crossing at 45 degrees to the horizontal.

Seismic response impact of incoherent SSI analysis by new hard-rock coherency model (5-1874)

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Many earthquake recordings show the response motions at building foundations to be less intense than the corresponding free-field motions. To account for these phenomena, the concept of spatial variation, or wave incoherence was introduced. Several approaches for its application to practical analysis and design as part of soil-structure interaction (SSI) effect have been developed, which require a coherency model compatible with the soil condition of the given SSI model. However, conventional coherency model didn't reflect the characteristics of earthquake data from hard-rock site, and their application to the practical nuclear structures on the hard-rock sites was not justified sufficiently. Seismic response impact of hard-rock coherency model proposed in 2007 on the incoherent SSI analysis is discussed in this study.

Case studies are performed to investigate the effects of site condition, location, foundation type and spatial variation of input motion. The ground condition represents typical medium-hard rock and hard-rock sites that have shear wave velocity of 3,500 ft/s and 8,000 ft/s, respectively. A typical reactor building of pressurized water reactor type is converted into three-dimensional-beam-stick model. To identify the responses of rocking and torsion behavior, additional edge points and rigid beam elements are added to the model. The foundations of two different models are placed on surface and embedded into the ground, respectively. Site specific response spectrum is defined as input motion, which is constructed through near field earthquake sources considering the effect of a fault around the site. Acceleration time histories composed of two horizontal components and one vertical component are artificially generated, and those spectral accelerations comply with 5% damped site specific response spectrum. Each ground motion has total duration of 24 seconds with interval of 0.005 seconds. Seismic response from incoherent SSI analysis is also compared with that from U.S. NRC Regulatory Guide 1.60 spectrum to identify the spectral impact due to site specific response spectrum in high frequency range.

All seismic responses from coherent and incoherent SSI analysis are obtained through SASSI computer code and INCOH module developed by Tseng to implement the effect of incoherent spatial ground motion. INCOH was also revised to add the new hard-rock coherency model developed by Abrahamson in

2007. Seismic response from new hard-rock coherency model is compared to that from the conventional coherency model developed also by Abrahamson in 2005. Seismic responses for comparison are calculated applying mode superposition of each component output and multiple input motions to reflect randomness of input motion.

Several facts could be concluded from the case study results.

- (1) Response reduction by wave incoherence is more obvious for embedded foundation model than surface foundation case.
- (2) The response caused by rocking and torsion effect due to incoherent motion does not increase remarkably compared to the coherent case.
- (3) Response having the unique peak at lower than 10 Hz does not show response reduction at even higher than 10 Hz range.
- (4) Response reduction effect at high frequency range due to incoherent motion can be expected under hard-rock site as well as medium-hard rock site.

Experimental study on modal identification and dynamic amplification of a steel frame structure (5-1875)

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The seismic qualification of electrical equipment in nuclear power plants can be normally performed by shaking table tests. However, these tests cannot be used for the equipment that have already been installed in operating plants since the equipment cannot be moved from the plants and mounted on shaking tables. This case requires another method of seismic qualification, i.e., analysis or combination of analysis and test. In addition, if complex equipment is to be seismically qualified by analysis, modal identification test is needed to consider reasonably the dynamic characteristics of analytical model. The dynamic properties of electrical cabinets, in most cases, are calculated from an analysis using finite element method. In some cases, experimental data obtained from either an in situ modal testing or a shaking table testing has also been used to estimate the cabinet dynamic properties.

Specifically for comparatively complex equipment with small devices which is not easy to be mathematically modeled, the testing method is preferably selected for the qualification. In the course of seismic qualification test program, an identification test for the dynamic characteristics of the equipment, usually called exploratory test, is performed prior to the main seismic proof tests to get useful information for the determination of the best method and interpretation of results of qualification tests. The modal identification test is also frequently used for the verification of analytical models used in seismic qualification by analysis.

In this paper, the modal parameters of a steel frame model with 1-bay and 3-stories are identified by impact hammer tests. The shaking table tests are also performed to analyze the dynamic amplification of the model. Forced vibration tests have been conducted using harmonic input motions with varying frequencies. Blind predictions and post-correlation analyses have been performed for the forced vibration test. The results obtained from detailed finite element analyses are verified with the corresponding results from shaking table testing of the frame model.

The paper presents a structural identification procedure for the finite element model updating in steel frame structures. This study consists of three parts: (i)

impact hammer tests and shaking table tests for harmonic loading, and modal identification, (ii) baseline FE Modal updating using the stiffness modification technique, and (iii) estimation of the dynamic amplification.

The effectiveness of the proposed procedure has been verified through an experimental study on a steel frame structure with 1-bay and 3-stories. This paper is also purposed to give a guideline for effective analytical modeling of cabinet-typed electrical equipment by comparing the test results with the analysis results from several different models.

The test specimen is a simple steel frame fabricated with square pipe with size of 620 mm × 420 mm × 1500 mm. The total weight of the specimen is 56.4 kg. The specimen was welded on a base plate with a thickness of 4 mm which was affixed with bolts to the shaking table. The shaking table has a maximum loading capacity of 500 kg.

The results of the modal analyses show good agreement with those obtained by modal tests. And the analysis results of the updated finite element model and the harmonic excitations are improved. Dynamic amplification factors of the structure under El Centro earthquakes obtained respectively by experimentally and numerically also show good agreement.

Performance of X-plate elasto-plastic dampers. A passive seismic supports for nuclear piping under cyclic loading (5-1877)

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Keywords: passive supports, Snubbers, locking, EPDs, radiation hazards, damping

In the nuclear power plant (NPP) piping design, major loads considered are pressure, dead weight, seismic and loads due to restraint to thermal expansion. The thermal stresses and seismic stresses are contradictory to each other. To reduce the former, piping should be flexible and for the later it should be rigid. Hence it becomes very tedious to meet these two contradictory requirements using conventional supports. In this condition snubbers are used, which allow the gradual thermal expansion and arrest the sudden motion due to earthquake. From the past experiences snubbers have proved to be very costly, expensive and need frequent maintenance, leakage problem in hydraulic snubbers and they also congest the space because of more space requirement for installation. Sometimes it is also observed that the mechanical snubbers lock during normal operation and cause undue thermal stresses in the piping and nozzles. Recently a trend has been started to use dampers in place of snubbers. Normally X- shaped plate is chosen as an elastoplastic energy absorbers such that the strain is constant over the height of the device, thus ensuring that yielding occurs simultaneously and uniformly over the full height of the damper. X-plate elastoplastic dampers (EPDs) are preferred because of their high seismic energy absorbing capacity, simple design, low cost and maintenance free operation. EPDs are based on plastically deforming steel components or layered laminated plate in flexure, shear, torsion or a combination thereof. For critical applications like NPP, where safety of public and environment from undue risk of radiation hazards is prime concern, it is necessary to evaluate the performance of supports under seismic loads by testing before implementation. In the present paper testing of 6 mm thick X-plate EPDs made of SS316L material has been performed for evaluation their performance under cyclic loads at different frequencies and tip displacements. By testing it was found that they can sustain many cycles of stable yielding deformation, resulting in high levels of energy dissipation (damping).

Inelastic axisymmetric analysis of BARC prestressed concrete containment model (5-1885)

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Reactor Safety Division (RSD) of Bhabha Atomic Research Centre (BARC) has initiated an experimental program at BARC Tarapur Containment Test Facility to evaluate the ultimate load capacity of Indian Pressurized Heavy Water Reactor (PHWR) containments. For this study, BARC Containment Model (BARCOM), which is 1:4 scale representation of Tarapur Atomic Power Station (TAPS) unit-3&4 540 MWe PHWR Inner Containment of Pre-stressed Concrete has been constructed. The model includes all the important major design features of the prototype containment structure and simulates Main Air Lock (MAL), Steam Generator (SG), Emergency Air Lock (EAL) and Fueling Machine Air Lock (FMAL) openings. The design pressure (P_d) of BARCOM is 1.44 kg/cm^2 (g), which is same as the prototype containment structure. For the experimental program of the ultimate load capacity evaluation it is desirable to identify the critical locations of various types of sensors on concrete and steel members of BARCOM. In addition, it is also desirable to understand the behavior of containment model under internal pressure and study the various failure modes and the expected elastic/inelastic response at the identified critical locations, which are important for instrumentation / monitoring during the experiment. In the present work the pretest analysis of BARCOM has been performed with finite element axi-symmetric modeling. The analysis has been performed using 2D axi-symmetric finite element model representing the 157.5 degree azimuth. This free field region is free from local stresses due to discontinuity of buttresses or penetrations and has been developed to predict the structural response for static over-pressurization load. The concrete structure is modeled with 8-node continuum axi-symmetric elements. The hoop reinforcements and tendons are modeled as rebar elements, which are represented as steel layers of equivalent smeared thickness in a particular continuum axi-symmetric element. These rebar elements have uni-axial behavior resisting only the axial force in the bar direction, which is the hoop direction in the present model. The longitudinal reinforcements and tendons were modeled as embedded axi-symmetric membrane elements with orthotropic material properties so that all the bars carry stress only along their individual axial directions. The thicknesses of the steel

layers have been calculated so that it represents the BARCOM reinforcement and pre-stressing tendons in the axi-symmetric model. A constant pre-stress has been applied as initial stress in the hoop rebar elements and embedded longitudinal tendon elements. It has been assumed that there is no slip between the concrete and steel rebar / embedded members. To consider the effect of the reinforcement, the tension stiffening is used in concrete material model. The structural response and the various failure modes of BARCOM were assessed through non-linear analysis. Elastic/inelastic properties of the concrete and steel members were taken into consideration to trace the response of the structure during over-pressurization. It was concluded that the BARCOM has an ultimate load capacity factor of 3.54 P_d for the identified failure modes which were studied on the axi-symmetric finite element model. Further analysis of the BARCOM with 3D shell element model is reported for the ultimate load capacity evaluation after considering the influence of various openings.

Flow induced vibrations for reactor internals of PWR – Less art, more science (5-1883)

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Aim of the study

In the design process of any component, some aspects are always considered as ‘too difficult to be really predictable’ and rather wisely, engineering wisdom leads to favor robust design over fine-tuned design. As a general rule, such grey areas occur at the interface between engineering disciplines, and FIV most surely belongs to that category.

Within the scope of Reactor Pressure Vessel Internals, it is the wish of the authors to shed some light on the matter firstly by reviewing the important features of FIV analysis and secondly by recognizing which particular topics can be adequately described using current tools and which advances may be desirable and/or achievable.

Short description of the work

This paper will first cover a review of available analysis techniques, from hand calculations to fully detailed numerical simulations. After discussing the compared merits of these increasingly complex approaches, some considerations will be made about the parameters needed to significantly enhance the reliability of such evaluations.

The different aspects covered will include the whole calculation process, from forcing functions models to structural response evaluation. The key assumptions under each method will be reviewed and, where applicable, sensitivity studies will be performed in order to evaluate the potential bias induced by the simplifying and idealization procedures routinely followed.

A particular emphasis will be put on the possible synergy between experimental and analytical approaches, and their relative strengths and weaknesses.

Conclusions

After having reviewed the physical phenomena involved, and how adequately they can be captured by various approaches, the primary contributors to a robust evaluation of FIV levels are identified. For each of them, a ‘best practice’ approach is described and some recommendations for future development are provided.

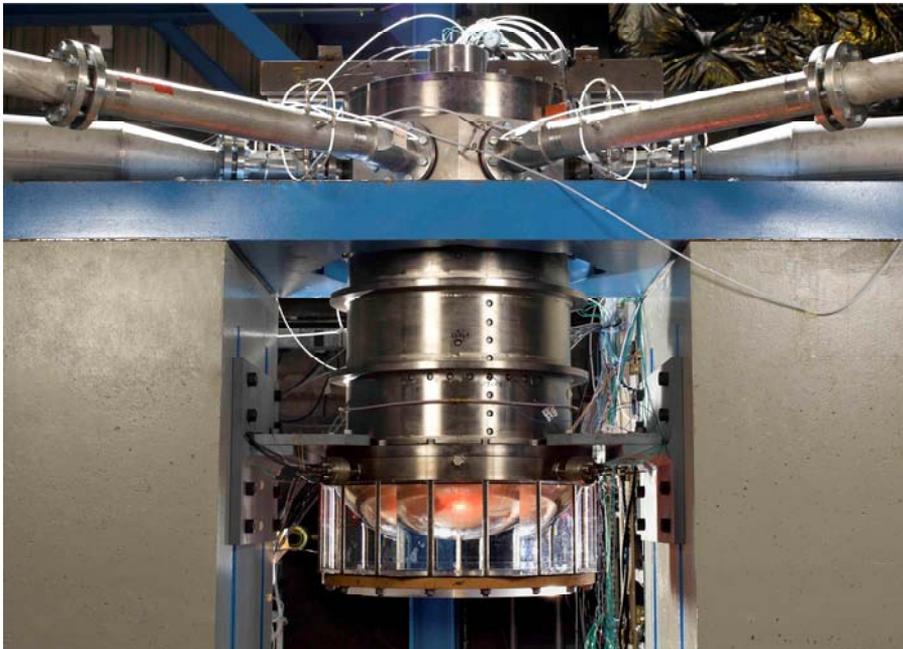


Figure 1. Example mockup used for hydraulic testing.

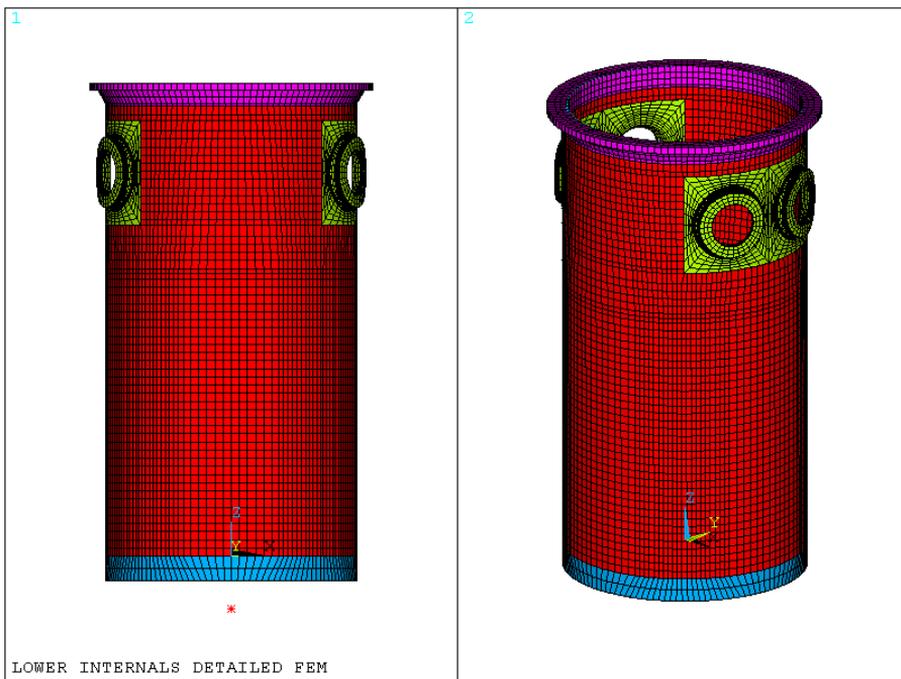


Figure 2. Typical FEM model used for a modern Core Barrel FIV analysis (4-loops PWR).

Structural modeling and analysis of the SMART-2008 shaking table specimen (5-1892)

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The problem of analysis and design of 3D buildings prone to relevant torsional effects when subjected to seismic excitation is still an open one. These effects can be particularly relevant for the reinforced concrete (RC) structures typical of the nuclear industry. For this reason the two French companies *EDF* and *Commissariat à l'Énergie Atomique* have recently launched the SMART-2008 project (Seismic design and best-estimate Method Assessment for Reinforced concrete buildings subjected to Torsion and non linear effects). A 3D scaled model (scale 1/4th) representative of a nuclear RC structure is currently being tested on the Azalee shaking table (at CEA research center in Saclay, France) under a series of bi-directional accelerograms of increasing amplitude and able to induce non linear behavior in the structure. The 3-storey building has an irregular shape in plan and several openings in the walls: both factors are able to emphasize the torsional effects under seismic loading. An international benchmark¹ contest has been launched in parallel to the experimental tests. The participants have to predict the structural behavior under different seismic loading, to the aim of: (a) modeling and analyzing typical buildings of nuclear power plants; (b) predicting the excitation experienced by equipments within the building; (c) determining fragility curves for structural components.

In this paper the initial work done by the authors within the context of the benchmark will be described. A brief review of the European codes prescriptions on 3D effects is presented. A few linear analyses have been performed according to code prescriptions, making use of both simplified plane models and complete 3D models. The capability of plane models in predicting the dynamic behavior of the specimen is thus assessed. High stress levels are expected in the structure elements at high amplitude levels of seismic excitation: the numerical environment of the ANSYS FEM code is adopted to describe the structural non linear behavior. The building is discretized in finite elements so as to perform the blind prediction of the structural behavior, which is the main goal of the first phase of the benchmark problem. The walls and the slab are firstly modeled as solid linear elastic elements to tune the structural model and avoid the difficulties one can meet in the analysis of a complex structure. The linear model is representative of the specimen under low levels of excitation. After the first linear elastic step, the nuclear specimen is modeled by nonlinear elements which

consider the real behavior of the material when subjected to strong ground motions. In particular, the reinforced concrete is modeled through the ANSYS available material model, which allows to simulate the concrete behavior with the presence of steel bars. The concrete model describes the failure of brittle materials; both cracking and crushing failure modes are accounted for. Geometric nonlinearities are also considered. The model is analyzed in the dynamic range: both spectral and time-history analysis are adopted in the linear range, while only time-history analyses are performed in the non linear range.

The numerical results, both in the linear and in the non linear range, will provide useful indications on the modeling and analysis strategies to be adopted for RC structures prone to 3D effects when subjected to seismic excitation. The comparison with experimental results, when available, will suggest possible improvements in the numerical models and analysis, further increasing the usefulness of the numerical analyses here presented.

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Structural dynamic analysis of a non symmetrical RC building within the scope of a blind prediction contest – Project SMART 2008 (5-1900)

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The Commissariat à l'Énergie Atomique (CEA) has launched a blind prediction contest project named SMART 2008 (Seismic design and best-estimate Methods Assessment for Reinforced Concrete buildings subjected to Torsion and non-linear effects) in May 2007 which will be concluded with a final workshop in 2010. The objective of the benchmark is to predict the dynamic behaviour of a non-symmetric 3-dimensional RC building that is designed according to the French nuclear methods and subjected to seismic excitation. More than 30 international teams consisting of industrials, nuclear/energy corporations and research facilities are taking part in this blind prediction contest. The project has got two phases: phase one – seismic analysis of the nuclear reinforced concrete building and phase two – variability quantification and fragility assessment. Phase 1 includes a blind predictive benchmark, an experimental test and a prediction upgrading.

Subject to this paper is the numerical seismic analysis for the structure carried out during phase 1 which is taking place from May 2007 to December 2008 by the team HSK (Swiss Federal Nuclear Safety Inspectorate) and Basler und Hofmann Consulting Engineers. The team belongs to the group of industrials participating in the contest. Methods and results of the best estimate analysis and the post-adjustments due to the findings during the experimental tests will be described.

The best estimate structural dynamic analysis has been carried out by means of modal time history analysis using different specified synthetic and real accelerograms. A 3-dimensional model has been developed for the concrete structure. The properties of the numerical model have been modified in order to account for the specimen's real behaviour during dynamic excitation. Non linear material behaviour due to higher loads is taken into account.

The experimental tests have been carried out at the 3-dimensional RC specimen on the AZALEE shaking table at CEA. The results that are presented to the participants can be compared to the results predicted by means of the numerical analysis. Insights into the genuine characteristics of the model, findings about when exactly strength degradation starts and about the actual amount of energy dissipation can be gathered. The comparison of the test results and the results of the participants' modal analysis show that the shaking table must have an important influence on the mode shapes of the structure. The effect of the shaking table as well as the other previously mentioned effects have been evaluated and taken into account in the refined numerical analysis.

Dynamic analysis of a concrete shear-wall (5-1908)

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Introduction

Current trend of earthquake resistant design is performance based one which limits the maximum displacement under the load. To evaluate the effectiveness of the displacement control under the near-field ground motion due to earthquake, IAEA initiated CRP program. As a first step, they performed an analysis of a shear wall and compared the result with the shaking table test results. In this paper, we use better concrete and rebar model to regenerate the test results using ABAQUS, a general purpose nonlinear FE program, and compare the result with other calculations.

Analysis

We used 'Concrete Damage Model' embedded in ABAQUS 6.4.1, which is originally proposed in Lubliner et al. (1989) and further developed in Lee and Fenves (1998).

The model of the concrete shear wall came from the previous report KINS/GR277 (Hyun et al. 2004) by KINS. Figure 1 described the original shear wall for the real test and Figure 2 presents the two dimensional model of the shear wall with rotational and translational springs. The concentrated masses were put as the Figure 3 with the same intervals. A dynamic analysis on this model resulted in the 3 initial modes of the structure (Figure 4), which are similar to the modes of beam-stick model in KINS/GR277 report.

Conclusions

The model analysis results are similar to the previous results, which show that current material model is in good agreement with test specimen within elastic limit. The strength of the structure is related to behavior beyond the elastic limit such that we calibrated the material property parameters to regenerate the dynamic response of the real test structure under serial earthquake loading, where the nonlinear damage model plays an important role.



Figure 1

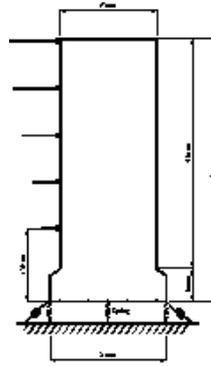


Figure 2

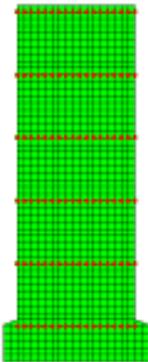


Figure 3

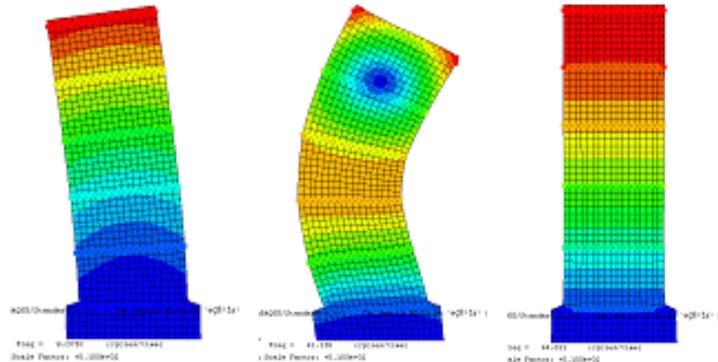


Figure 4

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Application and evaluation of “design by rule” procedures applicable to nuclear power plant ASME B & PVC section iii class 2 and 3 piping (5-1911)

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The text of the abstract is placed here. The abstract should be concise and should present.

This paper describes a “Design by Rule” procedure that could be used to seismically design nuclear safety related cold (to $< 150^{\circ}\text{F}$, 66°C) piping. The procedure pre-engineers the location of transverse pipe supports, which will maintain the piping system within applicable code stress limits which include seismic design loads.

Commercial nuclear power plants and other nuclear material and waste processing plants typically contain over 150,000 feet (48,000 meters) of cold safety related piping requiring seismic design. The engineering effort using conventional stress analysis procedures typically requires over 400,000 man hours per plant. By use of the “Design by Rule” procedure suggested herein this engineering man hour effort could be reduced by at least 70 percent with no loss of design conservatism.

The “Design by Rule” procedure consists of locating transverse pipe supports as multiplier of dead weight support spacings. Typical piping construction codes such as ASME B&PVC Section III, Subsection NF and ASME B31.1 recommend dead weight support spacings which result in a prescribed longitudinal dead weight stress in the piping typically defined as around $0.1 S_c$ where S_c is the allowable stress in the pipe.

By specifying multiples of these dead weight support spacings it is possible to determine the following:

- The dominate frequency of the piping of the piping in transverse and vertical directions
- The resultant seismic force to be applied to the piping taken from the applicable seismic response spectra
- The limiting seismic and total longitudinal stress in the piping system as a function of the support spacings.

With this information and the procedure developed in this paper it is possible to demonstrate that a piping system is within code allowable limits without the

effort to prepare an analytical computerized model of the piping system and to determine the frequency of the piping system.

The paper will demonstrate the application of the “Design by Rule” analysis procedure for typical piping system materials and layouts to include detailed analytical finite element response spectral analysis of piping systems which verify its applicability.

Numerical studies on pre-stressed impact loaded concrete walls (5-1921)

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The various protective concrete barrier walls of nuclear power plants are required to withstand the effects of impacts by accidental or intentional missiles. Structural systems and solutions are under development both in building framework and in detail level, which require more sophisticated tools for different design phases. For example detailing of shear reinforcement is under development. Therefore numerical methods have been developed and taken in use for predicting the response of pre-stressed shear reinforced concrete structures subjected to impacts by hard projectiles. The impact load function on reinforced concrete wall caused by a hard missile is studied. Predicted impact loads are further used in structural analyses. Alternatively, the dynamical contact between the projectile and the target plate is modelled with nonlinear FEM.

Structural behaviour of the impact loaded pre-stressed walls has been predicted both by analytical methods and by involved non-linear FE-models. Analysis methods to predict associated damage mechanisms like crater formation, penetration, shear cone formation and perforation are examined. Experimental data is needed in order to verify the accuracy of numerical models. In this paper, numerical results obtained using different kinds of methods are compared with experimental data and observations on impact loaded pre-stressed reinforced concrete walls with shear reinforcement and with pre-stress levels of practical applications.

An experimental set-up has been constructed at VTT for medium scale impact tests. The main objective of this effort is to provide data for the calibration and verification of numerical models intended to be used in full scale practical applications.

Seismic analysis and upgrading of suspended ceilings and air ducts over main control room, emergency control room and control and protection systems of units 3 & 4 of WWER-440 MW NPP Kozloduy (5-1931)

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The objective of the seismic analysis of the suspended ceilings and air ducts was to certify that the existing structure can / can not withstand an earthquake defined as Floor response spectra (FRS) from Review level earthquake (RLE) and Local earthquake (LE) for the specified location of installation (elevation +9.60) at Units 3 & 4 of Kozloduy NPP. The system of suspended ceilings and air ducts was classified as seismic category I because of direct danger for the personnel and the equipment for control of the nuclear reactor situated under the ceilings in Main control room (MCR), Emergency control room (ECR) and the Control and protection systems (CPS). The main bearing structure of the ceilings was built of double T steel sections in longitudinal direction united with U profiles in transverse direction which were hanged to reinforced concrete girders via hangers made mainly of angle and U profiles with length 3–4 m. For ECR on the bottom of double T sections were welded gratings of bars to which Rabitz net was attached and then lime-cast to the net from below. For MCR on the bottom of double T sections were welded gratings of T profiles to which gypsum plates were mounted. Part of the ceiling over MCR was additionally lowered with 1 m by using bars for hangers and gypsum plates below. The air ducts had rectangular shape and were also attached to the girders by hangers made mainly of bars. The vertical branches of the air ducts had contact with the ceiling structure via angle profiles closely fit from all four sides to each vertical branch and welded to the bearing structure of the ceilings.

Two separate models of the steel structure were made on SAP 2000 software – one model for the suspended ceiling over MCR and one over ECR and CPS. The air ducts were also modeled and the contact to the ceiling structure was taken into account. As the contact of the suspended ceiling on its periphery to the surrounding concrete walls and columns was via the lime-cast or the gypsum

plates that have very small rigidity, the horizontal supports in the models were neglected. Because of the lack of vertical X braces and relatively small rigidity of the hangers in case of horizontal excitation (earthquake), the calculated displacements were relatively high which lead to breaking of the lime-cast and the gypsum plates at the periphery which is inadmissible. At the same moment the internal forces in the hangers were very high and the calculations made according to ANSI N690 code showed that they did not have enough bearing capacity in case of seismic excitation defined with FRS for I category structure.

A lot of upgrading measures (connecting of the steel structure to the reinforced concrete walls via anchors, additional vertical X braces, especially for the lowered ceiling with 1 m over MCR, additional supports of the air ducts etc.) were made for Units 3 & 4. The upgraded models were run. The displacements were reduced to minimum and the forces in the elements lowered as well. The calculations for the bearing capacity of the existing and the new upgrading steel elements of the suspended ceilings and air ducts over MCR, ECR and CPS showed that the system of suspended ceilings and air ducts can bear the prescribed seismic excitation defined for I category structure of Units 3 & 4 of NPP Kozloduy.

Seismic analyses of safety important piping systems situated in reactor building of units 5 and 6, Kozloduy NPP (5-1932)

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The aim of the task was to qualify seismically the existing safety important piping installed in the reactor building. The need for seismic qualification came from reassessment of review level earthquake (RLE) established for Kozloduy NPP site. The pipelines should be properly qualified according to the new higher seismic level. The task was included in the Modernization Program developed for Units 5 and 6. The program was executed in two basic stages: Basic Engineering Phase (BEP) and Main Contract Phase (MCP). The selection of the piping to be seismically qualified was done in BEP, together with some preliminary analyses by Framatome ANP. The final analyses, and the detailed design for improvement of the seismic stability of piping were performed by EQE Bulgaria experts, during the MCP.

Some of the safety important systems pipelines, which were analyzed, are located in the containment of the units (pipelines from systems TQ, TG, TK, TX, YR and YP), and the others are in the reactor building outside the containment (TQ, VF, TG, TC, UJ). The systems listed are: reactor core cooling, containment spray, primary circuit purification, emergency feedwater for steamgenerators, pressurizer – bubbler surge lines, spent fuel pool cooling, fire extinguishing and essential service water.

For the analysis of the pipelines, a spectral method was used with suitable for the purpose calculating software, which allows the use of response spectra analysis and uses the ASME Boiler and Pressure Vessel Code, Section III: Rules for Construction of Nuclear Power Plant Components, for pipelines Class 1, 2 and 3.

For the analysis of Class 1 pipelines, the software PepS was used, which incorporates the processing software of DST Pipestress and the graphic interface for input and output by Tractebel Engineering's EditPipe.

For the analysis of Class 2,3 pipelines the Algors's software Pipeplus was used.

The loads and the load combinations incorporated into the models are according to the requirements of the ASME code. The results of the analyzed pipelines were compared to the AMSE criteria.

For the input dynamic loads from seismic, a reduction coefficient F_{μ} was used, which takes into account the ductile behavior of the pipeline systems. The used floor response spectra were for Review Level Earthquake (RLE) with local component LLE, which were developed for Kozloduy NPP.

As a result from performed analyses the pipelines are divided into two groups – the first one include those pipes which need upgrade of support system, and the second one group comprise the pipes, which models demonstrated adequate capacity to load combinations, including seismic load corresponding to latest defined level. In general, 1/3 of the analyzed pipelines did not satisfy the criteria, totally 46 out of 132 lines. Most of the “weak” pipelines belong to the fire extinguishing system.

Corresponding measures for upgrade of pipe support systems were proposed.

Strain-based acceptance criteria for section III of the ASME boiler and pressure vessel code (5-1940)

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Modern finite element codes used in the design of nuclear material transportation and storage casks can readily calculate the response of the packages beyond the elastic regime. These packages are designed to protect workers, the public, and the environment from the harmful effects of the transported radioactive material following a sequence of hypothetical accident conditions. Hypothetical accidents considered for transport packages include a 9-meter free drop onto an essentially unyielding target and a 1-meter free fall onto a 30-cm diameter puncture spike. For storage casks, accident conditions can include drops, tip-over, and aircraft impact. All of these accident events are energy-limited rather than load-limited, as is typically the case for boilers and pressure vessels. Therefore, it makes sense to have analysis acceptance criteria that are more closely related to absorbed energy than to applied load. Strain-based acceptance criteria are the best way to meet this objective.

The U.S. NRC has a long history of assuring the safety of the public from the potential hazards associated with the transportation of radioactive material. For most of this history, the design of the packages used to transport this material has been based upon the ASME Boiler and Pressure Vessel Code and guidance has been provided by U.S. NRC Regulatory Guide 7.6. For the past decade, the section of the Code that is most relevant to the design has been Section III, Division 3. This section of the Code is based upon the concept of stress intensity, which is twice the maximum shear stress. The allowable stress intensities vary according to loading case and type of stress. For some of these, the allowable stress intensity is larger than the yield stress of the material, a tacit approval for a limited amount of plasticity. This approach was necessary when stresses were determined with hand calculations and was still beneficial during the early days of finite element analyses. As finite element calculations became more detailed, it has become possible to determine the stress state at any point in the package and the associated strains. Since the Code has allowed limited plasticity, modern package designers would prefer to use inelastic analysis techniques to calculate the stresses and strains that result from the required loading conditions. There

are two ways to implement inelastic analysis: continue using stress-based acceptance criteria, or; develop strain-based acceptance criteria.

This paper will briefly discuss the efforts within the ASME, detail the advantages of using strain-based criteria, discuss the problem areas associated with establishing strain-based criteria, and provide insights into inelastic analyses as applied to radioactive material transportation and storage casks in general. The views expressed represent those of the authors and not necessarily those of their respective organizations or the ASME.

A simple dynamic model for estimating the effect of gaps on response of a spent fuel transportation cask closure lid during a drop impact (5-1941)

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During an impact event, gaps between the various components of a spent fuel transportation cask may create secondary impacts that result in higher dynamic loads than would have occurred if the gaps had not been present. A condition of particular interest is the gap that may exist between the fuel assemblies and cask closure lid and the effect this gap may have on amplifying the response of the closure lid during an impact.

When spent fuel is transported in a transportation cask, gaps exist between the cask closure lid (the containment boundary) and the cask internal components (fuel assemblies, fuel basket, etc.). If a transportation accident was to occur these gaps may lead to a secondary impact on the lid that could significantly increase the response of the lid above the values that would have occurred if the gap had not been present. For the 30 foot drop, this is why the regulations in 10 CFR Part 71.73(c) (1) require the cask to be dropped “in a position for which maximum damage is expected.” Position is made up of both the orientation of the cask as well as the geometric position of the cask and its internal components relative to one another. Thus to comply with the regulation, gaps whose size is sufficient to significantly influence the dynamic response of the closure lid or internal components must be incorporated in tests and finite element analyses of transportation casks.

Through the use of a simple dynamic model this paper investigates the effect of a secondary impact due to a gap between the cask internals and the cask closure lid on the response of the closure lid during a 30 foot end drop (or c.g. over corner drop). The dynamic model consists of five components (parameters): (1) The mass of the internals traveling at the impact velocity for a 30 foot drop (44.4 ft/sec), (2) the gap between the internals and cask lid, (3) the cask lid, assumed to be a simply supported circular plate, (4) the modal mass of the lid, and finally, (5) an impact limiter that applies a constant deceleration to the cask overpack. In addition, the dynamic model assumes elastic behavior. This is consistent with the Standard Review Plan (NUREG-1617), which recommends that the closure lid bolts and closure lid system within the region of the lid bolts remain elastic in order to demonstrate leak-tightness by finite element analysis.

The response results are presented in terms of the Dynamic Load Factor (DLF) for the closure lid. Response is shown to be a nonlinear function of the impact limiter deceleration, gap size and closure lid diameter and thickness. These results provide valuable insights into the parameters that effect response and show the conditions under which gaps may be of sufficient size to significantly influence response. The NRC Staff plans to compare these results with drop results from detailed LS-DYNA models.

Numerical model of the thermal and mechanical behavior of a CANDU 37-element bundle (5-1942)

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²Reactor Thermalhydraulics Branch, CNSC, Ottawa, Canada

Investigation of the bowing deformation of fuel elements is important for assessing the integrity of fuel and the surrounding structural components under different operating conditions including accidents. The bowing of a fuel element is defined as the lateral deflection of the element from the axial centerline and the magnitude of bow is the maximum deflection between points of restraints. Bowing can have significant effects on performance and safety of the nuclear reactor. For instance, during normal operations, bowed fuel elements could reduce sub-channel flow area resulting in poor heat transfer due to local coolant starvation, causing these elements to deflect as a result of local overheating. Under accident conditions, bowing of fuel elements can lead to safety concerns for fuel coolability and the pressure tube integrity.

The integrity of the pressure tube under accident conditions is a requirement for effective trips in CANDU reactor design. As a result, an understanding of all of the behaviors of the pressure tube in various different accident scenarios is crucial for understanding the trip effectiveness. One of the important pressure tube behaviors that needs to be investigated is the coupled thermal-mechanical interaction between the fuel bundle and pressure tube. In the early stages of a loss of coolant event, it is known that fuel bundles deform or barrel so that the outer elements bend outward. But the magnitude of the distortion is not readily quantifiable. The bundle barreling could result in the bundle coming into increased contact with the pressure tube. If this were to happen, the contact force between bearing pads and the pressure tube would be increased which increases the contact area on the bearing pad and pressure tube, and enhances the heat transfer from the fuel elements to the pressure tube, creating localized hot spots and high stress areas. The combination of higher temperature and higher local stress could lead to pressure tube failure. If this were the case, then the likelihood and numbers of pressure tube failures in loss of coolant and other accident conditions would be greater than presently predicted. This would significantly affect the current understanding of margins to severe core accidents.

The numerical simulation of the fuel bundle to pressure tube interactions requires a reliable numerical bundle model which is capable of predicting macroscopic thermal and mechanical behavior of the fuel bundle assembly under different accident conditions. In order to obtain realistic solutions, all the key

thermal and mechanical features of the fuel bundle, such as temperature-dependent nonlinear material properties, fuel-to-sheath interactions, endplate constraints and contact between fuel elements, need to be included. In this paper, we present a finite element based numerical model for nuclear fuel bundles, verify its accuracy and demonstrate its suitability for being utilized to investigate fuel bundle to pressure tube interaction in future nuclear safety analyses. The present study was carried out in three stages.

The first stage involved development and verification of a finite element-based numerical model for predicting temperature distributions and thermal-induced barreling deformations of a complete 37-element fuel bundle. For structural analysis, a 3D beam model was developed which was able to compute thermal-induced macroscopic deformations caused by axial and radial temperature variations in fuel elements and endplates. On the other hand, a 2D heat transfer model was used to predict asymmetric temperature distributions over the mid-section of fuel elements due to the effect of neutron flux gradient and uneven coolant conditions. The results from this 2D heat transfer model were then combined with a previous axi-symmetric temperature solution to form a complete 3D approximation of the temperature field in fuel bundle. Both the structural and heat transfer analysis models were verified against available experimental data and numerical solutions, but no comparison of model predictions against an integrated bundle deformation data was attempted.

The second stage of the present study involved application of the bundle model to predict static deformations of a fuel bundle to various steady-state power and coolant conditions. Setting up the heat transfer problem required extraction of asymmetric coolant conditions from the results of thermo-hydraulic calculations and adjustment of power generation rate to account for neutron flux gradients. The endplates were assumed to be at the coolant temperature, which was normally variable over the bundle cross-section. The approximate 3D temperature distribution was then applied to the structural model to predict bundle deformation and solutions were obtained for two different steady-state thermal conditions. These solutions indicated that all the fuel elements bowed in the anticipated direction and the overall pattern of the deformation was consistent with the temperature distribution. Element-to-element interactions through the spacer pads played a very important role in maintaining the shape of bundle cross-section. If these interactions were not considered, the maximum barreling deformation of each fuel element occurred at the mid-span and the magnitude of deformation was nearly proportional to the temperature gradient over the fuel element, suggesting that the temperature gradients in fuel rods are the primary driving force for bundle barreling.

In the third stage, the steady-state solution procedure was extended to handle transient thermal conditions, such as the loss of coolant accident (LOCA) conditions. This required implementation of the nonlinear visco-plastic material model for Zircaloy-4 into the beam finite element and generalization of the axial scaling technique for combining the planar and axisymmetric heat transfer

results to form a full 3D transient temperature field of the entire bundle. An investigation was conducted to identify the most important mechanism for barreling deformation of the fuel bundle and it indicated that although the differential on rod elongations played a role on bundle barreling, the most important driving force was the temperature gradients in the fuel rods.

Numerical studies presented in this paper demonstrate that the present finite element based fuel bundle model is reliable and predicts credible bundle behavior in response to accident type conditions in a fuel channel. Note, however, the ability of the present model to predict accident behavior is limited by lack of knowledge of many of the sub-phenomena; two examples are fuel column rigidity and effect of endcap welds on end plate rigidity. The intended application of this model is to assess pressure tube integrity in large break loss of coolant accidents. An evaluation of the completeness of the models and how the application should be performed is the next step towards this goal.

A rational seismic design approach for reinforced concrete walls for nuclear power plants (5-1943)

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Most nuclear power plant buildings are box-type structures consisting of several interconnected shear walls. In recent practice, it is common to use refined finite element models involving shell and plate elements for structural analysis of this type of structures to accurately capture the response of walls and slabs to seismic and other loads. Results from these analyses, in the form of element forces and moments are used by the structural engineer during the design process. In the United States such design is typically conducted according to the requirements of ACI-349 provisions, which are mostly derived from ACI-318 building code. ACI-318 provisions for shear walls are based on experimental results obtained from individual wall specimens tested up to global failure for in-plane forces and moments acting on the wall panel. The ACI design equations (and the tests backing these equations) do not directly address the wall adequacy for simultaneous in-plane and out-of-plane forces/moments; the assumption being that the two sets of demands are likely out of phase (which is often true for earthquake excitation along a single axis). Results from 3D seismic analyses involving simultaneous multi-directional excitation (using 100-40-40 or SRSS combination rule) however often show that a wall may be subject to significant in-plane and out-of-plane demands simultaneously. In the absence of clear rules, the designer is however left to perform independent checks for the in-plane and out-of-plane demands, which disregards the effect on wall adequacy under such simultaneous demand conditions.

Full sectional resultants (global) are required when conducting the design of shear walls according to ACI-349 code. Nevertheless such resultants are not readily available from the finite element models used for typical nuclear structures; they must be calculated using section cuts by numerical integration of

individual elements results. Some mainstream structural analysis programs provide postprocessing capabilities for these purposes. Nevertheless, such options are not available in soil structure interaction programs widely used in nuclear applications such as SASSI2000. Therefore custom postprocessing tools must be developed by the structural engineer for such purpose; or alternatively the shear wall design can be directly conducted using the results from individual finite elements. In this paper shear wall designs conducted using global and element based approaches are discussed and compared against experimental and numerical results. In particular, experimentally tested shear walls available in the literature are redesigned according to the aforementioned methodologies to resist the experimentally observed ultimate load. It is shown that both global and element based methodologies result in wall designs meeting ACI-349 code requirements. Nevertheless, their outcome is different in terms of reinforcement ratios and distributions. Therefore different performance levels can be expected for each design approach. Nonlinear finite element simulations considering concrete cracking and reinforcement yielding are conducted in order to investigate such performance. Wall designs are evaluated in terms of drift and cracks distribution at different loading levels. Conclusions and recommendations are drawn from this evaluation and used to guide the design of walls in box-type nuclear structures. The practical implementation of these recommendations is presented for a typical wall in the Ultimate Heat Sink (UHS) building of a generic nuclear power plant. The design seismic loads are obtained from soil structure interaction (SSI) analyses conducted for the design-basis event or Safe-Shutdown Earthquake (SSE). In addition, sensitivity analyses are carried out in order to account for uncertainties in the soil properties. The results of the seismic SSI analysis are summarized for design purposes in three different forms: first as static equivalent seismic loads based on average peak structural accelerations; second as absolute peak element force and moment resultants; and third in the form of time histories of element resultants. The use of each set of results for design purposes is addressed and discussed. In particular the applicability of ACI-349 provisions is discussed pointing the advantages and limitations of using a particular set of results at the design level.

Response and failure criteria of large cylindrical vessels to rapid pressurization in CANDU severe accidents (5-1945)

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Severe accidents in CANDU reactors are typically categorized into two distinct classes of accident progression. One class of accidents involves a slow progression to core damage and the second, lower likelihood class involves rapid progression to core damage. Due to the positive void reactivity in operating CANDU reactors, postulated accidents do exist in which engineered safeguards (i.e. reactor shutdown) are required to mitigate a reactivity transient that is either due to or a consequence of some initiating events. Depending upon the effectiveness of reactor shutdown, the subsequent progression to severe accident core damage is characterized by either very rapid core degradation (loss of shutdown) or a slow degradation of the core resulting from a power-cooling mismatch following loss of a significant number of mitigating emergency heat sinks.

Since CANDU reactors have two independent and dedicated shutdown systems, Level 1 Probabilistic Risk Assessments (PRA) indicate that severe accidents involving a loss of shutdown are very low frequency and are not dominant contributors to the total core damage frequency. However, due to the rapid core degradation that occurs following an unmitigated reactivity transient, this accident class has the potential for early releases and must be considered outside of the scope of PRA by undertaking consequence analysis. Understanding the response of the reactor system to large and rapid energy releases, and the resulting dynamic loads imposed on the large calandria vessel, is critical to assessing the nature of the initial consequences and the potential for subsequent mitigating actions following core degradation.

Loss of shutdown accidents are characterized by the near-simultaneous rupture of a large number of fuel channels. The discharge of coolant from ruptured channels with ensuing rapid pressurization of the calandria vessel is sufficiently large to be beyond the capacity of the pressure relief system for the vessel.

This paper presents a detailed finite-element model for the dynamic structural response of the calandria vessel following fuel channel failure. This detailed model is used to assess:

- The dynamic response of the calandria vessel to impulse loading transients typical of two-phase coolant discharge arising from multiple, near-simultaneous fuel channel failures,

- The impulse loading threshold to induce calandria vessel failure,
- The location of calandria vessel failure, and
- Potential pathways for subsequent fluid discharge following calandria vessel failure.

In order to evaluate the dynamic response of the vessel, a separate hydrodynamic model is developed that calculates the pressure propagation in the vessel originating from failed fuel channels, including the effect of collapse of calandria tubes onto their pressure tubes due to overpressure. This model provides the transient hydrodynamic pressure loading on the vessel wall. Short term interaction between the hydrodynamic transient and displacement of the vessel walls is taken into account using an elastic representation for the vessel. The finite element model is subjected to the calculated loading and dynamic estimates of strain are generated. In order to account for uncertainties associated with mechanical feedback from the vessel on the hydrodynamic loading transient, a series of parametric runs are performed to evaluate the sensitivity of this coupling. Results show that there is relatively low sensitivity arising from this coupling due to the fact that:

- The initial pressure wave acts as the dominant impulse loading of the vessel wall,
- Subsequent reflected pressure waves are significantly attenuated by the steam bubble formed by the ruptured channels, and
- The transient displacement of the vessel wall is governed by the dominant structural modes of the vessel.

High stresses are generated at the weld discontinuity at either end of the main vessel shell where it connects with a smaller diameter sub-shell extension. The stresses at the welds exceed ultimate tensile strength and the vessel fails by a tearing rupture around the circumference. A critical number of near-simultaneously failed channels are required for the vessel failure to occur.

Additional sensitivity analysis is performed to quantify the uncertainty in both the likelihood and extent of vessel failure. The extent of vessel failure is an important factor which governs the rate of moderator fluid displacement out of the vessel and the subsequent pressurization of containment.

Update of ASCE standard “Seismic analysis of safety-related nuclear structures and commentary” (5-1947)

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The American Society of Civil Engineers Standard 4 (ASCE 4) has been the main guidance document for the seismic analysis of the nuclear safety-related facilities and other critical or important facilities in USA for more than two decades. The Standard was last revised in 1998. This standard was developed mainly for U.S. Department of Energy non-reactor nuclear facilities. A working group of the ASCE Dynamic Analysis of Nuclear Structures Standards Committee undertook a task in 2005 to update the standard to implement recent developments in seismic analysis of these facilities. Recently, a resurgence of the nuclear power industry has made it even more important to have up-to-date seismic provisions.

ASCE 4 is a companion document to ASCE Standard- 43, “Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities.” ASCE Standard-43 has been used as a reference document by several utilities in submittal of combined license applications (COLS) for New Reactors.

The working group responsible for this revision is part of the ASCE Dynamic Analysis of Nuclear Facilities Committee. Its members are drawn from the industry, academia and governmental organizations, thus encompassing the potential users of the standard in different groups. It is expected that the revised document will be available in 2009.

The paper summarizes the changes being made to the document. The entire standard has undergone a major editorial change in order to make it more user-friendly. The standard now is composed of 7 Sections, appendices, and related commentary. Section 1 was expanded to highlight the “target performance goal” approach adopted both in ASCE 4 and its sister standard, ASCE 43. Performance based seismic design criteria have been implemented for Department of Energy Facilities for many years. Only recently, has the nuclear power industry employed such criteria. The ASCE 4 provisions for determining seismic demand include sufficient conservatism that when combined with the seismic design provisions in ASCE 43, the probabilistic target performance goals are achieved. The target performance goals are expressed as annual

frequency of exceeding unacceptable behavior for structures, systems, and components being designed. . Note that these criteria provide many levels of criteria in a graded approach that can be implemented based on the hazards and importance of the facilities. As a result, these criteria are especially valuable for nuclear facilities such as fuel processing facilities where the hazards are significant but much less than those associated with nuclear power plants. Such criteria have not existed in the past and these facilities have not been designed in a consistent manner.

Section 2 brings in the latest developments in seismic ground motions and the various input ground motion definitions which have come to use in recent years. Guidelines for utilizing the output from a probabilistic seismic hazard analysis to establish the input to a seismic soil-structure interaction analysis are presented. Section 3 discusses modeling for seismic analyses, providing guidance on issues such as mesh sizes, material properties, stiffness, damping and requirements for special structures.

Section 4 describes acceptable methods of analyses that range from simple equivalent static approach to complex nonlinear dynamic analysis. This section has been expanded to include pushover analysis and frequency domain method. Section 5 provides a significant update on soil-structure interaction analysis with emphasis on sub-structuring method and three-dimensional response calculations. Section 6 continues with the sub-system analysis, offering guidance on input motions, in-structure response spectra, in-structure time history motions, and coupled system analysis.

Section 7 is devoted to analysis of special structures, including buried structures, earth-retaining walls, above-ground vertical tanks, distribution systems, sliding and rocking calculations and base isolation. Seismic design provisions for distribution systems and for unanchored components provided in this Standard are strongly needed by the nuclear industry. In addition, the subject of dynamic soil pressures on underground walls has been significantly expanded to consider responses from soil-structure interaction analyses.

An appendix addresses the evaluations beyond design basis and presents available methods for such evaluations. Both the probabilistic seismic risk and seismic margin assessments are discussed and compared.

In summary, the revised standard will reflect the state-of-the-art approaches for determining seismic demands on nuclear safety-related structures, systems and components. This document is expected to become the definitive source for seismic analysis in the nuclear industry and resolve many outstanding controversial issues.

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Investigation of building structures response to heavy item drop (5-1951)

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Introduction

The Ignalina Nuclear Power Plant (NPP) has two RBMK-1500 graphite moderated boiling water multi-channel reactors. The Ignalina NPP Unit 1 was shutdown at the end of 2004 while Unit 2 is foreseen to be shutdown at the end of 2009. The projects of dismantling and decontamination of equipment of Unit 1 were initiated. One of important parts of such projects is analysis and assessment of all hazards that are possible during dismantling and decontamination activities. The equipments installed in NPP are heavy. The dropping of heavy dismantled item can destroy dismantling building and adjacent buildings and equipments. Therefore the drop of heavy parts of items should be evaluated during decontamination. The methodology of the evaluation of consequences to building structures of heavy item drop is presented in this paper. The structural integrity analysis of the building structures in case dropping heavy part was carried out. The finite element method was used in this analysis.

Analysis

This paper presents the structural integrity analysis of building structures in case drop of heavy dismantled item during dismantling and decontamination activities in the of Ignalina NPP Unit 1. The drop of heavy item potentially can damage building structure and as result can cause collapse of building. The maximal hazard, which could be caused by the load drop during dismantling activities in the building of Ignalina NPP, is drop of the cut ring of Emergency Core Cooling System (ECCS) pressure vessels. Potential damage of the building slab and adjacent structures due to drop of a cut ring of ECCS pressure vessel on to this slab has been assessed in this paper. The maximum possible drop height (16 m) was chosen in order to evaluate the loading to the slab at level 0.0 m of the building. Finite element model for the cut ring of ECCS pressure vessel drop force calculation was developed. The state-of-the art computer code ABAQUS/Explicit was used for load analysis. The maximum possible drop forces were calculated from this analysis. This maximum force was used for the static analysis of structural integrity of this compartment using finite element software ABAQUS/Standart.

Summary and conclusions

According to analysis results it is possible to conclude that the impacted reinforced concrete slab at level 0.00 m and supporting columns of this compartment will experience cracking of concrete, but the structural integrity of these slab and columns will be maintained during impact of a cut ECCS ring. The analysis results show that the structural integrity of the building will be maintained, it will not collapse and it will be capable of performing its intended function.

LS-DYNA impact analyses of nuclear power plant structures for tornado missile risk analysis (5-1961)

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The TORMIS methodology was developed to estimate the probability of damage to nuclear power plant structures and components from debris missile impacts in extreme winds. A critical component of performing this risk analysis is the assessment of the damage characteristics to the power plant structures from various missiles and the critical velocity at which this damage results. For some target/missile combinations, analytical methods have been historically used to predict damage. However, for more complex structures and missiles, either testing or more detailed analyses are required.

This paper details LS-DYNA finite element analyses that were conducted to determine the critical speeds for various missile/target combinations. LS-DYNA is a nonlinear explicit finite element code for the dynamic analysis of structures, and is particularly well-suited for impact and penetration analysis. Targets in this study included various parts of Emergency Diesel Generator (EDG) Exhaust Vents, EDG Fuel Oil Day Tank Vents and Steel Floor Plates. Critical speeds for vent structures were defined by the degree of closure of the vent cross section, which restricts the flow of exhaust gases. To account for uncertainties in the as-fabricated target strength, two welded joints were considered; a 'strong weld' that has the same strength as the baseline material and a 'weak weld' that fails in

shear at the design allowable strength. These two cases bound the actual strength of the joint and were necessary due to the lack of sufficient data on the welded joint characteristics.

In these analyses, both the missiles and targets were modeled explicitly to correctly model their interaction and the resulting impact damage. A variety of missile types, including metal pipes, concrete pavers, wood beams, steel grates, metal siding, plywood panels, and storage bins were considered. Impact analyses were conducted to determine the impact response for a large variety of missile impact orientations. Results from these analyses were used to determine the critical orientations and speeds for each missile against each type of target. Over three hundred separate analyses were conducted in various missile orientations at several speeds.

Finally, the feasibility of scaling the critical speeds for the various missiles by their initial kinetic energy was examined. This approach had mixed success because of the large differences in stiffness and strength of the missile types as well as changes in critical orientation. In general, however, missiles of similar size and stiffness to the targets took the least energy. Missiles of similar size but softer and weaker required higher energies and the softest and weakest missiles required the greatest energy. A more general scaling rule would require consideration of the missile strength and geometry characteristics.

Thermo-mechanical analysis of a helium cooled divertor of a fusion reactor (5-1963)

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The helium cooled divertor based on modular design concept is envisaged within the framework of a fusion power plant conceptual study. Several modular design variants have been developed and analyzed in the past few years at the Forschungszentrum Karlsruhe (FZK) [1, 2]. An advanced modular divertor concept with multiple helium cooling jets (HEMJ) was found to be the most appropriate for several reasons. Helium cooled divertor should master high heat flux of more than 10 MW/m^2 to ensure that the material constraints are not exceeded. The main design requirements of the helium cooled divertor are to increase the heat removal capability of the divertor and to minimize the pumping power for the coolant. In the presented work thermal and stress loadings on the divertor components cooled by multiple helium jets are investigated numerically taking into account different boundary conditions. Two different thermal loadings on the plasma-facing side of the divertor are analysed: a) stationary loading with a constant heat flux and b) cyclic heat-flux loading. Thermal and structural analyses are carried out with the code ABAQUS. Thermal boundary conditions at the fluid-facing side of the divertor are obtained from the CFD solutions of the turbulent helium flow [3]. The experiments at high heat flux and high mass flow rate, performed at EFREMOV Institute in Sankt Petersburg are simulated in our study. Results of thermal and stress loadings in the divertor structures are provided.

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A failure mode evaluation of a 480V MCC in nuclear power plants at the seismic events (5-1970)

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A 480V MCC Cabinet is one of major equipment system in Nuclear Power Plant. For the shaking table test, a real MCC cabinet was rented from the manufacturing company. For the evaluation of a failure mode for Motor Control Centers (MCCs), a shaking table test was performed. For the shaking table test, two kinds of seismic input motions were used. One is an artificial seismic input motion based on the NRC Reg. guide 1.60 design spectrum and the other is also an artificial seismic motion based on the Korean Nuclear Power Plant site specific Uniform Hazard Spectrum (UHS). The UHS motion was selected for an evaluation of a High frequency effect of the electric equipment in a NPP. PGA levels for shaking table test were scheduled by 0.2 g to 5.0 g but the test was stopped at about the 2.5 g level because of the chattering of the relay systems. The shaking table tests were performed with a one dimensional shaking which was a front to back direction (horizontal) and a vertical direction.

Functional and structural failure modes were also evaluated by this shaking table test. For the evaluation of a relay chattering, the electric signal was measured at several points. There are two kinds of ground fault relays and thermal relays are installed in the MCC. It is impossible to measure the electric signal of all relays, only some of the relays and electric equipments were considered. In the case of the US NRC spectrum, 480 V AC power was supplied so it can measure the signals of the equipments related to a power system like a power transformer. But in case of UHS spectrum, 480 V AC power wasn't supplied because of the safety of experiment. Therefore, only the signals from the relays were measured. For the measurement of a relay system, an arbitrary input power was supplied to the MCC. Also, for the evaluation of structural failure modes, in-cabinet responses and response amplifications of MCC, acceleration responses were measured at major points of cabinet.

Through this test, several kinds of functional failure modes can be found and the chattering effect of several relays in the MCCs can be certified. As a result, it can be recognized that the 480 V MCC has a sufficient seismicity as a SSE level earthquake. But in the case of a higher level earthquake motion, a chattering happened for both seismic motions, moreover both a horizontal and vertical shaking cause a relay chattering.

Seismic FEM analysis of reinforced concrete structure in SMART-2008 project (5-1974)

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In order to assess the seismic tri-dimensional effects and non-linear response of reinforced concrete buildings, a reduced scaled model (scale of 1/4th) of a nuclear reinforced concrete building was tested during year 2008 on AZALEE shaking table at Commissariat à l’Energie Atomique (CEA Saclay, France). This test, supported by CEA and Electricité de France (EDF) is part of the international “SMART-2008” project.

The first part of the project was a blind prediction of structure behavior subjected to the ground motion levels from 0.1 g up to 1 g. As continuation to the first part of the project participants were given some test results so that the participants could adjust their analysis models. This conference proceeding presents the approach and the methodology adopted by the Fortum Nuclear Services Ltd research team to adjust a finite element method (FEM) model to match the test results.

The main difficulty in the analysis was the accounting of concrete surface cracking in relatively low value of the ground excitation. This difficulty was overcome by the calibration of the brittle cracking concrete model parameters so that obtained displacement results corresponded better to the expected displacement result values. Also the damping played major role in the adjustments procedure.

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Salient aspects of analysis and design of large integrated safety related structures (5-1973)

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Introduction

Nuclear Power Plants often involve buildings which are integrated into a single large structure due to the functional requirements. Rigorous structural analysis and design that has to be performed for such large safety related structures, under various environmental conditions including static, seismic and transient conditions, poses certain typical issues to be addressed / resolved as enlisted below,

1. Reducing time and cost of the analysis and design performed.
2. Handling certain typical design conditions / loads, and
3. Re-evaluating/re-looking at the different analysis and design parameters and analysis methods adopted so as to reduce some of the conservatism involved and move towards better solutions.

This paper takes cognizance of the experiences gained from engineering performed for a Proto-type Fast Breeder Reactor (PFBR) under construction at Kalpakkam in INDIA. It comprises of a single large integrated building, Nuclear Island Connected Building (NICB) (93 m × 83 m × 74 m) integrating eight buildings, viz, Reactor Containment Building, Steam Generator Buildings, Fuel Building, Rad waste building, Electrical and Control Buildings.

Critical aspects of analysis and design have been reviewed in this paper to identify aspects for possible improvements with suggestions.

Critical areas addressed

1. Soil-structure interaction (SSI) plays a significant role in affecting the performance of such structures, in which uncertainties are involved in
 - Assessing the Stiffness of soil
 - Assessing damping ratio of soil
 - Idealizing soil in the FE model
 - Distribution of soil stiffness.

Results of analysis performed with soil stiffness determined using Vesic's equation and distributed uniformly, and, those obtained based on analysis with stiffness variably distributed based on pressure acting (using Boussineq's equations) have been compared. While Boussineq's formulation entails extensive computation time, it will be prudent to assess the advantage of Boussineq's formulation with respect to Vesic's formulation from practical design considerations, under the condition of rocky strata.

Uncertainty due to possible variations in measured value of soil stiffness has been taken care of by performing a range analysis adopting Soil Stiffness as $\frac{1}{2}$ K, K and 2 K, findings of which shows variation analysis need not be performed in case of a rocky strata as the effects are in-significant, details of which have been elaborated in the paper.

Soil damping evaluated based on provisions of ASCE code, with values limited to 7% for rocking / torsional motion, 20% for horizontal motion and 30% for vertical motion have been considered. Effect of usage of a uniform damping of 7% in all directions has been studied and found significantly conservative. Possible ways of adopting a composite soil damping in future have been brought out in the paper.

Effect of embedment of structure has been studied and found in-significant for the type of structure and strata considered.

2. Slabs in such large structures play a significant role in distributing the horizontal shear forces between different vertical elements and integrating them together, there-by altering the structural behavior. They are to be designed adequately to carry the in-plane forces, which will be especially significant under seismic and thermal loads.

With the available computational capacities and to take care of certain other aspects, it was sought to go for modeling the slabs purely as diaphragm elements, in NICB. But, with the current improvements in computational capacities, it could be possible to go for modeling slabs using a relatively coarser mesh of *Thick Shell* elements, instead of *Thin Shell* diaphragm elements. The effect of harnessing the out-of-plane bending stiffness (of the portion of slab acting along with beams) and corresponding economy in beam designs can be ascertained.

3. Efforts to reduce the size of the model need attention in eliminating the un-warranted computational costs in such large structures. In NICB, a) all slabs have been modeled using relatively coarser diaphragm elements, b) items / equipments that could be de-coupled have been identified and decoupled and c) stick models for vessels and equipments have been consciously used and coupled with the global model where possible, and d) refined individual sub-structure analysis has been performed for certain components which require a refined analysis, as measures towards reducing the un-warranted size of the model. Further efforts could be a) towards reducing the size of dynamic analysis problem by using super-elements and dynamic condensation techniques,

- b) Performing convergence analyses and determining appropriate elements sizes that could be adopted for different regions based on load variation, thickness etc, and c) study on effects of usage of certain lower-order elements for shells and beams in possible areas.
4. Load due to Initial drying Shrinkage of concrete plays a significant role in such large structures, due to build-up of stresses over large dimensions of slabs and walls, if applied in the entire FE model of the structure as a single load step. Efforts to reduce their effects could be towards application of drying shrinkage loads in the form of construction stage loads and evolving methods to estimate material properties of concrete under shrinkage and thermal loads etc.

Conclusion

This paper identifies the critical areas of concern that are to be addressed while performing structural analysis and design of large structures like NICB. Principles that had been adopted in the present project to address the issues, and possible scope and alternatives available for improvement on the process of analysis and design have been brought out in this paper.

Indian PHWR pre-stressed concrete containment performance evaluation with BARCOM and round robin analysis program (5-1977)

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The BARC Containment (BARCOM) test model is a 1:4 scale representation of Tarapur station 540 MWe Pressurized Heavy Water Reactor (PHWR) pre-stressed concrete inner containment structure. This test model has been constructed at BARC Containment Test Facility-Tarapur for the functional and structural performance evaluation of Indian containments. Bhabha Atomic Research Centre (BARC), Trombay has coordinated an international round robin analysis program to carry out the ultimate load capacity assessment of BARCOM with participants from research, academic and industrial organizations. This is the largest containment model which has been built so far in the world and would be shortly tested up to its ultimate load capacity and the experimental results would be analyzed with the inelastic numerical code predictions obtained from around 15 registered participants in this round robin exercise during the pre-test and post-test phases.

Indian nuclear power plants have double containment with outer and inner containment structures of reinforced and pre-stressed concrete construction respectively. The nuclear containment system of Indian PHWRs is defined as the primary Inner Containment Wall (ICW) and the secondary Outer Containment Wall (OCW) enclosures as well as the systems and components provided to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment. It is desirable to assure that the integral double containment system meets the functional requirements, which are important for safety under various postulated design basis accident (DBA) conditions like Loss of Coolant Accident (LOCA) or the Main Steam Line Break (MSLB) due to the largest break in the primary system. Performance of nuclear containment with improvement in engineered safety features and demonstration of safe technology for public acceptance of nuclear power program have been backed up with containment safety research at BARC, Trombay. Indian PHWRs of 220/540 MWe designs have inherent safety features with large containment

volume, low volumetric power density, low temperature heat sink, low excess reactivity and long prompt neutron life time. Hence the global threat to the containment is of lesser severity for the extremely low probable postulated beyond the design basis severe accidents compared to the PWR and BWR plants where such scenarios have been postulated and widely studied. However, with a view to demonstrate the adequacy of this ultimate barrier of the double containment system and to improve the existing design of the PHWR containments, structural safety assessment for both the design basis and beyond the design basis load conditions has been a thrust research area for Indian PHWR power program.

The objective of the present test program is to obtain the pressure, displacement and strain data related to the various failure modes of BARCOM. Following observations would be recorded that would aid in understanding the functional and structural behavior of the containment model up to the ultimate pressure.

- First appearance of concrete cracking
- First through thickness cracking
- First yielding of the reinforcement
- First significant loss of leak tightness
- Maximum pressure sustained by the test model before significant leakage
- Ultimate pressure sustained by the containment test model.

In order to predict the ultimate load, elaborate tests have been planned on soil, concrete, reinforcement and pre-stressing tendons as detailed below

- Soil investigation of the site to determine the geo-technical parameters (bearing capacity, Poisson's ratio, bulk modulus and shear wave velocity etc.)
- Compressive and tensile strength of the proposed concrete trial and pour mixes for the construction of the test model and the associated standard deviations
- Fracture energy characteristics of plain and reinforced concrete on beam specimens
- Tensile strength of reinforcement bars to be used for the construction of the model
- Tensile strength and relaxation characteristics of the pre-stressing strands
- Wobble and friction coefficient of sheath and pre-stressing system assembly.

More than 1500 surface and embedded sensors have been installed on BARCOM which include vibratory wire embedded and spot-welded strain gauges, electrical resistance strain gauges, dial gauges, earth pressure cells, tilt meters and camera systems for structural response, crack monitoring and fracture parameter

measurement to evaluate the local and global behavior of the containment test model. The leakage rate measurement shall help to establish the degree of imperviousness at different milestone pressures reached by the test model and validate the available analytical and semi-empirical models for the prediction of the leakage rate with respect to the pressure and crack width data.

All the functional and structural failure modes of BARCOM shall be studied. Benchmarking of the in-house/commercial non-linear finite element codes and inelastic analysis procedures for the postulated design basis and extremely low probable beyond the design basis severe accidents shall be carried out. This study would give an opportunity to address some of the structural code design and safety issues relevant for pre-stressed unlined containment structures.

The present paper describes the features of BARCOM and the activities of design, analysis, construction, engineering and instrumentation shall be presented for the mega size experiment that has been planned under this Indian containment research project.

The FSI seismic analysis for FBR core assemblies (5-1983)

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Introduction

In this paper, A FEM code CASTEM, is used in the analysis. The dynamic responses of one row assemblies under the Horizontal Seismic load are researched. Comparing the results with other similar analyses and the experiments, the model and the method is reasonable and effective. On this basis, considering the Fluid Structure Interaction, the dynamic responses of one row and three row 2D assemblies are analyzed. One single horizontal seismic analysis of 3 row assemblies shows that the interaction between neighboring rows has very small effect on each other. The estimate response characteristics of 3 row models or even the whole core under the single horizontal excitation can be replaced by one center row model.

Analysis model and method

Some pre-analyses on one single assembly are finished before the analyses on multi-assembly. Based on that one center row of assemblies simulated by variable cross section beam elements are selected to be performed one single horizontal seismic analysis. The model is shown as Fig. 1. The shock between the neighboring assemblies is simulated by combination of one gap element. The acceleration-time histories are taken as seismic input data. The modal basis method is adopted. The structural Rayleigh damping ratios and shock stiffnesses between the neighboring assemblies are assumed. Comparing the results with other similar analyses and the experiments, the model and the method is reasonable and effective. On this basis, considering the Fluid Structure Interaction, the dynamic responses of 2D assemblies are analyzed. The modal is showed in Fig. 2. For realizing the effect of interaction between the neighboring rows, 3 neighboring center rows of assemblies are selected to carry out horizontal seismic analysis. The seismic responses have been compared with those for one row model.

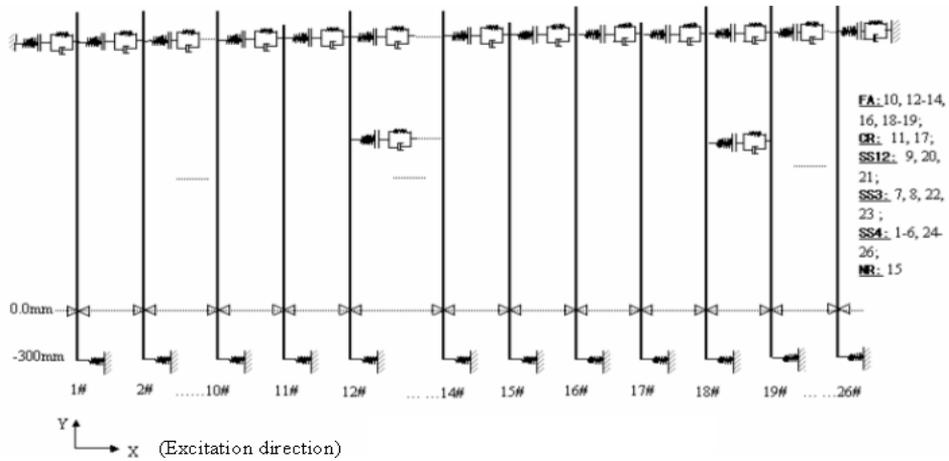


Figure 1. One single horizontal seismic analysis model of one row of assemblies.

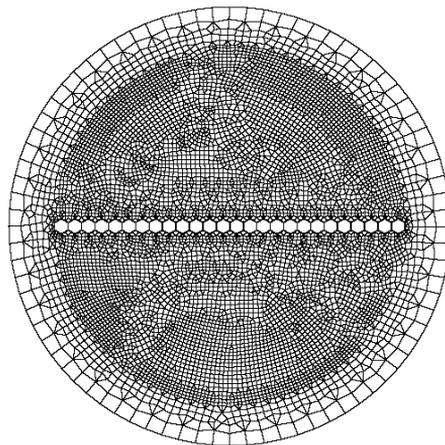


Figure 2. The analysis modal of 2D one row assemblies.

Summary and conclusions

The analysis modal and method of CFR core assemblies are researched, considering the fluid structure interaction and some assumed parameters, the dynamic analysis of one row and three rows assemblies under the horizontal seismic excitation are finished. We can draw a conclusion as following:

5. Modeling, Testing and Response Analysis of Structures, Systems and Components

- a, Comparing the results of analyses with other similar analyses and the experiments, the model and the method is reasonable and effective.
- b, The different type elements are adopted to analyze, owing to the reference coefficient are not same, and the beam element result is bigger than the 2D shell element.
- c, Comparing the results of model consisting of three rows of assemblies with that of one row, effect between neighbor rows is very small. Roughly speaking, we can estimate conservatively the response of three-row model or even the whole core under one single horizontal excitation by using one center row model.
- d, The analysis of 2D structures fluid structure interaction is finished, and the displacement and impact force are calculated, which will offer important indemnification for the farther analysis of all core 3D assemblies.

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The effects of seismic spectrum on seismic analysis (5-1988)

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In this paper, the basic principle of seismic analysis is explained and the factors which affect the seismic response are analyzed. In the seismic computation of regenerative heat exchanger of steam generator blowdown system of Lingao II nuclear power plant, one problem occurs, which is there's only a little difference between the seismic spectrum of the same component, but there is great difference between the seismic response of the same component in different plants, so the effects of the peak change of seismic spectrum and the magnitude of the seismic spectrum on results of seismic analysis are analyzed.

The results show that if the first few frequencies of the component are close to the inherent frequency of the plant, then the modal response of this mode is magnified remarkably, and these modal responses will contribute most to the whole structure response. Also, as to the same component, the principle of the seismic spectrum value change is identical to that of the results of the seismic responses. So this paper can be served as a reference of analyzing and solving those problems concerned with seismic analysis.

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Seismic analysis of primary sodium system components for the loop type fast breeder reactor (5-2000)

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For 40 MWt (13.5 MWe) loop type fast breeder test reactor (FBTR) operating with an unique plutonium rich carbide fuel, seismic analysis the primary sodium system components is reported towards seismic reevaluation purpose. The main components of the primary sodium system are the reactor vessel, two intermediate heat exchangers (IHX) and two sodium pumps. As far as sodium pipings are concerned, the hot pipe lines coming from the reactor joining to intermediate pipe line between IHX & pump, cold line running from pump to 'Y' junction, called 'cullotte' and finally reactor inlet pipe are the main sodium pipings. The main pipe is provided with double envelope, throughout its length. There are hangers attached in the double envelope of the hot pipings and bellows in the double envelopes of the pipelines. At a few locations the double envelopes are welded to the main pipes. At few more selected locations, there are only guides which allow the axial sliding while constraining all the radial directions, between them. Thus, the system is complex because of strong coupling of components, pipelines and double envelopes. Hence, there is a need to analyse them together with appropriate boundary conditions, which need special kinematic relations to be implemented in the computer code. To comply the design code requirements, the analysis is carried out for the dead load, internal pressure and seismic excitations. For the purpose of seismic reevaluation, review base ground motion (RBGM) spectra were generated at the ground level. Subsequently, floor response spectra (FRS) at the primary system support elevations are generated from the seismic analysis of civil structures. FRS generated at the elevation of reactor supporting elevation in two horizontal and one vertical directions corresponding to 5% damping are applied in such a manner to yield conservative results.

The analysis is aimed at to determine displacements and stresses to check the functional and design code limits. For preventing mechanical interactions between main component / piping and their respective double envelopes, the relative radial displacements are limited to gap between main and double envelopes at respective locations. For ensuring the structural integrity of bellows, the effective axial deflections of bellow are limited to the respective

limits prescribed by the bellow manufacturer. Stresses are limited by the primary stress limits recommended by RCC-MR (2002 edition).

While pipelines have 1 D feature, components, especially at the junctions and branch pipes call for 3 D treatment. Addressing these issues, seismic analysis is carried out by following an integrated approach. Finite element method is used for the entire analysis, with the computer code called 'CAST3M' issued by CEA France. The analysis is completed in three steps. In step-1, global analysis is carried out to determine the deflections, forces and moments due to dead load and seismic loadings using straight pipe elements and bends. The deflections are used for verifying the deflection limits. The forces and moments are used for the computation of P_m & (P_m+P_b) , following either step-2 or step-3. In step-2, P_m & (P_m+P_b) are computed using by using the correlations recommended in RCC-MR for the pipes, bends and branch pipes. The correlations for the tees recommended in RCC-MR are used for the branch pipes by assuming that the dimensional restrictions for the fillet radius, etc. are respected, based on which critical branch pipes are identified for the detailed FEM analysis in step-3.

As the first phase, natural vibration analysis is carried out to determine natural mode shapes and associated frequencies, which have been extracted up to 50 Hz. Based on the analysis, it is concluded that the seismic behaviour of components in east and west loops including double envelope are similar. The deflection limits to prevent the mechanical interaction between the main and respective double envelopes are met with comfortable margin. The maximum net axial deflections are found to be less than minimum acceptable values. The stresses induced in components namely, reactor vessel, IHX and pumps including their double envelopes are small. As far as pipings are concerned, the hot lines are critical, particularly the shell nozzle junctions. However, stress limits are met with detailed FEM analysis. The pipe bends including cullotte are meeting the design code limits.

In summary, all the main components in the primary sodium systems in the as built conditions meet the seismic design requirements.

Qualification of creep, fatigue and fracture design of PFBR components based on tests (5-2002)

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In 500 MWe Prototype Fast Breeder Reactor (PFBR), the critical out-of-core components are main vessel (MV), control plug (CP), inner vessel (IV), intermediate heat exchangers (IHX), steam generators (SG) and hot pipelines. Except for SG which is made of modified 9Cr-1Mo (G91), austenitic stainless steel (ASS) is used for other components. The salient structural mechanics features are large size thin walled shell structures, relatively low operating pressure (< 1 MPa, except SG which operates at 17 MPa), high operating temperatures (820 K for hot pool) and large thermal gradients (ΔT of 150 K between hot and cold pool). These components are designed as per French Design Code RCC-MR (1993)¹, for the design life of 40 y. As per the code, the design is done by analysis for which mainly numerical techniques by finite element is followed.

The components will be manufactured indigenously. In order to ensure that the design, analysis, indigenous material and indigenous manufacturing technology comply with the design and construction code rules, tests are carried out on a few important full scale components and mockups having component features such as welds, multiaxiality and stress concentration effects under simulated loading conditions. Particularly in the domain of creep, fatigue and fracture design, a series of tests were conducted in Structural Mechanics Laboratory (SML) with the objectives of qualifying the performance of components in the reactor and the fracture assessment procedure for the FBR application and for demonstrating leak before break (LBB) argument for MV, sodium piping and SG.

This paper highlights the summary of theoretical analyses that have been carried out on creep, fatigue and fracture design of critical components. Subsequently, the paper deals with a few of the experimental investigations that have been carried out essentially to qualify the creep-relaxation behaviour of IHX tube to tubesheet joint, creep rupture strength of SG tubes, fatigue and fracture assessment of SG tube bends and LBB justification of a typical full scale Tee of secondary sodium circuit.

Creep-fatigue and fracture analyses completed for PFBR components are validated by systematically planned experiments. The following are the important outcome of these experiments.

- Rolled and welded tube to tubesheet joints of IHX are stronger than the basic tubes. Hence, in IHX, tubes decide the pullout strength rather than the joint even after relaxation.
- Creep rupture life of SG tubes, extrapolated from the test data using appropriate Larson-Miller parameter, is more than ~ 2000 times the design plant life without accounting for wall thinning due to corrosion.
- A few thermal duty cycles impose stress range in SG bend tubes, more than $3S_m$ limit of design codes. However, tests conducted on 12 tubes at room temperature as per the experimental route recommended by ASME-Section III, reveal that the tubes can withstand more than 80,000 load cycles (minimum). After applying a factor of safety of 8.6, computed as per the code, the allowable number duty cycles is ~ 9300 , which is much more than the design load cycles (~ 860).
- French guide A16 on LBB assessment, yields reasonably accurate prediction of crack propagation and global instability of SG tubes. Hence, A16 rules which are validated mainly for austenitic steel applications, can be applied confidently for the modified 9Cr-1Mo steel. LEFM approach is found to be sufficient for this material.
- For the large size pipe Tees and bends, LBB argument can be applied comfortably. However, buckling should also be considered in the collapse load assessment procedure for the pipe bends and Tees in general, and Tees in particular.
- The creep damage estimation procedure specified in RCC-MR: Appendix A16 which is based on the σ_d approach is critically investigated and improvements needed for A16 were recommended, mainly by application of (i) appropriate multiaxial creep damage criteria, (ii) improved Neuber's rule for predicting elastoplastic stresses and (iii) relaxation of equivalent stresses. The improved procedure predicts the experimental creep initiation life satisfactorily.

Inelastic strain at sliding joint between primary ramp and primary tilting mechanism of prototype fast breeder reactor (5-2004)

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Inclined Fuel Transfer Machine (IFTM) is one of the fuel handling machines of PFBR fuel handling system, which transfers the core subassemblies from in-vessel transfer position (IVTP) to ex- vessel transfer position (EVTP) and vice versa. Primary Ramp (PR) and Primary Tilting Mechanism (PTM) are two important components of IFTM. Both are subjected to high temperature environment, as they are located inside the hot pool of sodium. PR is fixed at the top of the roof slab whereas PTM is fixed on the grid plate. Both are connected with a sliding joint to facilitate the smooth movement of transfer pot. PR and PTM have differential thermal movements w.r.t. each other as they are fixed at two different locations. Movement parallel to their axis is allowed at the sliding joint and hence no restriction of axial differential thermal movement is considered. Due to differential thermal movements of their support locations there will be significant movement perpendicular to their axis which is restricted in the sliding joint. Due to this restriction, bending of PR or PTM may occur and localized deformation of the component at the edge inside the sliding joint may occur.

To study the deformation of edges of the PR and PTM inside the sliding joint, analytical and experimental investigation of a simplified sliding joint is carried out at room temperature. The model of sliding joint is subjected to pure bending moment. The elasto-plastic analysis of the model is carried out to understand the local deformation, ovality, plastic strain etc.

To validate the analysis methodology analytical prediction and experimental observations are compared. The results are matching well between analysis and experiment. The same methodology of applied boundary conditions has been applied to PFBR IFTM where there is a sliding joint between PR and PTM. Elasto-plastic analysis has been carried out to find out the maximum inelastic strain at the location of the sliding joint for 50 mm relative displacement between primary ramp and primary tilting mechanism. The maximum local strain obtained is 0.12% which is well within allowable limit of 5% (local). The ovality on the primary ramp after applying relative displacement is found to be negligible. More details of analytical and experimental investigations will be presented in the paper.

Structural integrity assessment of DHX under CDA pressure loading (5-2006)

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The reactor assembly (RA) of Prototype Fast Breeder Reactor (PFBR) consists of various components like Main vessel (MV), Inner Vessel (IV), Intermediate Heat Exchanger (IHX), Decay Heat Exchanger (DHX) and Primary Sodium Pump (PSP) and Reactor Core etc. The entire primary sodium circuit is housed within the MV, which contains about 1100 t of radioactive primary sodium. Hence the integrity of the entire reactor assembly is to be demonstrated even for most extreme and unlikely events like earthquake and Core Disruptive Accident (CDA). Among these loadings, CDA resulting from core meltdown is a very low probability event in an FBR and hence it is considered as beyond design basis event. Nevertheless as a defense in depth approach, the structural integrity of the RA is demonstrated through complex numerical analysis, which involves calculations of fluid transients, structural response and fluid structure interaction effects [1, 2] and also through experiments on scaled down models. Experiments simulating the CDA conducted on 1/13th scale model of RA demonstrated the structural integrity of major components like IHX, DHX and PSP under the extreme transient pressure loading developed as a result of CDA. This paper give the details of the numerical analysis carried out on a 1/13th scale model of the DHX which is used in the CDA mockup studies under transient pressure loading which simulates the pressure distribution obtained form the mockup study and response have been compared with the prototype which undergoes a pressure loading expected during a CDA event predicted from the CDA analysis.

The modeling and analysis of both the prototype and scaled down model of the DHX has been done using finite element code CAST3M. Fast transient analysis of reactor assembly of PFBR for an energy release of 100 MJ has been carried out using in-house code 'FUSTIN' and the transient loading in the vicinity of DHX has been extracted which is used for the analysis of DHX prototype. Similarly pressure pulse in the vicinity of DHX model has been obtained from experiments simulating the energy release of 100 MJ for the 1/13th scaled down model of reactor assembly using explosives.

In order to verify the scaling laws maintained for the model and prototype and to verify the dynamic amplification during the transient loading condition, natural frequency analysis has been carried out for both DHX model and prototype and frequency of bending mode has been compared (5.1 Hz and 67 Hz for prototype and model respectively).

Response has been calculated for the transient pressure loading for the model and prototype of the DHX based on Newmark time integration. Maximum response viz displacement and stress have been extracted at different location of interest. From the analysis it was found the stresses developed in the prototype under actual CDA loading condition is less than that of the model under the test condition there by proving that the results obtained from the mockup studies of 1/13th scale model of RA under simulated CDA loading conditions is conservative. It can be noted that these stresses are the most pessimistic estimation without considering the communication effect of fluid inside and outside of the shell. In real case the net stresses will be considerably less than these values due to the differential pressure acting on the shell. The realistic stresses acting on the DHX model and prototype therefore can be obtained by multiplying the stresses with the reduction factor ' α '.

Dynamic response analysis has been carried out for the 1/13th scale model as well as prototype of the DHX of PFBR to determine the dynamic response and stresses under transient pressure during a CDA event. Transient loading acting on the DHX is idealised to be a triangular pressure pulse. Along the circumference of the shell pressure is assumed to have a cosine distribution. Analysis results show that the maximum stress in 1/13th scale model is higher than that of the prototype. So it can be concluded that the results of the mock up study conducted on 1/13th scale model of the RA gives a conservative estimate of the actual CDA loading. It is worth mentioning that DHX was not damaged in the experiment and hence the structural integrity of DHX is ensured for the CDA loading for PFBR.

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Experimental compression behavior of Stiffened Steel-Plate Concrete (SSC) structures under compressive loading (5-2008)

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Introduction and background

The importance of the research on the steel-plated concrete (SC) structure is now ever increasing in nuclear structures. The SC structures can be applied to area such as the first primary shield wall in nuclear structures to reduce the construction period dramatically. It is necessary to understand the compression characteristic of the new stiffened steel-plated concrete (SSC) structure under the compression loading. Compression characteristic is one of important key factors among design variables. This paper introduced and evaluated the compression characteristics of the stiffened SC structures under vertical monotonic loading.

Aim of work

The SSC structures were consisted of headed stud bolts, concrete and steel plates stiffened by W-shaped section. The primary aims of the work are to find and suggest the empirical compression equation considering the creep effect of concrete in the SSC structures. The maximum compressive forces resulted from the experimental test results were also compared with the one of theoretical finite element method (FEM). To understand of buckling behavior of the stiffened steel plates in SSC structures are the secondly important aim of this paper. The buckling patterns of the steel plates are compared with the results from the eigen-value analysis using FEM. The tension stresses of the steel plates were directly measured from the experimental works during the compressing

loading period. At the same time, the initial stiffness and the effective length factor were evaluated in this paper.

To achieve those phenomena of the SSC structures, three types of B/t ratio were selected. The B/t ratio stands for the ratio between the pitch of headed stud bolts and thickness of steel plate. The three types of B/t ratio were 25, 33, and 50 that represent inelastic and elastic stress behaviors. The compressive strength of concrete was 42 MPa and the two types of steel plates are SM 490 and SS 400 in the test. The thickness of steel plates was 6 mm. The diameter and length of the headed stud bolt are 9 mm and 71 mm, respectively. The compression loading was loaded monotonically to the top of the specimen using the Universal Testing Machine, 10000 kN. The LVDT and strain gauges are installed to measure stresses at many locations.

Essential results

The experimental works were carried out to suggest the empirical estimation of compression force including creep and shrinkage effects. First of all, the buckling shapes of the steel plates after the completion of the test were very similar the one of FEM analysis, Fig. 1.

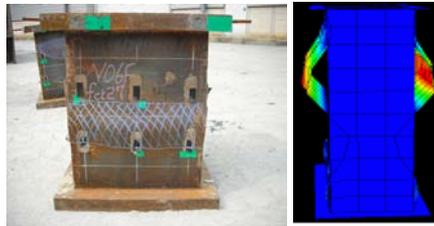


Figure 1. Buckling of Steel Plates.

The compression equation can be estimated by Eq. (1) through (2) and well agreed with the experimental results.

$$P_n = F_{cr} A_s + F_{ys} A_{ss} + 0.85 f_{ck} A_c \quad (1)$$

$$F_{cr} = (1.55 - 0.125\lambda - 90\varepsilon_n) F_{yp} \quad (2)$$

In equation (2), the buckling stress of the steel plate included the creep and shrinkage effects. The quantitative value of compression force for the SSC structures well agreed to the suggested Eq. (1) in average. The initial stiffness of the SSC structure in elastic range was very similar to the one from FEM analysis. The buckling stress measured at the steel plate directly to acquire the initial stiffness.

Conclusions and discussions

- The buckling shapes of the steel-plated concrete structure stiffened by W-Shape section placed through the lateral direction. The patterns are well agreed to the one of FEM analysis.
- The empirical compression equation was suggested and the compression forces were well agreed to the resulted force from the suggested equation.
- The increase of the yield strength of the steel plate does not affect on the increase of the maximum strength in most cases.
- The calculated effective length factor between the headed stud bolts was around 0.5 for the SSC structures.

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Developing a numerical model to describe the mechanical behaviour of die-formed expanded graphite for valve stems sealing (5-2009)

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Introduction

Stuffing box packings are very widely used for sealing valve stems, in the chemical and petrochemical industries as well as in nuclear industry processes. Following the banishment of asbestos in the 80's, compressed exfoliated graphite has become the most commonly used material for the designing of these seals. However, little knowledge has been gathered so far on the precise mechanical behaviour of this material.

As part of a project that aims to simulate the operation of nuclear power plants valves in nominal conditions, the development of a numerical model that could depict the main characteristics of the mechanical behaviour of CEG is studied. In consideration of the main objectives of the project, we do not intend to build a complex representation that would give a detailed account of the “microscopic behaviour” of CEG ; we are more interested in a simplified model that could predict the response of the packings, regarding the loadings they endure in service.

Experimental approach

Compressions and discharges tests are performed on CEG samples on a 10000 daN hydraulic press. These samples are cylinders of diameter 32 mm, prepared by axial compaction at a given pressure of a graphite “ribbon” of thickness 25 mm wrapped in a die. The tests are performed on the samples whether placed in the die or free to expand in the radial direction. In the first case, we measure the axial load and displacement imposed on the sample, as well as the lateral strains of the outer wall of the die (from which we deduce the pressure applied by the sample on the inner wall of the die). In the second case, we measure the axial load and displacement, as well as the radial expansion of the sample.

Finite element analysis

Based on the geometry of the test rig, a 2D axisymmetric finite elements model was developed using Code_Aster. The Cam Clay model was chosen to represent the mechanical behaviour of CEG. This model, more commonly used for soil mechanics, requires the knowledge of eight parameters, four of which being accessible through the experiments. Two parameters have to be fit numerically, from which we deduce the two last parameters.

The friction between the stem of the valve and the packing placed around it is of first importance regarding the operability of the valve. The pressure applied on the packing must be sufficient to prevent leakage, but one has to ensure that this pressure makes it possible for the actuator that drives the stem to open or close the valve whenever required. Thus, the Cam Clay model was further investigated modelling a whole packing placed between a stem and a stuffing box. The model includes manoeuvres of the stem, contact and friction between the packing and the stuffing box on one side, between the packing and the stem on the other side. The results are compared with previous experiments carried out in our labs.

Main results

Compressed exfoliated graphite (CEG) shows a very non-linear elastic-plastic behaviour, resembling strongly that of certain kind of rocks, like clays, or cohesionless soils, like sands. This similarity can partly be explained by the porous nature of the material. The key characteristics of this complex behaviour are: non-linear elasticity, apparition of permanent deformations at low levels of pressure, linear hardening at higher pressures, and accommodation. As seen for clays, this behaviour is usually well represented by such models as the Cam Clay model.

The numerical results obtained with the Cam Clay model show a good agreement with the “in die” tests, although the model does not account for the accommodation. Furthermore, the computed axial forces required to drive the stem are close to those measured experimentally, which reinforces our confidence in the model.

Conclusions

The Cam Clay model is a first step towards a reliable representation of the mechanical behaviour of CEG. The FE analysis based on this model shows quite a good agreement with the first “static” tests performed on CEG samples. The model has now to be validated through complementary “dynamic” tests, which will provide more information on the exact values of the friction ratios between

the packing and the stem. Future work will also investigate the influence of temperature and the presence of fluid.

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Tests on reinforced concrete slabs with pre-stressing and with transverse reinforcement under impact loading (5-2015)

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Introduction

Current empirical formulae for design of concrete elements under impact loading are based on tests carried out on reinforced concrete elements with longitudinal reinforcement and they are applicable only to such elements (References 1, 2). The influence of pre-stressing or transverse reinforcement is not taken into account. However, the code provisions (References 3, 4) for punching resistance of concrete elements under conventional loadings take into account beneficial effect of these parameters. In addition, the design acceptance criteria according to these formulae are based on the damage of a concrete element. More precisely, they are based on the visual damage of the rear side of the concrete element and formulated as scabbing of the concrete surface or missile perforation through the element. There is no quantification of the damage in terms of width and depth of the scabbed area or in terms of deflection of the element.

Aim of the work

The aim of this test campaign (Reference 5) carried out at VTT testing facility in Espoo, Finland, was to assess the influence of longitudinal, bidirectional, pre-stressing and transverse reinforcement, separately and combined, on punching resistance of concrete elements under impact loading. Six tests presented in this paper were carried out on low aspect ratio concrete slabs ($l/h < 10$; l is the slab span, h is the slab thickness) under medium velocity (around 100 m/s) hard missile impact. The variables were: the level of pre-stressing introduced using threaded bars and the transverse reinforcement in form of T-headed bars. The reference test was carried out on a reinforced concrete slab with longitudinal reinforcement only. In two following tests, in addition to the longitudinal

reinforcement, pre-stressing was introduced using threaded bars. The slabs were pre-stressed introduced in both longitudinal directions. The level of pre-stressing was 5 MPa in one slab and 10 MPa in the other. In three following tests the same slab design was used with additional transverse reinforcement in form of T-headed: one slab was reinforced concrete slab with longitudinal and transverse reinforcement and two remaining slabs with mentioned pre-stressing levels and transverse reinforcement. Two last tests were performed to asses the combined effect of pre-stressing and transverse reinforcement.

Essential results

The results are presented in terms of scabbed concrete area and permanent deflection for each tested slab. It was observed that the punching resistance of pre-stressed concrete specimens (without transverse reinforcement) was lower than the reference reinforced concrete specimen, which is not consistent with code provisions for conventional loadings. However, the difference in damage between two tested levels of pre-stressing (5 MPa and 10 MPa) was not significant. On the other hand, the transverse reinforcement, in form of T-headed bars, increases punching resistance of concrete elements under impact loading. Therefore, the results are consistent with code provisions for punching resistance under conventional loadings. The transverse reinforcement combined with pre-stressing significantly improves punching resistance of concrete elements for tested missile parameters. However, it was observed that combination of these two parameters modifies the failure mode. A punching cone, which is a current failure mode for target (concrete slab) and missile characteristics used in this campaign, is reduced to a punching cylinder with a diameter comparable to the missile diameter. Further tests are needed to evaluate ultimate resistance of slabs with pre-stressing and transverse reinforcement.

Conclusions

The tests performed on concrete slabs under impact loading showed that introducing pre-stressing in slabs with longitudinal reinforcement does not increase (even decrease) their punching capacity. Introducing transverse reinforcement in form of T-headed bars have beneficial effect related to the punching resistance and the transverse reinforcement combined with pre-stressing increase significantly the punching resistance for given slab and missile characteristics. Further tests are needed to asses the benefits of these two parameters related to ultimate resistance of concrete slabs under impact loading.

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Protection of seismic structures using variable FPS typed TMD system (5-2016)

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Introduction

Passive tuned mass damper (TMD) is a widely used control device in reducing structural dynamic responses due to environmental excitations such as winds, earthquakes, and man-made loadings [1–3]. A TMD system consists of an added mass together with properly designed spring and damping elements that increase the controlled modal damping for the primary structure. By attaching a TMD to the structure, the vibration energy of the structure can be transferred to the TMD and dissipated by the damping mechanism of the TMD. The design and application of the traditional TMD (linear typed) systems are well developed, but nonlinear typed TMD systems are still developing.

It is known that energy dissipation technology using friction mechanisms is an effective means for vibration mitigation of seismic structures. Some researchers suggested the application of the friction pendulum system (FPS) typed TMD system. For a FPS typed TMD, the shape of the sliding interface is made spherical, so that the TMD gravitational load applied on the slider will provide a restoring stiffness for the TMD to return to its original position after an earthquake. This restoring stiffness can be determined by the radius of curvature of the spherical sliding surface, which is a fixed value. Compared with the traditional TMD, the FPS typed TMD has following advantages: (1) Mass independent self re-centering ability. (2) Energy dissipating by friction mechanism. (3) No need to install an extra spring and damping device. (4) No need to install a suspended mechanism, the space demand is thus much smaller.

On the other hand, since the slip force is a pre-determined fixed value, the FPS typed TMD will start to slip (be activated) and dissipate seismic energy, only when the friction force exerted by the seismic TMD motion exceeds the constant slip force. A FPS typed TMD is not different from an added mass of the primary structure, if it is not in the slip state. In other words, the TMD may lose its tuning and energy dissipating ability when the TMD is in its stick state. The determination of the slip force level is always a key issue in the design of structures with FPS typed TMD. In the design practice, an earthquake with a predicted intensity is usually assumed, and the passive FPS typed TMD is then

designed according to this intensity. As a result, the TMDs may perform well under earthquakes of the predicted intensity, but may not perform well under others.

Aim of the work

In order to overcome the problems mentioned above, a variable FPS typed TMD system is evaluated in this paper. The friction force of the variable FPS typed TMD system is controllable. A non-sticking friction (NSF) control law [4, 5], which is able to keep the TMD activated and in its slip state throughout an earthquake with arbitrary intensity, was conducted. The disadvantages of the passive FPS typed TMD system can be improved.

Conclusions

The possibility of using a variable FPS typed TMD system for the protection of seismic structures is considered in this study. The performance of the variable FPS typed TMD with NSF controller for protection of seismic structures was investigated numerically. The numerical results show that the proposed variable FPS typed TMD system can improve the disadvantages of the traditional passive friction typed TMD system.

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Crack opening in steam generator tubes submitted to an internal pressure: experimental and numerical modelling (5-2019)

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In the framework of a systems reliability project in steam generator (SG) which aim is to estimate leak rate in SG tubes, we study the longitudinal crack opening modelling in the hard rolling transition zone of a SG tube submitted to hydrotest conditions.

To reach this goal, an experimental system (see figure 1) has been specifically developed in order to measure crack opening area in a SG tube [1]. Hard-rolled and virgin tubes with one longitudinal crack are tested. The tubes which are partly rolled are embedded in a metallic thick-walled cylinder whose interest is to limit the expansion of the tube due to the rolling and to the internal pressure applied. It represents the tubesheet's effect in SG. The first experimental results show that for low loadings and small crack sizes, the crack opening behaviour is reversible. Beyond some crack size and a certain internal pressure value, the crack keeps one residual opening. Besides, it has been shown that for cracks with a sufficient length, the crack opening does not depend on the zone where it is (straight section or roll transition zone).

In a second step, finite element calculations using finite element code Code_aster [2] have been performed in order to estimate the crack opening in hard-rolled SG tube. The simulations take into account residual stresses due to the hard-rolling and the effect of the tubesheet on the tube behaviour. Numerical results are compared with the experimental ones and a simplified model is proposed in order to estimate the crack opening of hard-rolled tubes in SG.

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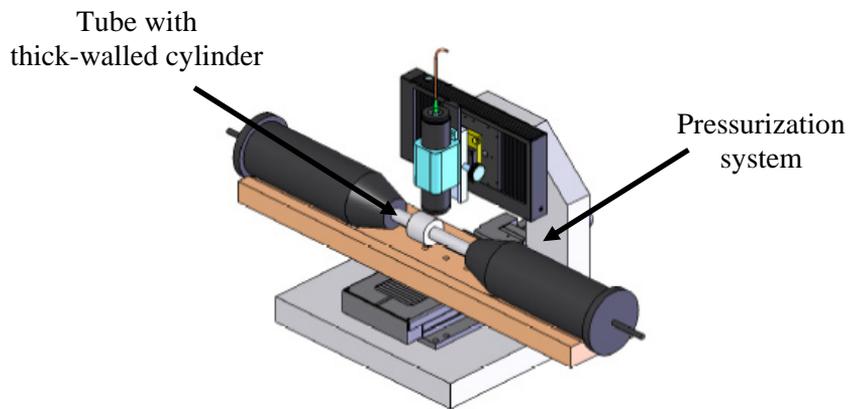


Figure 1. Experimental system specifically developed to measure crack opening area in SG tubes. An original pressurization system allows to applied a internal pressure to the hard-rolled tube.

Numerical simulation of plain concrete fracture experiments with fictitious crack model (5-2035)

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Fracture simulation of quasi-brittle material like concrete cannot be carried out in a generalized manner, unless the material constitutive parameters are identified and methods for their experimental determination are specified. A variety of numerical methods to predict crack formation and crack propagation for concrete has been proposed in recent few years. This paper presents case studies on numerical simulation of fracture behaviour of plain concrete beam specimens of grades used in Indian nuclear containments. Three point bend beam specimens with different sizes and notch parameters are simulated using the fictitious crack model (FCM). This has been accomplished with computer program for nonlinear finite element analysis of reinforced concrete structure named ATENA version 3.30. The experimental test beams are of M45 grade concrete (Characteristic cube strength). The ATENA formulation of constitutive relations is considered in the plane stress state. A smeared approach is used to model the material properties, such as cracks or distributed reinforcement. This means that material properties namely the tensile strength and fracture energy are appropriately defined within continuum mechanics framework for crack initiation and its subsequent propagation upto the final collapse for the general finite element analysis defined for a material point are valid within a certain material volume, which is in this case is associated with the entire finite element. The constitutive model is based on the stiffness and is described by the equation of equilibrium at a material point. The specimen details are as follows.

Beam Nomenclature	Specimen sizes (mm)			Notch depth (mm)	Thickness of notch (mm)
	Length	Depth	Width		
D1P20UB01	376	94	47	23.5	2.0
D2P20UC02	752	188	94	62.0	2.0

The load vs CMOD (crack mouth opening displacement) obtained from experiment and numerical simulations are compared for a typical D2 specimen identified above in Fig. 1. These studies shows good agreement between the experimental and numerical simulation, which confirm the constitutive material models used in the code ATENA is powerful to capture the fracture process of concrete which is also size dependent.

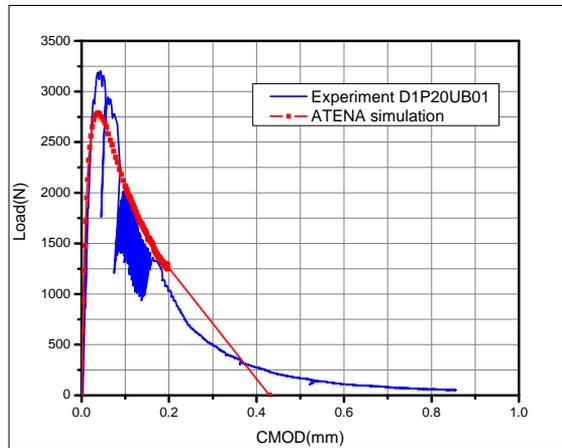


Figure 1. Comparison of Experimental vs Numerical Simulation of Fracture Process of D2 specimen.

Study of incident water hammer in an engineering loop under two-phase flow experiment (5-2042)

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Introduction

Water hammer is a common phenomenon when sub-cooled water flow in pipes or other containers filled with steam or steam-water mixture. They also appear as the consequence of fast closing or opening actions of valve or break in the pipelines conveying single or two-phase flow. In the latter case, shock waves are generated. In all the cases however, strong dynamic stresses are induced in the wall of the closed system or equipment. These stresses are very different from the stresses caused by static loading. The lack of experimental data obtained for well-defined geometric boundary conditions is a significant obstacle for validation of codes on fluid-structure interaction problems.

Two incident of loud sound, with in a span of four months was reported by operators of a full-scale engineering loop erected especially for carrying out experiments on loss of coolant accident and natural circulation. As the loop was required to cater for large number of engineering experiments it was instrumented with vibration and shock transducers and capture sound wherever it happens in the loop. Condition was recreated for steam condensation induced water hammer in the loop and the loop parameters were closely studied to estimate the arrival time and time delay between multiple shock.

Engineering loop

The loop mainly consists of a vertical fuel channel simulator (FCS), steam drum (SD), down comer, header, feeder and riser. The FCS is made of 54 electrically heated fuel rods cluster [1]. The loop simulates the elevation, pressure, temperature, velocity and time scales of prototype to carry out study on thermal hydraulic issues of natural circulation of main heat transport system (MHTS). The heat sink for the test facility is provided in the form of a secondary coolant circuit comprising of jet condenser, pool boiling coolers and feed pump.

The design pressure and temperature of the loop are 100 bars and 315 deg centigrade.

When the steam is let out from the steam drum, it flows down to jet condenser where it encounters jet of sub-cooled water and condenses. The condensed water is pumped back into the steam drum. From lower bottom of the steam drum, the down comer (26 meters) conducts hot water to a header and from header the water is fed to FCS through a feeder (16 meters). A separate loop feeds sub cooled water to jet condenser. A pool boiler connected in this circuit helps in maintaining spray water temperature.

Incident of water hammer

When the loop was in operation at 2% power (52 KWth) for shut down cooling experiments, the operators heard a loud sound. The heaters of FCS were switched of and the loop was brought to low temperature and pressure. After about a week of operation, the loud sound was again heard. After confirming the incident with all the crewmembers of the loop, it was decided to investigate the source and location of loud noise.

Possible mechanisms

Three possible mechanisms were considerer.

- Water hammer due to fall of big water slug in the tail pipe.
- Condensation induced water hammer in the steam line. [2], [4]
- Water hammer due to sudden condensation of steam in the jet condenser.

Recreation of condition for water hammer

The loop was fully instrumented to capture pressure, temperature, flow rate and fluid levels in various sections of the loop in addition to vibration sensors to locate the origin of water hammer. The loop was operated to simulate the condition of all the three possible mechanisms.

When steam line was opened for the steam to flow from SD to JC and spray water flow was started. The spray water temperature at the start of spray was 43 deg centigrade and the flow rate adjusted 100 LPM. With in a delay of 2 minutes, a series of five shocks were recorded one after the other. Every subsequent shock was felt with increasing intensity. After the 5th shock, the JC spray was stopped and the loop was brought to zero power.

Signal Analysis and role of super heat in the steam for time delay between the shocks.

An attempt was made to explain the reason for time delay different between the events through energy balance equation and the time required to quench the quantity of super heat in the steam was estimated.

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Advanced analysis of gasketed pressure vessel closure systems (5-2043)

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Introduction/background

The pressure vessels have closure systems allowing visiting the inner of the component or used for maintenance activities during its operation. For these closure systems, in particular for nuclear components, their mechanical resistance and their leak tightness must be demonstrated.

Aim of the work

The aim of the work was to develop a method to model completely a closure system, including the studs, the gasket, the nuts, the flange or cover and the part of the pressure vessel, for a 3-dimensional non-linear analysis taking into account the contacts between the gasket and their seating surfaces as well as the non linearity of the gaskets and the tightening preload of the studs.

Main results

Therefore, with such a complete representative non-linear 3D model we can:

- (a) perform the stress analysis in all parts of the assembly for all the operating conditions (temperature and pressure) and demonstrate their mechanical resistance (stress limit, fatigue, fracture, etc.)
- (b) analyze the behaviour of the contact area between the gasket and its seating surfaces, for all the operating conditions. By doing so, we can assess the history of the compressive forces on the gaskets in order to demonstrate that the necessary conditions for the leak tightness are met,

particularly the metal-to-metal contact for multi-ring (metallic-graphite-metallic) gaskets and as an important consequence, the re-calculation of the necessary tightening preload applied to the studs by the tensioner. This method was used, for example, for the behaviour analysis of man holes, hand holes and eye holes of nuclear components.

Summary/conclusion

The results are meeting the requirements of the nuclear regulations and standards.

This method is applied to openings of nuclear components for the justification of leak tightness of the in-service operating conditions.

In particular, the method was applied for the calculation of the initial tightening load performed in hydraulic test condition and for which no leakage was observed. This test can be considered as a successful scale test.

Update and comparative study on seismic wave incoherence in soil-structure interaction (5-2048)

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Introduction

Spatial incoherency of strong ground motions has the effect of lowering translational earthquake input motion at building foundations but introducing rocking and torsional behavior.

A methodology to incorporate spatial incoherence of seismic ground motions in seismic soil-structure interaction (SSI) analysis and practical engineering application guidelines were developed in a previous study (Tseng and Lilhanand, 1997). Along with the methodology, a computer program module INCOH was also developed which works together with the SSI analysis computer program SASSI. The combined SASSI-INCOH program is capable of: (1) calculating the transfer function vector relating the incoherent motion vector to the motion at a reference station conforming to a prescribed coherency function for motions within the structural foundation, (2) modifying the coherent free-field motion vector into the incoherent free-field motion vector, and (3) calculating the SSI response of structures with arbitrary foundation configurations due to the incoherent free-field input motion in the same way as the SSI analysis for the conventional coherent ground motion input (ICEC, 1998).

Aim of the work

This study describes an update on development of the computer program INCOH to incorporate: (1) the empirical spatial coherency models based on the Pinyon Flat array data at a hard rock site, and (2) a coherency model based on an extensive set of array data at soil sites (Abrahamson, 2007).

Results and conclusions

The new development is illustrated with an SSI study of a coupled Auxiliary and Shield Building (ASB), Steel Containment Vessel (SCV), and Containment Internal Structure (CIS) of a nuclear power plant. The coupled structure complex studied corresponds to a combined nuclear island structure of the AP1000 nuclear power plant located on a rock site. The results of this study are compared with a recent publication (Short, Hardy, Merz and Johnson 2007). The comparison shows very good agreement. Results of the study also serve to demonstrate the effect, especially in the high frequency range, of incoherence of seismic input motion conforming to the newly updated spatial coherency models for rock sites on the SSI responses of the typical new generation of nuclear power plant structures.

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Effect of geometrical defects and cracks on the collapse of heat exchanger U-bent tubes submitted to external pressure (5-2049)

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The tube bundle of heat exchangers or steam generators used in nuclear power plants is very often composed of U-bent tubes. These tubes are generally characterized by low diameter to thickness ratios (D/t) and high strength materials. D/t ratios as low as 10 to 20 are currently considered. In some cases, it is possible for the tube bundle to be exposed to a high external pressure, so that an adequate margin against buckling is an important design criterion.

For such a range of D/t , the collapse under external pressure is determined by the inelastic behaviour of the tube material. Moreover, the real geometry of the tubes has to be considered: during the manufacturing process, initially straight tubes have to be bent, and their cross-section becomes more or less elliptical in the curved part. The assessment of the true collapse pressure is quite problematical, due to the presence of both geometrical and material nonlinearities. The buckling pressure of a tube can be estimated conservatively using pressure vessel codes, such as ASME Boiler and Pressure Vessel code [1], RCC-M [2] (French nuclear code) or the German code [3]. In general these design codes yield to conservative results, because their most common feature is to provide a simple calculation formula, which of course cannot account for the complex behaviour of an elastoplastic, imperfect, U-bent cross-section.

The most complete and reliable procedure for calculating the collapse pressure of the oval tube is to perform geometrical and material non linear finite element analysis, knowing that both nonlinearities interact in the problem. Extensive numerical studies have been conducted to clarify the buckling of thick cylindrical tubes submitted to lateral pressure or hydrostatic pressure. Effects of geometrical initial imperfection (initial ovalization) are considered for different amplitudes and in both configurations, the curved pipe configuration or the straight one. Different strain hardenings are considered to gauge the effect of material law on the collapse behaviour. In addition, the effect of other imperfections, such as local wall thickness variations associated to corrosion or wear against tube supports, cracks due to stress corrosion, are examined in the light of the strong dependence on the inelastic behaviour associated to material properties and the presence of initial ovalization.

Considering this large parametric numerical study conducted with different FE codes (CAST3M, ABAQUS, Code_Aster), the methodology of the design and recommendations are proposed. The obtained design curve is then compared to

current design codes. Finally, this study establishes clear and accurate design criteria for the collapse of tubular structures under external pressure for heat exchanger applications.

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Study of liquid dispersal from a missile impacting a wall (5-2050)

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Fuel release and spread from impacted projectile are of interest for the determination of fuel spread and fire risk following an airplane crash on a structure, and for applying and validating simulation techniques. So far, little representative experimental information can be found from the literature. Thus, liquid dispersal processes have been studied at VTT in the selected medium-scale IMPACT tests where deformable steel or aluminium projectiles are filled with water and impacted on a solid concrete wall or a steel force plate [1], [2], [3], [4]. Most of the “wet” missile tests have been conducted using a cylindrical missile, but in few tests a more prototypical 3-D projectile which consists of representative fuselage and wings has been used.

The earlier paper [5] concentrated on the measuring methods and procedures used in the liquid dispersal study, and the main results of the preliminary simulations of liquid spread using the Fire Dynamic Simulator (FDS) computer program. Now the focus is on the major experimental findings of liquid release and spread from ruptured missile impacting a wall. Related measuring procedures are also dealt with. The main parameters of the liquid phenomena measured are the velocity of the liquid front coming out from the ruptured projectile, spreading angle and direction of liquid release, extent of liquid spray from the target, water pooling area on the floor, and droplet size of the liquid spray.

The test results show that the initial velocity of liquid spurting out from a missile may be much higher than the missile impact velocity, but the speed of droplets of the liquid spray slows down rapidly due to atomization processes and relating drag.

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Structural evaluation of drop load effects on buried structures (5-2057)

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Replacement of steam generators is likely to happen at least once during the service lifetime of pressurized-water reactors. The replacement of old steam generators requires the use of large rigging systems for lifting and moving of the old steam generator out the containment structure and of the new steam generator into the containment structure. These operations are generally performed over an area adjacent to the containment structure where many buried structures and utilities are located. Safe load paths are planar regions determined for each lifted component during the various stages of lifting operation so as to avoid damage to the buried structures to the extent that would compromise functionality of the critical buried structure, in case of an accidental drop of a lifted component. The critical buried structures are those structures that are needed to remain functional during the generator replacement operation, such as discharge and intake pipelines and electrical instruments and control panels related to cooling system of the fuel elements.

This paper describes an approach that includes a combination of dynamic finite element analyses and structural evaluation of the buried structures, to perform structural evaluation for drop loads occurring during steam generator replacement and arrive at final load paths for safe lifting operations.

The procedure is based on development of a three-dimensional dynamic soil-structure finite-element model, using explicit transient finite element analysis. The source of elastic wave resulting from impact of the drop load and propagation through the soil media is simulated through low velocity impact of objects on the free surface of foundation or soil. Energy dissipation is accounted for through Rayleigh soil damping and nonlinear response of the dropping objects, the targets and the buried structures. The transfer of dynamic analysis results to the standard structural analysis software for structural evaluation is discussed. A case study is presented in which the structural safety and the extent of damage and its effect on the ability to function of the underground prestressed concrete cylinder pipelines, ductile iron pipelines, and of a reinforced concrete wall is assessed for drop components from a nuclear power plant steam generator replacement lifting operation. Structural evaluation of the prestressed concrete cylinder pipe was performed in both circumferential and longitudinal directions based on limit states criteria used for the design of the pipe using serviceability, damage and different modes of partial or total failure using

AWWA C304 (2007) and AWWA M9 (2008). Structural evaluation of ductile iron pipe was performed in both circumferential and longitudinal directions based on yielding and buckling limit states using AWWA C150 (2002). Structural evaluation of the reinforced concrete wall was performed using ACI 318 (2005) and ACI 349 (2006) and considering damage and was based on evaluation of strains and the extent of cracking and deflection of the cracked wall.

In conclusion, a procedure based on finite element analysis of soil-structure interaction is developed for structural evaluation of buried structures subjected to drop load impact effect in both Defueled and Power Operation Modes. The procedure is used to classify the damage levels to buried structures as: “Code Compliant”; “Not Fully Code Complaint, but Undamaged”; “Damaged, but Functional”; or “Failed,” and is used to verify, modify, and/or provide new safe load paths for lifting operations to avoid failure of the structural systems that need operate during accidental drop loads. The procedures developed herein are applicable for the evaluation of the effects of the impact of the accidental drop loads during steam generator replacement on all types and forms of buried structures.

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Experimental and numerical simulation of radiolysis gas detonations in BWR exhaust pipes and mechanical response of the piping to the detonation pressure loads (5-2062)

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Introduction

Radiolysis gas ($2\text{H}_2+\text{O}_2$) can accumulate in BWR steam piping in case of steam condensation, with an ensuing detonation of the radiolysis gas being the likeliest cause of a pipe and/or valve rupture. In the German Reactor Safety Commission (RSK) recommendations for radiolysis gas control in BWR installations it is demanded to determine the reaction pressure for the highest possible radiolysis gas concentration that could arise. In the current work we consider a typical BWR exhaust pipe, which connects the high pressure steam piping with the ambient atmosphere, under the following “worst case” scenario: (a) accumulation of radiolysis gas in an exhaust pipe, (b) fast valve opening to the high pressure system with steam at 70 bar, and (c) adiabatic pressurization of the radiolysis gas by the steam. Taking into account a water mirror level of 6 m from the open end, this leads to an equilibrium state of 20 bar pressure and 602 K temperature for the pressurized radiolysis gas.

The main purpose of the current work is an experimental and numerical evaluation of the maximum pressure load plus the integrity of the BWR exhaust pipe in case of a detonation of the pressurized radiolysis gas.

Experiments

Detonation experiments of radiolysis gas were performed in a real scale exhaust pipe of 12.25 m length made from stainless steel Nr. 1.4541. The tube consisted of two parts: (i) with 510 mm outer diameter and 14 mm wall thickness, and (ii)

with 419 mm outer diameter and 20 mm of wall thickness. To reproduce the "worst case" conditions for radiolysis gas after the steam pressurization we used a radiolysis gas mixture at 10 bar and 293 K which energetically (with respect to the pressure load) equals the pressurized BWR mixture at 20 bar and 602 K. The inert part of the tube was filled with nitrogen instead of steam to enable experiments at ambient temperature. The radiolysis gas was included in a thin shell vessel to separate it from the inert gas, with ignition at a metal plate side used as an imitation of the gas-water interface in real conditions. 16 strain gauges were installed along the tube to measure longitudinal and circumferential deformations under the detonation process, and dynamic pressure loads were measured using 4 pressure transducers installed on flanges.

Four experiments were performed: at 1.6, 5.0 and two tests at 10.0 bar. The experiments showed that maximum deformations occurred at the end of the radiolysis gas vessel. The maximum dynamic strain was measured to be 0.75% for the radiolysis gas detonations at 10 bar, with the maximum remaining deformation, using strain gauges and direct measurements, being about 0.15%. This means that the exhaust tube remains intact even under this "worst case" detonation scenario of the pressurized radiolysis gas mixture.

Numerical simulations

To simulate detonation of the radiolysis gas mixture at 20 bar and 602 K with steam as an inert gas, the 3-dimensional CFD code DET3D [1] was used. This (explicit) code solves the 3D Euler gas dynamic equations for a reacting gas system taking into account effects of precursor shock waves, reflections and heat losses on the resulting pressure. After validating the code against available experimental pressure records, showing very good consistency of the calculated pressure records, pressure profiles were calculated at all strain gauge positions. Then, using a simplified 1D model for the mechanical response of a cylindrical pipe under an internal dynamic pressure load, the dynamic strain corresponding to these calculated pressure signals was determined. A comparison of these calculated strain values with the experimental signals showed very good agreement. This made it possible to use the DET3D code to predict the detonation pressure loads for the original BWR gas mixture under the worst case scenario, and to evaluate the resulting dynamic deformations and the integrity of the exhaust pipe. It turned out that the differences in the calculated pressure profiles for the 10 bar (293 K) and the 20 bar (602 K) radiolysis gas detonations were sufficiently small so as to result in basically the same mechanical pipe response, confirming the validity of the chosen experimental approach.

Summary

Worst case scenario of radiolysis gas detonations in a BWR exhaust pipe at pressures up to 20 bar was analyzed both experimentally and numerically.

It was demonstrated that the tube can withstand the detonation pressure at such conditions. The maximum remaining deformation was measured to be not more than 0.15%.

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A study on seismic behavior of nuclear power building in strong nonlinear area and fragility evaluation using 3 dimensional FEM. Part 1. Ultimate seismic condition of building (5-2074)

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The evaluation based on probabilistic safety assessment (PSA) is expected for nuclear power buildings because the risk of the occurrence of seismic ground motions beyond the design assumption cannot be denied. For the assessment, seismic ultimate analyses of the building are necessary.

In this paper, the seismic ultimate behavior was evaluated using an accurate three-dimensional nonlinear FEM model. In the model, the basemat and the soil were modeled by solid elements, and shear walls of the building were modeled by nonlinear layered-shell elements. The uplift behavior was estimated using joint elements between the basemat and the soil. The response analyses considering the maximum horizontal acceleration up to 3500 Gal was done. Then, the influence on the response given by the vertical ground motion and the basemat uplift was evaluated. Moreover, the response was compared with that of the lumped-mass model, which is generally used for current seismic design. From the study, the following results were obtained.

- 1) The building reached the ultimate condition at 7 times of the design basis ground motion input. Shear failure was occurred 3500 Gal input.
- 2) The horizontal response of the structure for simultaneous horizontal and vertical input was almost the same as for horizontal only input, thus the effect of vertical input was relatively small. The vertical response of the structure for simultaneous input agreed well for vertical only input.
- 3) The effect of basemat uplift on the horizontal response was relatively small. However, the effect on the vertical acceleration was not small. The difference was considered as the vertical induced motion.
- 4) The shear strain of the lumped mass SR model exhibited almost the same level as the FEM model in O/S. However in E/B, the SR model overestimated damage compared to the FEM model.

A study on seismic behavior of nuclear power building in strong nonlinear area and fragility evaluation using 3 dimensional FEM. Part-2. Fragility evaluation (5-2075)

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The evaluation based on the probabilistic safety assessment (PSA) is expected for nuclear power buildings because the risk of the occurrence of the seismic ground motions beyond the design assumption cannot be denied. In this paper, the building fragility evaluation of the seismic PSA was carried out using the 3 dimensional nonlinear FEM model based on the result of part-1.

As the fracture modes, the shear failure of the web wall and the flexural failure and the compressive failure of the flange wall were assumed. The fragility curves of the FEM model and lumped mass model in each analysis case were calculated as follows. First, the failure probability was plotted on a diagram for each input acceleration level where analysis was conducted. The failure probability is calculated by considering the aleatory uncertainty of the response and strength value. The plotted points are approximated by a lognormal cumulative distribution function using the least squares method, which is taken to be the fragility curve. From the study, the following results were obtained

- 1) In terms of the failure mode of the envisioned reinforced concrete seismic walls, shear failure led to flexural failure and compressive failure.
- 2) In the evaluation of shear strain, the difference between the fragility evaluations for horizontal input only and simultaneous horizontal and vertical input was small, and the effect of vertical input was relatively small.
- 3) The effect of basemat uplift on fragility evaluation was relatively small.

In the fragility evaluation, the lumped mass SR model exhibited a tendency to estimate damage largely compared to the FEM model.

The effect of foundation embedment on seismic SSI response of EPR nuclear island structures (5-2076)

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Introduction

The Evolutionary Pressurized Reactor (EPR) is an advanced nuclear power plant design developed by AREVA NP, Inc. The EPR consists primarily of a nuclear island (NI) and several other significant buildings outside the NI. A plan view of EPR structures is shown in Figure 1. The structures within the NI consist of the reactor building (RB), fuel building (FB), safeguard building 1 (SB1), safeguard building 2/3 (SB2/3) and safeguard building 4 (SB4). The nuclear island is embedded approximately 11.85 m below ground surface. The EPR standard design certification considers ten generic soil profiles that range from hard rock to stiff soil to soft soil conditions that are likely to be encountered at plant sites in the United States. Three sets of free-field seismic ground motions representing the rock/soil outcrop motions in the eastern United States are considered for the standard design. Each soil profile is associated with one or, and in a few cases, two of the seismic control motions.

This paper presents the seismic soil-structure interaction (SSI) analyses and results of the embedded EPR NI common basemat structures for the ten generic soil cases and three postulated seismic ground motions.

Analytical methodology

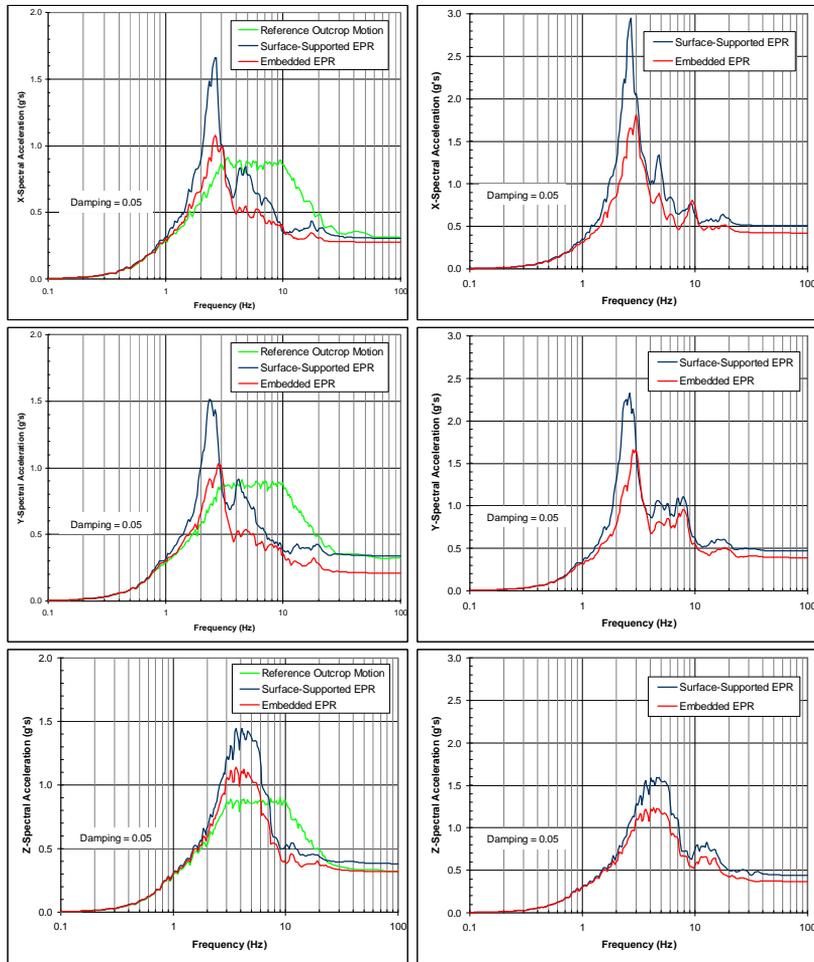
The soil structure interaction analyses were performed using the computer program SASSI. SASSI uses finite element and complex frequency response method to calculate dynamic SSI response of structures supported in horizontally layered soils system over uniform half-space. The primary soil material nonlinearity is the strain-compatible soil shear modulus and damping ratios. The structure model consists of interconnected stick models of RB, FB, SB1, SB2/3 and SB4 (NI sticks) connected rigidly to a common NI rigid

basemat (see Figure 2). To model the effect of embedment on the sticks, a series of horizontal rigid links connect the side soil walls to the sticks below ground surface. In developing the embedment model, it is assumed that basement walls that are in contact with soil are rigid in the out-of-plane direction. The foundation is modeled with horizontal soil layers over uniform halfspace with control motion applied as outcrop motion at the basemat level.

Discussion of results

The results of the SSI analyses in terms of the envelop of maximum accelerations and 5%-damped acceleration response spectra for different soil cases are presented and compared for the embedded and surface-supported EPR model at the foundation and several selected locations in the structure. In general, the effect of embedment is to reduce the in-structure maximum acceleration and acceleration response spectra across the frequency spectrum. However, the degree of this reduction depends on the soil profile, NI structures properties and input motion. In general, more reductions in the response due to embedment effects seem to be associated with the softer soil profiles except where the spectral response is affected by the structural frequency shift. As the soil profiles become stiffer, the responses of the internal structures appear to be less influenced by the embedment effects since they are not directly connected by the side soils. In general, the structures that have direct contact with the side soils (SB1, SB2/3, SB4 and FB) appear to be more sensitive to the embedment effects regardless of the soil stiffness. Typical results are shown in Figure 3.

5. Modeling, Testing and Response Analysis of Structures, Systems and Components



(a) Basemat

(b) Reactor Building Internal

Figure 3. Comparison of Acceleration Response Spectra, Soil Case 2sn4u/EUR Medium Motion.

Component mode synthesis based SSI analysis of complex structural systems using SASSI (5-2089)

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Introduction

Three-dimensional seismic soil-structure interaction (SSI) analysis of nuclear power plants (NPP) is often performed in frequency domain using programs such as SASSI. This enables the analyst to properly a) address the effects of wave radiation in an unbounded soil media, b) incorporate strain-compatible soil shear modulus and damping properties and c) specify input motion in the free field using de-convolution method and/or spatially variable ground motions. For large, complex structural systems with several hundred thousand degrees of freedom and large foundation impedance matrix associated with deeply embedded foundations, the conventional sub-structuring analysis approach employed in SASSI often results in a coefficient matrix that is too large to solve with currently available computer resources. To address this problem, the method of component mode synthesis (CMS) is employed in the SSI analysis. This involves partitioning the structure into several interconnected components, calculating the reduced-order model of each component, and then assembling the reduced-order component models into a global model of the total SSI system. This paper presents the formulation of component mode models, and their implementation into the global SSI model.

Analytical methodology

The soil structure interaction analysis is performed using the computer program SASSI. SASSI uses finite element and complex frequency response method to calculate dynamic SSI response of structures supported in horizontally layered soils system over uniform half-space. The CMS method has been implemented in SASSI utilizing the super element capability. According to this implementation, the structure is first partitioned from the foundation and analyzed as one or several interconnected components using the ANSYS program to compute the component mode properties that are used to form super elements. These super elements are input into the foundation/soil model and analyzed by SASSI to calculate the foundation response. The foundation response that includes the SSI

effects is then imposed onto the structural model to calculate the response of the detailed structure.

Discussion of results

The effectiveness of this procedure is demonstrated by comparing the results of seismic SSI analysis of a detailed typical NPP model (see Fig. 1) subject to horizontal excitations. The total SSI system is analyzed with SASSI using the conventional approach to compute the target solution. Following this, the structure is partitioned to several components and re-analyzed using CMS method implemented in SASSI. The results in terms of 5%-damped response spectra and maximum accelerations at selected key locations (see Fig. 2) in the structure are computed and compared. Comparison of the responses show close agreement between the target solution and those obtained using component mode synthesis.

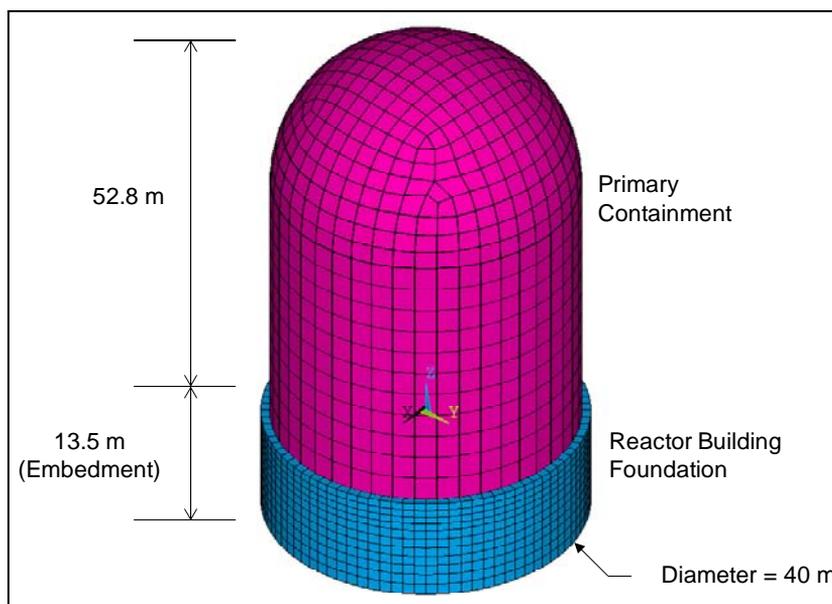


Figure 1. Reactor Containment Building.

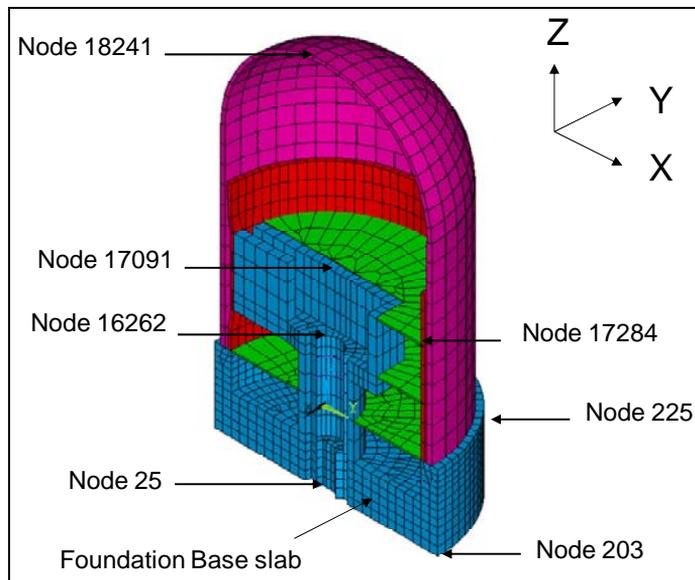


Figure 2. Interior View of Reactor Containment Building.

Seismic capacity test of overhead crane under horizontal and vertical excitation – element model test results on non-linear response behavior (5-2148)

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Design and Construction Issues

The validation of seismic capacity regarding the structural strength or operative function of equipment has been a key issue for seismic safety assessments of nuclear power plants. Since the new safety review guidelines for seismic design of nuclear power plants in Japan, which were established in 2006, required dynamic response analysis in the vertical direction, it has become an urgent issue to obtain the seismic capacity data of equipment under vertical excitation.

An overhead crane used in a BWR reactor building is typical equipment that has lower rigidity and higher response behavior in the vertical direction than in the horizontal direction. High level horizontal and vertical excitation may cause the nonlinear response behavior (slipping, leaping and landing) of the overhead crane, resulting in its derailing from the track.

Accordingly, JNES initiated seismic test programs in fiscal 2006 for the overhead crane. Seismic capacity tests of a 1/2.5 scale overhead crane model under horizontal (traveling and traversing direction) and vertical excitation were planned using the world's largest high-performance shaking table (E-Defense). The primary purposes of those tests were to evaluate the structural strength of an overhead crane and its nonlinear response behavior, and to validate the structural retrofit of components for prevention of its derailing from the track under high level excitation, whose retrofit was applied in some of existing plants in Japan. In order to minimize the seismic capacity test conditions of the scaled overhead crane model, three types of element model tests were conducted in advance, by focusing on the effects of key factors on the nonlinear response behavior. The

key factors were: i) the location of a trolley mounted on a girder (center and end of a girder); ii) the level of a carrying weight with different rope length (top, middle and bottom in height from the floor); and iii) rebounding characteristics of a wheel system after its landing on the rail. Numerical analyses were also performed to simulate the nonlinear response behavior observed in the element model tests.

Experimental determination of the interaction of blast waves proceeding in air and ground (5-2465)

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Introduction

Gneiss is considered to have good foundation characteristics. The characteristic feature of gneiss is its structure: the mineral grains are elongated, or platy, and banding prevails. The paper deals with the experimental and numerical evaluation of the influence of simultaneous ground shock and air blast forces on structural response of containment shell structure founded on gneiss.

Aim of the work

A surface explosion generates both ground shock and airblast pressure on nearby structures Wu et al. (2007). The ground shock usually arrives at a structure foundation earlier than airblast pressure because of the different wave propagation velocities in geomaterials and in the air. However, ground shock and airblast might act on the structure simultaneously, depending on the distance between the explosion center and the structure. Therefore, the precise analysis of structure response and damage to a nearby surface explosion should take both ground shock and airblast pressure into consideration. The current practice usually considers only airblast pressure. Many empirical relations are available to predict airblast pressure. Most of them, however, only predict peak pressure values.

Essential results

In this experimental study, the experimental relationships of simultaneous ground shock and airblast forces have been obtained which can be easily applied in structural response analysis.

The numerical model including both free air and gneiss properties were programmed and linked to sap2000 (2008) as its user provided subroutines. The arrival time, peak particle acceleration (PPA), duration and the principal frequency of the ground motion time history have been determined. The numerical and experimental scaled model results demonstrate good agreement

with each other. The full scale simulation of a typical reactor containment has been subjected to surface explosions for structural response analysis. The variation in stress values owing to time lag between air blast pressure and ground shock to structure founded on gneiss, structural height and curvature has been determined.

Conclusions

It is found that the structural damage will be critically underestimated owing to neglect of simultaneous ground shock and air blast force. The methodology can be employed to evaluate the blast response of concrete shell type containment structure and estimating the extent of cracking.

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Spectra-compatible time histories for the ACR NPP in Eastern North America (5-2471)

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The standard design of the Advanced CANDU Reactor®, (ACR®), is developed by Atomic Energy of Canada Limited to be the next step in the evolution of the CANDU product line. It is based on the proven CANDU technology and incorporates advanced design technologies.

In recent years, it has been established that ground motions occurring in Eastern North America (ENA) are richer in high frequency content than those occurring in Western North America. This is due to the impact of smaller earthquakes at shorter distances that govern the computed seismic hazard of sites in the Eastern North America region.

This phenomenon is particularly relevant for the seismic qualification of safety related structures, systems and components of the nuclear power plant, which are characterized by high frequencies. The ACR standard design takes this phenomenon into account by using a uniform hazard spectra characterizing the near-field motions in ENA, in addition to the ground response spectra defined for soil and rock sites per the Canadian Standard Association.

Three sets of artificially generated time histories have been developed to represent the ENA earthquake design spectrum for ACR Nuclear Power Plant. Different techniques are applied to generate the three sets: spectral matching, stochastic finite-fault simulation method and frequency-dependent scaling. There are 38 time histories generated in total.

This paper presents the approach followed in generating the time histories compatible with the ENA ground response spectra. In addition, the paper presents the results of the analyses performed to ensure that the generated time histories meet both Canadian and international acceptance criteria.

Soil-structure analysis for ACR nuclear island (5-2472)

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The standard design of the Advanced CANDU Reactor®, (ACR®), is developed by Atomic Energy of Canada Limited to be the next step in the evolution of the CANDU product line. It is based on the proven CANDU technology and incorporates advanced design technologies. The ACR nuclear island consists of the reactor building and its adjacent safety-related auxiliary and service buildings supported on a common base slab.

The ACR standard design is based on a standard uniform hazard spectra characterizing the near-field motions in Eastern North America, and on two ground response spectra defined for soil and rock sites per the Canadian Standard Association. In addition, the standard design is performed for seven soil profiles, enveloping a wide range of foundation conditions representing potential site conditions.

The nuclear island seismic response due to the three design ground response spectra is determined, taking the soil-structure-interaction (SSI) effects into account. The SSI effects are taken into account using the flexible volume method in the frequency domain. A mathematical model representing the dynamic characteristics of the nuclear island is developed. The foundation medium is represented by horizontally layered soil profiles resting on a rigid half space. The displacement and acceleration responses of the nuclear island structures as well as the base shear and overturning moments are determined. In addition, floor response spectra are generated at representative locations inside the nuclear island structures.

This paper presents the approach adopted in determining the seismic response of the ACR nuclear island structures due to the specified ground response spectra while considering the SSI effects.

Reactor head stand evaluation using simplified non-linear analysis (5-2474)

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Introduction/background

Diablo Canyon Power Plant intends to replace the existing Reactor Vessel Closure Head (RVCH) with a new RVCH and Integrated Head Assembly (IHA) that is heavier than the existing configuration. During refueling, the new combined RVCH and IHA structure is stored on a structure composed of four individual stands, 90 degrees apart. Since the head stands are only loaded during refueling outages, they are not considered to be safety-related structures. The Diablo Canyon design criteria allow such systems to have inelastic deformation under a seismic event if the behavior does not adversely impact any safety related structures, systems or components.

Previously, each stand had been modified to raise the elevation of the stored reactor head in order to allow for access to the bottom of the reactor head. The modification is composed of steel members that are welded to each other creating portal frames that are bolted to each of the original short storage stand legs. The original storage stand legs are welded to concrete embedment plates. The portal frames are not attached to the concrete slab and only act in bearing. The RVCH rests on the top of stand and is connected to the stand through (a) shear pins to resist horizontal forces and (b) bolts to resist vertical uplift. The total weight of the new RVCH and IHA is heavier than the original assembly, and the available design margin in the existing analysis is limited. Therefore, conservatism in the existing evaluation must be reduced to avoid structural modifications.

Objective

Several analysis techniques are presented to increase the design margins in order to qualify an existing structure without the need for structural modifications. These methods include developing higher damping seismic response spectra, inelastic spectra generation, decoupling of non-linear systems, simplified non-linear modeling and the use of the 100-40-40 combination method.

Approach

One of the conservatisms in the existing seismic evaluation was the use of 4% damping Hosgri response spectra because no higher damping value response spectra were available. NRC Regulatory Guide 1.61 (1973) permits the use of 7% damping for bolted structures under Safe Shutdown Earthquake. The 7% damping is more appropriate since the reactor head is bolted to the stand, and each head stand also has bolted connections. Therefore, a simplified methodology is developed to generate the 7% damping response spectra based on the available 4% damping response spectra using the relationship given in Newmark (1971). The seismic response spectra are further reduced in the low and intermediate frequency ranges by assuming a limited nonlinear (elasto-plastic) behavior exists in the structural system. This method was proposed by Newmark (1982) and is also discussed by Gupta (1992).

The RVCH head stand exhibits non-linear characteristic at the interfaces between the RVCH and head stand, and between the head stand portal frame and concrete slab. The non-linear characteristic is due to the different stiffness when the connection is in compression or tension. A methodology is developed to account for the different compression and tension stiffness to better predict the behavior of the structure. This method is then simplified by using a decoupling technique to separate the structure into two models at the interface of RVCH and head stand.

The reduced inelastic seismic response spectra is used in finding the reactions for an ANSYS stick model of the reactor head, CRDM, IHA, and the bolts between the head and the stand (noted as Model 1 in Figure 1). The non-linearity of Model 1 is due to reactor head-to-stand bolt stiffness in tension being smaller than the head-to-stand bearing compression stiffness. The reactions at the base of the ANSYS stick model are combined using the 100-40-40 method. Unlike the Square Root of the Sum of the Squares (SRSS), the 100-40-40 method maintains the sign of the reaction forces. This allows the analysts to distinguish between the different structural behavior, i.e., in tension and compression. As such the 100-40-40 reduces conservatism by qualifying members based on their actual response in tension or compression. The reactions of controlling combinations from Model 1 are then applied statically on a 3-D non-linear model of all the support frames comprising the stand structure (noted as Model 2 in Figure 1). This model depicts the behavior of the system and distributes the loads between the stand frames. The non-linearity of Model 2 is due to the fact that the portal frame (W14) is not attached to the concrete slab (i.e., act in bearing only). The resulting forces and stresses from Model 2 are evaluated using the design basis allowable criteria and code of record, AISC (1989), for acceptance.

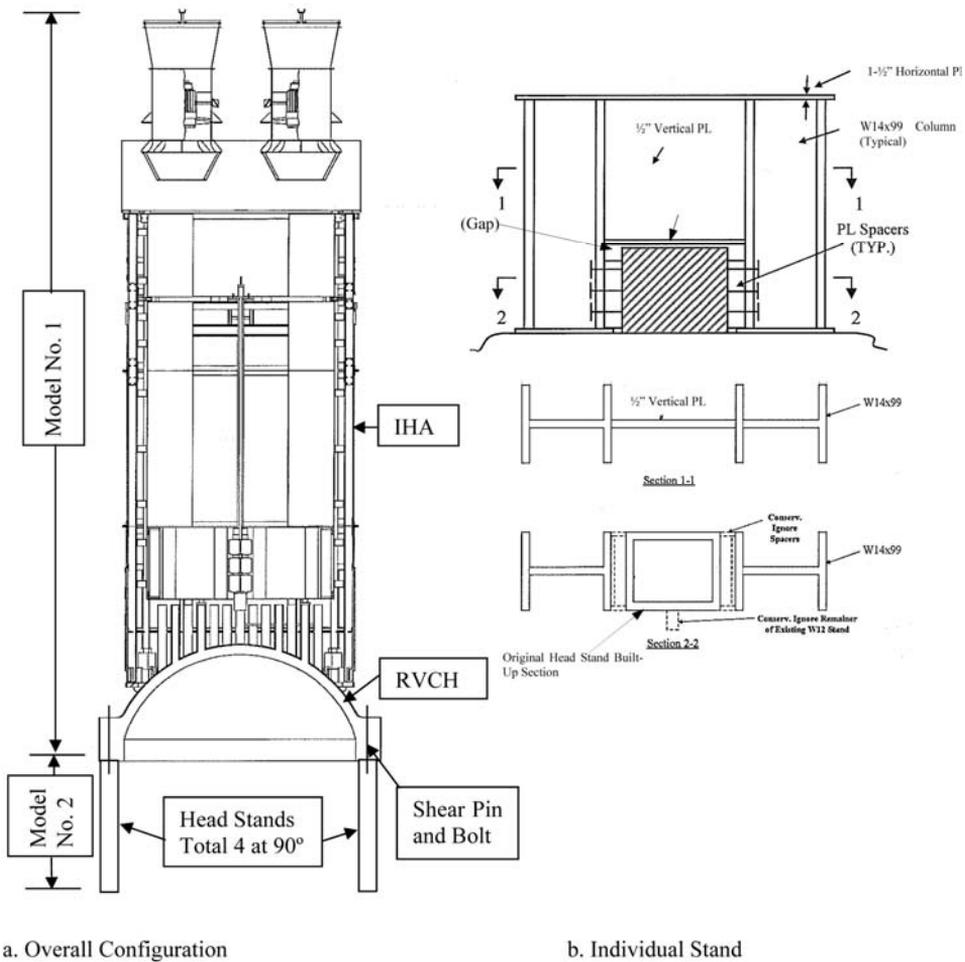


Figure 1. Reactor Vessel Closure Head (RVCH) Stand.

Conclusions

The existing head stand structure was qualified without the need for structural modification using simplified non-linear analysis techniques. These techniques include developing higher damping spectra, inelastic spectra, decoupling of the structure, simplified non-linear model, and the 100-40-40 seismic combination methods.

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NPP seismic protection against shock and vibration loads (5-2479)

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The paper presents the results of theoretical and experimental study of a seismic isolation system based on 3D elastic-plastic dampers (3D EPD) designed for overall protection of nuclear power plants against seismic, shock and vibration loads.

Choice of the material for 3D EPD elements, their sections and spatial configurations provides for standard level of load transfer to the object to be protected, operational stability of 3D EPD at a given number of loading cycles.

Using 3D EPD makes it possible to create a compact system for the object protection against active spatial impacts. Together with the above specific features 3D EPD preserves all good properties of unidirectional plastic dampers: they preserve their characteristics irrespective of the environment conditions, their long-duration operation does not require maintenance.

Designing of the 3D EPD depends on the initial 3D EPD yield surface, which is to be found as the solution of rigid-plastic problems for the 3D EPD strength model. A new method for these problems solving had been developed. When the initial 3D EPD yield surface is defined we can evaluate the dynamic movement of the object protected and the loads acting under given external impacts.

More exact solution for the problem of seismic protection based on EPD requires to consider the following: reinforcement effect, loading recurrence (recurrent reinforcement, low-cycle fatigue), elastic operation of non-plastic-deformed segments of 3D EPD elements and their cross-sections, elastic operation of 3D EPD elements when the load has been removed, plastic deformation influence on changes in the 3D EPD elements geometric parameters.

Thus the 3D EPD types proposed had passed the whole cycle of development: calculation, dynamic and static tests of elements and pilot samples. Therefore using the EPD for protection against extreme loads is effective.

The gearbox for the helium cycle of 10 MW high temperature gas-cooled reactor (5-2508)

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The direct helium cycle has much high efficiency to generate the electricity for the power conversion unit of high temperature gas-cooled reactor (HTGR). In order to validate the helium cycle technology, INET initiates a project to couple a helium cycle with 10 MW high temperature gas-cooled reactor (HTR-10) to replace current steam generator. Such a cycle consists of a turbine-compressor system and a power generator. While the speed for the turbine-compressor system is 15000 rpm and the speed of the power generator is 3000 rpm. A gearbox is chosen to connect the turbine-compressor system and the power generator. The paper studies the special features of the HTR-10 helium cycle as well as the current gearbox technology worldwide, provides the technical requirements for the gearbox, and then makes the preliminary design for the gearbox based on the structural mechanical analysis. The center distance between the turbine shaft and the generator shaft is 300 mm. An vertical gearbox with a speed ratio of approximately 5:1 is selected to connect the helium turbine-compressor system and the power generator to transfer the power of 2500 KW. The mechanical analytical results show that the gearbox design successful satisfies the technical requirements and the specification.

Reactor building 3D-model for evaluating the pressures on concrete regularization and foundation waterproofing membrane (5-2514)

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Introduction

Angra dos Reis site in Brazil has already 2 operating PWR NPPs. Unit 3, with identical design to Unit 2, also a 1350 MW PWR, is expected to have its construction started in 2009. The reactor building of Angra 3 is a complex concrete structure with several thickness and dimensions. In a general point of view it is founded on a base plate having a thickness of about 2 m and radius about 30 m. The top level of this foundation is -0.85 m and the ground level is 6.15 m. This new plant shall be founded directly on a hard sound rock. The first step is to prepare this rock surface with a concrete regularization and a foundation waterproofing membrane.

Aim of the work

This study presents a 3D model approach of the corresponding reactor building to verify the maximum loads acting on this surface. ANSYS Mechanical Release 11.1 is used for this analysis.

Dead weight, permanent and live loads, Safe Shutdown Earthquake (SSE) combined with Burst Pressure Wave (BPW) from the Feedwater Tank and temperature are taken into account.

Foundation stiffnesses corresponding to the hard sound rock and to the waterproofing membrane are represented separately by elements of linear elastic spring COMBIN14. Stepped foundation is also represented by additional springs. The springs are in the three orthogonal directions. Rock stiffnesses are obtained from elastic half-space theory and membrane stiffness is based on test results. The superstructure is represented by linear shell elements – SHELL63.

The maximum pressure for each load case is obtained for vertical and horizontal directions that correspond to compression and tangential reaction loads.

Results and conclusions

The results are compared with a more simplified analysis performed before, showing a good agreement in global values.

The 3D model permits to show a more realistic pressure distribution at every foundation specific detail, proving that the results obtained of combined loads, 1192 kN/m^2 for compression, 179 kN/m^2 for shear effects, are below the allowable limits.

On the generation of inelastic secondary system seismic response spectra (5-2526)

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Introduction

In the electric power reactor industry all safety related systems are designed to resist and to keep the operability during and after a postulated earthquake. The diversity and the large number of the secondary systems in a NPP lead to the response spectra methodology for the seismic analysis.

In general the secondary system seismic design is based on floor response spectra, using the assumption of linear analysis, although, it is useful to evaluate the plastic reserve due to secondary system ductility.

For piping systems, the evaluation of this reserve can be performed by the comparison of the linear response spectra to the nonlinear inelastic ones obtained on particular models of such systems.

Aim of the work

It is suggested a procedure for the generation of in-structure seismic response spectra for secondary system design, which includes a probabilistic approach and considers coupling effects between primary and inelastic secondary systems. First, the ground excitation PSD is transferred to a SDOF model of the secondary system conveniently attached to the primary system. Then, a uniformly probable coupled response spectrum is obtained using the first passage analysis, /5/.

A global ductility factor, relating the plastic to the overall work done by the seismic external forces on the secondary system, and a specified yielding factor allows one to obtain transposition factors from elastic to inelastic response spectra.

Summary of the methodology

The proposed script is developed in frequency domain. A set of computer programs is developed to be used with the SASSI2000 system modules to

consider three-dimensional models and their responses for a generic base excitation, acting in 3 orthogonal directions.

The methodology includes the representation of coupling effect between primary and secondary systems, and the influence of the secondary system multiple supports relative displacements, to produce uniformly probable coupled response spectra, i.e., its ordinates represent the maximum response of a single degree of freedom (SDOF) system with equal probability of non exceedance along the usual frequency range of analysis

With the use of the mentioned transposition factors, it is possible to evaluate the plastic reserve of the secondary piping systems.

The application of this methodology is a practical and consistent solution. It is practical because it only needs the earthquake PSDF and consistent since it may produce also coupled response spectra whose ordinates maintain a probabilistic commitment with the PSDF amplitudes.

Conclusions

Although the great computational efforts, with the proposed methodology one can achieve the following advantageous:

- Better representation of the damping effects, considered directly in the soil-structure interaction formulation, because neither the use of modal damping nor the definition of the Rayleigh coefficients are necessary.
- The choice of the frequency for which the response spectrum is calculated is oriented by the Transfer Function peak values. It requires a lower number of calculations points than if all modal frequencies are used besides that 75 specified ones [US NRC- RG 1.122 – 1978].
- Superposition and combination of different responses can be obtained in probabilistic ways.
- Probabilistic response spectra, obtained directly from PSD, are much smoother than those obtained deterministically from time history samples, and the errors can be evaluated, leading to more analysis reliability.
- Evaluation of the plastic reserve of secondary piping systems is achieved, by the use of transposition factors.
- The degree of automation, allows the production of response spectra including modeling refinements, reaching a more realistic analysis, without additional efforts beyond those already required by the usual methodology.

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Seismic response of a two-degree-of-freedom system with friction based on the mass ratio (5-2542)

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Introduction

Recently, massive earthquakes such as the Niigata-ken Chuetsu-oki earthquake (2007) or the Tokachi-oki earthquake (2003) have occurred frequently in Japan. Large-magnitude earthquakes seriously damage not only ordinary houses, but also structures in industrial facilities. In order to prevent hazardous material spills and secondary disasters, industrial facilities are required to have high aseismic performance compared with houses.

On the other hand, the frictional isolator has attracted attention for the reduction of seismic response in industrial facilities. This system has the effects of shifting the natural frequency away from the predominant frequency of the seismic wave and dissipating seismic energy. These effects can improve the aseismic capacity of a structure in industrial facilities and reduce the cost of seismic design. When a structure is a multi-dof system with friction, however, the seismic response of this system can be obtained by only non-linear time history analysis [1]. A great deal of time is necessary for non-linear analysis. In general, immediate and easy estimation of the seismic response is required in the seismic design of industrial facilities. In the previous studies [2]–[4], the authors proposed a response spectrum for a 1-dof system with friction. This spectrum can simply estimate the seismic response of a 1-dof system with friction.

Aim of the work

The present study deals with the easy estimation of seismic response for a multi-dof system with friction using the response spectrums method. The 1-dof linear system and the 1-dof friction system are used in this method. This estimation is based on the modal analysis method. The authors also proposed a method of modal separation to use one slip direction for a piping system with friction [5]. In addition, Hanawa et al. proposed a method that is available for any slip direction [6]. When we consider the simple 2-dof system with friction, however,

the method of modal separation for same directional friction force has not yet been proposed.

In the present paper, the method of modal separation for a 2-dof system with friction is proposed when the mass ratio of the structure and the support is nearly zero. The 1-dof system, which is converted to modal space using this method, calculates the time history response. The time history response on modal space is then translated to real coordinates. An application of the proposed method is discussed by comparing the results of modal analysis for a friction system with the 2-dof non-linear time history analysis.

Essential results

This paper show the one mode of the vector of the friction force can be regarded as approximately zero when the mass ratio is nearly zero. However, the accuracy of the proposed method depends on the mass ratio, the frequency ratio and the friction force. The proposed method is compared with 2-dof non-linear time history analysis, and the accuracy of the proposed method is calculated using the sinusoidal wave and seismic waves. The present study defines the area within an error of 10% as the applicable area of the modal analysis method for a friction system and shows the range of error over 10%. From this area, we can judge whether the proposed method is applicable for each parameter.

Conclusions

In the present study, the method of modal separation for a 2-dof friction system using the ratio of elements of the modal matrix is considered. The proposed approximation method can convert dynamic equations of a 2-dof friction system into modal space. Analytical errors of the maximum acceleration at the structure and support obtained by the proposed method are calculated and compared with the 2-dof non-linear time history analysis. The results of the present study revealed that an area of error of over 10% for the modal analysis method for a 2-dof friction system. Therefore, the range of application of the frequency ratio and the mass ratio can be easily estimated using this error area.

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Response of graphite dowel-socket structure under various loads (5-2564)

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Keywords: HTR, graphite components, dowel-socket, equivalent stiffness

The graphite components are the main part of the core structures of high temperature gas-cooled reactor (HTR). Unlike metal components which can be welded or riveted together, those graphite components should be only connected by dowel-socket and key-keyway structures to transfer horizontal forces and to restrict correspondingly horizontal motion, therefore, to ensure the integrity of the whole core structure.

Among the assembly of graphite components, small clearance remains to accommodate the thermal and fast neutron irradiation strains. Under dynamic loads, such as seismic loads, the impact would occur among those graphite components due to the given clearance. Therefore, in addition to the research of the overall response of graphite dowel-socket structure under various loads, it's necessary to study the influence of different gaps of dowel-socket structure on the load-deformation response of the graphite structure. Since the final goal is to investigate the integrity of the whole core structure, it is necessary and important to get the equivalent stiffness of dowel-socket structure that consists of two graphite blocks, which is considered as the basic component of the whole core structure.

In this paper, with commercial finite element code ANSYS, the load-deformation response of the graphite dowel-socket structure is studied, and the influence of different gaps of dowel-socket structure on the response is discussed in addition. Besides, the corresponding results and equivalent stiffness of dowel-socket structure are given.

Internal pressure capacity evaluation of prestressed concrete containment buildings considering multiple aging effects (5-2580)

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Introduction

Since the accident at Three Mile Island nuclear plant in 1979, it has become necessary to evaluate the internal pressure capacity of the containment buildings for the assessment of the safety of nuclear power plants [1–3]. According to this necessity, many researchers including Yonezawa et al. [4] and Hu & Lin [5] analyzed the ultimate capacity of prestressed concrete containments subjected to internal pressure. In these studies, the ultimate capacity analyses are performed for the containments under fresh condition. However, most of nuclear power plants are exposed to the severe environments such as costal area and ambience irradiation. Hence, the aging effects on the structural system caused by these environmental conditions should be considered for the estimation of internal pressure capacity in a quantitative manner. Especially in Korea, some containment structures were built in the late of 1970 or early 1980, so that the degradation of their structural performance also must be explained in the procedure of the of the internal pressure capacity evaluation.

Therefore, in this study, we developed the degradation models for the structural components of prestressed concrete containment buildings, and evaluated the internal pressure capacity considering multiple aging and degradation factors. The target containment building types were PWR (Pressurized Water Reactor) and CANDU (CANada Deuterium Uranium) type containments which are the most typical reactor buildings in Korea.

Methodologies and results

There exist many degradation and aging factors in the prestressed concrete structures. For the concrete material, the most degradation factor can be

classified into physical processes such as cracking, freezing & thawing, irradiation, fatigue, settlement, and chemical processes such as sulfate & biological attack, acids, aggressive water, etc. In general, these degradation factors finally cause a loss of section and tensile/compression strength of containment wall. We modeled such aging effect and considered for the internal pressure capacity estimation. In the prestressed concrete containment building, the loss of prestressing force is recognized another important aging factor. Hence, we also estimated the amount of prestressing loss for the evaluation.

For the modeling of containment buildings, there exists a burden process on the modeling of tendons at the hatch area since that its geometry is somewhat complicated. In this study, we developed a program which can automatically generate the tendon components considering the hatch areas. The FE-based general-purpose structural analysis program, ABAQUS [6], was adopted as an analysis tool and developed models were implemented into the input files. We modeled PWR and CANDU type containments as 3D FE models. For the modeling of containment wall, dome, buttress and slab, solid elements were used. The reinforcement bars and tendons were modeled using embedded surface and truss elements, respectively. The material nonlinearity of concrete was implemented by introducing the concrete damaged plasticity model [6]. The tri-linear plasticity model and piecewise linear stress-strain model were used for the material nonlinearity of steel rebars and tendons, respectively. In the preliminary analysis, the critical points under an internal pressure load were a top of the hatch and a middle of the containment wall areas. Figure 1 shows one of the results for the ultimate internal pressure capacity evaluation for the PWR type containment. The elastic capacity almost linearly decreased as the prestressing loss increased, while it does not vary during the degradation of concrete strength. It can be also found that the loss of concrete strength effect only to the elastic stiffness of containment walls.

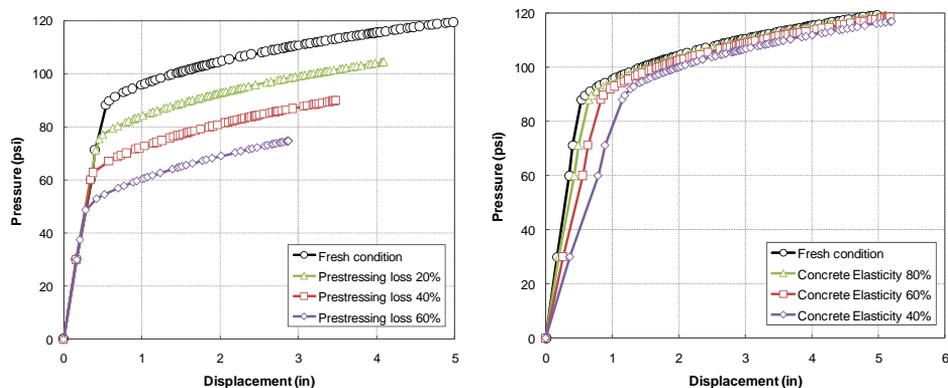


Figure 1. Internal pressure capacity evaluation results considering the prestressing loss and the degradation of concrete strength.

Conclusions

We developed the degradation models for the structural components of prestressed concrete containment buildings, and evaluated the internal pressure capacity considering multiple aging and degradation factors. The FE-based general-purpose structural analysis program, ABAQUS was adopted as an analysis tool and developed full-3D nonlinear FE models were implemented. The results show that the elastic capacity almost linearly decreased as the prestressing loss increased, while it does not vary during the degradation of concrete strength. More detailed results for the various multiple aging effects will be discussed in the full paper.

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Qualification seismic test on control rod driving mechanism of CEFR (5-2588)

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Background

China Experimental Fast Reactor (CEFR) is the first fast neutron breeder reactor in China. The Control Rod Driving Mechanism (CRDM) of CEFR was designed and fabricated by Russian engineers. The CRDM has been qualified strictly by tests according to Russian nuclear codes, including seismic test. Because of the difference between Russian codes at that time and Chinese codes or international nuclear codes, the CRDM was qualified by seismic test according to the Chinese codes and international codes. The latter seismic test is introduced in this paper.

Aim of the work

The control rod assemblies (CR) of CEFR consist of three types: Safety Assembly (SR), Compensatory Assembly (CA) and Regulatory Assembly (RA). The CRDM is the equipment of nuclear safety class 1 and seismic qualification of class I. The CR should be inserted in the core during and after the earthquake to make sure that the reactor be shut down safely. The factors that influence the inserting are very complicated. The CRDM is excited by not only one point but four points during the earthquake. The moving component of the CRDM impacts on the inner wall of the guiding pipe. The most part of the CRDM is immersed in the liquid sodium and fluid structure interaction (FSI) occurs during the earthquake. The test is performed to validate that the CA can be inserted in the core during and after the earthquake to shut down the reactor safely.

Results and conclusions

The qualification seismic test had been performed according to the Chinese nuclear codes. The original CRDM and full scale CRs was adopted and the liquid sodium was substituted by water at room temperature in the test. The earthquake was simulated by multipoint (4 points) displacement time history excitation. The 5 tests on Operation Based Earthquake (OBE) and one test on Safety Shutdown Earthquake (SSE) had been finished for every type of CR. The CRs were validated to be inserted in the core in the time during and after the earthquake. The longest time to inserted in the core for SA is 0.687 second (shorter than 0.7 second which is designed) and for CA-RA 1.789 seconds (shorter than 2.5 seconds which is designed). And no rebound occurred. The test improved that the seismic design of the CRDM was certified and valid.

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Seismic assessment of the sellafield B38 mobile caves (5-2615)

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Introduction

The B38 building at Sellafield originally consisted of six concrete silos with an overbuilding, and was commissioned in 1964. The first extension, a further six silos, was commissioned in 1974. A further building extension gave a total of twenty two silos. The silos were used for the storage of Magnox swarf, and miscellaneous beta gamma waste under water cover.

In time, storage of waste in B38 ceased and the plant lay dormant for many years. In the early 1990's concept designs facilitating the removal of waste from the silos, dispatching this to downstream plants started to be looked at by BNFL. This led to the design of retrieval machines known as Mobile Caves, of which there are three. These would locate over a silo, remove its roof plug, and by means of grab, tools and manipulators, remove waste. The waste would be placed into a skip internal to the Mobile Cave, which would then be removed through a gamma gate into a 50 Te flask temporarily located on the Cave. The flask would then be removed by building cranes for further downstream treatment. Despite being only moderately large in size (approximate envelope of 12 m × 5 m × 6 m), a Mobile Cave has a design mass approaching 400 Te – the equivalent mass of a 747-400 series jumbo jet fully laden at take off (Ref 1).

In the mid 1990's seismic analysis work was completed however, in time the Mobile Cave design evolved significantly, and after a period of mothballing of the project, it became clear that new seismic qualification would be required. Hence, the National Nuclear Laboratory, set up during 2008 as part of the restructuring of BNFL, was requested to provide seismic analysis support to the project.

Seismic analysis

1. Building structure

A finite element model of the building structure, concrete silos, and rails along which the Mobile Caves run was utilized. In this model, the Mobile Cave was represented as a rigid “brick” with the same mass and centre of gravity location as the Mobile Cave itself. The response spectrum and time history analysis

output from this work include the three orthogonal seismic accelerations at the Mobile Cave centre of gravity, enveloped for multiple retrieval locations. The above provided the inputs to the work described in this paper, which was the actual detailed qualification of the Cave structure, described below.

2. Mobile cave structure

A detailed finite element model of the Mobile Cave was built using the proprietary code Ansys (Ref 2). The main structure was modeled using 3D solid elements, but where appropriate 3D shell, 3D beam, and 3D mass elements were also utilized. To ensure that the effect of prying is included in the bolt load analysis, Ansys 3D surface-to-surface contact elements were used between mating faces. In all locations where a bolted connection is made, co-incident nodes have been positioned in the model, and these are coupled in the three translational degrees of freedom to simulate bolt load transfer. By isolating the various assemblies from the rest of the model in post-processing runs, the bolt loads and hence bolt stresses were determined.

A separate finite element model of the seismic isolation bearings (SIBs) was used to confirm that the properties used in the main model were appropriate. Hand calculations determined appropriate section and material properties required in the model in order to replicate the known stiffness properties of the SIBs. Before inserting these calculated values into the main structural model, simulation of the SIBs in a separate finite element model was used to confirm that the calculated properties correctly replicated the SIB stiffness

Allowable stresses were specified using ASME III (Ref 3). Defining allowables for the Mobile Caves in terms of ASME III is difficult, since the primary intent of ASME III is to assess pressure retaining equipment, piping and supports. The Mobile Cave structure is neither a pressure retaining structure, nor a support. The use of ASME III was however, retained in the assessment as there are parts of the code that can be used in principle to evaluate performance of both the main structural items, and the bolts holding them together.

Results

The finite element model was used to successfully demonstrate that all key structural items had stress levels within permissible values. The attachment bolts and fabrication welds were likewise demonstrated to be acceptable. Due to the modeling & assessment work described in this paper, the safety case seismic performance requirements of the Mobile Caves was satisfied. The Mobile Cave is expected to be commissioned and operational by 2011.

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Response and seismic margin of Kashiwazaki Kariwa Nuclear Power Plant building to Chuetsu-oki earthquake (5-3192)

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1. Introduction

The maximum earthquake acceleration observed in the reactor buildings of Kashiwazaki Kariwa Nuclear Power Plant during the Niigata-ken Chuetsu-oki earthquake in 2007 exceeded the design value. However, the structural members were generally within the elastic range, so it was considered that their integrity was ensured. This report discusses the causes of the above by comparing the design force with the actual seismic force by the Chuetsu-oki earthquake.

2. Structure of nuclear reactor building

Reactor building Unit 6 was investigated. Its major structure is of reinforced concrete, with four stories above ground and three stories under ground. The height from the base mat is 63.4m and that from ground level is 37.7m. It is of almost square in plan, measuring 56.6m (NS) x 59.6m (EW).

Its main earthquake resisting walls are the external box type wall and interior walls, and it is designed as a reinforced concrete containment vessel (RCCV). The seismic walls resisted the horizontal force that acted on the building during the earthquake.

3. Design seismic load of nuclear reactor building

The design dynamic and static seismic forces were both three times larger than those required by the Construction Standard Act in Japan. The dynamic seismic force was calculated from earthquake response analysis using the reference earthquake motion S1-D (195 Gal on the base mat).

The design seismic force ensured a tolerance by adding a safety margin to both the dynamic and static seismic forces. This margin resultantly became within a range in which the effects of burying into the ground, taken into account in the static seismic force calculation, could be ignored.

4. Comparison with seismic force due to Chuetsu-oki earthquake

The seismic force acting to nuclear reactor building Unit 6 during the Chuetsu-oki earthquake (320 Gal at the base mat) was estimated from an elastic seismic response analysis using the observation record. The maximum shear stress during the Chuetsu-oki earthquake became larger than S1-D. However, it was at the same level or lower than the static seismic force, and it was sufficiently lower than the design seismic force.

The design seismic forces on the building's structural members were less than the allowable forces which have the members within elastic range. Because the seismic force was lower than the design seismic force during the Chuetsu-oki earthquake, the building was considered to be within the elastic range.

It was considered that an earthquake ground motion larger than the reference earthquakes ground motion S1-D and S2-D, had acted. However, the margin adopted for the static / design seismic force would have maintained the building's integrity.

5. Response on shear skeleton curve

The seismic force caused by the Chuetsu-oki earthquake was estimated from elastic analysis. However, attempts were also made to estimate it from elasto-plastic analysis. The nonlinear characteristics (skeleton curve) of shear force of the earthquake resisting wall was determined by the method specified in the guideline for seismic design of nuclear power plants JEAG4601. The conditions assumed for the above were the same as those for the elastic analysis. The results show that the response to the Chuetsu-oki earthquake was below the first turning point (concrete crack onset point) and was considered to be within the elastic range.

6. Conclusions

The response on the base mat showed that the earthquake ground motion caused by the Chuetsu-oki earthquake at the reactor building Unit 6 was larger than the design force. However, the structural members of the building were considered to be within the elastic range.

This is considered to be because a static earthquake ground motion three times that of ordinary buildings was assumed and a tolerance was incorporated into the design seismic force. Furthermore, the effect of the tolerance was taken into account in designing the reinforcing bar arrangement. Thus, the behavior was considered to be within the elastic range. The above is consistent with the relationship between the hysteresis characteristics and the response.

Simulation analysis of reactor buildings on Niigataken Chuetsu-oki earthquake at Kashiwazaki-Kariwa Nuclear Power Plant (5-3193)

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1. Introduction

Observation records were obtained at the reactor buildings of Kashiwazaki-Kariwa nuclear power plant during the Niigata-ken Chuetsu-Oki Earthquake, which occurred on July 16, 2007. These records were used to carry out simulation analyses on representative reactor buildings. This report presents the framing of the analyses and the analysis results for Reactor Buildings of Unit 1 and Unit 6.

2. Method of analyzing earthquake response

Observation records (acceleration time history) on the base mat slab were input to the earthquake response analyses for the reactor buildings specifically where such records were available. In the analysis, linearity was assumed for both the building and the soil spring. Two horizontal directions (North-South and East-West) were analyzed independently.

The simulation analysis method is as follows.

First, the dynamic responses of the ground at the embedded part of the building were calculated from the analysis based on the one dimensional wave propagation theory.

Second, the dynamic responses were applied to the building for the simulation analysis in the horizontal direction as the input motions. As a result, the

responses of the building were evaluated taking into account the soil structure interaction. With the responses of the building, transfer function between each floor and the base mat slab was obtained.

Third, the earthquake responses of the building's individual parts were obtained by multiplying the transfer function calculated as above by the Fourier spectrum obtained from the Fourier transform of the observation records for the base mat slab.

3. Earthquake response analysis model

The analysis model considering soil structure interaction to be used for the earthquake response analysis was determined to be an embedded sway-rocking model consisting of a lumped mass system for the building and the ground spring.

The building model is described as lumped mass model with weight concentrated at mass point located at each floor, taking into account bending and shear stiffness. In the building model, auxiliary walls that were thought to increase stiffness during an earthquake were taken into account in addition to the main shear walls considered in the initial design. Stiffness of concrete was evaluated based on the measured compressive strength of test pieces sampled from actual building walls. The model also took into account the stiffness estimated in accordance with the design code for reinforced concrete structures in Japan (Young's modulus and shear modulus).

The building's damping constant was determined to be 5%. The soil springs for the base mat slab employed a constant spring (horizontal and rotational) based on Tajimi's vibration admittance theory. The soil springs for the side plane of the building's embedded part employed a constant spring (horizontal and rotational) based on the NOVAK's method. The ground properties took into account the strain-dependency of the stiffness and damping based on laboratory tests. Building-specific decreasing stiffness and including damping factor were provided in accordance with the strain level of the ground.

4. Outline of earthquake observations

Seismometers were installed on the base mat slab of the building's lowest floor and on the intermediate floor in the building's upper portion in each building. The seismometers recorded the acceleration time histories as the observation record. Sampling frequency was 100Hz. A sufficient number of observations were acquired for the earthquake response analysis.

5. Results of simulation analysis

Simulation analysis was carried out using the acceleration time histories observed at the Reactor Buildings of Unit 1 and Unit 6 as input motions. The results were compared with the observation records in terms of the distribution of the maximum response accelerations in building height direction, the wave form of the acceleration time history, the acceleration response spectrum, etc. As a result, the simulation analyses using the analysis models that took into account the realistic conditions during the earthquake showed relatively good agreement with the earthquake observations.

6. Conclusions

Simulation analyses were carried out on reactor buildings using observation records at reactor buildings of Kashiwazaki-Kariwa Nuclear Power Plant obtained during the 2007 Niigata-ken Chuetsu-Oki Earthquake, which occurred on July 16, 2007.

Those for Reactor Building of Unit 1 were based on the design model. They employed the sway rocking models incorporating realistic conditions of building and ground. It also utilized observation records from the base mat slab. As a result, the following items were found.

- The simulated distribution of the maximum response acceleration closely followed the trend of observation records and showed good agreement.
- The simulated floor response spectrum of the intermediate floors in the building closely followed the frequency characteristics of the observation records.

6. Design and Construction Issues

Design, manufacture and construction of fuel and core structure, pressure vessel; steel and concrete containment structures; other concrete structures (storage, processing, etc.); piping and major components, fuel cask design. Capacity, ductility, redundancy and quality considerations. Design of passive safety systems. System isolation and energy absorption. Construction management.

On the design of pipe supports and steelwork regarding revised German nuclear safety standards (6-1587)

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The paper deals with recent revisions of technical standards for structural steelwork and supports in the conventional, non-nuclear area and their influence on the state of the art concerning nuclear safety standards in Germany.

Since several years efforts are made in the European Union to find a common basis concerning the set of technical standards and to leave the path of individual standards in the single nations. One of these efforts concerns structural steelwork in the non-nuclear area, where the EUROCODE 3 /1/ was established in civil engineering and it is planned to withdraw the valid German DIN 18800 [11/90] /2/ in the near future, as it was already done with former German standard DIN 18800 [03/81] /3/ some years ago. Both the EUROCODE 3 /1/ and the German standard DIN 18800 [11/90] /2/ are based on a semi-probabilistic procedure considering different partial safety factors for actions as well as for material properties in contrast to the obsolete German standard DIN 18800 [03/81] /3/ which was based on a deterministic procedure using allowable stresses and global safety factors for the resistances.

In Germany the mechanical engineering parts (including their support structures) of nuclear power plants are based on atomic regulations (and related nuclear safety standards) but the civil works of nuclear facilities (including for example anchorages and dowels) are additionally based on conventional buildings regulations (and related conventional standards). Many of the existing supports with importance to safety in German nuclear power plants were installed in the construction phase of the power plants. This was a time when a deterministic concept was the basis for the related design calculations. Today the deterministic procedure is still implemented in German nuclear safety standard KTA 3205 /4/-/6/ for supports. But the deterministic design concept is today not more allowed for new proof calculations concerning the power plant buildings and anchorages. Therefore two different design concepts (deterministic and semi-probabilistic) meet at the interface between the building and the supports of mechanical engineering equipment. This is especially to be seen for the building structure interaction loads acting on anchorages of passive and active components, because there is no safety factor for actions in the deterministic design concept but there are different partial safety factors for permanent, variable and accidental loads in the new semi-probabilistic design concept.

Current efforts are made in standardization committees dealing with German nuclear safety standards KTA to find an interface agreement considering the state of science and technology. In this paper an overview about the German nuclear safety standard KTA 3205 part 1 to 3 /4/-/6/ for supports is given with respect to recent works concerning the revision of part 2 of KTA 3205 /5/. Furthermore differences between German standards DIN 18800 [03/81] /3/, DIN 18800 [11/90] /2/ and EUROCODE 3 /1/ are regarded. An example of a steel construction is presented to illustrate the design concepts. The results obtained show that different solutions concerning stress utilization ratio and interaction loads can occur. Caused by the new German and European design standards for steelwork a revision of current German nuclear safety standard KTA 3205.2 /5/ is necessary. In this revision requirements resulting from mechanical engineering and from civil engineering have both to be taken into account.

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Active control of vibrations in piping systems (6-1658)

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There are various dynamic load cases that can lead to vibrations in piping systems. Typical vibration sources are, for example, turbulences in pipelines with through-flow, pressure surges resulting from actuation of valving, vibrations of directly connected machines, machine faults or earthquakes.

If the resulting vibrations exceed permissible values remedial measures have to be taken. Common measures for reducing vibrations in piping systems are reduction of excitation, detuning, damping or passive tuned mass dampers.

The efficiency of conventional passive methods for reducing vibrations is limited. The application of passive dampers for example requires a fixed support. Tuned mass dampers demand a mass ratio of about 10% to reach a good attenuation. Therefore tuned mass dampers are often unfeasible due to a high static load. In addition, tuned mass dampers are restricted to one resonance frequency.

To overcome problems with conventional passive methods an Active Vibration Absorber (AVA) for vibration reduction of industrial piping was developed, manufactured and tested. The design and the realization of the AVA shows Fig. 1. The AVA is based on a 2 DOF active system that is able to reduce vibrations over a large frequency range.

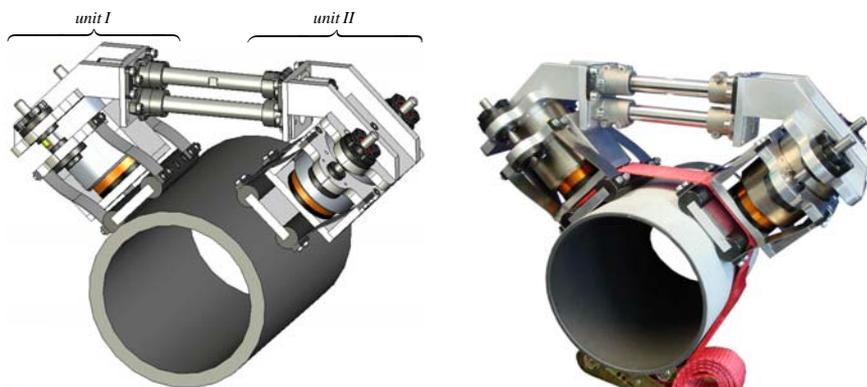


Figure 1. Active Vibration Absorber (left: design, right: realization).

The AVA's function is based on the principle that an accelerated inertial mass generates a reaction force in the supporting structure:

$$F = -m \cdot \ddot{x} \quad (1)$$

The reaction mass is connected to the structure by means of a spring. An actuator is located parallel to the spring. By accelerating the reaction mass, using the actuator in combination with an appropriate control algorithm, a resultant force for vibration reduction is obtained.

The AVA was investigated on a mock-up of a piping system. The developed decentralized controller shows an excellent performance in the considered frequency range. In addition, it could be demonstrated that the control performance is not sensitive even to distinct parameter changes of the structure. Fig. 2 shows typical frequency response functions measured at the mock-up with and without the AVA.

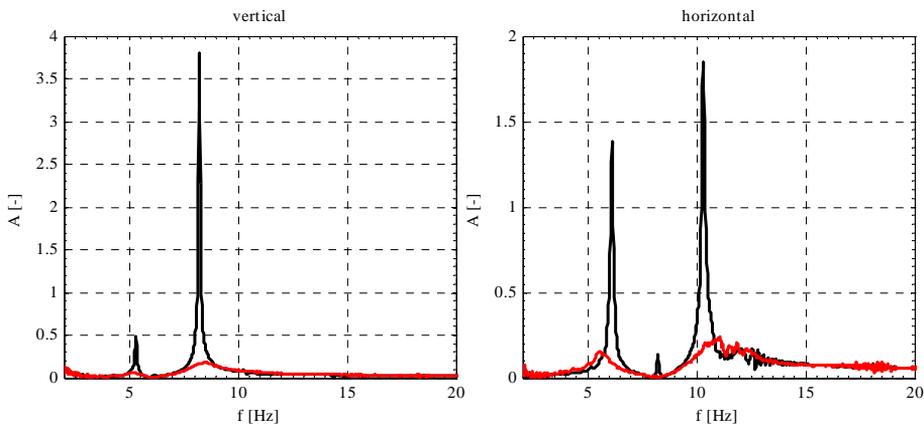


Figure 2. Typical frequency response functions (left: vertical, right: horizontal).

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Performance-based design of SSC wall in fire (6-1675)

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Introduction

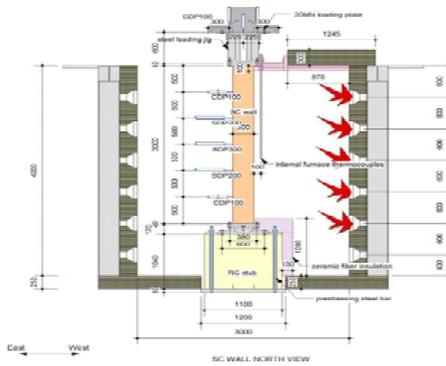
This paper describes proposed design method based on the fire performance test results of stiffened steel plate concrete (SSC) wall with ribs. SSC wall is a composite structure consisting of surface steel plate with stud and rib instead of rebar of reinforced concrete wall. In fire conditions, surface steel of SSC wall is directly exposed to fire attacks, therefore load-carrying capacity of the member is reduced due to degradation of material property. Although the reduction of stiffness and strength is expected, a minimum fire resistance rating of 3 hours should be provided for application of SSC wall in nuclear power plant structure.

In this study, the testing of a loaded SSC wall under ISO fire conditions was carried out in generic transient state testing way and also a simple design method was proposed based on fire performance test result to ensure required fire resistance rating. The design specifications are included in SC structure standard for nuclear facilities in Korea.

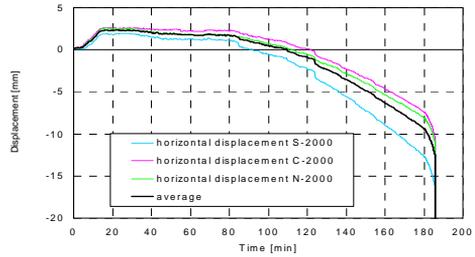
Fire test of SSC wall

Fig. 1 shows arrangement for fire testing SSC wall and the typical test results. In the testing, axial load is applied to the structure first and then held constant and the structure is exposed to ISO fire attack. The test is terminated when one of the specified failure criteria is reached. Horizontal displacement at 185 minutes after fire exposure is drastically increased in test result and the maximum deflection exceeds $L/20$, where L is the span of the specimen. Fig. 2 shows the relations of vertical displacement and fire exposure time. The axial contraction at 185 minutes exceeds limiting value defined in ISO 834 as $h/100$, where h is the initial height in millimeters. Fig. 3 shows the temperature distribution on unexposed surface and the maximum temperature rise at the locations is 87°C . This temperature is less than 135°C which is maximum allowable temperature on unexposed surface of fire barrier wall.

6. Design and Construction Issues



(a) Test Set-up



(b) Measured Displacement

Figure 1. Test Set-up & Measured Displacement.

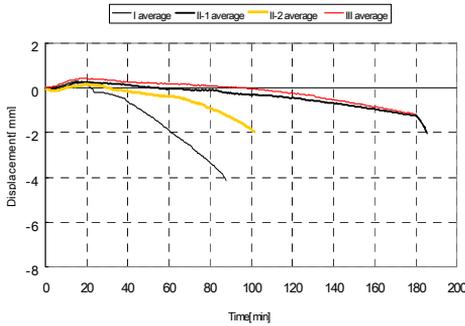


Figure 2. Vertical Displacement.

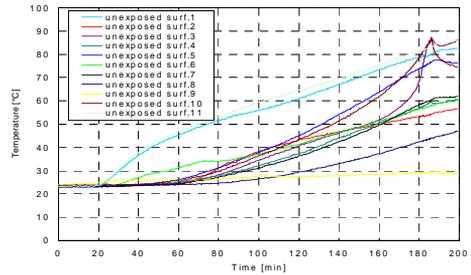


Figure 3. Temperature on Unexposed Surface.

Performance-based design method

SC wall design specifications for the fire conditions consists of the design requirements for fire conditions, mechanical properties of steel and concrete at high temperature, temperature distribution at cross-section, finite element method (or fiber model analysis), segment analysis, simple design method, and required detail.

Based on the fire test results and the theoretical studies the specifications are developed for SC wall design under fire condition. In this specifications, SC

wall with the thickness (t) of above 300 mm and the ratio of below H/t (wall height/wall thick)=13 shall be considered the fire barrier structures satisfying license requirements if the axial force ratio is less than 40%. There are three design methods for the fire barrier design of SC wall such as the fire model analysis, the segment analysis, and the simple design method. The simple design method is the safe design method. Moreover, the simple design method is adapted because structural components in the nuclear power plant are typically thick enough to satisfy a three-hour fire resistance requirement even without a precise design method (Korea Electric Association, 2008). In proposed simple design method for SC wall, a damaged depth for the stiffness calculation is assumed as 100 mm based on the test data of Kodaira et al. using SC wall with a thickness of 200 mm and the test result of the authors using SSC wall with a thickness of 300 mm. A damaged depth for the strength calculation is assumed as 50 mm based on the same test results.

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Upgrade and modification of fuel handling equipment in Korea (6-1693)

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The upgrade and modification of the fuel handling system are currently in progress in Korea. Twenty (20) Nuclear Power Plants are in operation and four (6) units are under construction and two (2) units are being designed. The fuel handling system consists of equipment used for receiving and transporting fuel assemblies and acts as critical processes in the refueling outage. The refueling machine, the fuel transfer system and spent fuel handling machine are used for hoisting and transferring a fuel assembly between the core of the containment building and the fuel storage rack of the fuel building. The improvements for the equipment under construction and in operation have been studied to enhance the operation reliability and efficiency. One of these improvements is to upgrade the hoisting and transferring capability of the equipment during the refueling outage. For the upgrade and the modification of the equipment, the Programmable Logic Controller (PLC) based control system and high speed motor drive system and advanced operating procedures are introduced. In this study, the scope of the improvements for the fuel handling equipment is reviewed and the performance of the equipment during the fuel loading process are analyzed and evaluated. We can expect the fuel transferring capability of the equipment per hour and recommend further improvements based on this study. Resulting from applying the advanced fuel handling equipment, operating efficiency of the plant will be increased by reducing the refueling time and accompanied by reduction of radiation exposure during the refueling outage.

Damping values for seismic design of nuclear power plant SC structures (6-1697)

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Background

Structural damping values for seismic design of nuclear power plant structures are specified in Regulatory guide 1.61 for reinforced concrete structures of 4% (OBE) and 7% (SSE), and for steel structures of 3% (OBE) and 4% (SSE), but not for steel-plate concrete (SC, hereinafter) structures. SC structures have been being developed in the worlds for long time, but no research investigates the damping values except that Akiyama et al. concludes the identical damping value of 5% for both RC and SC structures for nuclear power plants. However, the experimental tests conducted by Akiyama are static cyclic loading tests and hydraulic-shaker vibration tests without any mass in the test specimens of both test types.

Objectives and investigations

This paper describes the research work of experimental testing method, analysis of test results and proposed damping values for seismic design of nuclear power plant SC structures.

The concept of this research is to investigate the relative difference in damping values between RC and SC structures rather than to find out the absolute values of SC structures, so that tested are 4 specimens of RC-S, RC-M, SC-S and SC-M where S stands for shear-govern and M for moment-govern. As described in Table 1, the moment-govern specimens are higher than the shear-govern ones, but all the specimens are 1,700 mm wide and 1,700 mm long with wall thickness of 240 mm.

Table 1. Descriptions of 4 specimens.

RC Specimens				SC Specimens			
Symbol	Height (mm)	Re-bars		Symbol	Height (mm)	Surface Plate	Stud Diam. (mm)
		Vertical	Horizontal				
RC-S	1,650	2D22@129	2D16@133	SC-S	1,650	3mm thk.	Ø6
RC-M	2,850			SC-M	2,850		

Conducted method is free vibration testing by rupturing a brittle steel plate which links an actuator and the center of 50 ton mass as shown in Figure 1. Rupturing load levels on each specimen are controlled by pre-designed tensile strength of the linking steel plate.

Results

Natural frequencies of 4 specimens are determined from the experimental test results with respect to the load level as illustrated in Figure 2. It is noticed that those frequencies are similar to the values in design practice.

Figure 3 shows an example of time vs. acceleration curve measured for specimen SC-S at rupturing load of 420 kN, together with fitted curves of exponential function representing free vibration. Consequently, damping value is determined from the exponential function and its natural frequency. Similar analyses are performed to determine damping values of 4 specimens at the different rupturing load.



Figure 1. Test Setup of SC-M Specimen.

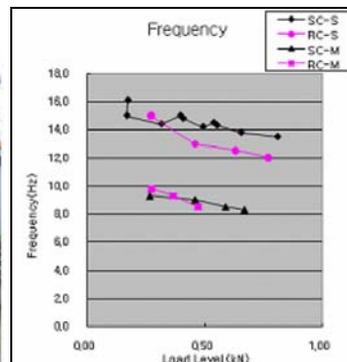


Figure 2. Natural Frequency of 4 Specimens.

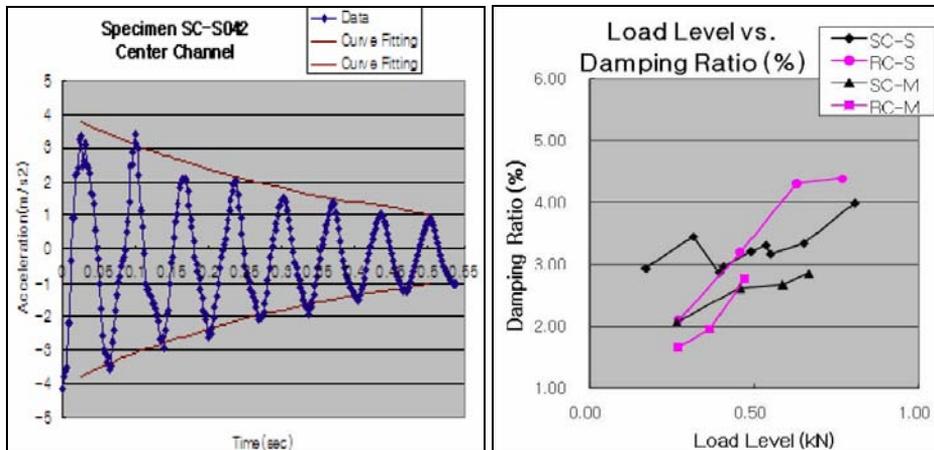


Figure 3. Example of Time vs. Acceleration. Figure 4. Damping values of 4 specimens.

Figure 4 shows comparison of damping values of 4 specimens with respect to load level, where the load level is the rupturing load divided by designed load which corresponds to specimen design strength.

Conclusions

By examining the relative differences in damping values of 4 specimens, it is proposed for SC structure to use the same damping values of 4% as RC at OBE, but 1% less value than RC resulting in 6% at SSE.

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Performance-based fire design of half SC slabs in nuclear power plants (6-1698)

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Background

One of primary elements in steel-plate concrete(SC hereafter) structures is a half SC(HSC hereafter) slab which consists of concrete, top reinforcements, bottom surface steel plate, bottom steel ribs and shear studs. Meanwhile, required fire resistance rating is 2 or 3 hours based on ISO fire conditions for all the slabs of nuclear power plant structures in Korea. It is obvious that HSC slabs are weaker than SC slabs in fire design since bottom reinforcements of HSC slabs are basically exposed surface steel plate and attached steel ribs only.

Michikoshi(1) et al. performed experimental tests by simulating both end-fixed condition for HSC slabs subjected to required loading at fire condition as shown in Figure 1, and found out that 300mm thick HSC slabs meet the requirement of 3 hour rating by relying on end moment capacity.

There is a need for additional fire testing of thicker HSC slabs which are connected to SC walls as well in order to provide a performance-based fire design specification of not only both end-fixed condition but also one end-fixed and simply supported ones.

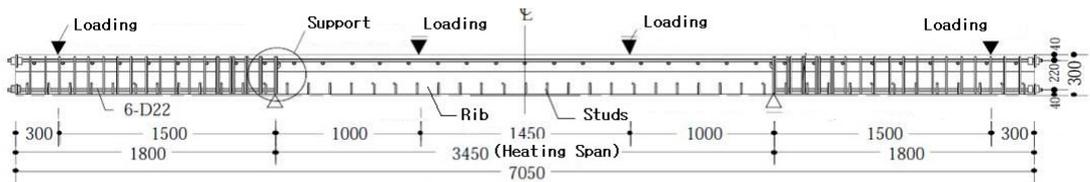


Figure 1. Cross-section of Michikoshi's Specimen for Fire testing.

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analyzed for negative bending at the end and for positive bending at the mid span at the time of 3 hour rating by using segment analysis. Such analysis results for Michikoshi's specimen are compared with test results resulting in good agreement.

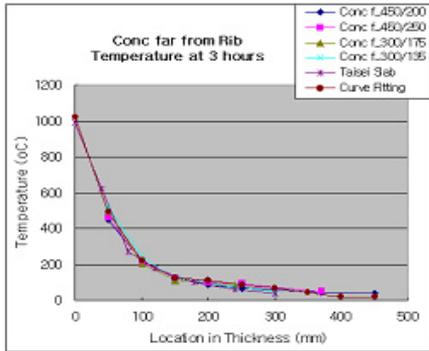


Figure 4. Temperature Profile on Concrete far from Steel Ribs.

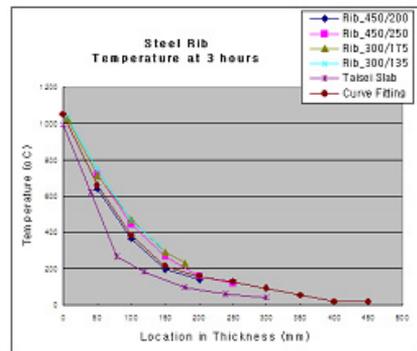


Figure 5. Temperature Profile on Steel Rib.

Conclusions

Based on this research work, Korea Electric Power Industry Code SNG specifies a segment analysis method and a simplified method for performance-based fire design for the 3 hour rating of HSC slabs in both end-fixed, one end-fixed and simply supported conditions.

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Serviceability limit state and crack width analysis of concrete structures in nuclear power plants (6-1706)

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In nuclear power plants typical for reinforced concrete structures is that they are thick and massive compared to structures in ordinary buildings. The needed tightness or long-term stiffness of several critical structural components made of reinforced concrete is the reason that the serviceability is perhaps more stringent requirement than the ultimate capacity in designing of reinforced concrete structures. However, their design is largely based on codes developed for non-nuclear applications, and the challenge is the application of beam theory based code formulas to complicated multidimensional structures of nuclear facilities, where the temperature changes assumed may produce substantial part of loading. A special feature of reinforced concrete is that the effect of temperature changes depends on the stiffness of the structure, which on the other hand caused by cracking. Thus, it is important to have a design tool by which it is possible to update the stiffness of the structure due to cracking and to consider the effect of stiffness changes in redistribution of stress.

The paper presents the in-house design tool comprising of several independent programs that are chained to analyse and dimension three dimensional reinforced concrete structures iteratively. The main parts of the design tool are a commercially available program based on the general finite element method and the in-house programmes that define the cracking, update stiffness and calculate reinforcement bars in different directions. The results of the tool are actual amounts of reinforcement required in each element of the finite element mesh. The amounts can finally be transferred to drawing programs. The program can be used either for dimensioning of reinforcement of a new structure or for analysis of crack widths of an existing structure. The influence of the order in which the loads affect on the structure can also be analysed by the tool.

A specific object of the paper is to compare different design codes in the serviceability limit state design. The design codes compared are the National Building Code of Finland "B4: Concrete structures" and the European Standard

“Eurocode 2: Design of concrete structures”. The calculation formulas of the different design codes are compared and their differences are demonstrated by practical test examples and real structures of nuclear power plants. The results reveal that these codes lead a little bit different reinforcement even if they both are official codes in Finland presently. For example with the same crack width requirement, the amount of required reinforcement according to Eurocode 2 is for bending a bit smaller and for pure tension notably larger than according to B4. If the comparison is made using the same exposure class, the amount of required reinforcement according to Eurocode 2 is smaller than according to B4 both for bending and for pure tension, because the crack width requirement of B4 is often stricter than the requirement of Eurocode 2.

Demands on anchor systems for concrete structures of nuclear facilities (6-1709)

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With respect to nuclear power plants or other nuclear facilities the safety aspects of anchor systems are of great importance especially in view of extraordinary action effects like earthquake actions. In the last two years this importance has become obviously especially in Germany by the outage of several nuclear power plants because of not correct installed metal anchors. In the concerned plants extensive repairs with the exchange of several thousands of metal anchors were necessary.

Generally the anchoring of mechanical components in concrete structures will be realized by steel anchor plates with welds for the connection between the plates and the components. During the erection of nuclear facilities for the so far known fastening points cast-in fasteners in form of steel anchor plates with welded studs will be preferred. After the erection of the buildings for modification measures so called post-installed anchors are the only way for subsequent connections. Supplementary metal anchors can be used for such fastening points. Three types of metal anchors seem to be suitable for the application in nuclear facilities:

- **expansion anchor**: anchor with friction connection for the anchoring of tensile forces
- **undercut anchor**: anchors which develops its tensile resistance from the mechanical interlock provided by undercutting of the concrete at the embedded end of the fastener
- **bonded anchor (chemical anchors)**: threaded bar embedded in the bore holes by an adhesive mortar.

In German nuclear facilities expansion anchors have been used before the development of the undercut anchors. Nowadays undercut anchors will be preferred in Germany for such applications. Until now no bonded anchors have been qualified for the utilization in German nuclear facilities because of the very high demands on the resistance in cracked concrete. But in recent years bond expansion anchors were developed to offer a suitable high performance adhesive anchoring system in cracked concrete.

Metal anchors are building products with high quality demands on the utilization in nuclear facilities. Regarding these demands three aspects have to be considered:

- **anchor product** regarding the licensing aspects for building products which can be used in nuclear facilities
- **design** regarding the verification format for the different limit states especially for the ultimate limit states
- **installation** regarding the requirements during the installation process including the necessary installation protocol.

In Germany the different demands have been established in the DIBt-guideline [1] with the specifications for the licensing of metal anchors and for the design of anchor connections. This guideline represents an extension of the EOTA-guideline [2] to consider extraordinary action effects typically for nuclear facilities like earthquake actions. So for example the load bearing of anchors has to be guaranteed in cracked concrete structures with crack openings of 1.5 mm in a single crack considering cyclic loading.

Because of the new partial safety concept in combination with the new definition of requirement categories for nuclear facilities [3] a revision of the DIBt-guideline has become necessary. In view of this revision started in 2008 latest scientific findings about the anchor behaviour in cracks (see [4]), the gained experience and new knowledge about installation proceedings as well as the nuclear specific demands like the limitation of anchor displacements will be discussed in the contribution.

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Establishment on slip coefficient of slip resistant connection (6-1712)

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A slip critical joint has various values to adopt the proper slip coefficient on various conditions of faying surfaces in following codes: AISC, AIJ and Eurocode 3. However Korean Building Code still regulates the unique slip coefficient, 0.45 regardless of diverse faying conditions. In this study, the slip resistance test including five kinds of surface treatments were conducted to obtain the proper slip coefficients available to steel plate KS SM490A. The faying surfaces were comprised of clean mill, rust, red lead paint, zinc primer, and shot blast treatment. The candidates of high strength bolt were torque-shear bolt, torque-shear bolt with zinc coating, and ASTM A490 bolt. Based on test results, the specimens with shot blasted surface and rusted surface exhibited ks, 0.61, 0.5 respectively. It is recommended that the specimens with zinc primer exhibited ks. The clean mill treated surface were prominently lower values, 0.27. For red lead painted treatment, thickness of coating affected on the determinant of slip coefficient, it is necessary to establish the minimum of ks of 0.2, with coating thickness, 65 μm . For 1,000 hours relaxation, the uncoated surfaces exhibited the loss of clamping force behind 3%, while the coated surfaces within a certain limited thickness exhibited the loss of clamping with a range between 4.71% and 8.37%.

The SWR 1000 containment – civil design aspects in view of high robustness (6-1714)

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Operational lifetime of nuclear power plants of the new generation is normally provided to be 60 years currently. Taking into account a construction period of about 4–5 years together with a decommissioning period of about 10–15 years, time periods of about 80 years are newly requested, where safety-related requirements on construction members have to be guaranteed.

The containment of the SWR 1000 is characterized by a lot of passive safety systems and therefore by a robust behaviour in abnormal operational conditions. This concept will be realized not only in plant engineering but also in civil engineering: The SWR 1000 provides a passive safety concept in its constructional members which fulfil the requirements of high robustness.

Due to these requirements on robustness, materials and structural members are preferred which are visible and therefore easily accessible for monitoring. This results in basic consequences with regard to the application of e.g. pre-stressing system (especially combined with a steel liner), tightness of joints using sealing strips, bituminous waterproofing systems, joints between separated fire compartments. These are only few provisions which can be taken as examples.

The containment structure of the SWR 1000 is therefore provided without pre-stressing, using only non-pre-stressed reinforcing bars. Monolithic construction methods avoid joints, result in a higher bearing capacity of the global containment structure and avoid also difficulties with joint sealants with regard to fire protection and pressure differences. Watertight structural members, e.g. waterproof concrete basements, do not require bituminous sealing.

Construction engineering design with regard to high robustness has to be supported by corresponding codes and standards. These rules should be based on a unified safety concept, providing design procedures for robust structural members and addressing also the global integrity of the reinforced containment combined with a robust liner construction, penetrations and airlocks.

Parallel to the development of the SWR 1000-containment in Germany, the new codes DIN 1045 (reinforced and pre-stressed concrete structures) with parts 1–4 and DIN 25449 (reinforced and pre-stressed concrete components in nuclear facilities) have been issued. A working group has been established in the year 2008 to develop a revision of DIN 25459 (reinforced and pre-stressed concrete

containments) to achieve a harmonization and adjustment to the other codes mentioned above. These three codes, based all together on the Eurocodes regarding a unified partial safety concept, represent for the first time in Europe a self contained and complete set of codes and standards for the design of reinforced concrete containments.

The design of a robust containment will be demonstrated, representing the construction of the SWR 1000-containment. Furthermore, the current state of applicable codes and standards and future developments for new rules are illustrated, to enable the design of robust containments.

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Guidelines and dataware for life cycle management for NPP pipeline supports (6-1774)

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The present development of the Russian nuclear industry requires an increase of power, life duration and reliability of design for new nuclear power plants (NPPs). These goals necessitate the upgrade of existing nuclear industry guidelines and the development of new methods and technologies for life cycle management of NPP components. The development of a “Code of Rules and Guidelines on Support Structures for NPPs with WWER” (SPiR-O-2008) is a part of such work.

The paper describes a structure and basis for SPiR-O-2008 that is created in such form in Russia for the first time. Methodic approaches, which form the base of SPiR-O-2008, cover both design and post-design assurance of capacity of support structures, namely:

- Division of requirements in dependence on safety classes;
- Limit analysis;
- Taking into account of actual finite stiffness;
- Conformance evaluation by calculations or tests;
- In-service inspection (structural health monitoring and damage detection).

The important feature of SPiR-O-2008 is its development on a base of modern information technology. The document is created in the form of hypertext and includes electronic catalogues and 3D models of standard supports and hangers that can be used directly by designers of pipeline systems.

SPiR-O-2008's information maintenance is based on product life cycle management (PLM) – technology recently introduced into the Russian nuclear industry. It involves the use of an integrated information medium for all stages of the life cycle and for all cooperating participants. It also considers standards of information exchange and justification of engineering decisions on the basis of mathematical modeling. There is a logical link created between the mathematical model and the in-service structural health monitoring. Data from monitoring is needed to update the mathematical models of support structures

that take into account their actual state. The last point is important for the substantiation of decisions concerning extension of life duration, repair or replacement.

Another specific feature of SPiR-O-2008 is its reconciliation with foreign Codes and Guidelines (ASME Code, KTA Rules, VGB Guidelines) that provide a possibility of international cooperation within the frame of Russian NPP projects. Standard support structures of LISEGA AG (Germany) are considered as an example of such cooperation in a NPP project with WWER designed by SPbAEP. The paper describes problems arising in this design in connection with Russian design practices.

A case study on a radiation shielding structure for the cold neutron guide at HANARO – focused on a mixed proportion design and fabrication of heavy weight concrete for a radiation shielding (6-1791)

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This paper aims to describe the key factors regarding the design and fabrication of a heavy weight concrete as a shielding material. The differences, when compared to the concrete shielding blocks that are normally stacked around neutron guides, are not only the need for a seismic integrity as a structural material, but also a radiation shielding effect for the neutron and gamma rays from the cold neutron guides. In general, heavy weight concrete is not available as a normal product. That is why this study was conducted, namely to fabricate a shielding material that meets the design requirements and technical standards.

The cold neutron guide should be installed inside a biological shielding that is assumed to be an absorber for neutron and gamma rays. Based on the suggested material, a simulation for the shielding effect has been carried out to evaluate the appropriateness of the proposed design. According to the simulation by help of the Monte Carlo Method, the shielding materials were designed by using a heavy weight concrete in 3.5 g/cc density. Because many kinds of instruments are operating around beam ports, there is a lack of space to accommodate a shielding structure in the reactor hall. So, the maximum shielding thickness should be equal or lower than 800 mm. There is no other way except that the concrete must have a higher than 3.5 g/cc density and a 35 MPa compressive strength at 28 days.

The radiation shielding effect would be mostly affected by the physical and chemical characteristics of the concrete. Generally, it is known that the shielding effect depends on the kinds of aggregates to be composed in concrete. American National Standard provides several cases as an example of typical compositions of representative concretes after curing. Magnetite was considered, but it did not have a proper density as a coarse aggregate, namely at least higher than 4.0 g/cc. Therefore, barite composed of over 90% BaSo₄ was adopted as a coarse aggregate for the heavy concrete. The mixed proportion design for the heavy

concrete was carried out by focusing on the homogeneous density at any part of a structure and a quality control. After that, a mockup test to investigate the possibility of segregation, a cracking by shrinkage and a thermal stress by hydration heat, and workability was carried out. The mockup test was prepared with a half scale at the same height and thickness as the shielding structure. From the results, there is no evidence of a crack induced by shrinkage and a thermal stress after a three months curing. Also, the heavy weight concrete, which was made by a commercial batcher plant with a 100 ± 25 mm slump, can be transferred into a mold at a nearly 60 m length by using normal pumping equipment. The distance is simulated with the same pouring conditions of the reactor building.

In order to increase the density of the concrete, it was better that the air content is lower than the normal case. ACI provides that the air content is defined as $4.5 \pm 1.5\%$ for a normal case and 3.5% for a maximum size of an aggregate at 20 mm and a mild exposure. Also, in ACI 301, the air content may be additionally reduced by 1% in case of a compressive strength above 35MPa. So, it has been determined that the air content at $2.5 \pm 1\%$ might cause the density to be an average of 2% higher than that of the normal case. It has been found that the heavy weight concrete satisfied the design requirements regarding its density. And, the other characteristics for a structural material such as the slump, compressive strength, air content, and chloride content have also satisfied the design requirements. On the other hand, because this shielding structure was designed with a thickness of 800 mm, we should consider controlling the thermal stress induced by the hydration heat up of the mass concrete. Based on the mockup test results, a thermal cracking might be induced at 1.5~2 days after a pouring. It turns out that the heat up decreases after 8 days. So, it should be resolved with protection layers, which are made of a fabric, normally used in curing concrete, to retard the dry up speed of the water in the concrete just after a pouring.

Conclusively, a fabrication procedure has been derived from the mockup test and an analysis. The radiation shielding structure was installed at the designated place for the neutron guides in August 2008. Although there are many difficulties in a radiation controlled area, it sufficed that the work was successively conducted without any technical violation by using the developed procedures. It is hoped that this study will be a useful practice for other cases that require a high density heavy weight concrete to satisfy not only a seismic resistance but also a radiation shielding effect in nuclear facilities.

An efficient structural form for concrete containment structures (6-1806)

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Background

Containment system of a nuclear power plant is an engineered safety feature to meet one of the basic safety requirements of containing radioactivity. It derives importance not only amongst the scientific community as the ultimate barrier against release of radioactivity but also as a readily perceptible and hence psychologically soothing protection as viewed by the general public. Structural design of containments is governed by the design pressure value and the leak-tightness requirement. Other important design loads are those associated with the external events such as earthquakes, wind, airplane crash, and blast shock waves. Generally a secondary containment is provided to resist loads associated with the external events, except earthquake that pervades through every structure, and for other functional aspects such as limiting the ground level releases during an accident.

Large dry concrete containments are the most popular because of the inherent advantages of large volume and less maintenance. Since shell type structures resist pressure loads efficiently, these containments are built as prestressed or non-prestressed reinforced concrete (RC) shells. Concrete is strong in resisting compression but very weak in resisting tension. To overcome this deficiency, either pre-compression is introduced by prestressing to balance the tensile stresses from pressure loading or large cross-sections are adopted to keep the tensile stresses low and prevent through-thickness cracking. A metallic liner is introduced, particularly for non-prestressed RC constructions, to improve leak-tightness. Major drawbacks of prestressed RC constructions are the design and construction complexities that add to the cost and time of the project, besides additional monitoring requirements associated with the prestressing system.

Aim

The aim of this paper is to introduce a structural form that can be used for large dry RC containments with significant advantage in terms of structural capacity and leak-tightness. In the current proposal, the shells are so oriented that the internal pressure loading induces net compressive stresses across the cross-sections and keeps the inner face of the general region of the containment under

compression that eliminates the need for any prestressing system. However, inner faces near discontinuities develop tensile stresses. A structural framework of columns and beams is introduced to support this structural form and to localize the extent of inner face tensile regions around discontinuities, thus limiting the need for metallic liner to small local regions. The framework may be integrated with the outer secondary containment to augment its capability to resist dynamic missile loading due to airplane crash.

Results

The proposed typical structural form is shown in Figure 1. Analysis results for internal pressure loading concurrent with self-weight are shown in Figure 2. It can be seen that the tensile stresses in the inner faces of the primary containment are localized around structural framework members. The need for improving leak-tightness is thus limited to these localized zones and this can be achieved by providing metallic liner in these zones only.

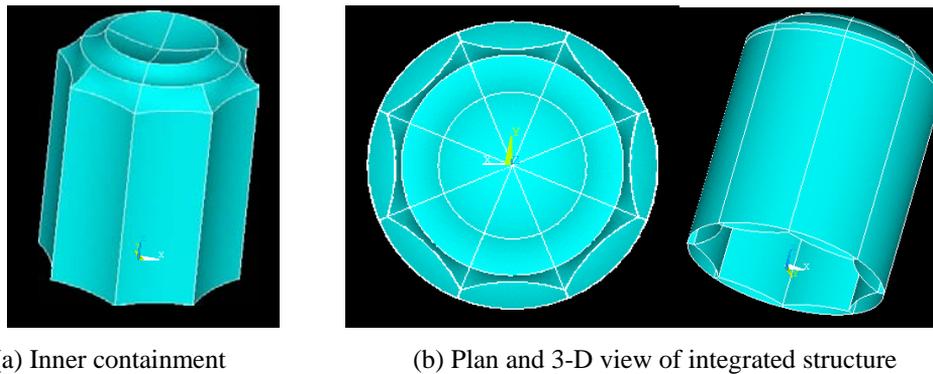


Figure 1. Proposed structural form for containment structure.

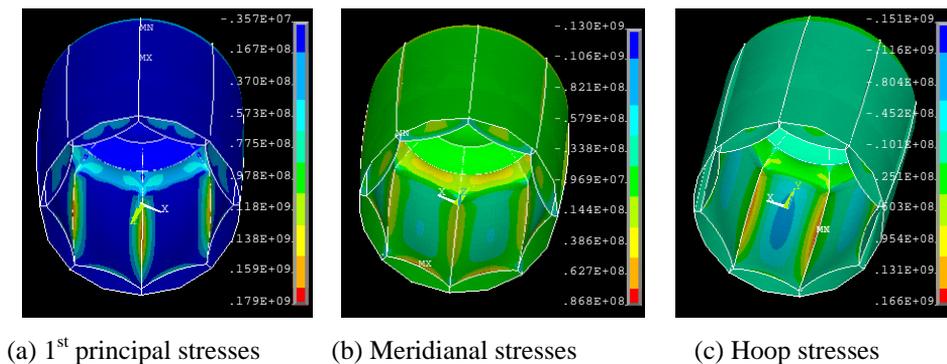


Figure 2. Stresses under internal pressure.

Important considerations for design against missile effects of airplane crash are the global vibration effects, the local structural deformations and the local perforation. Global vibratory effects are hardly a design issue for containments. Providing sufficient concrete thickness based on empirical relations ensures safety against perforation. The structural performance of the outer containment shell may be augmented against the effects of local deformations by integrating the proposed structural framework of the primary containment to the outer containment. Transient analyses carried out for missile loading associated with airplane crash on cylindrical plane shell segments and stiffened shell segments indicate that the stiffening effect of the structural framework can augment the structural performance against local deformations.

Conclusions

An efficient structural form for containment structure is introduced that enhances its structural and leak-tightness capabilities and augments resistance to loading effects of airplane crash in an integrated structural complex of the primary and the secondary containments. It needs to be explored whether the structural framework could serve as the primary seismic resistance system so that strengthening the framework without altering the standardized containment geometry could accommodate any change in site dependent seismicity. Another area for further work is to explore the effect of edge stiffening on the perforation resistance of RC shells, slabs and walls.

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Timber mat protection design for buried utilities subject to impact loads (6-1809)

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The heavy rigging operations necessary at nuclear plant sites during construction, routine maintenance, and special activities such as steam generator replacement require assessment of safety related underground utilities for the effects of postulated accidental impact loads in the form of dropped loads, crane boom drops, etc. Safety related utilities located directly beneath postulated impact locations may require protection, which frequently is in the form of timber crane mats. Sand bags, and soil or gravel mounds have also been used. To design timber mat protection for a buried utility requires understanding of the dynamic behavior and response of the coupled system comprised of impacting element, timbers, supporting soil and buried utility. Calculation of such system response is complicated by a number of factors, specifically a) the nonlinear behavior of the crushed timbers which exhibit elastic, near perfectly-plastic behavior up to a locking strain at which high-modulus strain hardening occurs [Ref.1], b) the stacking arrangement of timber levels which results in tensionless interstices on the timber sides, c) beam-on-tensionless-foundation behavior of the timbers, d) soil compliance at the interface with the timber mat, and e) soil-structure interaction between utility and soil. No evaluative process purporting to address the combined system has been reported on in the literature. Quick reference design aids are suitable for such temporary protective measures. Indeed, alternative load paths would be opted for, before resort to the otherwise necessary detailed analytical or FEM approaches.

This investigation develops design tables and charts, based on studies of cross laid timbers resting on an elastic half-space, and dynamically impacted, end on, by common impacting elements encountered during nuclear plant maintenance. The analysis model for the soil/mat is comprised of a bar with locking material behavior to represent the local dynamic crushing of the top wood timbers [Ref. 1], timber beams on elastic foundation to represent the underlying layers of timber, and Winkler springs to represent the supporting soil. This model is used to relate the rigid impacting element geometric and dynamic properties to the forcing function at the soil/mat interface. The interface forcing function is then be used to determine a free-field soil response [Ref. 2] which is applied to a soil-structure interactive representation of the soil and utility to determine the utility response [Ref. 3].

Based on parametric evaluations for typical and realistic postulated missiles, design tables and charts associating ground surface pressures and underground pressures with impact energies are developed. These representations enable the

rapid design selection of the appropriate number of timber layers. An example illustrating the application of the results to the evaluation of an underground utility is provided.

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Design of modular composite walls subjected to thermal and mechanical loading (6-1820)

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Keywords: Modular, composite walls, thermal load, pressure, interaction

Modular composite (MC) walls consist of steel plates on the exterior to serve as stay-in-place formwork and equivalent rebar. This type of construction is attractive for reducing the schedule and field labor associated with massive concrete construction projects (e.g., nuclear facilities). This paper presents analysis and design approach for MC walls subjected to simultaneous thermal loading and transverse applied loads. The applied loads can be due to accident pressure or a seismic event. Thermal loading can be due to severe temperature gradient across the wall thickness. Such gradient can be determined by performing heat transfer analysis.

ACI 349 provides a simple treatment for determining the effects of temperature gradient on a conventionally reinforced concrete wall. For MC walls, similar techniques are not available that account for actual behavior under thermal and mechanical loads. Based on experimental and analytical research, the authors have developed an approach for determining the thermally induced moment on MC walls. Unlike the ACI method, the proposed method does not require determination of an equivalent (approximate) linear thermal gradient; rather, the actual thermal gradient, which has a sharp nonlinear profile within the first few to several inches of the wall thickness, can be directly used for determination of the thermally induced moment.

For structural analysis of walls/compartments subjected to combined thermal and mechanical loading, the authors have developed an approach that takes into account the self-limiting nature of the thermally induced moment. The subsequent design of the walls is thus not unduly conservative.

The effects of design parameters on the thermal response of an LBE capsule (6-1821)

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The development of a SFR (sodium fast reactor) as one of the advanced reactor systems in Korea requires high temperature irradiation tests of new fuels, claddings, and structural materials. To characterize the performance of these new materials, it is necessary for us to have leading-edge technology to satisfy the specific test requirements such as the conditions of high neutron exposures (~ 200 dpa), high operating temperatures (390–700 °C) and a specific chemistry (Na). The existing design concept of a capsule using Al thermal media, however, is not satisfactory for these high temperature tests. Thus, literature surveys about the system design characteristics of various irradiation devices being developed or used in foreign research reactors (i.e. ATR, MITR, JHR), which are helpful in understanding the key issues for the on-going R&D programs related to a SFR, were conducted to develop new design concepts. For the high temperature irradiation tests in the HANARO reactor, the candidate thermal media as well as internal structural materials as one of the capsule components should have a high thermal conductivity, a high density and a very low reactivity which is needed to obtain the required specimen temperatures, and results a small temperature difference within the specimens. From an extensive survey of the literature, one of the candidate thermal media is selected as an LBE (lead bismuth eutectic alloy) for the high temperature irradiation devices.

Under the current HANARO capsule design practice, in order to evaluate the relative significance of the various parameters on a thermal response, the temperature calculations for the concept of a capsule using an LBE were performed using a finite element analysis program, ANSYS. The concerned design parameters such as the gap between the holder and the specimens (G1), the gap between the LBE container and the external tube (G2), and the thickness of the specimen holder, which are designed to effectively control the temperature of a specimen, are considered as variables. The analysis model for a circular cylinder with multi specimens is generated by the coupled-field elements of PLANE223 with a 2-D structural-thermal field. The results of these studies indicated that the gap between the LBE container and the external tube can have a great impact on the thermal response. However, variations in the gap size between a specimen and a specimen's holder and the thickness of a holder material seem to have no significant effect on the specimen temperature, and the LBE capsule concept can be applied to a high temperature irradiation of new SFR materials in the HANARO reactor.

An investigation on the fuel assembly structural performance for the PLUS7 fuel design (6-1824)

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The extreme level of fuel assembly bow can be the main cause of IRI (Incomplete Rod Insertion), adverse effects on the nuclear design, or handling difficulties impacting nuclear plant performance. In order to better understand the mechanism of in-core fuel assembly structural performance, a computer code (SAVAN) and methodology have been developed by ENUSA. The SAVAN code analyzes the fuel assembly growth and bow using fuel assembly design characteristics and in-core conditions. KNF, Westinghouse and ENUSA jointly developed a new fuel assembly growth and bow computer code (SAVAN2D) for the prediction of in-core deformation behaviour of the PWR fuel assemblies. The SAVAN2D can be efficiently used to facilitate fuel design development, core loading pattern optimization, fuel structural behaviour prediction, and fuel loading/unloading sequence optimization.

The PLUS7™ skeleton and fuel assembly models for SAVAN2D analysis have been developed based on the test results. The PLUS7™ core model has been developed using fuel assembly model as a basic model. The out-core mechanical characteristics of skeleton and fuel assembly and the in-core structural behaviour of fuel assembly were analyzed using SAVAN2D computer code and models. The load-deflection characteristics and deflection shapes of the PLUS7™ skeleton and fuel assembly were compared with the test results to verify the models. The in-core analysis results were compared with the measured data to estimate the growth and bow characteristics of the PLUS7™ fuel assembly. The analysis result shows a good agreement with the test result and measured data and the PLUS7™ bow analysis results were very depend upon the magnitude of initial bow and gap. It was concluded that the PLUS7™ fuel assembly and core models can be utilized for the PLUS7™ out-core and in-core structural performance analysis.

Friction coefficient measurement test on 13MN class tendon of PC strands for prestressed concrete containment vessel (PCCV) (6-1825)

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In recent years, the pressure by the DBA of the containment vessel of nuclear power plants tends to increase because the power generation capacity is getting larger. Along with it, it is necessary that the tendon capacity used for the PCCV is to be enlarged.

The tendon capacities used for the existing PCCVs are 10 MN or less class and the values of the friction coefficient of 10 MN class tendon used to design were obtained experimentally. However, the values of tendon which capacity is more than 10 MN class have not been obtained experimentally yet.

Therefore, in order to confirm the friction coefficient of the 13 MN class post-tensioning tendon which is intend to be used for future PCCVs, the friction coefficient measurement tests using a full-scale mock-up structure of the PCCV have been carried out.

The outline of the test is shown below.

- (1) The friction coefficient measurement tests for the 13MN class tendon were carried out using a full-scale mock-up structure of a PCCV constructed.
- (2) The mock-up structure is cylindrical RC structure of 23,960 mm in internal radius, and the wall 700 mm thick and 2,750 mm high. The friction coefficient measurement tests were executed using three tendon sheath ducts embedded in this mock-up structure. As for the tendon sheath ducts, each duct has 160 mm in internal diameter with zinc electro-coated.
- (3) The tendons used for the tests, whose capacities are 13MN class, are made with 49 bundled No. 15 strands of ASTM A-416 Grade 1860MPa. The primary coating of the corrosion preventing material is to be applied to the tendons before the conducting of the tests.

- (4) The tests are to be executed according to the following procedures.
 - 1) The tendon is installed by pushing the strands one by one through the tendon sheath ducts embedded in the cylindrical wall over 360° . Then the both ends of the tendon are anchored by the anchor heads at the both sides of the buttress.
 - 2) The hydraulic jacks are set at the both anchorages of the tendon. Then the tendon is stressed about 1MN by the jacks on both sides concurrently.
 - 3) After that, the stressing load of the jack on the active side increases until 10MN by 1MN. Receiving load at the anchorage of the passive side is measured at each loading point.
 - 4) The friction coefficient is calculated from the relationships between the stressing load of the jack at the active side and the receiving load measured at the passive side.
- (5) The friction coefficients for curvature obtained by the tests results for three tendons are 0.108/rad, 0.121/rad and 0.124/rad when the friction coefficient for wobble is assumed 0.001/m.

Analytical study for failure probability of PCCV under pressure load after seismic experience (6-1826)

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Purpose of the study

Recently, the importance of Probabilistic Safety Assessment (PSA) is increasing because of growing the social demand to the nuclear safety, and the methods of PSA are developed and improved. Meanwhile, almost all of the seismic PSA have been executed as only the level-1 (the evaluation of the core damage frequency) in the industrial world. The evaluation of the seismic PSA exceeding the level-1 will be needed in the future. In the seismic PSA exceeding a level-1, failure probability of containment vessel has to be evaluated. But, there are only few examples and knowledge about it. In this study, analytical evaluation about the failure probability of containment vessel is conducted and it aims at establishment of the evaluation method in preparation for the seismic PSA exceeding the level-1.

Contents of the study

In the seismic PSA exceeding the level-1, discharge probability of the radioactive material has to be evaluated. Therefore, it is necessary to take local failure of containment vessel into consideration to evaluate the failure probability. In this study, using the FE method, the analyses in consideration of local failure were conducted. The contents of analyses are shown in the following.

(1) Analytical Model

Analytical model was three-dimensional model in consideration of material nonlinearity for actual PCCV. In order to evaluate the structural performance of the PCCV as probability distributions, variations of the elastic coefficient and the strength of construction material were taken into consideration.

(2) Behavior of PCCV by FE analyses

Using the three-dimensional FE model, the analyses were conducted for three cases: the seismic case, the severe accident case and the severe accident after the seismic case. In the seismic case, the seismic response analyses were conducted for the various levels of seismic load, and the whole and local behavior of the PCCV were evaluated in each seismic level. In the severe accident case, the inner pressure load by the severe accident were applied to the model statically, and the whole and local behavior of the PCCV were evaluated in each pressure level. In the severe accident after the seismic case, after the seismic response analysis, where the remaining strain and the crack of construction material were held, the pressure load of the severe accident were applied to the model. Then, the whole and local behavior of the PCCV were evaluated.

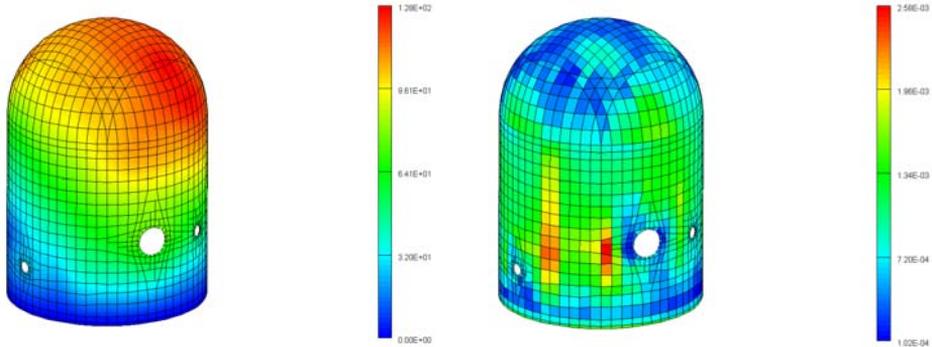
(3) Failure probability of PCCV

Evaluating the failure probability of the PCCV, it is important what kind of condition is to be supposed the failure of PCCV. In this study, by the evaluation of the PCCV behavior obtained from the FE analysis, various failure modes were set up and the failure probability were evaluated for each seismic level and each inner pressure level. In addition, the fragility assessment conducted for each of the seismic case, the severe accident case and the severe accident after the seismic case.

Results of the study

By the three-dimensional FE analysis, the whole and local behavior of the PCCV in process of reaching to the failure condition were able to be clarified, in the seismic case, the severe accident case and the severe accident after the seismic case. Moreover, from these behaviors of the PCCV, failure probability of the PCCV was evaluated for various failure modes. Consequently, the effective evaluation method was shown in preparation for the seismic PSA exceeding the level-1.

6. Design and Construction Issues



Deformation Contour
Severe Accident after the Seismic Case
(1.0MPa after 1000Gal)

Liner Strain Contour (Von-Mises)
Severe Accident after the Seismic Case
(1.0MPa after 1000Gal)

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Civil engineering experiences from the oversight of Olkiluoto 3 (6-1850)

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Introduction

This paper presents civil engineering experiences, which Finnish Radiation and Nuclear Safety Authority (STUK) has received from the oversight work of Olkiluoto 3. Feedback from the Olkiluoto 3 is presented in order to bring some useful information to be considered in future projects for nuclear power plants. Experiences presented in this paper are from nuclear safety authority point of view.

Background

STUK is overseeing the construction of nuclear power plants by inspections and supervision of design, component manufacturing and construction at the site. Quality control and assurance by all players of Olkiluoto 3 has brought lot of information in a form of non-conformance reports and audit findings. Some of the non-conformance reports have been required by STUK based on inspection findings, most of these reports are initiated by licensee and main supplier of the power plant based on their own quality control and assurance.

Nuclear safety relevant and other essential technical questions are dealt with correspondence between STUK and licensee of the Olkiluoto 3 in order to ensure, that all solutions fulfil nuclear safety requirements and from licensee point of view that these solutions are accepted by STUK. Important part for efficient dealing of different, mostly complicated technical questions has been a close cooperation between licensee and supplier of the nuclear power plant so, that also specialists from STUK have participated in technical discussions.

STUK has also ordered independent studies and research work for ensuring the quality requirements [1, 2]. A Finnish research programme SAFIR [3, 4] has an important role for giving the state of art level scientific background in different phases of nuclear power plant delivery.

STUK has been active on developing its' own oversight of Olkiluoto 3. Certain tangible changes to the methods and scope of inspections and supervision have been made in order to be more proactive. Also the cooperation with independent inspection organisations has been developed.

Essential results

Feedback from STUK's studies and decisions and from non-conformance reports of Olkiluoto 3 construction has been collected so, that it can serve in future projects as well as in the further development of regulatory guides on nuclear safety [5, 6].

Civil engineering experiences are presented by structures so, that is possible to understand also the correspondent design criteria and cooperation in delivery chain. Such structures and construction works are excavation works, common base slab of nuclear island, prestressed reinforced protective shell of inner containment, steel liner of inner containment and protective structures against aircraft crash.

Important issue for civil engineering is cooperation between different technical domain areas, like cooperation with process and piping design. Vibration specific questions relating the framework of buildings and vibration resistance of safety equipment of the nuclear power plant are also mentioned. Civil engineering is part of the fire protection of nuclear power plant. Experiences from Olkiluoto 3 fire protection issues are described in this paper as well.

Summary

Experiences from Olkiluoto 3 civil engineering is the main issue of this paper. In order to see the total picture, brief description of how these lessons have been learnt in STUK will be presented. Feedback is collected by structures so, that is possible to understand also the correspondent design criteria and cooperation in delivery chain.

Selective references from standards, regulatory guides and research studies, which are directly related to the reported feedback/experiences:

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Seismic motion incoherency effects for AP1000 nuclear island complex (6-1852)

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The paper addresses the seismic motion incoherency effects on the soil-structure interaction (SSI) response of the AP1000 nuclear complex. The paper addresses both theoretical and practical aspects of seismic incoherent SSI analysis. The paper describes briefly the theoretical basis and specific implementation aspects related to the stochastic and deterministic incoherent SSI approaches. These incoherent SSI approaches were benchmarked by Electric Power Research Institute (Short, Hardy, Merz and Johnson, 2006 and 2007).

Two different structural models of AP1000 NI complex are considered: i) the AP1000-based stick model (used in the EPRI studies) and ii) AP1000 NI20 finite element model (used by Westinghouse for computation of in-structure response spectra). The AP1000 NI20 model was assumed for sensitivity studies with both flexible and rigid basemat. Using AP1000-based stick model, comparative results are shown for a hard-rock site and a soft soil site. Hard-rock high frequency and Regularity Guide 1.60 ground spectra were considered for the two site soil conditions. The recent Abrahamson plane-wave coherency models for hard-rock and soil conditions were applied. However, it should be noted that the soil coherence function is not accepted by US NRC at this time. Only the hard-rock coherence function is permitted by US NRC.

The effect of foundation flexibility on the coherent and incoherent SSI responses is discussed using the AP1000 NI20 model with flexible-basemat and rigid-basemat, respectively. Finally, few incoherent SSI analysis recommendations are stated.

The paper also describes the theoretical basis and key specific implementation aspects related to the incoherent SSI approaches benchmarked by EPRI (Short, Hardy, Merz and Johnson, 2006 and 2007) for performing incoherent seismic SSI analyses for new nuclear plant designs. The implementation of all incoherent SSI analysis approaches is based on the spectral factorization of the coherency kernel. In AP1000 SSI studies, we considered both stochastic and deterministic incoherent SSI approaches. In addition to stochastic simulation approach, three deterministic approaches were considered: i) linear superposition, or algebraic sum, of the scaled incoherent spatial modes (AS in EPRI studies), ii) quadratic

superposition of the incoherent modal SSI complex response amplitudes (transfer function amplitudes) assuming a zero-phase for the incoherent SSI complex response phase (SRSS in EPRI studies), and iii) quadratic superposition of the incoherent modal SSI complex response amplitudes (transfer function amplitudes) assuming a non-zero phase for the incoherent SSI complex response that is equal to coherent SSI complex response phase (not used in EPRI studies). The last implementation is an alternate version of SRSS approach that does not neglect the complex response phase.

For *rigid foundations* the incoherency-induced stochasticity of the basemat motion is driven by the global or rigid body spatial variations (integral variations) of free-field motion and, therefore, is less complex and random than free-field motion. The rigid foundation motion has a smoothed spatial variation pattern since the kinematic SSI interaction is large. Thus, the differential free-field motions are highly constrained by the rigid basemat, and because of this (rigid body), the foundation motion complexity is highly reduced in comparison with the complexity of the local motion spatial variations. For *flexible foundations*, the incoherency-induced stochasticity of the basemat motion is driven by the local spatial variations of free-field motion. The flexible foundation motion has a less smoothed spatial variation pattern since kinematic SSI is reduced. Thus, the differential free-field motions are less constrained by the basemat, and because of this, the (flexible) foundation motion complexity is similar to the complexity of the local motion spatial variations.

Based on our investigations, we noticed that due to their stochastic modeling simplicity, deterministic SSI approaches are limited to rigid foundation applications, as shown in EPRI studies. For flexible foundations, the stochastic simulation approach is the only choice since it accurately captures the statistical nature of the local free-field motion spatial variations. For flexible foundations, the free-field motion local spatial variations are directly transmitted to the flexible basemat motion. Deterministic approaches are not capable of capturing the local phasing of the interaction motions. Thus, only the stochastic incoherent SSI approach was used by Westinghouse for the final incoherent seismic SSI analysis of the AP1000 nuclear island complex founded on hard-rock sites.

Based on the AP1000 NI seismic SSI studies done so far, we state two important conclusions:

- 1) The effects of motion incoherency on the computed SSI response are significant for both the rock and the soil sites. The in-structure response spectra (ISRS) amplitude reductions were twice larger for rock sites than for soil sites, as shown herein. We believe that more study is worthwhile to propose and gain acceptance for the use of soil coherence function by the US NRC.
- 2) The basemat flexibility reduces the kinematic SSI under incoherent waves, and by this slightly decreases the motion incoherency effects on the computed ISRS.

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Out-of-plane shear strength of steel plate concrete walls dependent on bond behavior (6-1855)

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Introduction

Double skin steel plate concrete wall structures (SC walls) have been considered as one of viable options for speedy construction of nuclear power plants. For reliable design and construction, structural behaviors of SC walls under service loading condition including extreme cases need to be investigated by 3-year research program in Korea.

Aim of the work

As one of research topics on their structural characteristics of SC walls, shear strength of SC walls under out-of-plane shear focuses on difference of shear transfer between ordinary reinforced concrete structures and SC walls. Review on the design formulas proposed by JEAG 4618 requires further understanding of shear behavior dependent on bond stress generated by studs and development of shear strength taking into account of difference between ordinary reinforced concrete members. This study proposes a modified shear strength models based on a new interpretation of the effects of bond on the development of tensile forces in steel plate due to arch action as well as the diagonal stress field between plates. The proposed model focuses on the role of bond on arch action which is considered as a primary out-of-plane shear transfer in SC walls of short a/d ratios. The diagonal stress field for truss action combined with a single strut for arch action requires us to consider reduced effective compressive strength of concrete in struts.

Essential results

To this end an arch action dependent on bond action by shear studs is experimentally investigated and effectiveness factors of compressive strength of

concrete of direct strut for arch action are proposed. The contribution of tensile strength and compressive strength of concrete to shear strength due to arch action in short shear span ratio is investigated by assuming diagonal compression field involving a strut. Series of experimental programs in this study prepared specimens of different shear-span-to depth ratios, web reinforcement steel ratio, and spacing of shear studs for verification of proposed shear strength models with test results. Two different types of SC walls were fabricated: specimens without ribs and specimens with ribs. In the first year experimental program, tests for non-ribbed SC wall were performed focusing on three important parameters: shear span ratio, plate thickness, and shear reinforcement ratio. Also two different loading patterns were applied: double curvature systems and single curvature systems within shear span.

Conclusion

The proposed formula show good agreements with the test results including those from JEAG. With consideration of bond and its effect on strut strength for arch action the proposed shear strength model is able to predict shear strength of SC walls. The strength models consider effects of bond strength on the interface between concrete and plates with determination of the width of strut and effectiveness factor for strut in biaxial stress state. Also, it is necessary to clearly define shear span to beam depth ratio for the slope of diagonal strut explaining arch action irrespective of single or double curvature of moment distribution. Otherwise, conventional a/d ratio has confused engineers blindly following design formula without understanding of shear transfer by arch action. Increase in flexural capacity by ribs requires higher shear strength for ductile failure. Strain distribution in plates indicates that assumption on average bond stress, μ_{avg} , in the proposed model be reasonable. Experimentally observed strain distribution in shear bars and ribs showed stress states expected from shear strength model.

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Development of the simplified fuel assembly model for the fuel assembly SSE and LOCA analysis (6-1858)

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Introduction

The load on the fuel assembly under accidents and postulated events like Safe Shutdown Earthquake (SSE) and Loss of Coolant Accident (LOCA) events shall not results in permanent deformation that would prevent effective emergency cooling of the fuel or that would prevent safe reactor shutdown. Under the SSE and LOCA events, the fuel assembly lateral deflection and grid impact force between fuel assemblies are obtained by the dynamic transient analysis for the reactor core finite element model.

The impact behavior between fuel assemblies shows non-linear characteristics, because fuel assembly shows non-linear dynamic characteristics and its structural geometry is complicated. Furthermore, since a reactor core consists of a large number of fuel assemblies, the dynamic behavior of the core under the postulated events is very difficult to analyze due to the nonlinearity and huge model size. Therefore, it is necessary that fuel assembly model is simplified with considering the dynamic non-linear characteristics in core analysis. Until now many researches for the simplification of fuel assembly model have been performed.

In this study, a simplified fuel assembly finite element model for the Westinghouse type 17×17 RFA has been developed. To obtain the simplified model, the optimization algorithm of ANSYS code was used, and the configuration of the model was determined by the sensitivity study. The simulations for static test, pluck vibration test, pluck impact test were performed using the model. The simplified fuel assembly model was verified by comparison with fuel assembly mechanical test results.

Analysis results and discussions

The object function of optimization problem for the simplified fuel assembly model was defined as the minimization for the difference between the test natural frequency of fuel assembly and optimized natural frequency of the

model. The constraints of the problem were defined as tolerance range for frequency at the each mode.

In order to obtain the optimized model, the design variables of model considered the beam rigidity (I) for fuel rods and guide thimbles, and the rotational rigidities (R_T and R_B) at top and bottom end to simulate the interface rigidities between top/bottom nozzles and core plates. And, the rotational springs (K_T : top grid, K_B : bottom grid, K_M : mid grid, K_I :IFM grid) between each grid and top and bottom ends were considered to simulate the friction resistance between grid spring and fuel rod. The configuration of the model is shown in Fig. 1.

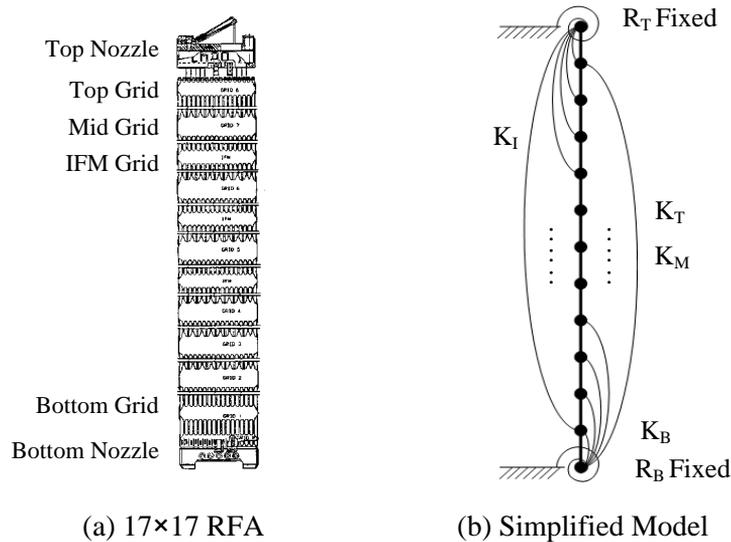


Figure 1. Configuration of Simplified Model.

The sub-problem approximation method in ANSYS was used for the optimization of model, and the design variables for beam rigidity and the stiffness for each rotational spring were optimized. After preliminary optimization, the fixed boundary conditions at top and bottom end were considered, because R_T and R_B were shown too high. The optimized results of the model are shown in Table 1 and Fig. 2.

Table 1. Comparison of Natural Frequencies with Test Results.

Mode		1 st	2 nd	3 rd	4 th	5 th	6 th
Frequency (Hz)	Test	3.65	7.8	12.3	17.7	24.2	30.5
	Analysis	3.65	7.89	12.53	18.44	24.38	34.12

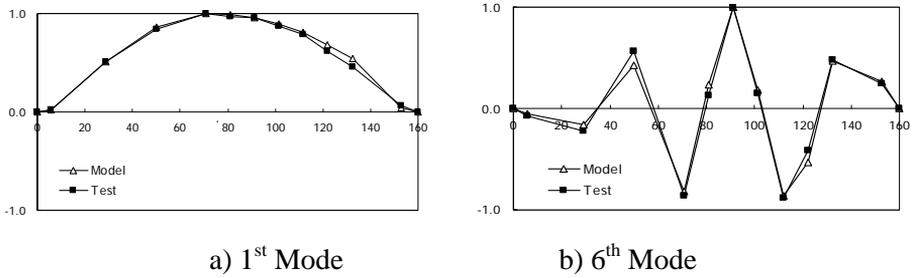


Figure 2. Comparison of Mode Shapes with Test Results.

In order to verify the model, the simulations for static test, pluck vibration test, pluck impact test were performed. The results are shown Fig. 3, Fig. 4 and Table 2. Transient analysis for core model was performed too. The maximum impact force was shown to be 2% differential compare with current design code results.

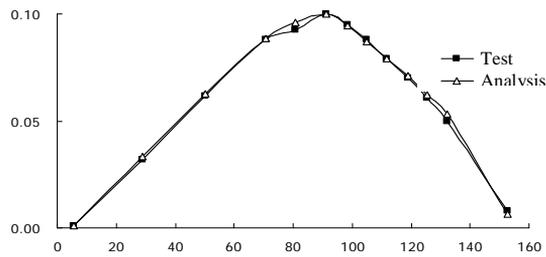


Figure 3. Comparison of Static Analysis with Test Results.

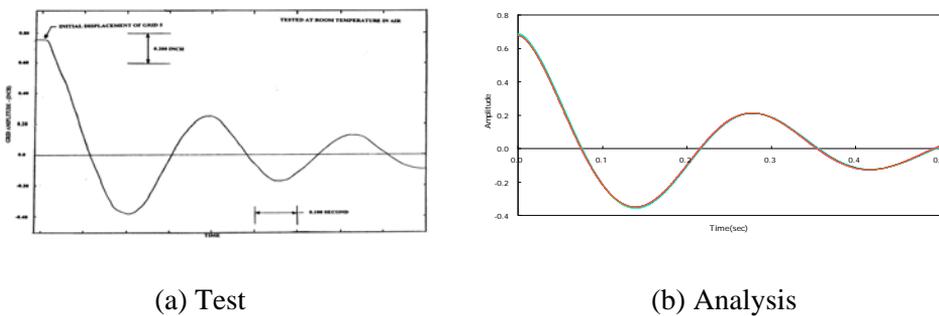


Figure 4. Comparison of Pluck Vibration Analysis with Test Results.

Table 2. Comparison of Pluck Impact Force with Test Results (unit: lbs).

Items	Grid 3	Grid 4	IFM 1	Grid 5	IFM 2	Grid 6
Test	1,255	1610	295	1325	-	910
Analysis	1431	1544	334	1517	284	1309

Summary

The simplified fuel assembly model for fuel assembly SSE and LOCA analysis has been developed with ANSYS code to evaluate the structural integrity of 17×17 RFA fuel assembly under SSE & LOCA events. The model has been developed using optimization method and design variables have been determined by sensitivity studies. The optimized model has a good correlation with fuel assembly mechanical test results and in-reactor impact behavior of current model.

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Concentration of plastic strain in the steel liner near the equipment hatch in a 1:4 scale prestressed concrete containment model (6-1903)

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A substantial part of nuclear reactor containments in US and Europe are designed with an outer bearing concrete structure and an inner sealing consisting of a tight-welded steel liner. The liner constitutes the ultimate leak-barrier which prevents leakage at high internal pressure loads.

It has been shown in containment scale tests that the global displacement measured at liner failure do not correspond to the critical strain level for the liner. A general conclusion from these results is that some type of strain concentration has to take place to get this “early” liner failure. In work carried out at Lund University non-uniform deformation of concrete and discontinuities such as penetrations is concluded to be important issues giving concentration of strain in steel liners (see [1], [2] and [3]).

In year 2000 a 1:4-scale containment model was tested by over-pressurization at Sandia National laboratories (Sandia 1:4, see [4]). The first leak in this test was concluded to be caused by tears in the steel liner found near the equipment hatch. In work made at Lund University, a finite element model of the region near the equipment hatch show that high localized strains will develop in the vicinity of the bend line (see [2] and [3]). In this local study the liner anchor profiles (connecting the liner to the concrete) is assumed to follow the uniform global expansion measured in the containment test model, i.e. the influence of non-uniform displacement due to concrete cracking is not included.

In the study presented in this paper the model presented in [2] and [3] is improved by including the non-linear behavior of concrete. The sub-model is also improved by using more realistic boundary conditions given from a global finite element model of the Sandia 1:4. The global FE-model was made by Scanscot Technology within the ISP-48 project (see [5]).

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Structural design of replacement emergency core cooling filtration system (6-1907)

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Keywords: modular Finned Strainer™, AECL, ECC, strainer, seismic, submerged, hydrodynamic, LOCA, thermal

This paper summarizes the results from recent design experience of ECC (Emergency Core Cooling) strainers. ECC strainers are designed for use in emergency situations (LOCA event) to filter debris from the water used to maintain the cooling of the reactor core. After the incident at Barsebäck-2, a Swedish BWR, nuclear regulators around the world started re-evaluating ECC systems in place at operational NPPs. Considerable research has been done to determine adequate design parameters. Based on new debris definitions it was determined that present filtering systems do not have sufficient capacity to provide reliable coolant circulation in case of a LOCA. The existing filtration surface was found to be very small compared to the one needed per new research. As a result of this new design solutions had to be implemented and AECL has developed its modular Finned Strainer™ design. This strainer consists of porous fins attached to a common header and can be modified to fit a wide variety of conditions. The design process for replacement of containment sump strainer is a very complex one due to limited spacing in RB basements and the need for compact and at the same time adequate in suction surface strainers. The challenge for complex design was not only space limitation, but also significant suction pressure and temperature variations. In addition to this, design requirement for considering a seismic event during or following a LOCA imposes that seismic design be done for submerged under water conditions. The performance of the equipment had to be evaluated for all those loads, the major of which are suction pressure, temperature and seismic in submerged conditions. Each one of those is challenging on its own, but their simultaneous presence further added complexity to the problem. The suction pressure major challenge was the need for relatively large surfaces with limited options for stiffening due to hydraulic flow limitations to avoid flow blockage. Thermal elongation was significant and because of the need for long suction trains, modularization and special consideration to sealing had to be given, together with allowing thermal expansion within each module. Seismic design under submerged condition requires that hydrodynamic loads due to fluid – structure interaction be considered in addition to seismic inertia load due to selfweight. In the area of fluid structure

interaction, a number of research studies have been published, however, there are no prescriptive design standards. The event that causes the most critical loading combination is a Safe Shutdown Earthquake occurring during a Loss of Coolant Accident (LOCA) i.e. while the strainer is in a submerged condition.

The ability of the strainer to perform its safety function during and after this event has been demonstrated by analysis. Its ability to function during and/or following one safe shutdown earthquake (SSE) event, preceded by a number of operation basis earthquake (OBE) events has been demonstrated as well. Special design of interconnecting ducts and pipes to the pumps had to avoid imposing any additional loads on the pump inlet. Due to congested environment, drilling limitations and the presence of Reactor Building liner, supporting and anchoring the equipment was a challenge itself. This paper describes the design experience of the Finned StrainerTM on a number of NPPs. As a result of approaches used all challenges were successfully overcome and a reliable and robust design was produced by AECL.

Assessing the reliability of seismic base isolators for innovative power plant proposals (6-1918)

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Most innovative power plant proposals are conceived so as to eliminate or reduce considerably, at the design level, the occurrence of internal accidents. As a consequence, the core damage frequency associated to internal events is significantly decreased, to the point that values below 10^{-8} events per year often are declared. In this situation, external events (such as earthquakes) become dominant and reducing their damage probability presently is a challenge. In particular, for seismic events a still unsettled problem is the definition of the design earthquake, expected to be significantly more severe than nowadays SSE's, to be used when so low values for the damage frequency are demanded.

A possibility that presently is under study refers to the introduction of base isolators, consisting of laminated rubber bearings capable of very large deformations. Such isolators have been successfully employed in several instances for buildings, bridges or tanks operating in seismic areas and it is felt that similar benefits could be gained in the nuclear context as well. The experience gained so far is significant, but is limited to ground motions associated to the earthquakes that are presently considered for design purposes. Whether such devices would maintain their effectiveness for more severe seismic inputs is a question still open, which is felt to deserve some study. This paper is intended as a first step in this sense. Rather than facing the problem of the definition of the design earthquake, a probabilistic reliability assessment is performed on a typical bearing, with the aim of determining the maximum peak ground acceleration (compatible with a given design spectrum) that can safely be undergone.

Analyses are performed on a simplified model which considers rubber as an incompressible hyperelastic material undergoing large strains and metal disks as rigid. The model is simple enough to permit the expression of displacements, stresses within the rubber and at the rubber-metal interface as functions of the material parameters by means of comparatively simple formulas, for different values of the vertical and horizontal forces acting on the isolator.

Such forces were computed for different ground acceleration time histories, referring to earthquakes of increasing amplitude and are available as input data. They are used to define families of fragility surfaces for the isolator, to be used as a basis for its reliability assessment under different earthquakes. As limit situations, the following are considered:

- Excessive horizontal displacement
- Limit strength in the rubber
- Strength of the rubber-metal interface bonding.

Such limits are treated as random variables, as the material constants and the parameters governing the internal dissipation of the rubber. A Monte Carlo procedure is adopted to evaluate the failure probability of the device for different peak ground accelerations.

The results presented refer to preliminary computations, only a first step toward a reliable assessment of the earthquake magnitudes that do not jeopardize the isolator functionality. Even at this early stage, however, the effectiveness of the procedure is apparent.

Low-activation concrete design methodology for reducing radioactive waste. Categorization of low-activation concrete by low-activation factor (6-1923)

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Introduction

Concrete is very valuable and inexpensive material, however it can be changed to be expensive and hard to deal with in use of a reactor after long operation. One of the counter plans for the above is to use low-activation concrete instead of the ordinary concrete, that will reduce radioactive waste and could be even below clearance level in decommissioning and that is very useful in term of life cycle cost. Instrumented neutron activation analysis showed that Co and Eu were the major target elements which decide the radioactivity level of reinforced concrete in decommissioning stage, and a several material were selected as a low-activation raw material from wide survey of raw materials for concrete (typically aggregates and cements). With the candidate of raw materials, several low-activation concrete were proposed for various portion of the reactor, which reduction ratio were 1/10 to 1/30 which were mainly consist of limestone and low heat cement or white cement, comparing to the ordinary concrete in $\Sigma Di/Ci$ unit, where “Di” indicates concentration of each residual radioisotope, Ci defined by IAEA as a clearance level, and suffix of “i” indicates each radioisotope.

Aim of the work

National funded project for development of low-activation design method for reduction of radioactive waste below clearance level were started from 2005 to 2009 with aiming (1) development of a database on the content of target elements, which transform radioactive nuclides, in raw materials of reinforced concrete, (2) development of calculation tools for estimation of residual radioactivity of plant components, and (3) development of low-activation

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materials for concrete such as cements and reinforcing steel bars for structural components. So, development for low-activation concrete design reducing radioactive waste have been conducted in the project mentioned above, and 13 papers were presented in the last SMiRT19. For the applying low-activation concrete to the real reactor portion, effective evaluation of every uncertainty between the designed concrete and execution concrete are necessary. So, the comparison of calculated activation for the mix proportion based on raw data, to the concrete by the mix trial was conducted in this paper.

Essential results

The projects have conducted several results, such as development of new low activation cement, additives, and reinforcement bar, material data base including more than 2000 raw materials, low-activation material development system and activation mapping system. In additions, about 100 mixture proportions of low activation concrete by tons of execution experimental works for several types of low-activation concrete, which were conducted to categorize as conventional, high performance and boron added, with reduction rate of radioactivity to the ordinary concrete from 1/10 to 10000.

The materials use for Low-Activation concrete are, white cement and low heated cement for the cement, pure limestone for the aggregate, and limestone powder and silica fume for the additives. These materials were selected by the wide survey of instrumented neutron activation analysis. Pure limestone was used for fine and coarse aggregate with the different density. The trial mixings were conducted for 2006 to 2008, and materials were ordered two times to four times. Physical properties as a raw material for concrete were tested with the confirmation of Low-activation by Instrumented neutron activation analysis.

Low-Activation Factor (LAF) is tentatively defined as the ratio of the $\Sigma D/C$ for Low-Activation concrete to the ordinary concrete in the assumption of certain neutron field condition with Clearance level by IAEA. On the other hands, each trial mixing test produced many samples for several tests, in order to confirm the proper physical properties as structural concrete in the reactor, as well as Low-Activation performance. This paper described the relationship of Low-Activation Factor to the cement content per unit volume of concrete in the mix proportion, compressive strength of concrete, and dried density. The three distributions of above relationship to the LAF has similar tendency that higher LAF become the lower values (the cement content per unit volume of concrete in the mix proportion, compressive strength of concrete, and dried density). In additions, the comparison of Low-Activation Factors of measured data by instrumented neutron activation analysis of executed trial mixing concrete and those of calculated data by instrumented neutron activation analysis of raw materials and mix proportion were conducted, and average value of the ratio of LAF for measured data to that for calculated data is close to 1.

Conclusions

More than 50 kinds of developed Low-Activation concrete were categorized by defined Low-Activation Factor (LAF). The relationships of LAF to the typical physical properties of concrete, which were the cement content per unit volume of concrete in the mix proportion, compressive strength of concrete, and dried density, were evaluated. It was clarified that these relationships had similar tendency.

This work is supported by a grant-in-aid of Innovative and Viable Nuclear Technology (IVNET) development project of Ministry of Economy, Trade and Industry, Japan.

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Concrete shrinkage taken into account as crack width assessment (6-1924)

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Context

In very large buildings with connected walls, such as large Nuclear Power Plants, concrete shrinkage strains have to be considered because of elements differential strains, as required in the new European construction code (EC 2). The fastest engineering method consists in considering shrinkage as an equivalent thermal strain, which is in fact computed as internal forces. In EPR, for Flamanville 3 conditions, this first method led to enormous reinforcement ratio in the lower part, so EDF with its partners proposed a new methodology based on crack width assessment.

Methodology summary

At first, we compute the shrinkage differential strains, “ ϵ_s ”, depending on moisture conditions and elements thickness. Then we fix a reinforcement section as a calculation hypothesis in order to estimate the distance “ S_s ” between cracks, which is independent from the loads. Consequently, we can assume that a certain crack width value “ W_s ” is consumed by the shrinkage itself, with $W_s = \epsilon_s S_s R_{ax}$ (R_{ax} restriction factor). So the available crack width for the other loads is the remaining crack width. From cracking theory and according to EC2; we can deduce steel stress σ_d . So the structure design (reinforcement mainly) can be undertaken with this allowable limit value σ_d , in the load combinations where shrinkage as to be considered.

Conclusions

This new methodology is more physical than the one with a thermal equivalent load, so safety requirements are satisfied in focusing on crack width assessment, which is a performance approach. It has allowed EDF to size a reinforcement ratio, which is compatible with concrete technical rules, especially in the areas near the raft. Of course this method remains in accordance with durability hypotheses and other requirements connected to nuclear specificities.

Design of suspended ceilings in main control room of units 5 and 6 of Kozloduy NPP (6-1933)

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In 2007 was realized EQE Bulgaria project for new suspended ceilings in Main Control Room of Units 5 and 6. The project purpose was to be replaced the old suspended ceilings with new ones, which are modern with broken up design, on different levels and with curved outlines. Ergonomic lighting with controlled brightness was built in the ceilings. It was also improved the air exchange and the climate in the rooms, thanks to the new built in air-conditioning system.

The bearing structures of the new suspended ceilings were qualified for new seismic loads.

The project was implemented in short terms and resulted in improved comfort and ergonomic working environment for the operators in Units 5 and 6 of Kozloduy NPP.

Soil remediation for seismic design of independent spent fuel storage installation (ISFSI) pad (6-1935)

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Keywords: ISFSI Pad, seismic, soil remediation, soil mixing, compaction, grouting

Interim spent fuel storage, using a U. S. Nuclear Regulatory Commission (NRC) approved Dry Cask Storage System (DCSS) is an acceptable means of spent fuel management, until the U. S. Department of Energy accepts and stores the spent nuclear fuel in a high-level waste repository. The DCSS' for Independent Spent Fuel Storage Installation (ISFSI) are massive steel and/or concrete structures, loaded with spent nuclear fuel, and stored (in most cases unanchored) outside on reinforced concrete pads. The storage cask vendors have specific requirements for critical soil parameters under the reinforced concrete pads.

Requirements for critical soil parameters under the reinforced concrete pad foundation supporting the DCSS have to be met to ensure stability of the pad when challenged by natural phenomena such as seismic events. These requirements vary for different vendors and are described in the vendor-specific Certificate of Compliance (CoC) issued by NRC. In instances where the existing soil is vulnerable to potential liquefaction, and/or settlement because of a design-basis seismic event, various approaches could be used by licensees to stabilize the natural soil. The reinforced concrete pad foundation is required to satisfy the safety objectives of Title 10 of the Code of Federal Regulations (10 CFR) Part 72. Regulatory Guides (RGs), NUREGs, Standard Review Plans (SRPs) and other guidance documents are available to assist an applicant in complying with the regulations.

This paper will: 1) provide an overview of selected approaches that could be used for meeting the seismic demand on the storage pad for ISFSIs licensed under the provisions of 10 CFR Part 72.210; 2) discuss the design process and compare the relative merits of these approaches; 3) investigate the potential effects of soil remediation; and 4) discuss the resulting soil-structure interaction effects and the seismic input ground motions appropriate for the design of the foundation.

Improving constructability of the new generation nuclear construction through improvements in design efficiency and use of high-strength reinforcement (6-1937)

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Preliminary indications are that new generation nuclear structures will need significant amount of reinforcement in order to make them conform to the current regulatory requirements, Codes and Standards. The concern is that amount of reinforcement required is likely to cause congestion and consequent concrete placement problems thus impacting constructability of new generation nuclear plant structures. The factors that contribute to the reinforcement requirements include:

1. Creeping loads due to revised seismic hazard criteria/ground motion data.
2. Standard plant concept which requires that a standard plant be designed for a suite of soil conditions ranging from very poor soil to rock site.
3. 2-Step Method of Analysis which involves using the envelop of seismic forces (irrespective of the fact that they may occur at different times) from the soil structure interaction model analysis as static input to structural analysis model for design of concrete elements.
4. Interpretation of finite element results – because of lack of a rational methodology, the following process is currently adopted for design of concrete walls.
 - i. The design is based on envelop of element forces which may occur at different times during the analysis.
 - ii. The out-of-plane bending is assumed to occur simultaneously with the in-plane forces. Note that out-of-plane bending of wall is generally a high frequency mode and does occur simultaneously with in-plane shear.
 - iii. A sectional methodology is applied to design elements in the vertical and horizontal direction which are assumed to act independently as columns.
 - iv. In general, the reinforcements required for in-plane shear and membrane normal in-plane and out-of-plane forces are determined independently and added to get the total reinforcement for the walls.

The larger design loads combined with many layers of conservatism discussed above add to the required reinforcement and potential congestion. The paper discusses factors that will help reduce the overall conservatism in the analysis/design process. The incompatibility of using the FEM analysis results with design approaches prescribed in the Codes/Standards is discussed. Based on this discussion, a rational and transparent analysis/design approach aimed at reducing the overall conservatism is outlined.

In order to reduce the reinforcement congestion and improve constructability, use of high-strength reinforcement (Gr 75 as a minimum) is proposed in nuclear construction. Note that Grade 75 reinforcement is permitted in ACI 318 under ASTM A615 and has been in use for over 20 years now. The chemistry control and bend requirements of Grades 60 and 75 are essentially the same. The elongation requirement, which is used as a measure of ductility, is 7% for Grade 60 and 6% for Grade 75 for No. 9 and larger bars. Note that No. 9 and larger size bars are predominantly used in nuclear construction.

ASTM A615 Grade 60 reinforcement has been successfully used in nuclear and defence construction to resist both seismic as well as impact loads. Since nuclear structures are designed to remain essentially elastic during a design seismic event, a marginal reduction in elongation requirement [from 7% to 6% elongation] should not be an issue for use of Grade 75 reinforcement. For impact type of loads, the expected ductility of concrete elements with Grade 60 reinforcement is given in Appendix F of ACI 349. Although this appendix calls for use of A706 reinforcement, the allowable ductility values are still based on ASTM A615 reinforcement used previously. As a result of this, no significant changes are necessary for use of Gr 75 reinforcement for impact resistance. Note that European regulations/Codes also permit use of equivalent Gr 75 reinforcement in nuclear construction.

Use of high-strength reinforcement in nuclear construction is consistent with recent trends in the construction industry. Several high profile projects using up to 100 ksi reinforcement have recently been showcased in some major industry magazines. Furthermore, there is a proposal in ASTM to get ASTM A706 Gr 75 and Gr 80 reinforcement standardized. The ACI Task Group ITG-6 is proposing a design yield strength of 100 ksi for flexural tension reinforcement, 80 ksi for compression reinforcement, 60 (or possibly 80) ksi for shear reinforcement, and 100 ksi for confinement reinforcement in columns and shear walls. Note that 100 ksi for confinement reinforcement is already accepted in the current ACI 318 Code.

The design process improvement outlined in this paper will help reduce the overall reinforcement requirement thus improving design efficiency and reducing potential for congestion. The use of high-strength reinforcement will further improve constructability by allowing larger spacing between reinforcement to ensure quality concrete placement that is warranted in nuclear construction.

Implementation of high-performance concrete in the ACR-1000 containment structure for 100 year design life (6-1969)

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Introduction

The ACR-1000* reactor (Advanced CANDU Reactor*) is designed for a 100-year plant life including a 60-year operating life and an additional 40-year decommissioning period. The ACR-1000 containment structure consists of a vertical cylindrical perimeter wall, founded on a base slab and hemispherical dome at the top. The perimeter wall and the dome are prestressed concrete with post-tensioning tendons in the horizontal and vertical directions as well as reinforcing bars in both directions. The inside surfaces of the containment structure are lined with a carbon steel liner. The base slab is a reinforced concrete structure.

It is evident that the service life performance relies not only on the Ageing Management Program (AMP), but is also strongly influenced by the design strategy and material characteristics [1]. It is believed that improved performance during the design life of a structure can be achieved by implementing durability design criteria and improving material characteristics. Therefore, for a new plant, the Plant Life Management Program (PLiM) starts at the design process stage and continues through the plant operations and decommissioning stages.

Modern building codes are increasingly based on performance specifications for durability (Performance Based Design) [2]. In the development of the ACR-1000, particular attention is paid to specifying structural and long-term durability performance as part of the technical requirements. Many recent innovations in advanced concrete materials technology have made it possible to produce modern concrete with exceptional performance characteristics. The concrete

* ACR-1000® (Advanced CANDU Reactor®) is a registered trademark of Atomic Energy of Canada Limited (AECL)

performance strongly relies on four key factors a) material ingredients, b) mix design, c) concrete production and (d) curing.

Aim of the work

AECL has been active with an R&D in order to implement this new approach, including studies on mix design, analysis techniques, and construction procedures to reduce the risk of cracking at early ages. The long-term performance of the containment structure is strongly influenced by crack control, particularly at early ages. Successful implementation of the program recommendations will enable high performance and time-effective construction of the reactor building base slab, containment wall and dome. Implementation of a continuous-cast concrete base slab has already been addressed in an earlier phase of the research program [3].

Essential results

This paper provides the results of the research and development program for the containment wall and dome. Laboratory and field-trial tests, together with thermal and stress analyses for the probable range of concrete mixes have been carried out in order to predict the mechanical properties, temperature variations, thermal stresses and risk of cracking for a thick pour concrete containment wall and dome. Performance criteria for a 100-year service life and the type of specification required to meet the performance criteria have been established.

Laboratory and field-trial tests, together with thermal and stress analyses for the probable range of concrete mixes have been carried out in order to predict the mechanical properties, temperature variations, thermal stresses and risk of cracking for jump-form construction of the wall and dome. Mechanical and thermal properties, obtained from the field trials were employed in the finite element thermal-stress analysis program in order to predict the time-dependent risk of cracking at early ages of the wall and dome. Figure 1 shows the temperatures measured near the centers of the $1 \times 1 \times 1$ m insulated cubes that were cast as part of the field trials.

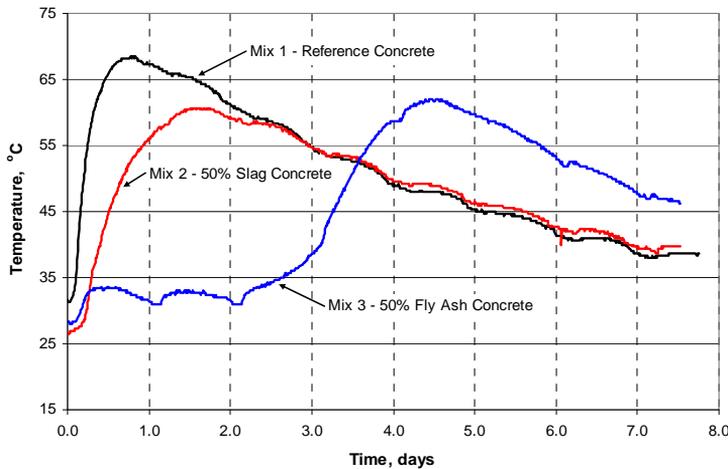


Figure 1. Measured temperatures near center of insulated cubes for different concrete mixes.

Summary

The development program provides the necessary recommendations and guidance on mix design ranges, analysis tools and construction techniques to build the containment structure including the wall and dome of the ACR-1000 reactor building.

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Structural analysis, design and detailing of reactor vault in prototype fast breeder reactor (6-1972)

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Introduction

Prototype Fast Breeder Reactor (PFBR) is the first Reactor of its kind presently under construction at Kalpakkam in INDIA. This paper deals with different aspects related to structural layout, analysis, design and detailing of Reactor Vault (RV), located in Reactor Containment Building (RCB) of PFBR, bringing out certain typical considerations made, analysis procedures adopted and constructability aspects.

General layout of reactor vault

The Reactor Vault housed in the Reactor Containment Building (RCB) form a major structural system. This Building forms one of the buildings constituting the Nuclear Island Connected Building (NICB), an integrated structure constituting of eight buildings, viz Reactor Containment Building (RCB), Steam Generator Buildings (SGB1 & SGB2), Fuel Building (FB), Rad Waste Building (RWB), Electrical Buildings (EB1 & EB2) and Control Building (CB) integrated to form a single integrated structural system. The Reactor Vault comprises of an inner wall and an outer wall (Circular concentric RC walls), and the Reactor Assembly (comprising of Reactor Vessel and the core) hanging from the outer wall. Inner wall is lined with metallic liner, and Biological Shield Cooling System (BSCS) is provided behind the liner for cooling purpose, (Figure 1). A Safety Vessel (SV) is provided outside the Main Vessel (MV) to act as a barrier against sodium leakage from the MV. A roof slab is provided for the Reactor Assembly, over which, different equipments / operating platforms get supported.

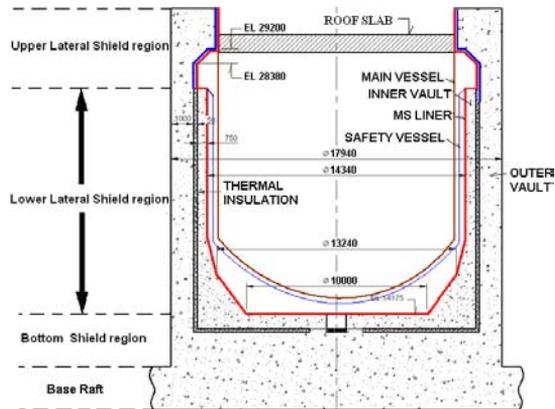


Figure 1. Schematic of Reactor Vault along with liner arrangement.

Structural analysis and design

Structural analysis of the Vaults involve performing analysis under normal conditions (where sodium is contained in the MV) and under an assumed condition of leakage in the Main Vessel (MV) and sodium occupies the inter-space between MV and SV (during which SV will be loaded with leaked sodium). Under such a condition, the structure can be subjected to seismic events (OBE and SSE). Condition of overflow of leaked sodium from SV into the inter-space between inner wall of Reactor Vault and SV is also considered in the analysis.

For performing analysis of the structure under normal conditions, when Sodium is contained in the MV, Reactor Vault, MV and SV have been coupled along with the Global FE model of NICB, and coupled model has been used for performing the analysis.

For performing analysis under condition of MV leakage, a separate isolated model of Reactor Vault along with the RA (MV with its internals) and SV, as extracted from the Global FE model, has been considered. Analysis is performed in-order to determine the additional forces that would be transferred to the civil structure when a seismic event occurs under the condition of sodium leakage. The leaked sodium mass is lumped in the walls of SV while performing this analysis, to represent the effect of extra force that gets transferred to the SV support.

Design is performed as per codal provisions of AERB, India for strength and crack-width based requirements.

The MV support zone in outer wall of RV comprises of a solid bracket of size 2.3 m (vertical C/S Dim) \times 2.3 m (horizontal cross-sectional dimension) running along the circular periphery of MV to support its flange. Reinforced Concrete (RC) in this area is subjected to highly varying stresses because of the geometry of the bracket and heavy and concentrated loads transferred from MV under the action of different loads. Separate FE analysis is performed for the projecting portion of Outer wall where the Flange of MV is supported, by means of refined

FE sub-model developed using 3-D solid elements. Moments and forces across different cross-sections of the model are determined by summing the element forces about the center of section considered, and design is performed for the summed forces and moments, using provisions of ACI 349 for Corbel design.

Thermal Analysis of the vaults have been performed for Heat-of-Hydration load using a Construction-stage Time-History Analysis and Standard Heat Transfer equations and the reinforcement required is verified to be less than the provided reinforcement.

The base supporting arrangement for inner wall of RV (IV) comprises of a central monolithic support transferring the entire seismic shear forces while the peripheral supports comprising of mirror-finished bearing pads, free to slide over each other, so as to allow for thermal expansion of the IV. The supporting system for the IV has been separately designed for the different forces transferred at the base of IV.

Structural analysis of liner along with the attached BSCS is performed for imposed deformation loads under different conditions, and different components of the liner system and BSCS (including liner anchorage) have been qualified for safety and stability.

Detailing aspects considered

Detailing had been done in a planned manner to avoid any interference between any of the EP / Liner anchors, reinforcement bars and embedded cooling pipes and other services, and congestion has been reduced to the extent possible. Typical reinforcement detailing provided in critical areas like projecting bracket zone of outer wall of RV, have been presented in the paper.

A full separation is provided between inner and outer walls of the vault, so that they behave independently. Expanded Polystyrene is provided in the gap between the two walls. At the base of inner wall of RV, provisions have been made such that the Inner wall is free to undergo radial thermal expansion, achieved by means of Manganese Phosphate coated bearing pads.

Mock-up tests have been conducted for different critical portions of the Reactor Vault assemblage, to study the possible construction problems that could be envisaged.

Conclusion

This paper presents different issues and aspects pertaining to analysis, design and detailing of Reactor Vault for PFBR. Critical loading environments pertaining to Reactor Vault and Analysis methodologies adopted have been discussed. Detailing issues have been addressed, and methods adopted have been discussed in the paper, with detailed sketches showing different critical areas where typical detailing has been provided.

Development and in-reactor verification of three types of advanced nuclear fuels for PWRs (6-1986)

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Keywords: Advanced, Nuclear, Fuel, In-reactor, Performance, Verification.

Three types of advanced fuels for PWRs are being verified in three Korean nuclear reactors: PLUS7TM for Optimized Power Reactor of 1000 MW class (OPR1000) and Advanced Power Reactor of 1400 MW class (APR1400), and 16ACE7TM and 17ACE7TM for 16x16 and 17x17 Westinghouse types of plants, respectively.

Each four lead test assemblies (LTAs) for each fuel type had been loaded to verify the irradiation performances in the commercial reactors. Four steps for in-reactor verification were being applied: the first one for assembly-wise examination in poolside after each cycle, the second one for rod-wise examination after disassembling in poolside, the third one for the rod examination in the hot cell test facility in detail, and the final one for the skeleton examination in the hot cell test facility in detail.

The first leading fuel, PLUS7TM, has completed 3 steps of verification and will start the final step from this year, while the 16ACE7TM has completed 2 steps and is waiting for the third and fourth steps until next year and the 17ACE7TM is being verified in reactor for the third cycle irradiation. The examination results up to now showed that all these 3 types of fuels were being irradiated successfully in the reactors.

The designs of these three types of advanced fuels are summarized and in-reactor performances on three types of advanced fuels are compared in this paper. In conclusion, all the irradiation performance parameters were within the expected design limits. In-reactor performances of these fuels are being verified continuously through the surveillance program during commercial implementation.

Investigation of possible corrective actions during manufacturing of fast breeder reactor components towards assessing the structural integrity (6-2003)

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The specifications of tolerances for the manufacturing of components are very important. While tight tolerances pose challenge to even big industries, liberal tolerance may have impact on design and integrity issues. Further, the actually achieved dimensions after manufacturing may not meet the initial specified value. One need to investigate the effect of deviation including functional and integrity requirements preferably at design stage to avoid problems of over / under specifying the tolerances or need for corrective action / rejection at a later stage.

In case of pool type fast breeder reactors, the ovality of large diameter shells (main vessel / safety vessel / inner vessel / thermal baffle) are important from weld mismatch considerations at manufacturing time apart from other integrity requirements including buckling during operations. The verticality of penetration shells in top shield is important from functional requirements. Another possible necessity of corrective action is mismatch in circumference of two matching parts such as a nozzle and a large diameter pipe to be welded with the nozzle. In case the pipe is manufactured in two halves with a final longitudinal weld, there is a chance that the circumference of the pipe is more or less than that of the nozzle to which the pipe is to be welded. All these requirements may call for corrective action or repair or rejection at manufacturing stage if adequate care is not taken.

A few possible deviations envisaged are discussed along with corrective actions that can be taken and the effect of corrective action is investigated. In case of ovality problem, one need to apply uniform force all around which means straining the component at manufacturing stage. The resulting additional stresses and their effect on integrity during operation are investigated. In case of circumference mismatch, (say pipe is smaller than the nozzle), one can attempt corrective action by mechanically expanding the pipe near the free end (weld end) using a mandrill. This means a shear force and bending moment are applied at the junction resulting in additional stresses than envisaged for operating load.

The resulting creep /fatigue damage due to the additional stresses are determined for typical examples / components taken as case study.

All these investigations form a basis for deciding the course of action during manufacturing of major components of 500 MWe Prototype Fast Breeder Reactor, which is under construction at Kalpakkam. The paper discusses the details of various possible examples, investigations carried out including finite element analysis and also brings out the actual manufacturing experiences in few cases.

Structural analysis towards erection of prototype fast breeder reactor components (6-2005)

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PFBR is a sodium cooled pool type reactor. The normal operating temperature is 670 K (397°C) for its cold pool components and 820 K (547°C) for the hot pool components. The hot pool and cold pool are separated by an inner vessel. The entire primary sodium pool is contained in a 25 mm thick main vessel of 12.9 m diameter and 13 m height. The main vessel is surrounded by a safety vessel which is still larger than the main vessel. The reactor has a design life of 40 years. Thus fabrication and erection of the large components are key activities for the construction of PFBR. The life of reactor is dependent on the structural integrity of these large components in both normal and accidental conditions. In view of this the fabrication and erection activities connected with PFBR have to be performed with high quality and safety requirements.

In PFBR higher safety restrictions are called for in view of the radiation hazards involved during operation as repairs and rectifications for the primary sodium circuit components is not possible. The large thin components are manufactured with very tight tolerances. Hence good care is essential to avoid errors and damage to the parts during erection. Also the functionalities and specification of the PFBR components are different from conventional industrial components, specialized techniques for erection are required. In this paper the erection sequence and methodologies adopted for the various reactor assembly components such as main vessel, safety vessel, inner vessel, thermal baffles, grid plate etc are brought out.

In PFBR the safety vessel and main vessel are to be placed in a pit type reactor vault, having very narrow gap between the vessel and the vault. The erection of components and the construction of reactor building have to be carried out simultaneously. Since the components are thin shell structure and cannot be repaired once erected special attachments and handling fixture are designed for the erection. The components are analysed for the loads occurring on them during handling process. In this process the safety vessel (diameter 13.5 m and 16 m height) has already been erected successfully on the inner reactor vault within a gap of 70 mm. Further erection of components is planned, analysed for erection loadings and mockups are performed where ever needed. The total sequence of erection for the various critical components, options studied, analyses carried out etc. will be discussed in depth in the paper.

SSI analysis for a reactor building with high frequency seismic ground motion (6-2040)

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Introduction

Current seismic hazard assessments express hazard in terms of Uniform Hazard Spectrum (UHS). High frequency content is present in the UHS for Nuclear Power Plants (NPP) in Central Eastern North America. It was found that this high frequency content in the UHS has significant effects on the seismic response of a structure when using conventional analysis methodologies. In several analytical cases, the high frequency content contributes to an increase in the floor response spectra (FRS), especially for those elevations close to the ground. In reality, however, it is well known that high frequency content of ground motion has much lesser damage effect to Structures, Systems, and Components (SSCs) of a NPP than low frequency content (except functional performance of some vibration sensitive components, such as relays). The challenge is how to reflect this reality in seismic analysis of a NPP.

In this paper, a literature review of existing techniques to mitigate the effects from the high frequency contents of the ground motion is presented. Two of these techniques, namely, seismic wave incoherence effects and multiple sets of artificial acceleration time histories (instead of the conventional one set of broad band envelop time histories), are selected to perform Soil-Structure Interaction (SSI) analysis for a typical reactor building. A simplified stick model (see Figure 1) is developed and is used in the SSI analysis. A free field UHS with significant high frequency content is used as the seismic ground motion input. The seismic wave incoherence effects on foundation and building response are considered using ACS SASSI, a computer software for SSI analysis. Multiple sets of time histories are derived from the UHS corresponding to different frequency contents. The effects on both hard rock condition and soft soil condition are compared. Floor response spectra at both the base slab of the reactor building and top of the internal structure are generated. The results from ACS SASSI and other seismic analysis computer codes (e.g., STARDYNE) are also compared.

Conclusions

In this paper, Soil-Structure Interaction analysis of a typical reactor building is presented using a UHS with high frequency content. Floor response spectra are generated at both base slab and top of the internal structure. The results are obtained using ACS SASSI and other seismic analysis computer codes (e.g., STARDYNE). It is seen that seismic wave incoherency has major effects on high frequency seismic response of the reactor building founded on hard rock as expected. Seismic wave incoherency has lesser effect on high frequency seismic response when the building is founded on soft soil condition. The application of multiple sets of time histories instead of the conventional spectrum compatible set can affect the seismic response of the reactor building for both soil and rock conditions. However, this concept of using multiple sets of time histories has to be rationalized.

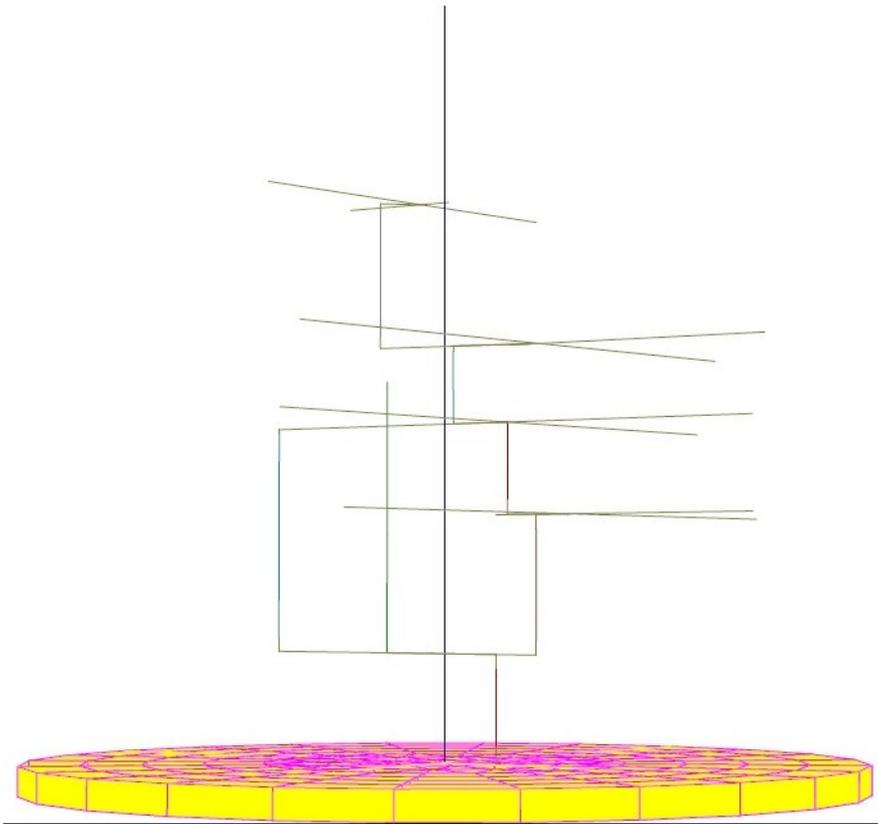


Figure 1. A simplified stick model of a typical Reactor Building.

Evaluation of local stresses at the vessel shell to nozzle intersection (6-2117)

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Local stresses at the intersection of the pressure vessel shell with a nozzle can be evaluated by Bijlaard or finite element methods. The first method, which is based on the classical shell theory, is still very common and described in many references like [1] and [2]. The second approach can be implemented by using shell or solid finite element model. This paper compares both approaches and discusses the limitations of the Bijlaard and shell finite element methods.

The paper demonstrates that for the purpose of ASME Code evaluation, a more practical approach is an automatic evaluation of the stresses from a solid element model in which one second-order element through-thickness is used. The advantages of this type of modeling have been discussed in [3]. The geometry of the model can be generated by a computer program whose input data are vessel and nozzle diameters and thicknesses. An example of a source code of such a program, generating an input deck for Abaqus finite element code, is provided.

The proposed approach not only allows for fast evaluation of the stresses but also visualization of stresses and deformation as well as investigation of various factors affecting the solution. The output produced by the Abaqus program can easily be post-processed automatically for the need of ASME Code evaluation.

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A study on optimization of seismic strengthening for the plant facilities in terms of plant management (6-2225)

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Introduction

At present, seismic design of chemical plant facilities mainly complies with “the Seismic Design Code of High Pressure Gas Facilities in Japan. In this notice the main purpose of seismic design is to ensure public safety at seismic events. In case of execution of seismic assessment and seismic strengthening, the code is applied to the existing plant facilities. If the existing plant is revamped, strict observance of the code is required. Although another design method should be proposed to execute the seismic strengthening from the view point of “Social responsibility of product supply” and “Maintenance of facilities and restraining of damage cost”, which are ones of the most important seismic risks of plant management.

In this study, from the view point of “Social responsibility of product supply” and “Maintenance of facilities and restraining of damage cost”, estimated term of shutdown and necessity of repair are focused on. And authors study evaluation method of seismic strengthening cost using damage levels which take them as the criteria for judgment. Hence damage levels after seismic events focused on the estimated term of shutdown and necessity of repair are established, and seismic strengthening cost at each damage level is calculated.

Authors have already proposed the method¹⁾ which can provide the information concerning improvement ratio of seismic performance, which is focused on the estimated term of shutdown and necessity of repair, at the selected damage level in case of determining the seismic strengthening cost. In this paper, the procedure to obtain the relationship between seismic strengthening cost and critical seismic coefficient on the ground at each damage level is proposed, and a calculation result of an example of a tower is presented. Furthermore this approach is effectively applicable for existing nuclear power plants and their related facilities instead of chemical plant facilities.

Aim of the work

The aim of this evaluation method is the following; if investment amount for seismic strengthening of the equipment and damage level at a seismic event are supposed, the portions required seismic strengthening and degree of strengthening can be determined. Accordingly improvement ratio of seismic performance of the subject equipment at the selected damage level could be assumed.

Therefore the damage levels focused on the estimated term of shutdown and necessity of repair are supposed, and the fault tree, whose fundamental events consist of damage modes, will be prepared for each damage level. Besides by means of approximation of the cost required for seismic strengthening and calculation of the seismic strengthening cost using fault tree, extraction of the portions to be strengthened, its required degree of the improvement, and its seismic improvement ratios can be assumed. Consequently the relationship between critical seismic coefficients on the ground and the cost of seismic strengthening is presented, and this relationship is applied for evaluation of seismic countermeasures. And here a self-supported tower on skirts is selected as the typical example of equipment in the plant facilities.

Essential results

The relationship between design seismic coefficients and seismic strengthening costs at each damage level is calculated. Examples of applications are presented as follows:

- 1) The seismic strengthening costs could be approximated by determining the expected damage level of the subject equipment and the design seismic coefficient.
- 2) In case of a given seismic strengthening cost, the improvement ratio of the seismic performance could be decided by selecting the damage level.
- 3) If the seismic strengthening cost and damage level are decided, the critical seismic coefficient on the ground of the equipment at the expected damage level could be assumed.

Summary/conclusions

In this paper, the estimated term of shutdown and necessity of repair is focused on and damage levels are proposed considering them, and an evaluation method of the seismic strengthening cost based on the damage levels is developed.

Using this evaluation method, in case of a given amount of seismic strengthening cost, the information of improvement ratios of seismic performance

focused on the estimated term of shutdown and necessity of repair can be provided, if the required damage level in terms of maintenance for the equipment is selected.

Additionally in this method extraction of dominant damage modes and appropriate distribution of seismic strengthening cost for plural damage modes, which are derived from analysis of fault trees, are possible. For an example, it is verified to be possible to extract dominant damage modes require seismic strengthening, in case of selecting damage levels and seismic coefficients on the ground.

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Comparing European and American codification in the field of NPP civil engineering (6-2493)

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Context

The opportunity to build an EPR in different countries during the next decades has inclined EDF Engineering Department presently involved in the construction of Flamanville 3 NPP to compare American nuclear codes in Civil Engineering to the French practice recently brought up to date for this project.

The needs for a new technical code for EPR

In France, the needs for a new design code for the Civil works of the EPR has come from the necessity to take into consideration the European Regulation (eurocodes) and the specificity of this project in term of safety improvement with respect to Severe Accident. This code entitled ETC-C (EPR Technical Code for Civil works) has been written so far by EDF.

Design of the containment steel liner

The design of a steel liner covering the inner surface of the containment has popped up the necessity to consider the blistering as a conséquence of the strongly pre-stressed concrete wall combined with a high temperature inside due to a hypothetical Severe Accident. This led the ETC-C to complete the mechanical analysis and reformulate the acceptance criteria used in the former codes.

Concrete structures design

For the concrete structures a comparison is made between the ETC-C and the American rules ACI 349 and ASME (section III Division 2).

An application of both reglementations to the determination of the reinforcement ratio on EPR concrete structures such as walls and decks of the reactor building has shown significant differences due to the codes applied as well as to the material properties, the loads and the engineering practice.

Conclusion

It is obvious that a new period of nuclear plant construction has created an opportunity in France to upgrade the nuclear codification taking into account the recent European reglementation in the fields of concrete and metallic structures. This article gives good examples of the differences between American and French standards applied to EPR.

Aspects of the design and construction of a new feedwater line for Angra 1 Nuclear Power Plant as a part of the steam generator replacement program (6-2497)

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Angra 1 Nuclear Power Plant Steam Generator replacement plays an important role in the intents of plant owner, Eletronuclear, for plant life extension and power uprating. As the feedwater nozzle position is now different from the original design, it brought the opportunity of modifying piping layout and support configuration of the main feedwater line inside containment. This paper presents the work performed and the improvements reached, when compared with the original design.

The main feedwater line piping has been re-routed to mach up with the feedwater nozzles attached to the new steam generators. The new routing of the feedwater lines has made it necessary to re-analyze the piping to demonstrate qualification to design code requirements. Stresses computed are within the allowable limits. New piping layout was also designed to avoid thermal stratification possibility in nozzle and line, and the new material of the piping, A335 P11, was chosen because of its corrosion resistant properties.

Since a new approach to postulated pipe break, based in present regulatory requirements, is now used, the concept of Arbitrary Intermediate Break, considered in the original design, was eliminated. According to the stress analysis, no intermediate break had to be considered in the new arrangement and thus pipe whip restraints are now used only at terminal points. Old pipe whip restraints are unnecessary and may be dismantled for increasing working area in that plant region inside containment.

The piping design was qualified according to the design code requirements (ASME code, section III, division 1, subsection NC, class 2 components) and to the Brazilian regulatory requirements.

Analysis of the stress-strain state of containment depending on temperature fluctuations in the environment (6-2560)

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Introduction

At the present time the design life of containments at several Russian Nuclear Power Plants (NPPs) is nearing completion that requires both analysis of the capability to prolong their operation and development of measures to extend service life of their structures. New NPP units under construction also face these problems due to the need for both reducing damageability of reinforced-concrete structures during operation and extending their service life.

The running experience of NPP containments collected over a protracted period of time has demonstrated that environmental effects (mostly seasonal and daily temperature fluctuations) are the key factors influencing their stress-strain state.

Taking account of the fact that environmental-temperature fluctuations may have rather large increments over short periods of time, the temperature-stress values could be also significant and would lead to degradation of strength properties of containment materials and, as a consequence, to loss in their operational functionality.

The aim of presented work is the calculation analysis of changes in the strain-stress state of NPP containments with VVER-1000 depending on seasonal and daily environmental-temperature fluctuations.

Results

Two specific cases are considered: 1 – real condition of containment without thermal insulation at the outer wall surface; and 2 – assumed coating of the outer containment surface with a foam-concrete layer (heat insulation). Two finite element models of the containment were developed corresponding to considered cases. The second model has two additional finite-element layers at the outer surface of both the cylindrical and the domical parts simulating heat insulation. The model consisted of 567594 finite elements and 604494 nodes.

The convective heat-exchange boundary conditions were assigned at the inner and the outer surfaces of the containment model.

In the temperature-field calculations with consideration for both seasonal and daily temperature fluctuations for normal operating conditions was taken the constant temperature inside the containment. The temperature outside the containment was assigned as a piecewise-linear dependence in accordance with the results of *in-situ* measurements performed every three hours in the course of the time frame under consideration (30–50 days).

According to the results of air temperature measurements close to the Kalinin NPP site maximum mean daily temperatures during summer months equal about 30°C, whereas minimum temperatures in winter months go down to minus 30°C. In spring, substantial daily air-temperature fluctuations (up to 20°C and more) are typical for the Kalinin NPP area, and in the nighttime air temperatures can even drop down to -5°C. Such large daily temperature fluctuations may lead to strength degradation in concrete due to destructive processes caused by freezing-thawing strains.

The calculations were conducted in the following order. At first, temperature distributions across section of the containment wall depending on fluctuations of both seasonal and daily temperatures were calculated. Two specific cases were considered: 1 – without heat-insulating layer; and 2 – with heat insulation of 10-cm thick (foamed concrete of $\gamma=600 \text{ kg/m}^3$ density) at the outer surface of both the containment cylinder and dome.

Next, calculations of the containment stress-strain state caused by the thermal load were conducted followed by those due to the integral effect of operating loads (dead weight, prestress, operating temperature). The stress-strain state was analyzed for both studied cases (i.e. without heat-insulating layer and with heat insulation at the outer surface of the containment cylinder and dome).

The fulfilled researches have showed that the following beneficial effects might be expected as the result of coating the outer surfaces of containments with heat insulation:

- diminution of negative environmental impacts (temperature fluctuations, humidity effects, frequent changes of freezing by thawing, impacts of deleterious substances etc.) on reinforced concrete would allow extending service life of NPP containments;
- decrease in variations of geometric containment parameters within the crane track area due to diminution of the temperature gradient inside and outside the containment;
- decrease in financial and manpower resources for NPP maintenance and repair thanks to reduction of the scope of crane-track-adjustment works;
- decrease in creep strain of concrete and consequently a diminution of losses in tendon efforts from creep strains of concrete; and
- diminution of compressive stresses in concrete at the inner containment surface and within the metal liner which develop under service-load impacts (taking account of creep strain of concrete) and in case of a maximal design-basis accident.

Using the stressed frame for blast resistant fenestration design of full containment structures (6-2601)

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Designing for internal accidental explosions in safety related structures (Nuclear and Demilitarization structures) are a challenge when one considers that safety related structures have to be designed for full containment. This means that for the maximum credible event, the structure not only survives the event but must survive in such a way that it continues to prevent the leakage of agent or release of radiation. So if the walls of the full containment structures need fenestrations for say utility openings, doors and HVAC, in a typical design, this usually necessitates substantial rebar construction around such penetrations. In some cases, the walls are designed with embedded pilasters integral to the wall and placed around these openings. The resulting design is highly prone to congestion of rebar around the openings and subsequent honeycombing of the concrete placed. One approach to prevent this inevitable rebar congestion is the use of the stressed frames. This approach has been more frequently used in the demilitarization projects but its use is being broadened for other safety related structures. Several uses of the stressed frame currently exist. The discussion in this paper is general in nature and does not reflect any one design in current use. This paper discusses what stressed frames are, how they work, how to design them and estimates cost savings and more importantly schedule savings expected.

Methodology

In this paper a representative wall, 25 inches thick is designed with the typical substantial rebar construction around openings. Previous problems of congestions and constructability are investigated and discussed. Then that wall is designed using stressed frames similar to that shown in Figure 1. The total containment criteria for the wall chosen is a deformation of not more than 1 degree support rotations when subjected to a detonation from a net equivalent TNT weight of 18 lbs. Since the hazards in the room are to be contained in the room the openings are covered openings. So the frames around the openings may have a gate, shadow shield, penetration plate or valve attached to them. The stressed frame is designed not just for the shears that result from the covering within the frame but are designed for the moments and membrane tension that

occur around these openings when the wall deforms. The uniqueness of the stressed frame, therefore, is that ability to take the stresses around the opening with the frame itself and not in the concrete. Since the stress concentrations are not in the concrete the need for additional reinforcement and the potential for rebar congestion are avoided. Figure 2 shows the stressed frame installed with the wall reinforcing.

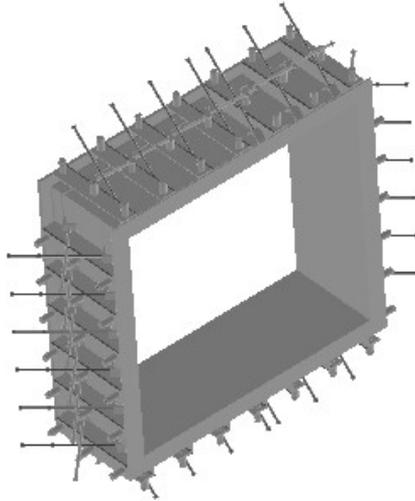


Figure 1. Stressed Frame around Utility Opening.

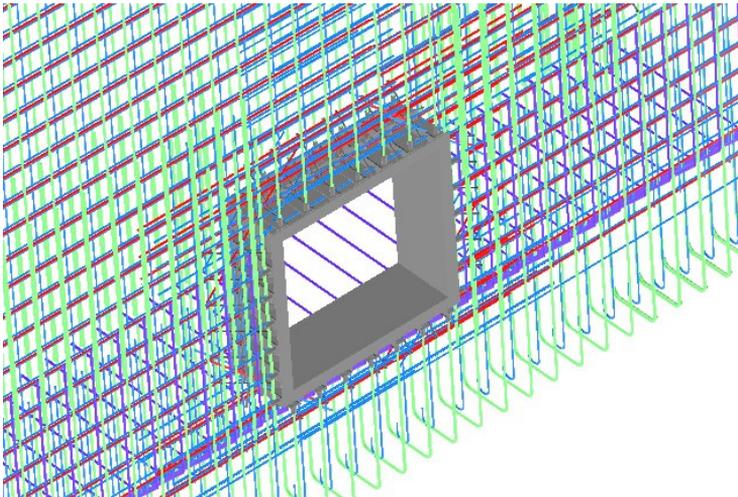


Figure 2. Stressed Frame with Wall Reinforcement.

A new device for the study of early-age cracking in massive concrete structures (6-3140)

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At early-age in massive concrete structures, cracking may occur during hardening. Indeed, hydration is an exothermic chemical reaction (temperature in concrete may overcome 60°C). Therefore, if autogenous and thermal strains are restrained, compressive stresses and then traction stresses rise, which can exceed the concrete strength (in an elastic finite element calculation). This cracking may increase significantly the concrete wall permeability.

The restrained shrinkage ring is a good method to determine the concrete behaviour (strain and cracking) due to the autogenous and drying shrinkage. In this study, a concrete mix which is representative of a nuclear power plant containment is tested. This test showed that at 20°C and without drying (cf. Figure 1), the amplitude of autogenous shrinkage is not high enough to cause cracking. Indeed, in this configuration, thermal shrinkage does not occur. Therefore, a new test has been developed to study cracking due to restrained thermal shrinkage.

This new test is an evolution of the restrained shrinkage ring test which allows taking into account the autogenous shrinkage and also the thermal shrinkage. Actually under development, it aims is to predict the behaviour, the cracking at early age of massive structures (especially the nuclear power plant containment). The new test principle is to increase the temperature of the steel ring in order to expand it. In this case, expansion of ring is restrained by the extern concrete layer, which induced compressive stresses in the ring and therefore tensile stresses in concrete.

The history of the ring temperature could be calculated by finite element in order to reproduce similar stress rate that can occur in a “real” massive wall (calculated also by finite element simulations, cf. Figure 2).

In order to verify that cracking will really occur, we have modelled the new test with a finite element code (Cast3M, developed by the French atomic energy commission). We use a model developed by Benboudjema and Torreti (2008). This one is a thermo-chemo-elastic-damage model.

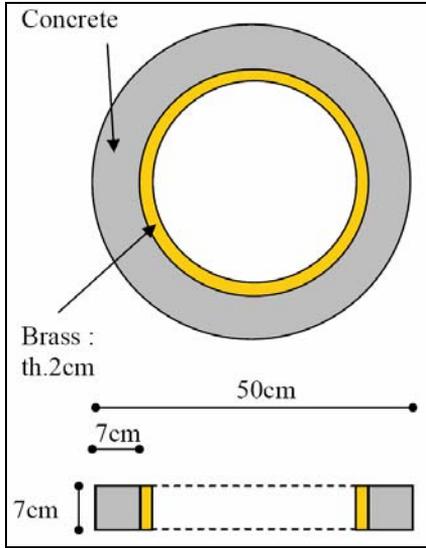


Figure 1. Geometry of the restrain shrinkage ring test.

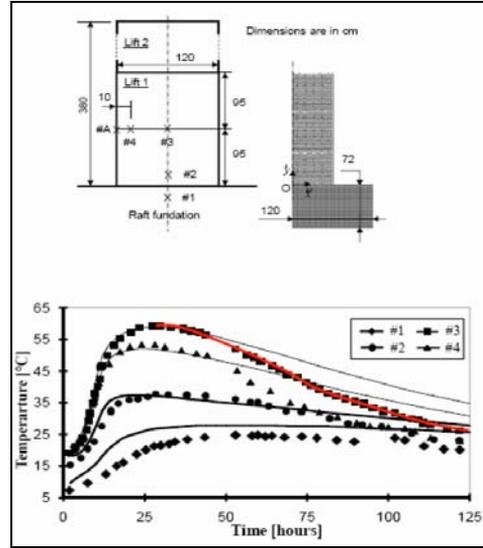


Figure 2. History of temperature measured in a concrete wall (1.2 m width).

The prediction of cracking needs a numerical resolution due to the complexity of the behaviour of concrete at early-age. In our model, the following phenomena are taken into account:

- The evolution of hydration: this is achieved here by the use of a chemical

$$\text{affinity (Ulm and Coussy, 1998): } \dot{\xi} = \tilde{A}(\xi) \exp\left(-\frac{E_a}{RT}\right)$$

- The evolution of temperature: the energy balance equation, which includes the release of heat due to the hydration reaction is solved:

$$CT\dot{T} = \nabla(k\nabla T) + L\dot{\xi}$$

- The evolutions of autogenous and thermal strains; $\dot{\epsilon}_{au} = -\kappa\dot{\xi}\mathbf{1}$ for $\xi > \xi_0$ and $\dot{\epsilon}_{th} = -\alpha\dot{T}\mathbf{1}$

- The evolution of Young modulus and tensile strength with respect to the hydration degree by the use of De Schutter laws (1997):

$$B(\xi) = B_\infty \left(\frac{\xi - \xi_0}{\xi_\infty - \xi_0} \right)^a \quad \text{where B is the mechanical variable}$$

- The description of cracking in tension: elastic damage model (Mazars, 1986)

$$\text{slightly modified: } \boldsymbol{\sigma} = (1 - D)\tilde{\boldsymbol{\sigma}} \quad \text{and} \quad \dot{\tilde{\boldsymbol{\sigma}}} = \mathbf{E}(\xi)\dot{\boldsymbol{\epsilon}}_{el} = \mathbf{E}(\xi)(\dot{\boldsymbol{\epsilon}} - \dot{\boldsymbol{\epsilon}}_{au} - \dot{\boldsymbol{\epsilon}}_{th})$$

6. Design and Construction Issues

To take into account these phenomena, several parameters have to be identified. Therefore, several tests were performed and some results from the literature were used. The results of the mechanical tests are shown in Figure 3 and table 1 (the tensile strength is determined by a Brazilian test).

Table 1. De Schutter laws parameters.

	Estat	Rc	Rt
ξ_0	0,115		
a	0.449	1.105	0.806

With this modeling we could know approximatively when the concrete ring will be able to crack so we could be able to improve the temperature history by decreasing the temperature rise rate and obtain a better permeability evolution.

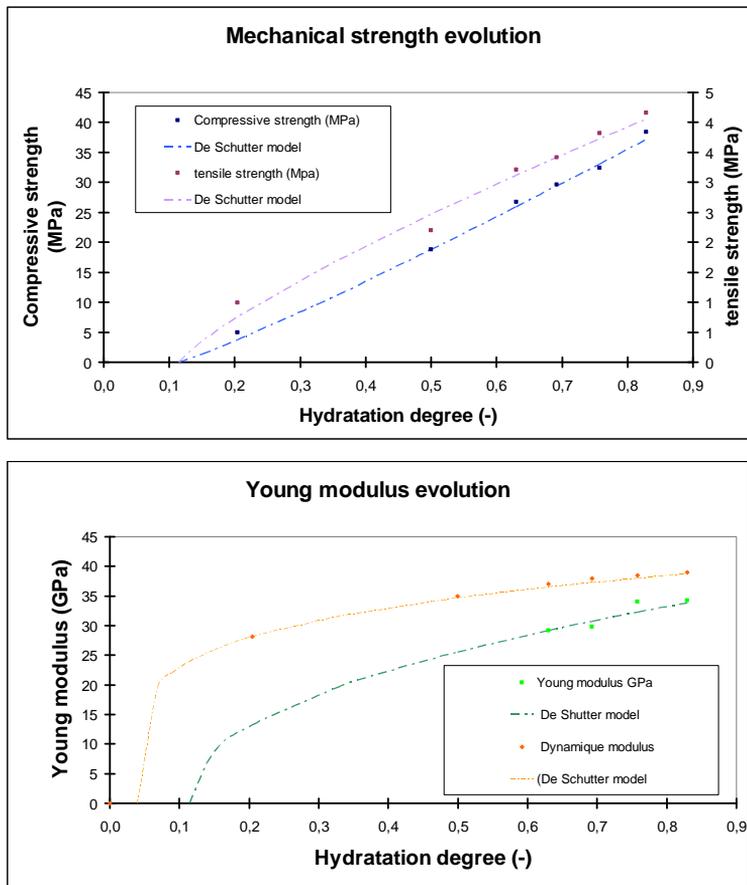


Figure 3. Mechanicals characteristics evolutions.

For a first approach, we have take a bilinear evolution of the temperature. At the beginning the ring temperature is maintained at 20°C. Then, at 25 hours (when the pic of temperature is attempted; see Figure 2), the brass ring temperature increases linearly with a rate of 0.35°C/h.

The model results show us (Figure 4: example of damage map for local damage) that with different sizes of mesh (7 mm and 5 mm) and with two types of computation (local and non local damage) we effectively obtain cracks.

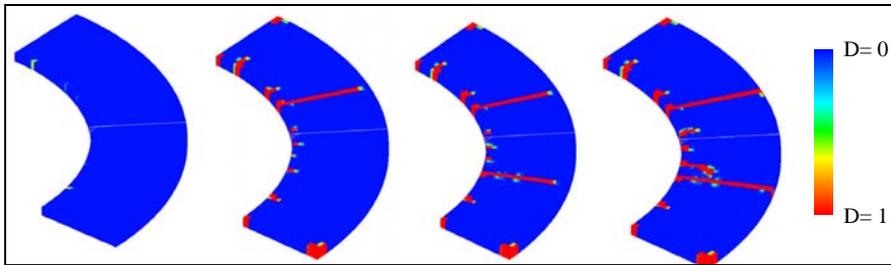


Figure 4. Example of damage map (local model; mesh size: 7 mm).

Although we didn't take into account the creep in this model, the first numerical results seem to show that we can effectively obtain cracking with our new test (for suitable values of temperatures) and that we will be able to study the early-age behaviour and cracking evolution due to restrained autogenous and thermal shrinkage.

7. Safety, Reliability, Risk and Margins

Reliability of structures, systems and components. Application of probabilistic methods to improve design, operation and safety of complex systems. Probabilistic safety assessment (PSA). Internal and external events. PSA and margin studies. Development of risk-informed approaches and risk-based regulations, codes and standards. Human factors engineering. Failure mode analysis. Severe accidents.

Adjusting the fragility analysis method to the seismic hazard input. Part I: The intensity-based method (7-1567)

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The integrated oversight process of nuclear power plants in Switzerland contains a significant amount of risk informed elements. National regulations require a full scope (all internal and external events) PRA for all power and shutdown operational modes at level 2. Thus, seismic PRA became an integrated part of this regulatory framework. The development of a seismic PRA typically involves the following the steps:

- Development of a site-specific seismic hazard input, usually based on traditional PSHA
- A detailed fragility analysis for all safety important structures and components
- Human Reliability Analysis for post-earthquake actions
- Development of a plant logic model
- Quantification, sensitivity and uncertainty analysis.

Although the methodology of seismic PRA to some extent has become standardized, different seismic PRA studies differ largely in their results. The main reason for this observation consists in the different approaches used for Probabilistic Seismic Hazard Analysis (PSHA) which defines the seismic hazard input for the PRA. While older studies used an intensity-based PSHA maintaining a close link between damaging effects of earthquakes (clearly and uniquely defined in a probabilistic way by the corresponding intensity scale), newer studies are based on SSHAC [1] type PSHA using ground motion models for peak instrumental characteristics of earthquakes for the developing of the hazard, which are not directly linked to damaging effects of earthquakes. This causes problems as is shown in this paper, because the traditional fragility analysis method used for seismic PRA is conditioned to a seismic hazard input which represents strong, damaging earthquakes. Empirical fragilities for structures and components are derived from field observations covering only damaging earthquakes. Fragilities for components derived from tests are also conditioned to the occurrence of damaging earthquakes, because qualification tests are based on time histories representing high magnitude seismic events. Therefore, the seismic hazard input derived from contemporary PSHA methods like

SSHAC [1] is inconsistent to the method of fragility analysis as used for seismic PRA. In case of the direct implementation of the results of the PEGASOS-project [2] into the seismic PRA of NPP Goesgen this inconsistency of methods leads to a potential overestimation of the seismic core damage frequency of up to a factor of 10 as is demonstrated in the paper. Therefore, a methodology has to be developed to adjust the fragility method to the seismic hazard input as delivered by seismologists, today. In this paper a procedure is proposed, which is aimed at a direct probabilistic correction of this inconsistency between seismic hazard input and the fragility analysis method. The main steps of the procedure are:

- Detailed deaggregation of the seismic hazard into a bivariate distribution of controlling earthquakes, characterized by their magnitude and distance for different spectral frequencies and different frequencies of exceedance (hazard frequencies)
- Development or selection of appropriate models for the conversion of controlling earthquakes into site intensities in the EMS-98 scale for different levels of frequency of exceedance (hazard frequencies)
- Development of an enveloping distribution for site intensities resulting from the distribution of controlling events.
- Calibrating the fragility analysis method to the developed distribution of site intensities to ensure the same statistical damage prediction by the fragility analysis method as predicted by the intensity distribution by using appropriate vulnerabilities for classified structures (according to EMS-98)
- Developing a distribution for an adjustment factor applied to the modelling factor considered in the traditional fragility method. This assures a more meaningful damage prediction by the adjusted fragility analysis and more reasonable PRA results.

The developed procedure fully accounts for available epistemic uncertainty and random variability by performing an appropriate statistical calibration of PSHA-results against the developed site intensity distributions. The application of this procedure for the fragility analysis of NPP Goesgen is demonstrated. In a company paper a more physical approach is suggested to complement this statistical approach to the adjustment of fragility analysis.

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Adjusting the fragility analysis method to the seismic hazard input. Part II: The energy absorption method (7-1568)

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The integrated oversight process of nuclear power plants in Switzerland contains a significant amount of risk informed elements. National regulations require a full scope (all internal and external events) PRA for all power and shutdown operational modes at level 2. Thus, seismic PRA became an integrated part of this regulatory framework. The development of a seismic PRA typically involves the following the steps:

- Development of a site-specific seismic hazard input, usually based on traditional PSHA
- A detailed fragility analysis for all safety important structures and components
- Human Reliability Analysis for post-earthquake actions
- Development of a plant logic model
- Quantification, sensitivity and uncertainty analysis.

Although the methodology of seismic PRA to some extent has become standardized, different seismic PRA studies differ largely in their results. The main reason for this observation consists in the different approaches used for Probabilistic Seismic Hazard Analysis (PSHA) which defines the seismic hazard input for the PRA. While older studies used an intensity-based PSHA maintaining a close link between damaging effects of earthquakes (clearly and uniquely defined in a probabilistic way by the corresponding intensity scale), newer studies are based on SSHAC [1] type PSHA using ground motion models for peak instrumental characteristics of earthquakes for developing the hazard. These characteristics are not directly linked to damaging effects of earthquakes. This causes problems as is shown in this paper, because the traditional fragility analysis method used for seismic PRA is conditioned to a seismic hazard input which represents strong, damaging earthquakes. Empirical fragilities for structures and components are derived from field observations covering only damaging earthquakes. Fragilities for components derived from tests are also conditioned to the occurrence of damaging earthquakes, because qualification tests are based on time histories

representing high magnitude seismic events. Therefore, the seismic hazard input derived from contemporary PSHA methods based on the direct use of ground motion parameters is inconsistent to the method of fragility analysis as used for seismic PRA. Hence a methodology has to be developed to adjust the fragility method to the seismic hazard input as delivered by seismologists, today. Detailed analysis has shown that the reason for the observed inconsistency consists in neglecting the energy dependency in traditional fragility formulations which are based on simple ground motion characteristics. In this paper a procedure is proposed, which introduces energy (magnitude) dependency into traditional fragility formulations. The procedure considers that the seismic demand in terms of energy absorption is completely different for small and high magnitude earthquakes defining a completely different damaging potential of these two categories of earthquakes. It is shown that this difference in damaging potential can be related to the ratio of a function of Arias intensity (or Cumulative Absolute Velocity, CAV) for the controlling earthquake of the seismic hazard to the same function of Arias intensity of a reference earthquake used to develop component-specific fragilities from empirical observations. The main steps of the procedure are:

- Development of scenario-specific controlling earthquakes for different frequencies of exceedance (hazard frequencies)
- Development of models for the evaluation of Arias-Intensity or Cumulative Absolute Velocity (CAV)
- Derivation of reference earthquakes for different types of components used for the development of generic fragilities from empirical observations.
- Development of a mean adjustment factor applied to the modelling factor considered in the traditional fragility method based on the difference in the different seismic demand with respect to energy absorption.

The developed procedure fully accounts for available epistemic uncertainty and random variability. The application of this procedure for the fragility analysis of NPP Goesgen performed for the implementation of the PEGASOS-hazard [2] into seismic PRA is demonstrated. In a company paper a statistical approach is suggested to complement this approach to the adjustment of fragility analysis. The statistical approach can be used for quality assurance.

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On the treatment of dependency of seismically induced component failures in seismic PRA (7-1581)

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The integrated oversight process of nuclear power plants in Switzerland contains a significant amount of risk informed elements. National regulations require a full scope (all internal and external events) PRA for all power and shutdown operational modes at level 2. Thus, seismic PRA became an integrated part of this regulatory framework. Experience from past seismic PRA studies at NPP Goesgen [1, 2] indicates that simplifying assumptions with respect to the treatment of dependent failures of multiple redundant components strongly affects the results of seismic PRA and shifts them to the conservative side. Typically, it is assumed that all components of the same type characterized by the same fragility function and located on the same floor of a building will fail together once the load capacity of a single, representative component is exceeded. Therefore, full dependency between the failures of similar components belonging to different redundancies of the safety system is usually assumed. As long as seismic risk was treated in a simplified way and was not found to be a significant risk contributor, such a simplifying assumption could have been tolerated. For older plants with a low number of redundancies (e.g. with a two-train safety system) and insufficient physical separation of redundancies this assumption may even be justified today. The situation is different for newer European plants (e.g. design of KWU, today AREVA). Here the treatment of dependency of seismically induced failures of multiple redundant components is of major importance because of the presence of multiple redundant safety trains with good physical separation. The question is also important for new designs. The dependency between seismically induced failures in the Goesgen seismic PRA is modelled by a β -factor common cause failure (CCF) model [1]. Generic β -factors have been applied to model the dependency. This procedure was revised to address the results of the revised fragility analysis and to incorporate the results of the PEGASOS PSHA study. The Goesgen NPP has a four-train safety system with an additional two-train bunkered special emergency safety system. Detailed analysis of the allocation of system trains and their supporting equipment did show that most of the systems could be represented by two groups of two components, where each group is expected to show a different response to the same seismic demand. Therefore, the dependency between the component failures can be divided into two steps:

- a first step treating the dependency between the two groups,

- and a second step treating the dependency within the two components of a common group.

Therefore, the problem can be reduced mathematically to the treatment of the correlation in failure occurrence of two objects using two different β -factor models in a staggered way. In the traditional fragility formulation the correlation of failure occurrence [2] is described by the following equation:

$$\rho = \frac{\beta_{R1}\beta_{R2}}{\sqrt{\beta_{R1}^2 + \beta_{S1}^2}\sqrt{\beta_{R2}^2 + \beta_{S2}^2}} \rho_{R1R2} + \frac{\beta_{S1}\beta_{S2}}{\sqrt{\beta_{R1}^2 + \beta_{S1}^2}\sqrt{\beta_{R2}^2 + \beta_{S2}^2}} \rho_{S1S2} \quad (1)$$

Here, ρ_{R1R2} , ρ_{S1S2} are the response and capacity correlation factor between object 1 and 2; β_{Ri} , β_{Si} the lognormal standard deviations of the response and capacity of component, i. The total correlation factor can be set equal to the required β -factor assuming a quasi-linear dependency model.

For the first step of the procedure (between the two groups) the β -factor can be derived from the simple assumption that $\rho_{R1R2} = 0$, while the second term in equation (2) is maximized. This yields a β -factor of 0.5 between the two groups of structural similar, redundant components.

For the second step a Monte-Carlo-procedure was developed, which evaluates the fraction of concurrent failures of two components given the same seismic load and assuming identical response of both components (first term in equation (1) is maximized and set equal to 0.5). The correlation factor ρ_{S1S2} is evaluated as the ratio of concurrent failures of both components related to the total observed failure number in the sample. The Monte Carlo procedure considers some minimal seismic diversity of the components caused by the uncertainty of the mass distribution of similar components by incorporating some Gaussian noise reflecting this uncertainty. Using the results, the β -factor can be evaluated from equation (1).

A set of calculations for different boundary conditions with respect to the seismic load, the combined uncertainty of the fragility function and with respect to the median capacity of components was performed to derive a set of tabulated β -factors for use in the Goesgen seismic PRA. The values can also be applied for generic applications.

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Time-dependent reliability of reinforced concrete beams considering variability in degradation due to reinforcement corrosion (7-1590)

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Keywords: corrosion, beams, degradation, reinforced concrete, structural reliability

Reinforced concrete (RC) structures in service may be affected by ageing, which may include changes in strength and stiffness beyond the conditions that are assumed in the structural design. Corrosion of reinforcement has been identified as being the most widespread and predominant degradation mechanisms. The associated reduction of steel area, spalling of concrete cover and/or the loss of bond with time result in the loss of safety and serviceability. Therefore, the evaluation of effects of corrosion damage on RC structures would help in deciding the possible repair and continued service where their replacement is economically unfeasible. Structural reliability evaluation will further help in the efficient allocation of limited resources for periodic inspection and maintenance of such structures.

In this paper, degradation of flexural and shear strength of RC beams due to reinforcement corrosion is considered. For corrosion-affected RC beams, a methodology is presented to generate time-dependent mean degradation function which represents the ratio of degraded strength at any time from the initiation of corrosion to the original strength. The various parameters that could affect the time-dependent strength are identified as material strength of concrete and steel, dimensions of the beam, covers to the reinforcements and the annual mean corrosion rate. By considering the variability associated with the parameters under consideration, the time-dependent variability associated with the degradation function is evaluated. Simple analytical formulations are presented for estimating the time-dependent failure probability considering variability in degradation function as well as the stochastic nature of load processes, and uncertainties in the initial strength. For illustration, time-dependent failure

probabilities are presented for a corrosion-affected RC beam for flexural and shear failure modes. It has been observed that the time-dependent failure probability considering the variability in degradation function evaluated from analytical formulations agrees well with the failure probability evaluated from Monte Carlo Simulation. Parametric analyses show that failure probability is sensitive to the variability associated with the degradation function if it is larger than about 5%.

Component degradation effect on seismic risk of NPP (7-1643)

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Introduction

Ageing related degradation of a nuclear power plant component is an important aspect to secure the long term safety of the plant, especially for the seismic safety, since the degradation of the components affect not only the seismic capacity of a component but the response of a component. This can cause the change the seismic margin of a component and over all seismic safety of a system. Finally this induces the reduction of plant level seismic safety and the increment of CDF.

Degradation of safety significant components

Total of 530 components in Korean NPP were investigated and a database was developed based on the investigation results. As a result of seismic walkdown, crack and corrosion are the typical aging related degradation which can reduce seismic safety of safety related equipment. In order to manage aging related degradation data collected through the seismic walkdown in effective and systematic database system is established.

KAERI (Korea Atomic Energy Research Institute) and BNL (Brookhaven National Laboratory) are collaborating to develop seismic capability evaluation technology for degraded components. To better understand the status and characteristics of degradation of components in NPPs, degradation occurrences of components in the U.S. NPPs were identified by reviewing recent publicly available information sources and the characteristics of these occurrences were evaluated and compared to the observations in the past.

Component degradation effect on seismic risk

The SPRA is an effective method to estimate a plant level risk quantitatively. The purpose of a SPRA is to determine the probability distribution of core damage due to the potential effects of earthquakes (ANS, 2007). SPRA is performed based on four steps: seismic hazard analysis, component fragility evaluation, plant system and accident sequence analysis, and consequence analysis.

The component degradation effect on the seismic safety of a nuclear power plant was performed using past SPRA model. The degradation effect on the initiating event frequency was estimated according to the degradation level of safety significant components.

Figure 1 shows the total CDF increasing ratio according to the component degradation level. As shown in this figure, it can be revealed that the degradation of major equipments can increase the seismic induced CDF. ECW is the most influential equipment according to the degrading level. Also the degradation of the Diesel Generator and the CS Tank also has greatly influenced the total CDF of NPP.

Conclusion

For the examination of ageing related degradation of a nuclear power plant component, several times of plant walk-down were performed for Korean NPPs. Total of 530 components were investigated and a database was developed based on the investigation results. To better understand the status and characteristics of degradation of SPCs in NPPs, degradation occurrences of SPCs in the U.S. NPPs were identified by reviewing recent publicly available information sources, and the characteristics of these occurrences were evaluated and compared to the observations in the past. KAERI and BNL developed a component aging database of US nuclear power plants using the NRC documents, NUREG report, LERs, LRA, and generic comments.

For the evaluation of degradation effect of major equipment which related to an initiating event, seven components and associated initiating events were selected. When the seismic capacity of the selected equipments was decreased from 10% to 50%, the initiating event frequency was determined. Through this study, it was possible to identify the safety significant component to secure a long term seismic safety of a plant.

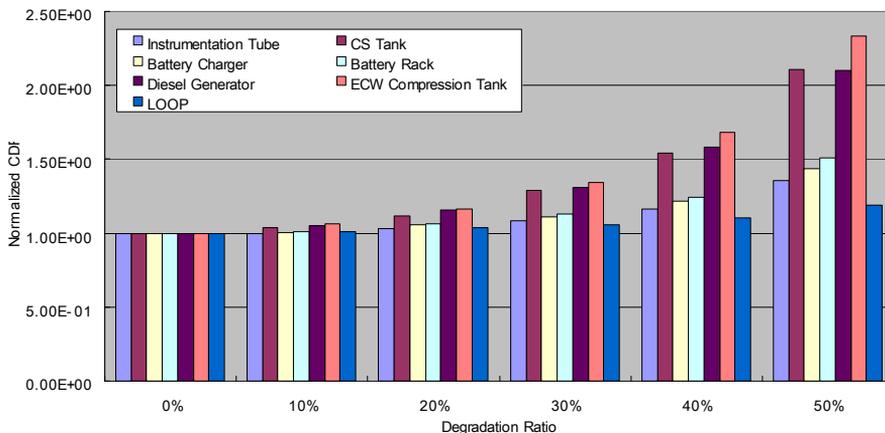


Figure 1. Component Degradation Effect on Seismic Induced Core Damage Frequency.

Acknowledgement

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The reactor coolant circuit strength and the safety and reliability issues (7-1651)

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The problem issues of performing justifications of the strength and reliability of the WWER-1000 coolant circuit components at the stages of designing, operation and extension of the operating period are considered. Distinctions in the approaches to substantiation of the coolant circuit integrity in the Russian and western regulatory documents are noted. The report deals with the most complicated problems by solving which the service life is defined. The basic differences in the approaches to the deterministic and probabilistic analyses of strength are formulated. The approaches to safety analysis on the basis of probabilistic assessments of the leaks and application of the LBB concept are considered as well.

As the safety analysis must be conservative, the deterministic calculations from the viewpoint of coolant circuit integrity must be taken to be conservative. Any kinds of calculation uncertainties are considered so that for problem statement the properties and characteristics of materials and media, the force and thermal impacts and simplifications of calculation models and approximate solution bring about the known worse results. Decreasing conservatism of calculations for strength within the framework of safety analysis is an important component of reducing the costs during designing, operation and extension of service life. Here it is shown that there is available a boundary in the process of decreasing conservatism and beyond this boundary there occurs growth of uncertainties and to overcome the latter the considerable costs are required as compared with benefits from properly decreasing conservatism.

The requirement of the regulatory documents on justification of operating reliability of the primary circuit components and systems during the design service life presumes for the reliability indices to be set related to probability of no-failure operation, failure interval, technical life, service life, etc. For the primary circuit equipment and pipelines, namely, for the vessels and pipelines, and also for the systems, i.e. for the reactor, steam generators, pumps, etc. such factors as damage to the walls, for example, formation of a through macrocrack or division into the parts and loss of tightness of detachable joints are assumed to be the basic reliability indices.

At the operation stage in the primary circuit components determining the coolant boundary in accordance with the different mechanisms of degradation of the materials there exist probability of formation and growth of the flaws and the technical survey and metal inspection are used to reveal these flaws. Detection

of the flaws being inadmissible for further operation in case of their obvious indication is performed using the non-destructive tests on the basis of both deterministic and probabilistic approaches. In this case the failure probability is determined considering statistical spread of data on unsoundness, mechanical properties and crack growth resistance characteristics. The approaches to the calculated support of the reactor plant operation are formulated including that one on the basis of computerized monitoring of residual cyclic life.

Improvement of the seismic fragility analysis by use of the methods of structural reliability and safety analysis (7-1655)

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The key task in a seismic probabilistic safety analysis (PSA) is the fragility analysis. Seismic fragility of a structure or equipment item is defined as the conditional probability of its failure at a given value of the seismic input response parameter. The peak ground acceleration (PGA) is commonly used as input response parameter. The objective of fragility evaluation is to estimate the ground motion capacity of the item and its uncertainty. Because there are many variables in the estimation of this ground acceleration capacity, the fragility is described by a family of fragility curves to reflect the uncertainty in the fragility estimation.

The mostly used method for fragility analysis is the scaling method. In this method the family of fragility curves is described by three parameters: the median ground acceleration capacity \check{A} , and logarithmic standard deviations β_R for randomness and β_U for uncertainty. The fragility parameters \check{A} , β_R and β_U are estimated by an intermediate random variable, the factor of safety F , which relates the acceleration capacity A to the earthquake level specified for design A_{SSE} . The factor of safety is the product of individual factors which describe the conservatism in the design. All factors of safety are assumed to be log-normally distributed, principally for its calculation convenience.

The use of lognormal mathematics in the scaling method is known to be an erroneous approach in the tails of the lognormal distributions, even when the lognormal shape adequately describes the data in the main parts of the distribution. The probability of failure for seismic events, however, is generally low. To improve this shortcut, the method of structural reliability and safety analysis as it is used for reliability analysis of civil structures is proposed. In this methodology the failure mode is defined by a function (limit state function) of deterministic and stochastic parameters. The stochastic parameters (basic variables) may be described by any arbitrary distribution function. The dispersion of the basic variables is split into the inherent randomness and model uncertainty. The probability of exceedance of the limit state function is calculated by an iterative approximation method as function of the peak ground acceleration as variable parameter. The advantages of this method are the appropriate calculation of the probability of failure also for low probabilities as it is common for seismic events. Furthermore this method allows the analysis of a complete structure as system by combination of the individual failure modes in a minimal cut-set representation.

The application of the method is demonstrated at a pipe run of the rapid shutdown system of a boiling water reactor as illustrative example.

The seismic fragility assessment of the feed water tanks plant using robust prediction concept of structural response (7-1664)

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Keywords: robust fragility assessment, structural response, uncertainty quantification

The computational task described in this paper is part of the updating effort of the Seismic Probabilistic Risk Assessment for Loviisa Nuclear Power Plant. Robustness in the prediction of structural response is an essential requirement for probabilistic fragility assessment. In order to develop the updated fragility curves for the feed water tanks in the Loviisa plant, which constitute the key element in assessing the core melt frequency of the plant, very detailed structural model incorporating the feed water tanks was developed. The fragility of a key plant component is defined as the conditional probability of its failure given a value of the response parameter, such as displacement, strain, stress or stress resultant. The first step in generating fragility is to develop a clear definition of what constitutes the failure of a component. It may be necessary to consider several modes of failure, and fragility is required for each mode. To assess the structural performance, it is important that robust predictions are made that treat all the uncertainties, from modeling applicable loads to modeling the structural behavior.

For large tanks, a failure of the support system or a plastic collapse of the pressure boundary is considered to be the dominant failure mode. Structures can be considered to fail functionally when the inelastic deformations under seismic loads are estimated to be sufficient to potentially interfere with the operability of equipment attached to the structure or fractured sufficiently for equipment attachments to fail. The event and fault trees should appropriately reflect these failure conditions.

The fragility of large components is directly developed from the seismic response analysis results. The component fragility for a particular failure mode is expressed in terms of the ground-acceleration capacity A . The fragility is therefore the probability at which the random variable A is less than or equal to a specified value, a . The ground-acceleration capacity is, in turn, modeled as

$$A = \underline{A} \varepsilon_R \varepsilon_U \quad (1)$$

In Eq. (1) \underline{A} is the median ground-acceleration capacity, ε_R is variable representing the inherent randomness about A , and ε_U is a random variable representing the uncertainty in the median value. It is assumed that both ε_R and ε_U are log-normally distributed with logarithmic standard deviations β_R and β_U , respectively.

As a result of the fragility evaluation the curves expressed with the aid of Equation 1 are given for the feed water tanks of Loviisa plant.

Longevity curves for probabilistic lifetime analysis (7-1684)

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Introduction and purpose

Seismic fragility analysis has become standard practice for assessing the seismic safety of NPPs for a range of earthquake magnitudes, extending beyond the design basis level. The methodology is well established, although it continues to rely on expert knowledge that is not widespread. Individual component seismic capacity is described by a family of “fragility curves”, defining the conditional probability of failure as a function of a ground motion parameter. The component fragility curves are often reduced to a high-confidence-of-low-probability-of-failure (HCLPF) capacity for each safety important component. The combination of component level HCLPF capacities, through boolean expressions describing safety system dependencies, provides a plant level HCLPF value that is viewed as the true reference seismic level up to which the plant can be considered as “safe”. Typically, a plant that is designed to a 0.3g design basis earthquake can normally be shown to have a HCLPF capacity of 0.5g, thanks to conservatism accumulated in the seismic analysis design process.

The purpose of this paper is to study the technical value and practicality of adopting a similar methodology in an entirely different field, namely the assessment of plant safety for long term operation in the presence of ageing; the expected outcome being that one would likely be able to demonstrate a “high-confidence-of-low-probability-of-failure” life duration largely exceeding the design life.

Technical approach

In order to achieve the above objective, the concept of “longevity curves” is introduced. Longevity curves are defined as the conditional probability of failure of a safety relevant component as a function of its age (see Figure 1). Thanks to the probabilistic formulation, an objective assessment is made possible of the time (age) at which a nuclear installation will start seeing its safety level degrading below its intrinsic, time-independent value, expressed in the normal

PSA practice as core damage frequency (CDF) or large early release frequency (LERF).

The component level longevity curves should be propagated through PSA modeling to obtain a plant level longevity curve or CDF over time. In this process, plant vulnerabilities associated with a particular aging degradation issue can be uncovered and possibly resolved through mitigation or component repair or replacement, so as to increase the plant lifetime horizon.

A number of parallels can be drawn between the well established seismic fragility analysis methodology and the proposed longevity analysis for plant aging assessment. They will be extensively discussed in the paper. It is sufficient here to note the following key features:

- the shape of the curves is similar and expected to be well represented by Weibull or Lognormal distributions
- code based criteria are usually deemphasized in a PSA-based analysis as they entail varying levels of conservatism
- uncertainties, being the result of randomness in component lifetime itself and/or of incomplete knowledge of the parameters, are explicitly quantified
- as in a risk-informed approach, the PSA model must be exercised to incorporate the safety significance of a particular component failure.

Longevity curves are associated with a particular age-related failure mode, affecting a long-lived component. They can be derived from a full probabilistic analysis using time-dependent properties and damage modeling. Alternatively, they can be estimated through a progressive analysis of design safety factors that are treated as random variables characterized by their median value and standard deviations. Such safety factors can be grouped in several categories such as:

- F_{MD} for material and design (physical component properties)
- F_{LE} for loads and environment (time dependent solicitations)
- F_{DT} for defect tolerance (safety margin beyond the appearance of a defect).

The longevity curves are then expressed as:

$$A = F_{MD} * F_{LE} * F_{DT} * A_{Design}$$

A_{Design} being the design life.

Contributors to each of those factors will be illustrated on particular examples of degradation mechanisms affecting long-lived components of PWRs.

The incorporation of longevity curves into the PSA and the critical aspect of acceptable risk increase due to age will also be examined.

Conclusion

The seismic fragility methodology provides a blueprint for developing a longevity analysis methodology, with necessary adaptations. The present paper will serve as a proof of concept. Longevity curves could provide a consistent framework for assessing the plant safety impact of individual component aging issues.

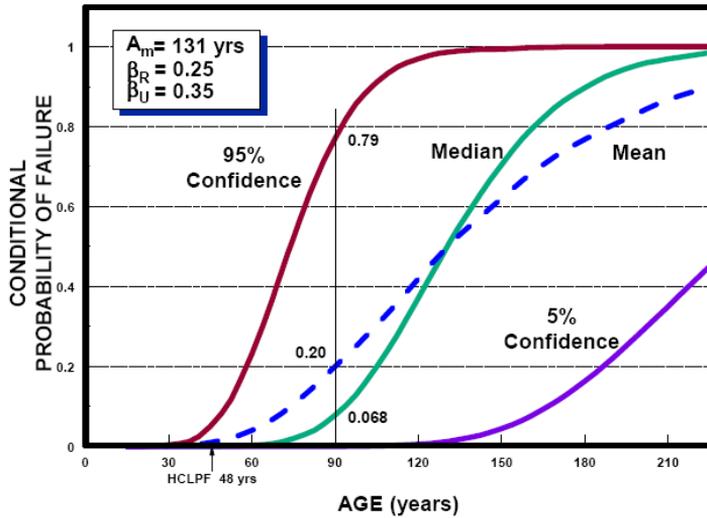


Figure 1. Mean, Median, 5% Non-Exceedance and 95% Non-Exceedance Longevity Curves for a Component.

Reliability and safety analysis of raft foundations under dynamic loading (7-1688)

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Moderately thick plates are frequently encountered in civil engineering. Typical plate structures on elastic foundations are machine foundations and raft foundations, concrete pavements of highways and airfields etc subjected to impulsive and harmonic type of loadings. These structures are predominantly subjected to dynamic loadings and can be modeled mathematically by assuming a thick to moderately thick plate resting on an elastic foundation.

Some of the most important types of dynamic loadings, nuclear reactors' raft foundations are subjected to are seismic loading due to earthquake ground motion and impulsive loading such as due to explosion, aircraft impact etc. These raft foundations have been modeled as thick plates on elastic foundations and the plates are analyzed using Reissner-Mindlin plate theory which considers first order shear deformation effects thus leading to better representation of actual behaviour of thick plates. It is very essential to ascertain in advance how reliable the raft foundations are during their operational life subject to the action of dynamic loading while exposed to severe environmental conditions. Dynamic loads may govern the analysis and design of raft foundations and knowledge of reliability of these structures beforehand will improve the structural design of raft foundation so as to sustain dynamic impulsive loads and severe environmental loads. The present study aims at developing formulations for ascertaining the reliability of nuclear reactors' raft foundations subject to impulsive type of dynamic loading and also deriving methods for determination of safety margin. The analysis and derivations have considered plate-foundation interaction. The foundation parameters characterize the compressive strain and shearing strain in the foundation.

For determining reliability first the modes of failure of the structure have to be assessed and probability of failure to be determined. Deflections of the structure have been assumed to be the governing parameter for evaluating failure conditions. Based on several factors viz. foundation stiffness, transverse shear deformation, plate aspect ratio, shape and duration of impulsive load, loaded area, and initial membrane stress etc the maximum deflections will be different for different loading combinations. To remain safe under the action of dynamic

loads, these maximum deflections should be less than the permissible deflections for respective load cases as per codes of standard applicable. In this paper two formulations have been investigated for studying dynamic behaviour of moderately thick plates. The first formulation is based on the Reissner-Mindlin plate theory, considering the first order shear deformation effect and including the plate-foundation interaction and thermal effects. The formulations have been extended to the case of large deflections of Reissner-Mindlin plates. The second formulation assumes Mindlin plate theory; the governing equations of dynamic equilibrium have been derived using Hamilton's principle and Euler – Lagrange equation of calculus of variation. Using these two formulations, programs have been prepared and validated for evaluation of deflection and bending moment. Failure equations have been developed based on values of maximum deflection and corresponding bending moment. A procedure has been suggested to determine the reliability of raft foundations.

The plate-foundation interaction has been considered in the analysis and the plate is assumed to vibrate and deform along with the foundation simultaneously. The study assumes flexibility of the raft as opposed to conventional engineering practice which assumes the raft foundation as fixed leading to larger stress resultants under dynamic loading conditions.

Challenges in the application of probabilistic safety goals for nuclear power plants (7-1769)

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The paper will deal with challenges in the application of probabilistic safety goals, as analysed in the Nordic (Sweden/Finland) project dealing with the use of probabilistic safety criteria for nuclear power plants (NPP). The project has relations to an on-going OECD/NEA WGRisk task on probabilistic safety criteria in member countries.

Safety goals are defined in different ways in different countries and also used differently. Many countries are presently developing them in connection to the transfer to the risk-informed regulation of both operating NPPs and new designs. However, it is far from self-evident how probabilistic safety criteria should be defined and used. On one hand, experience indicates that safety goals are valuable tools for the interpretation of results from a probabilistic safety assessment (PSA), and they tend to enhance the realism of a risk assessment. On the other hand, strict use of probabilistic criteria is usually avoided. A major problem is the large number of different uncertainties in PSA model, which makes it difficult to demonstrate the compliance with a probabilistic criterion. Further, it has been seen that PSA results can change a lot over time due to scope extensions or increases of level of detail, typically leading to an increase of the frequency of the calculated risk. This can cause a problem of consistency in the judgments.

This paper will give an overview of the current situation with probabilistic safety goals with emphasis on challenges in applying them. The following items will be discussed:

- use of safety goals in a strict manner (limiting values) vs. as targets (orientation values),
- definition of valid subsidiary safety goals at the levels of core damage risk (for level 1 PSA) and large release risk (for level 2 PSA)
- needs to internationally harmonize probabilistic safety criteria for NPPs
- qualification of PSA for the application of probabilistic safety criteria
- differences/similarities in the use of safety goals for new and operating NPPs.

Integrated soil-structure fragility analysis method for nuclear structures (7-1771)

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The Integrated Soil-Structure Fragility Analysis (ISSFA) method, presented in this paper, is intended to utilize the recently available computation power with the established analysis techniques to develop a more reliable foundation for the performance based seismic design.

Performance based seismic design of structures is the accepted design methodology required by the Nuclear Regulatory Commission (NRC) for nuclear safety related projects. One of the essential steps in the Performance based seismic design approach specified in ASCE 43-05 for the seismic design criteria for Structures, Systems and Components (SSC) in Nuclear Facilities is seismic fragility analysis. The goal in fragility analysis of SSC is to estimate the probability of failure – defined as unacceptable seismic behavior – of SSC as a function of some ground motion quantity representing the intensity of the seismic event. The failure of SSC is typically defined in terms of seismic force (or deformation) demands exceeding their corresponding capacities leading to an unacceptable seismic behavior. Thus, to determine the fragility function at each level of ground motion intensity, a realistic estimation of the probabilistic distribution of the demand and capacity parameters is necessary. While mean and dispersion of the capacity parameters may be prescribed, controlled and evaluated by design criteria, determining the probabilistic distribution of the demand parameters generally involve significant analytical effort.

The SSC seismic fragility functions are combined with the seismic site hazard curves to obtain the annual failure probability (i.e. performance) for the subject SSC. The NRC performance goal is stated as 1 in 10000 over the preclosure period for Yucca Mountain Project in the US. Considering a preclosure period in the order of 100 years, the performance goal can be interpreted as a target annual failure probability of 10^{-6} .

Traditionally, simplified methods for estimating both the mean seismic demand values and their corresponding dispersions are developed in building and nuclear industries. These methods, while attempting to minimize the computational effort, typically rely on empirically obtained and prescribed mathematical relationships for the fragility function and (linear) structural analysis at limited (often one) ground motion intensity levels, and are typically

overly conservative resulting in over-design of structures. In this paper, the fragility functions corresponding to Seismic Margin Analysis (SMA) Method defined in terms of High-Confidence-Low-Probability-of-Failure (HCLPF) capacity are used as a benchmark for comparing the results obtained from the featured ISSFA method.

In the past few years, prompted by newly available computing power, researchers have explored new techniques for determining seismic fragility functions in the building industry taking advantage of methods such as nonlinear Mont Carlo time-history analysis and nonlinear incremental dynamic analysis. An equivalent effort to make parallel explorations and advancements in the nuclear industry, however, is lacking. This topic is more challenging for nuclear industry structures considering that these structures are typically very stiff and the response nonlinearity when subjected to large seismic motions is mostly associated with the soil layers and soil-structure-interaction (SSI). Other sources of nonlinearity in the response of nuclear structures that are typically ignored in the current practice of fragility analysis are foundation sliding, rocking and separation between soil and foundation. The framework developed in this study integrates the aforementioned effects in fragility analysis for nuclear industry structures.

The input data are the site specific seismic hazard curve and rock motions at different earthquake intensity levels and representative randomized soil profiles as well as the structural model of the subject SSC. Latin Hypercube Sampling technique (LHS) is used throughout the analysis to propagate the input uncertainties through the SSI analysis. The output products are the fragility functions for specified limit states that in conjunction with the seismic hazard curve yield the annual probability of failure (performance) for SSC.

In this paper, the developed ISSFA methodology is outlined and discussed. Furthermore, a comprehensive example analysis is implemented and a comparison is drawn with the currently applied simplified SMA method.

Risk-informed implementation of results from modern seismic hazard analyses into the design of new buildings of the existing NPP's (7-1773)

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The modern seismic hazard studies like PEGASOS for the existing NPP-sites exhibit higher spectral acceleration and uncertainties compared to the former seismic hazard studies for the same probability level. The informed technical and scientific community are discussing since a half decade the related topics like methodologies, assumptions to source characterization, appropriate attenuation models, site response issues, aggregation of expert models, plausibility checks of hazard results, etc. It seems that many issues of the seismic hazard will not be resolved in the short time period. However existing NPP's continue their operation and need to demonstrate the fulfillment of the national and international safety criteria. After the completion of PEGASOS-Project, the Swiss NPP's reviewed results together with international experts. The main conclusion of the review is that the uncertainties could be reduced, for example by means of additional site investigations. Swiss licensees have already started detailed geological and geotechnical investigations at the existing plant sites. The results of site investigations will be implemented in the follow-up project namely in the Pegasos Refinement Project (PRP). However these investigations, new site response calculations and the conducting of the PRP using different expert groups with a high level of detail and appropriate quality will require about four years.

In the meantime, the PEGASOS-results were implemented into the Probabilistic Safety Assessment of the NPP Beznau [2]. According to the PSA-results seismic is the dominating risk contributor to the Core Damage Frequency (CDF) and the Large Early Release Frequency (LERF). Independent of the PSA results, some new buildings and equipments are being planned to replace the emergency power supply from a nearby hydro plant. To design the new buildings and new equipment, the seismic input has to be defined. In this regard this paper proposes a risk-informed approach to determine the seismic design input. The main goal is to result in a low contribution of the new buildings and equipment to the seismic CDF and LERF. The proposed procedure leads to a consistent seismic safety for several buildings of the existing plant as well as avoids especially for foundations high construction costs due to a disproportional seismic input.

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PSA Level 2 – Experience with the review process from the perspective of the independent evaluator (7-1778)

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According to the current German PSA guide dating from August 2005, in the framework of the safety review, a PSA Level 2 has to be delivered by the operator every ten years. This analysis has to be made in compliance with the “state-of-the-art” scientific methods.

In the meantime, several analyses have been performed in Germany – based on the common understanding that performing a PSA level 2 lies within the operator’s field of responsibility.– The legal status (regarding the influence of the results on the supervision and licensing process) remains open but a review of the analysis by the regulator and its independent experts was carried out.

TÜV SÜD Industrie Service GmbH, Energy and Technology, was approved by the regulatory authority with the implementation of the mandated reviews for several NPPs in the south of Germany (both for PWR and BWR types).

Based on mutual understanding the IAEA-TECDOC-1229 “Regulatory review of Probabilistic Safety Assessment (PSA) level 2” represented the guideline and evaluation criteria.

The aim of this work – shown in this presentation – was to evaluate a German PSA Level 2 according to international criteria, to gather experience with the review process in comparison with the “old way” of a detailed expertise generally applied to German supervision procedures and, of course, to prove the quality and the results of the submitted analyses. The most important initial step of the review process was to bring national demands into line with international requirements and procedures and to specify IAEA TECDOC 1229 in terms of appropriate procedures and requirements.

The essential results of our work were

- To demonstrate how to convert international to national procedures
- To confirm that the selected approaches and methods are “state-of-the-art” in science and technology
- To prove that the qualitative and quantitative results of the analyses are traceable and understandable

- To identify the optimization measures that should be implemented quickly, including hardware changes as well as changes to the operating rules of the plant.

In conclusion, the review procedure has shown to be successful from the perspective of the evaluator and the other parties. The high safety level of the plants was reflected by the results of the analyses. The chosen approach is suited to assess the analysis carried out with respect to methodical approach and quality.

Probabilistic fracture mechanics: PTS Screening Criteria for RT_{NDT} , application of FAVOR code to a German KONVOI plant (7-1785)

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In Germany the structural integrity and safety of reactor components, like reactor pressure vessel (RPV) or pipes, is done by deterministic analyses. Deterministic approaches use conservative assumptions (crack geometry, external loadings, material properties, etc.) to maximize the safety margins. Reasons for such conservative assumptions can be missing information (aleatory uncertainties) or missing knowledge of certain mechanisms (epistemic uncertainties).

A probabilistic analysis uses the same methods (e.g. calculation method for stress intensity factor, crack propagation) as a deterministic one, but addresses the uncertainty of required input data or mechanisms. Unlike a deterministic analysis, which criterion is the achievement of a critical or reference value (e.g. stress intensity factor reaches lower bound of fracture toughness), a probabilistic analysis gives a probability of flaw initiation or component failure. Therefore a probabilistic analysis of a reactor component gives an additional classification of the integrity (safety) of such a component under more realistic assumptions and helps quantifying governing parameters for the component failure, which can be useful for lifetime extension.

An example of such a probabilistic safety analysis is the FAVOR (Fracture Analysis of Vessels, Oak Ridge) computer program (see Williams et al. (2007)), which was developed in the US by the Oak Ridge National Laboratory, and is applicable to the core region of a RPV. FAVOR was applied to the RPVs core region of a German KONVOI-type nuclear power plant (NPP) with the aim of defining a Screening Criteria for the nil ductility transition range temperature RT_{NDT} assuming pressurized thermal shock (PTS) transients.

The application of FAVOR, version 06.1, to a German KONVOI-type nuclear power plant (NPP) is done with the help of the following adjustments:

- For the one dimensional finite elements stress calculation in FAVOR, the thermal hydraulic conditions (convective heat transfer and coolant temperature at inner vessel wall) obtained from KWU-MIX were averaged in axial and circumferential direction.

- A distribution of postulated flaws was developed from results of ultrasonic (UT) examinations of the RPVs core region.
- As most of the UT indications are assumed to be underclad defects, own calculated stress intensity factors for underclad defects were implemented. The linear elastic stress intensity factor for an underclad defect was calculated according to Marie et al. (2008) with a plastic correction obtained from RSE-M (2000).

Several FAVOR applications with increasing cumulated neutron fluence ($E > 1$ MeV) at inner vessel wall according to effective full power years were made, in order to correlate the maximum reached RT_{NDT} in the circumferential weld or in the two forgings of the core region with the failure frequency (failure probability per year). For the Screening Criteria a maximum allowable failure frequency of 10^{-6} was chosen. This leads to a maximum allowable RT_{NDT} in the circumferential weld of the core region of 285°C and in the two forgings of the core region of 170°C.

Without changing the computer code of FAVOR the user has to deal with some possible implicit restrictions or assumptions. For example the implemented Weibull model for initiation fracture toughness or the lognormal distribution for crack arrest toughness can lead to an over – or underestimation of flaw initiation or crack arrest. Therefore it might be useful to adapt the computer code or develop an own probabilistic fracture analysis tool for the application of interest and its potential unique boundary conditions.

As conclusion, probabilistic fracture analyses are powerful tools to define governing parameters for component failure like RT_{NDT} under realistic conditions. This also helps to define governing parameters for lifetime extension. To ensure that the application of interest will be done without any unnecessary conservative assumptions, existing PFM tools like FAVOR may need to be adapted or an own tool may need to be developed.

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Effects of AAR on seismic assessment of nuclear power plants for life extensions (7-1789)

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In Nuclear Power Plants, the containment structures provide an ultimate barrier to fission product releases to the environment. The design has been performed and the operating strategies are in place to ensure that this final barrier is effective. It is very important to understand that the containment capabilities and the margins in performance of the containment to resist beyond design loads and severe accidents and to improve containment performance are being considered.

Several NPPs are nearing the end of their original design life and as service life is being extended, containment improvements are performed in order to meet the Probabilistic Safety Assessments (PSA) related targets. As part of the PSA work, Seismic Margin Assessment (SMA) is performed as seismic requirement has increased for some plants.

To perform SMA work, new Floor Response Spectra (FRS) for the Reactor Building (R/B) containment structure had to be developed, which in addition to new seismic requirements had to consider current condition of the structure including possible aging related degradation. Alkali Aggregate Reaction (AAR) was identified as applicable Aging Related Degradation Mechanism (ARDM) for one of the plants. Alkali reactive aggregate was used for construction and, although measures were implemented to mitigate possible reaction, evidences of AAR were found in some parts of the concrete.

Investigation was performed in order to establish the means of accounting for AAR in the development of Floor Response Spectra (FRS) for R/B to be used in SMA work. Review of studies and experiments to determine effects of AAR on mechanical properties of concrete and evaluation of the effects of restraint provided by the reinforcing and prestressing steel was considered. The main findings of the investigation are described in the paper.

Based on this investigation and considering geometry and current condition of the post-tensioned R/B, it was concluded that no change in modulus of elasticity to account for AAR is necessary for generation of seismic FRS. However, it was considered prudent to include a possible reduction in modulus of elasticity, which was quantified as 15% in order to account for possible local variations in material characteristics, environment of exposure, and chemical reactivity.

Measured compressive strength of the R/B concrete was higher than the strength used in original design calculations. Thus, an increase in the modulus of elasticity needed to be considered for seismic FRS generation for R/B.

Site-specific ground motion models for soil sites with thick sedimentary layers (7-1795)

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As so far ground motion prediction models (GMPM) for soil sites might be misleading if the selection of recordings is restricted to the uppermost 25 m or 30 m of the subsoil (and the average shear wave velocity $V_{s,25}$ or $V_{s,30}$) and if the whole underlying geological depth profile is ignored. The concept of subsoil and geology-dependent ground classes (recently introduced into German Code DIN 4149) offers an alternative approach by the explicit consideration of the thickness of sedimentary layers and their effect on the site amplification as well as on the shape of site-specific spectra. As a whole, six respectively seven site-specific subsoil classes are distinguished. Despite the fact that alongside the river Rhine in Western and South-Western Germany, the thickness of the sediments is reaching several hundred metres, site-specific data and corresponding attenuation relationships are missing due to lack of earthquakes and strong-motion recordings.

Within a series of comprehensive instrumental site studies supported by USGS, Californian strong motion stations are classified with respect to their ground classes considering the characteristics of the uppermost 25 m of the subsoil overlaying the geological depth profile (Lang & Schwarz, 2006). The commonly used subsoil class “soft soil” (C) is replaced and differentiated by three classes (C-R, C-T, C-S), where C-R stands for soft soil above rock (with high amplification factor in a small plateau range), C-S for layers with more than 100 m thickness, and C-T for a transition range. Ground class C-S is connected with reduced site soil amplification factors (which can also be derived from site response analysis). Taking profit from an accompanying study (see Schwarz et al., 2007), as a whole 484 records from Californian “soil site” strong motion stations are considered. From this basic or primary dataset (DS I) spectral attenuation relationships are elaborated with different regression methods; the one with the smallest standard deviation is used.

The attribute of “site-specific” implements two further elements: the deaggregation of Probabilistic Seismic Hazard Assessment (PSHA) for the relevant design hazard level (i.e. $10^{-5}/a$ for N.P.P.) and the selection of recorded ground motion for the mean or modal (controlling) magnitude–distance combination.

According to these principles, the design ground motion (DGM) is regarded as site-specific if ground motion data for the local depth profile are considered, exclusively, and if the database corresponds with the deaggregated hazard from PSHA results. In the low seismicity areas of Central Europe and in cases where DGM for very low probability rates of exceedance is required, small magnitude and near field events (often neglected in common GMPM) are of importance. Therefore, further sets of data (DS II) are taken from aftershock recordings in Turkey during in 1999/2000 (DS II) and – as an innovative element of the whole approach – from seismic instrumentation at reference site(s) in Germany contributing to a third dataset (DS III).

While the number of records in dataset DS I is permanently growing due the events recorded from Californian stations, for dataset DS III a longer time window is required, not at least due the low level of the seismic activity. The key and linking element of the presented procedure is the instrumental subsoil classification making the different datasets comparable and unique. Ongoing studies are related to a further refinement of the selection criteria by comparing not only the peak and amplitude level of H/V-spectra but also the frequency-dependent shape by a cluster analysis of all stations related to the target site and its depth profile.

The whole procedure can be characterized as a „single-station“ approach considering records from stations of comparable subsoil profiles under the assumption of similar site amplification effects. The impact of the datasets, their combination and composition on the ground motion prediction models (GMPM) are studied. Besides the realistic description of site-specific ground motion it will be discussed to which extent the procedure is contributing to a reduction in the uncertainty of ground motion models.

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Seismic performance assessment for safety-related nuclear structures (7-1818)

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Seismic Probabilistic Risk Assessment (SPRA) was developed in the early 1980s and subsequently accepted by the United States Nuclear Regulatory Commission (USNRC) to be used in nuclear power plant (NPP) Individual Plant Examination of External Events (IPEEE). The most widely used SPRA procedure is the Zion method, which was first developed and applied in the Oyster Creek probabilistic risk assessment and later improved and applied in 1981 to estimate seismic risk for the Zion Plant (Pickard, Lowe, and Garrick et al. 1981). In the Zion method, the component fragility curves are defined in terms of ground-motion parameters (generally peak ground acceleration), although the failure of a component has a much improved correlation to structural response parameters, such as floor spectral acceleration and story drift.

Procedures for seismic performance assessment of buildings have been developed in the ATC-58 project and proposed in the 50% draft *Guidelines for Seismic Performance Assessment of Buildings* (ATC 2008) (termed the draft ATC-58 Guidelines hereafter). These procedures use fragility curves that are defined using structural response parameters. The procedures in the draft ATC-58 Guidelines cannot be used directly for performance assessment of NPPs because the methodology does not accommodate accident sequences, event trees and fault trees but provides the robust technical basis needed to develop an alternative procedure for seismic probabilistic risk assessment for NPPs.

This paper introduces a new procedure based on the Zion method and the methodology presented in the draft ATC-58 Guidelines for seismic performance assessment of NPPs (Huang et al. 2008, 2009). The new procedure improves the Zion method by using nonlinear response-history analysis and structural response-based fragility curves. This procedure is used to evaluate seismic risk of a sample NPP reactor building of conventional and base-isolated construction. The impact of the implementation of base isolation on the seismic performance of the sample NPP is identified.

The proposed procedure includes five steps. Step 1 involves the characterization of the seismic hazard. The procedure allows the seismic hazard to be defined using a user-specified *intensity* of earthquake shaking, a user-specified *scenario*

of earthquake magnitude and distance or a *time*-based representation considering all possible earthquakes. The final products of intensity – and scenario-based assessments are the probability of unacceptable performance of the NPP to the specified intensity and scenario of earthquake shaking, respectively. The final product of a time-based assessment is the annual frequency of unacceptable performance of the NPP. In this paper, only the results for the time-based assessment of the sample NPP are presented since the annual frequency of unacceptable performance is the most widely used index for risk assessment of NPPs.

The seismic risk for the sample conventional and base-isolated NPP reactor buildings was evaluated using the procedure described above with a focus on the secondary systems in the sample NPP. The mean annual frequency of unacceptable performance of the base-isolated NPP is nearly orders of magnitude smaller than that of the conventional NPP.

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Development of a reliability data handbook for piping components in Nordic nuclear power plants (7-1837)

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The Nordic PSA Group (NPSAG) has undertaken to develop a piping reliability parameter data handbook for use in risk-informed applications that involve the consideration of structural integrity of piping systems. The scope of the handbook is to establish high quality reliability parameters that account for the Nordic and worldwide service experience with safety-related and non-safety-related piping systems in a consistent and realistic manner.

While the work to develop the handbook is expected to be finalized during 2009, the planning for its preparation has been underway for well over ten years. An important step towards the handbook development project has been the international cooperative effort through the OECD Nuclear Energy Agency to create an event database (OPDE) on the service experience with piping in commercial nuclear power plants; an event database, which provides the necessary input to the work with the NPSAG handbook.

The paper will demonstrate the progress made since the initiation of the R-Book project in 2005. The paper also summarizes the results and insights from a pilot project to define the content and outline of the proposed handbook. Comments and recommendations for the R-Book development process were solicited from Nordic and international experts. Detailed information about technical considerations for how to derive realistic pipe failure rates from the available service experience data is documented in SKI Report 2008:01 [1] (January 2008, available from <http://www.ski.se>). Work is currently underway to produce the R-Book and a first edition is scheduled for release to project sponsors during the second half of 2009.

The paper and the presentation at SMiRT 20 will mainly be focused on presentation of results and insights made when analyzing different process piping's and experiences learned from working with the OPDE database.

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Benchmark exercise on risk-informed in-service inspection methodologies (7-1841)

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In 2005, the Joint Research Centre of the European Commission (JRC) together with the Nuclear Energy Agency of the OECD (NEA) launched a project for benchmarking various risk-informed in-service inspection (RI-ISI) methodologies. The project, called RISMET, had more than twenty participating organizations from Europe, U.S., Canada and Japan. The JRC acted as the technical coordinator of the project, and the NEA provided secretariat support.

The overall objective of the project was to apply, for the first time, various RI-ISI methodologies to the same case, i.e. selected piping systems in one nuclear power plant, with the idea of verifying whether they would lead to significantly different results. Also, a benchmarking exercise would ideally result in the identification of those phases in a methodology with the greatest potential to affect the outcome, and might suggest areas for further improvement.

Four systems from the Swedish PWR Ringhals 4 were selected for the benchmark exercise. The following criteria were used for selecting these systems: All safety classes should be covered; a variety of degradation mechanisms should be covered; good coverage of risk categories should be achieved; systems with a significant increase or decrease in the new inspection program (before/after applying RI-ISI) should be included; and balance between initiating and mitigating systems should be ensured. Based on these criteria, the following systems were suggested by Ringhals and approved by the project team as the scope of the exercise:

- Reactor coolant system
- Residual heat removal system
- Main steam system
- Condensate system.

The following approaches to define the ISI program were considered in the benchmark exercise:

- Swedish regulatory requirements (“SKIFS”)
- PWROG methodology
- PWROG methodology adapted to Swedish regulations (“PWROG Swedish”)
- EPRI methodology
- Code Case N-716, “streamlined RI-ISI”
- ASME Section XI (deterministic).

The application results were evaluated by five groups concentrating on the following issues: 1) Scope of application; 2) Failure Probability Analyses; 3) Consequence analyses; 4) Risk ranking, classification and selection of segments/sites to be included in inspection programs; and 5) Regulatory aspects. The evaluation included the identification of differences in the RI-ISI applications, the analysis of the importance of identified differences, and the comparison between RI-ISI and “traditional” inspection programs.

Even if the scope of the benchmark was limited to four systems, the variety regarding safety class, potential degradation mechanisms and pipe break consequences ensured a good coverage of issues for a comparative study. The risk-informed methodologies showed some significant differences and resulted in slightly different risk ranking and selection of inspection sites. However, the results of the benchmark indicated that the risk impact of these differences is small, and the RI-ISI approaches identify safety important piping segments that are ignored by approaches not using the probabilistic safety assessment (PSA). The results of the benchmark exercise RISMET improve the knowledge on differences in approaches and their impact on plant safety, and promote the use of risk-informed ISI.

This paper summarizes the results of the RISMET benchmark exercise.

CANDU pressure tube degradation and probabilistic safety criteria (7-1847)

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The aging management and plant life extension processes for commercial nuclear power plants have evolved over several decades. As nuclear power plant age, degradation of structure, systems and components can be expected to occur. The understanding of the effect of the age related degradation is important to ensure safe operation of the nuclear power plant. The wealth of service experience data from aging research programs in combination with plant-specific data provides a good foundation for risk-informed applications to assess the potential effects of aging management on plant safety and reliability. Aging or degradation phenomena may lead to time-dependant changes in engineering properties that can impact the ability of plant systems, structures and components (SSCs) to respond to anticipated and unanticipated challenges during routine and upset plant operations.

Probabilistic Safety Assessment (PSA) is an accepted tool for risk-informed decision-making. A key consideration of PSA applications is to demonstrate compliance with regulatory and corporate safety objectives during the whole lifecycle of a NPP, including the extended lifetime. Demonstration of compliance with applicable safety objectives is required for the NPPs that are subject to plant life extension. Incremental changes to core damage frequency (CDF), large early release frequency (LERF), Δ CDF and Δ LERF provide measures of how aging could affect plant safety.

The pressure tubes in CANDU reactors are critical components and its failure has consequences of concern. These cold worked Zr-2.5% Nb pressure tubes may degrade due to ageing mechanisms like creep and delayed hydride cracking. It is important to periodically monitor the serviceability of these tubes over the service life. This paper discusses the estimation of the allowable conditional failure probability of the pressure tubes for meeting the risk informed regulation criteria of the US-NRC regulatory guide RG. 1.174.

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Results and insights from interim seismic margin assessment of the Advanced CANDU Reactor (ACR) 1000[®] reactor (7-1849)

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The ACR-1000[®] reactor developed by Atomic Energy of Canada Limited (AECL) is a 1200-MWe-class light-water-cooled, heavy-water-moderated pressure-tube reactor, which has evolved from the well-established CANDU[®] line of reactors. It retains the basic, proven, CANDU design features while incorporating innovations and state-of-art technologies to ensure fully competitive safety, operation, performance and economics.

The objective of this paper is to describe the seismic margin assessment performed for the ACR-1000 reactor at full power operation. The design basis earthquake (DBE) is 0.3g peak ground acceleration (PGA). The seismic margin was assessed, and potential seismic failure modes as well as weak component links/functionality leading to severe core damage and widespread fuel damage were identified.

The Level I internal event at-power PSA models were reviewed and the systems required to bring a plant from a normal operation to a safe shutdown were identified in the seismic safe shutdown equipment list (SSEL). In the first approach, seismic capacities of the items on the SSEL have been developed using the ACR seismic design criteria and qualification criteria, past seismic experience and recent seismic probabilistic safety analyses and seismic margin assessments. The plant responses to seismic events were modelled in seismic event trees, from which the accident sequences potentially leading to severe core damage and widespread fuel damage were identified. These accident sequences determined a combination of the failures of frontline systems. There exist the dependencies between frontline systems and their support systems, and among support systems. These dependencies were included appropriately using system dependency matrices. Then the accident-sequence seismic capacities were estimated from the seismic failures of structures or components resulting in frontline systems and their support systems in terms of high confidence of low probability of failure (HCLPF). The plant HCLPF capacities for severe core damage and widespread fuel damage were then determined.

This assessment shows that the ACR-1000 design can reasonably achieve a seismic margin in terms of the plant HCLPF that is equal to or exceeding 0.5g PGA. Therefore, the ACR-1000 design is inherently capable of safe shutdown in response to a strong magnitude earthquake.

Application of CFD code PHOENICS for simulating CYCLONE SEPARATORS (7-1867)

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Keywords: separator, CFD, IPSA, PHOENICS, two-phase flow

Cyclone separators are widely used in the field of air pollution control, gas–solid separation for aerosol sampling and in many industries like power plants, sand plants etc. In cyclone separators the air flow enters the cyclone through a tangential inlet, generates a swirling flow that forces entrained particles radially outward and leaves via an axial outlet pipe at the top of the cyclone. The rotational fluid motion is generated from the energy obtained from the fluid pressure gradient. This rotational motion causes the particle to separate relatively fast due to the strong acting forces. The cyclone separator is very useful engineering equipment with no moving parts and virtually no maintenance. It enables particles of micrometers in size to be separated from a gas moving at about 15 m/s without excessive pressure-drop.

This work presents a computational fluid dynamics (CFD) calculation to investigate the flow field in a tangential inlet cyclone which is mainly used for the separation of the moisture from an air stream. Three-dimensional, steady state Eulerian simulations of the turbulent gas–droplet flow in a cyclone separator have been performed. Numerical simulation was carried out using CFD code PHOENICS for the given geometry of separators available in literature. The IPSA (Inter-Phase-Slip Algorithm) method has been utilized which entails solving the full Navier-Stokes equations for each phase. The turbulence was modeled with standard k- ϵ turbulence model. The liquid droplet was modeled as particle of size 10 μ and density 1000 kg/m³. The volume fraction of moisture was 1% at inlet and outlet volume fraction was predicted with CFD. The results were in good agreement with the reported results. This knowledge can be further extended for other two phase flow applications in nuclear industry.

Seismic fragility of a civil engineering structure (7-1871)

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Introduction

Evaluation of seismic fragility is an integral part of seismic probabilistic safety assessment. Seismic fragility of civil engineering structures are generally evaluated by analysis. A nuclear power plant has many safety related civil engineering structures, with different structural configurations. These configurations vary from a common framed architecture to complex shear wall type design.

A structure is considered as system, an assemblage of a number of elements. Seismic fragility for a particular failure mode is determined for each element. Strength based approach is common for fragility evaluation of element. Weakest link approach is generally adopted to determine the seismic fragility of the overall structure, wherein it is assumed that the structure/component fragility is same as that of the weakest element. Civil engineering structures are complex structures with high degree of indeterminacy. It may not be rational to apply weakest link approach for this type of structures. Herein the failure of one element may not cause global failure of the structure. Seismic fragility evaluation should account for this consideration.

Aim of the work

Seismic fragility, taking into consideration of global failure of structure, can be evaluated by means of deflection based approach. It is considered that a building frame could be designed to a given level of seismic forces – as long as the building could endure the distortions involved. One approach is to consider lateral storey drift as structural response parameter. Another approach is to adopt a displacement based analysis technique like pushover analysis. This analysis identifies critical elements of the structure which would result in the formation of a mechanism leading to failure of the structure. The latter approach seems to be a new one.

In the paper, seismic fragility of civil engineering structure is evaluated adopting three methods:

- Method – 1: Strength based approach,
- Method – 2: Deflection based approach considering storey drift as structural response parameter, and
- Method – 3: Based on pushover analysis.

Using an example problem of a representative bay from a RCC framed structure of a NPP, the paper compares the seismic fragility evaluated adopting the above three approaches.

Essential results

The paper will describe determination of seismic fragility curves by the three methods with numerical examples. Comparative study of the results obtained from three methods will also be presented.

Summary

Evaluation of seismic fragility of civil engineering structures should take account for consideration of global failure. This can be accomplished by adopting a global failure parameter as the basis for deriving seismic fragility or by adopting a displacement based technique like pushover analysis. Out of the three methods described in the paper, method based on pushover analysis technique is observed to be the rational approach for determining the seismic fragility of a civil engineering structure.

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Thermal-hydraulic analysis for accidents in OPR1000 and evaluation of uncertainty for PSA (7-1878)

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Probabilistic Safety Assessment (PSA) is a conceptual and mathematical tool to evaluate numerical estimates of risk for nuclear power plants (NPPs) and can be used to calculate the probability of damage to the core as a result of sequences of accidents identified. After the first comprehensive application of the method, reactor safety study, WASH-1400, PSA has become a standard tool in safety evaluation of not only NPPs but also industrial installation. When PSA is performed, thermal hydraulic analysis is necessary to obtain the basic data, from which system success criteria for construction of event tree and the allowable outage time for human reliability analysis are determined. Up to now, this analysis has been undertaken with various system codes such as RELAP, RETRAN, MELCOR and MAAP4. However, it is well known that deterministic assumptions and input values have been often applied to the analysis even if most of them are best-estimate codes. To acquire more realistic result, the analysis with nominal value and realistic assumptions needs to be carried out and the uncertainties from the result of the analysis are needed to be essentially quantified.

The aim of the present study is to develop a best-estimate thermal hydraulic analysis methodology applicable to PSA as well as able to quantify uncertainty. In the present study, Optimized Power Reactor 1000MWe (OPR1000), which is the standard nuclear power plant in Korea, was selected as the objective power plant. As the frozen code, MARS code was chosen, which is best-estimate code and has been developed at Korea Atomic Energy Research Institute (KAERI) by consolidating and restructuring the RELAP5/MOD3.2 code and COBRA-TF code. Korea Hydro and Nuclear Power (KHNP) Cooperation has already performed PSA of OPR1000 and made the accident sequence table in which the accidents have ranked along to the frequency of occurrence. Thus the accidents in the table as mentioned are required to be analyzed with the input made based on the realistic assumptions. Moreover the uncertainties from the results of analyses should be quantified. To do what are aforesaid, Phenomena Identification and Ranking Table (PIRT) of each accident is necessary since we cannot consider all parameter for reason of calculation time and cost every analysis. PIRT for each accident has been already made by the group of expert and is

desirable to be reconstructed if needed. With its own range and distribution, each candidate parameter in the PIRT was simulated in MARS code. On this occasion, it was assumed that the range had 95% confidential interval and acceptable assumption is applied only when the information about the distribution of parameter does not exist. A number of calculations by MARS code were performed repetitively with varying the input value of certain parameter within its uncertainty range. The peak cladding temperatures (PCTs) from the calculation results, with which it was determined if the core was damaged, were used to construct the response surface and quantify the uncertainties. Conventionally, there are several methods to quantify the uncertainties; Monte Carlo Method (MCM), Latin Hypercube Sampling Method (LHSM), Response Surface Method (RSM), etc. In the present, MCMs are the most widely used means for uncertainty analysis. However, after full consideration of time and cost for the present study, RSM is most suitable to perform the study since there are more than ten parameters for each accident and it takes too much time to use MCM to carry out uncertainty analysis. The regression equation for PCT was obtained by RSM and the randomly sampled values from the range of each parameter were substituted for the equation. As a result, the distribution of PCT of each accident was gained and it was used to assess the core damage frequency (CDF) from the PSA which is already performed.

Evaluation of the seismic damage index of structures using fuzzy logic (7-1890)

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This paper presents an explicit fuzzy logic based method for assessment of seismic damage of the structures. Uncertainties in earthquake ground motion parameters and in structural parameters modeling behaviour during earthquake is expressed linguistically by fuzzification. The number of this parameters and the choice of the membership functions for the parameters and the damage index are generally analysed.

The fuzzy rule base for assessment of seismic damage of the structures is formed for $n = 9$ input parameters, which take values in the set $M = \{1,2,3\}$. These numerical values are attached to the linguistic values of the parameters by the level of contribution to damage of structure. To the each system of nine values of input parameters (x_1, x_2, \dots, x_9) is attached a value of output f in the set of integers $\{1, 2, 3, 4, 5\}$, witch conventionally represent the five levels of damage, eventually with a percent degree of membership to this level. Then the response (output) f can take two values $\in \{1, 2, 3, 4, 5\}$ with the same probabilities p_i and q_i (equals to the degrees of the membership to two adjacent classes). If more than two parameters takes, independently, two adjacent $\in \{1, 2, 3\}$ each of them with the complementary probabilities (percents) p_i and $q_i = 1 - p_i$, then the single output (response) f take more than two values with the probabilities (percents) equals with the sum of the products of corresponding probabilities (by the law of alternatives or total probability). A generalised control rule utilizing weight functions for the input parameters is proposed.

The damage index is estimated finally by applying the centroid defuzzification method, which express the fuzzy linguistic variable by a crisp value. A program in MATLAB language for the logical inference is also presented. The proposed MATLAB program can be easily adapted for other fuzzy parameters or other number of parameters if their membership functions are defined.

A numerical example verifying this proposed method and a couple of final remarks close the paper.

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A procedure for the computation of seismic fragility of equipment components in NPPs (7-1904)

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A numerical procedure for computing the seismic fragility of equipment components in nuclear power plants is presented, focusing on the effect of the introduction of a base-isolation system encompassing HDRB devices (High Damping Rubber Bearings). It is assumed that: 1) no significant interaction exist between the dynamic behaviours of the building and of the equipment, so that a two-stage analysis is possible; 2) the dynamic behaviour of the reactor building can be linearized.

The procedure is based on the use of the Response Surface Method for modelling the influence of the selected random variables on the building response; given the system linearity the building performance can be described in terms of dynamic amplification. More precisely the ratio a/a_g of the peak acceleration at the component supports to the peak ground acceleration is considered for the non-isolated reactor building.

A Central Composite Design is used as experimental strategy; to account for random excitation, at each experimental point the dynamic analysis is repeated for a number of realizations of the seismic input. From the results, the mean and variance of the extreme value of the dynamic amplification R are computed; two Response Surfaces are built, modelling both the mean and the variance of R (“dual RS” approach), which are obviously function of the problem random variables. The system performance function can be subsequently expressed as the difference between a given amplification ratio a/a_g and the value of R .

Once R is defined, in probabilistic terms, the computation of the probability of exceeding a given amplification factor $P_{exc}(a/pga)$ can be obtained, via Monte Carlo Simulation, for all amplification values in the selected range.

A procedure for refining the RSs is also proposed, based on the computation of the seismic risk in terms of annual probability of failure for a prototype site and for a given value a_f of the support acceleration leading to collapse. By denoting with $p_{PGA}(pga)$ the PDF of the annual extreme of the PGA, the risk is equal to the convolution of the conditional probability $P_{exc}(a_f/pg_a)$ and the pdf $p_{PGA}(pga)$. By investigating the integrand function in the convolution, the PGA range delivering the largest contributing to the total risk is defined: from this the amplification range in which the RS's must be refined is stated.

An example of application is shown regarding the analysis of a preliminary design of the auxiliary building of within IRIS reactor (~335 Mwe pressurized light water reactor) under development by an international consortium which includes more than 21 partners from 10 countries, led by Westinghouse Electric Company. Installation in a site characterized by a low-to-average seismicity level is assumed. Fragility analysis is performed focusing on critical components inside the reactor vessel; the results show the practical applicability of the method, leading a reasonable balance between computational effort and degree of refinement.

When the introduction of the isolation system is considered, resulting in a drastic reduction of horizontal peak accelerations inside the building, attention is more focused, in the fragility analysis, on the behaviour of HDRB devices. A limit value of the relative displacement of the isolators is assumed and the annual probability of exceeding this value is addressed with the same criteria as above mentioned. The adopted linearized model of the HDRB devices is validated by comparison with the results obtained, in the dynamic analysis of the building, by means of a refined force-displacement hysteretic model taken from the literature and implemented in the building FE model. The results obtained are discussed, especially in light of the effectiveness of the HDRB devices and of the role played by the vertical excitation, which is not affected by the isolation system.

Detailed plant seismic walkdown of the Armenian NPP – Unit 2 (7-1949)

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The Armenian Nuclear Regulatory Authority (ANRA) and the Armenian Nuclear Power Plant (ANPP) requested to the International Atomic Energy Agency (IAEA) to provide an engineering service to be performed within the framework of the TC Project ARM/9/014 and in relation to one of the critical tasks of the seismic safety upgrading programme of the ANPP-Unit 2, corresponding to the final detailed plant seismic walkdown. This task has been committed to and performed completely by CKTI-Vibroseism (CVS) team consisted of S&A, CR and CVS experts with the project review by Dr. John Stevenson.

The ANPP is located in one of the highest seismic zones with ZPGA level of 0.35 g and seismic capacity of the plant was and still is a key issue of ANPP restart-up in 1990s and actual operation.

Current project was aimed to:

- (a) evaluating the seismic capacity of systems, structures and components (SSCs),
- (b) screening out those SSCs from further consideration because of their adequate seismic capacity, and
- (c) proposing the necessary upgrades (either “*easy fixes*” or “*non-so-easy-fixes*”) as required and appropriate.

Paper will introduce the main findings of ANPP seismic re-evaluation and upgrades and results of walkdown and SMA CDFM analysis.

In result of investigation of all previous activity and acquired data, comprehensive walkdown of 2589 items included in the Safe Shutdown Equipment List (SSEL) and analysis of HCLPF values for the most important components of the ANPP it was concluded that there is a good reason to believe, that the problems of ANPP seismic re-evaluation can be successfully resolved within a limited period of time if advanced analysis and test techniques are used, and also if necessary administrative solutions will be accepted.

A quantitative method for RI-ISI assessment (7-1975)

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The applications of risk-informed approaches to develop in-service inspection (ISI) programs are becoming increasingly common in nuclear power plants. When moving from a deterministic ISI program to a new risk-informed selection of inspection targets, the safety authorities ask for the assessment of the risk impact of the change. The risk impact evaluation is especially important in cases where RI-ISI program is used to justify a reduction in the total number of inspections.

Assessing risk impact of a RI-ISI inspection program requires three components: modeling the probability of pipe failures, assessing the consequences of pipe failures and modeling the effect of inspections in detection of degradation before failure. This paper presents a method for modeling each of these components and combining the results for a full quantitative assessment of any RI-ISI program.

The estimation of piping failure probabilities is based on a combination of probabilistic fracture mechanics (PFM) calculations to simulate crack growth and a discrete-time Markov process for modelling the inspection activities. The PFM analyses are performed with a tool based on the deterministic fracture mechanistic analysis code VTTBESIT, developed by the Fraunhofer-Institut für Werkstoffmechanik (IWM), Germany and by VTT. VTTBESIT was modified by adding probabilistic capabilities to the code. In the PFM tool the following randomised input parameters are used: exponential distribution for initial crack depth, exponential distribution for initial crack length and Poisson distribution for thermal load cycle frequency. Monte Carlo simulations are run by sampling from the probability distributions of the random parameters and using the respective crack growth (for instance fatigue or stress corrosion cracking) equation. The simulation ends either when the crack depth reaches the outer pipe surface, or the time cycles reach the end of plant lifetime, here taken as 60 years. Each run is then a 60-year simulation with the crack depth calculated at 1-year intervals, corresponding to the frequency of regular maintenance outages, and conditional on the existence of an initial flaw. The annual crack depth information for each simulation is transferred to the second phase of the analyses.

In the second phase, the analyses are based on Markov processes. First, the PFM simulation results for crack growth are used to construct transition matrices, where the states of the Markov process correspond to crack penetration depths in the material, and the transition probabilities from a lower state to higher states (deeper cracks) model the crack growth. The transition matrix can

be time dependent to take in to account different PFM calculations as a source for the transition matrix at different times throughout the plant lifetime. This makes it possible to have different crack growth characteristics in each, for example, 10 year interval of the plants lifetime. Inspections are taken into account with another transition matrix, where the probability of detection (POD) is a function of the crack size. The effects of inspections are included in the model as transitions from a crack state to a flawless state.

Results of a standard PSA analysis are used to find the risk consequences of any pipe failure. When the piping failure probabilities calculated in the Markov process are weighted with consequences of pipe failure, the full risk impact of any inspection strategy (selection of inspection targets, ISI intervals, ISI capability) can be assessed. One significant advantage of the approach is that for analysing different inspection strategies, the time-consuming PFM calculations need not to be re-run. Only the Markov analyses, that can be run very fast, are performed.

The model is based on a number of assumptions and limitations, related for instance to the randomised parameters of the PFM model. The absolute values are thus subject to large uncertainties. The Markov property assumptions made for the transition from one flaw depth state to another can be questioned, since for some degradation mechanisms, it might be justified to assume a “memory” (the future growth depends not only on the present flaw size, but also on how this size has been achieved). Further, the assumptions in the inspection process are highly simplified, assuming that detecting a flaw returns the component to a flawless state. Despite these simplifying assumption it is believed that the modelling approach gives reasonably realistic results for the comparison of alternative inspection strategies and for risk impact assessment.

The developed method and results are illustrated by applying them to a selected piping system in an existing Finnish NPP.

Evaluation for run-out distance distribution of rocks falling from slopes (7-1979)

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The Japanese “Examination Guide for Seismic Design of Nuclear Power Reactor Facilities” was revised in 2006. In the revised guide, it is described that the slope failure must be recognized as one of the potential phenomena triggered by earthquake occurrence. Although a large number of studies have been made on deterministic evaluation of slope stability, only few attempts have been made at effects evaluation of post failure of slopes due to earthquake ground motions.

For this purpose, it is important to analyze the factors which determine movements of rocks falling from slopes. The objective of this paper is to specify the factors which affect the run-out distance of rocks falling from slopes, and slope dimension, slope inclination and rock size are considered in this analysis. We adopt two approaches: one is stone falling test using shaking table and the other is numerical simulation using the analytical method for discontinuous body.

In order to conduct the test, the concrete slope, on which the steel box containing stones is attached, is established on the shaking table. The height of the slope is 1.2 meter. We conduct the following tests:

Test 1: Three hundreds of stones extracted of the same kind rocks are prepared. The stones are classified into the two groups: the stone size of the first group is between 20 mm and 40 mm, and that of the second is between 40 mm and 80 mm. First, every stone is fallen from the slope top (the edge of the box bottom), individually. Secondly, three hundreds of stones are fallen from the box, simultaneously. The histograms regarding the stone travel distance from the slope toe are obtained in a case where we change the slope inclination and stone size, respectively.

Test 2: By conducting the shaking table tests, the mass of stones falling from the slope top is measured corresponding to the distance from the slope toe. The size of the shaking table is 3.0 meter times 3.0 times, and sine wave is input as ground motions. Slope dimension, inclination, stone size are chosen as experimental parameters. In the current test, the slope model which side walls are installed to restrict of the stone movements in the slope traverse direction is called as the two dimension slope, and the slope model with no side walls is

called as the three dimension slope. Meanwhile, we perform the numerical simulations by using two-dimensional Distinct Element Method (DEM) ¹⁾ in order to compare the results from Test 2.

The following results are obtained from the experiments and the numerical simulations:

- (1) The falling stone mass from the slope of 60 degrees inclination is larger than from the slope of 41 degrees inclination. The maximum distance of falling stone from the slope of 41 degrees inclination is larger than from the slope of 60 degrees,
- (2) The histogram obtained from the numerical simulation, which represents the relationship between the distance from slope toe and the mass of falling stones, is compared with the histogram from the Test 2. The mode, which is the peak of the histogram, is almost equal and the maximum distance differs, each other.

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Reliability analysis of slope stability at nuclear power plant site (7-1982)

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It is well known that great uncertainty and variability exist in subsurface materials at nuclear power plant site regarding the layer uniformity and soil/rock engineering properties, as well as seismic loadings to be considered. Those uncertainties and variability not only affect the stability of the subsurface materials and foundations, but also affect the stability of the slopes that are usually present at the site, especially when seismic loadings are involved. This study focuses on the impact of the uncertainties of soil and seismic loads on the slopes stability analysis and how to determine the reliability of slope stability analysis results in term of factor of safety. In this study, uncertainties involved in the slope stability analysis were first identified, and then a procedure to cope with those uncertainties in the analysis was developed. After conducting sensitivity study, the parameters that have greater influence on the slope stability analysis results are identified. Probability analysis method that considers the variation of parameters caused by uncertainties and involved in the slope stability analysis was proposed with the use of reliability index as the reliability measurement of the factor of safety of slope stability. The proposed procedure was applied to slope analyses for a specific site using computer software GeoSlope. The analysis results illustrated that 1. The seismic loadings and soil internal friction angle parameters affect the slope stability analysis results more than other parameters involved in the analysis; 2. The proposed reliability analysis method can give a good indication of the degree of reliability of the factor of safety of slope stability; and 3. The degree of uncertainty of engineering properties of subsurface material and seismic loading conditions affect the reliability of slope stability analysis. Based on the results of this study, it is concluded that the uncertainties of site subsurface materials and site specific seismic loadings should be considered in the site stability analyses and their impacts on the analyses results need to be studied. This paper proposed a method to evaluate the reliability of site slope stability analysis results in term of factor of safety and this concept may be applicable to other site safety analyses.

Seismic damage assessment by probabilistic seismic demand models applied to NPP structures (7-1993)

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Introduction

This paper extends one of our contributions to the SMiRT 19 Conference, as well as of other earlier papers of ours [1 – 3] dealing with the probabilistic modeling of the seismically induced damages in structures, and also with the seismic fragility models for NPP structures. A couple of contributions to the 13 WCEE Conf. (Vancouver, August 2004) were based on such approaches involving the probabilistic demand and capacity analysis, or probabilistic performance-based demand models. Authors like Y.K. Wen, C. Alin Cornell, S. Janković and B. Stojadinović, J.W. Baker, K. Mackie, a.o. proposed models and methods going along such approaches for RC frame buildings but also for nuclear facility structures. Such investigations are based on the (pre)standards elaborated by FEMA and PEER in several reports starting with FEMA 350 & 356 (2000), PEER 2003/08.

Performance based seismic demand models

The probabilistic estimation of the demand on a structure from a given earthquake is essentially based on one (or several) intensity measure(s) – *IM*, and it consists in evaluating the probability that the earthquake will cause a certain level of demand in the structure as a function of such *IMs*. In our paper [4] we investigated several formal relationships between the engineering demand parameters (*EDPs*) and the seismic hazard parameters (like *IMs*) met in PBSM models. The conditional probabilities and the total probability theorem are basic notions that occur in PBSMs. A typical performance based seismic demand model involves an equation of the form

$$\lambda_{EDP}(z) = \sum_{\text{all } x_i} P(EDP > z \mid IM = x_i) \cdot \Delta IM(x_i) \quad (1)$$

where the *EDP* is compared with the level z , *IM* is an intensity measure with x_i as one of its possible values and the annual frequency of exceeding it given by $\lambda_{IM}(x_i)$.

PBSD models applied to NPP structures

In the present paper we also investigate some mathematical aspects of the PBSD models but we discuss some ways to apply them to seismic damage assessment and to performance based evaluation of NPP structures. Damage indices for concrete and steel, at FEM level, were presented by four Korean authors in their contribution [8] to the 13 WCEE, for the EQ damage assessment of NPP containment structures. We suggest the use of other damage indices besides those due to Roufaiel & Meyer and Miner. As regards the P-B evaluation of NPP structures, we investigate how the notion of confidence ratio (approached in [3]) can be used to a higher extent for the risk reduction ratio. This ratio is expressed as

$$\lambda_{\text{con}} = \frac{\gamma \hat{D}^{P_H}}{\phi \hat{C}} = \exp\left(-K_x \beta_{UT} + \frac{k}{2b} \beta_{UT}^2\right) \quad (2)$$

where P_H is the probability of exceeding a spectral acceleration hazard, D is the demand, C is the capacity, k is a slope of the log-log plot of P_H , β_{UT} is the total standard deviation due to uncertainty. This expression (2) is based on FEMA 350 report, but it can also be used to express the risk reduction ratio of FEMA 350 in terms of the confidence ratio by

$$R_R = \frac{1}{\lambda_{\text{con}}^{k/b}} \exp\left(\frac{k^2}{2b^2} \beta_{UT}^2\right) \quad (3)$$

Conclusions

Providing a method to compute a level of risk reduction implied by FEMA-356 structural evaluation provisions can allow for a proper calibration and rational use of these provisions for nuclear facility structures. Some ways to follow are sketched in the conclusions to paper [9] by Orbović *et al.*

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Safety margins in mechanical integrity assessments for passive NPP components (7-2014)

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NULIFE (NUclear plant LIFE management) is a European excellence network comprising nuclear utilities. It was launched in 2007, and aims at promoting methods to evaluate and manage the lifetime of NPPs, particularly the lifetime of materials and passive components. In this regard, the issues related to safety and risk assessment methodologies are an important concern; they become increasingly significant with the ageing of NPP components and with the industrial objectives of lifetime extension. They are investigated in the framework of NULIFE expert group n°4 called “Safety and Risk”.

One particular concern of both regulators and operators is the evolution of safety margins versus time for the components subject to ageing phenomena: the requirement of the regulator is to keep the same safety level, and therefore the same safety margins, during the whole component lifetime, and this requirement is a prerequisite for any possibility of lifetime extension. Therefore, it is necessary to identify what safety margins are affected by this requirement.

This paper presents the various meanings of “safety margin” that have been identified at EDF for the safety margins at the component level. The margins considered are basically those appearing in the mechanical integrity analysis of passive components of NPPs.

Firstly, the importance and significance of the use of safety margins is mentioned. As the first sense of this term denotes the overall safety factor, the limits of the use of overall safety factors are exhibited on a basic example. The mechanical integrity analyses of components rely on a mathematical model of the failure phenomenon, including possible degradation causing structural ageing. This model characterizes the structural state, which is compared to a limit (unacceptable) state: this is the failure criterion. Various types of “safety margins” appear. There is a distinction between implicit and explicit margins. These two types of margins contribute to the conservatism of the analysis, and therefore to the structural safety. Explicit margins are the safety factors or coefficients, that can be an overall safety factor or multiple partial safety factors applied to several input parameters of the analysis. Implicit margins are

encompassed in the characteristic values representing the input variables of the physical model: they are taken as pessimistic values.

Secondly, an understanding of the requirement to keep the same safety margins is given. It may apply to the safety coefficients. But even these regulatory coefficients should be considered as a formal reference and other equivalent sets of factors may be adopted under certain conditions of statistical knowledge. And the way to perform this equivalence of safety factors is of course to resort to probabilistic methods, that provide a rational and rigorous treatment of the uncertainties and explicitly refer to safety (reliability levels). Finally, what is important is that the safety level has to remain unchanged; and the appraisal of the safety level should at least include probabilistic highlights.

Finally, a current example of such methodological evolution including probabilistic considerations is given on the case of the reactor pressure vessel. RPVs are submitted to a risk of fast fracture due to the following characteristics:

- Existing manufacturing cracks
- Steel embrittlement due to cumulative neutron irradiation,
- Possible Pressurized Thermal Shocks (PTS), occurring under certain accidental loading conditions.

Two populations of cracks are considered: detected cracks, with known dimensions and location, and non detected cracks, whose characteristics are not known. The current deterministic treatment of this population leads to overly conservative assumptions, that may endanger the objectives of lifetime extension adopted by EDF. Therefore, an alternative, semi-probabilistic approach, has been developed. It takes advantage of the better statistical knowledge of the input variables like fracture toughness, RT_{NDT} shift, fluence and especially the non detected crack population. In the deterministic analysis, this population is represented by one single crack (“generic crack”) with the most pessimistic location and dimensions. Some assumptions are performed about the distribution of these parameters. It is therefore possible to compare at a given age the RPV reliability of the so-called generic crack, and the reliability of the population of non detected cracks, which is of course much better. Consequently, it is also possible to calculate the age at which the two reliabilities are equivalent. This gives the possible RPV lifetime extension. This approach is summarized in this paper.

Load factor in case when separating aleatory uncertainty and epistemic uncertainty (7-2023)

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Aim of the work

The uncertainty exists in the loads acting on a structure. In general, this uncertainty consists of contains both the aleatory uncertainty and the epistemic uncertainty. The former is virtually unavoidable and, the latter is caused by the limited ability and/or the imperfect information and knowledge. Thus, the load factor derived from the aleatory uncertainty is may be unchangedinvariable, while that derived from the epistemic uncertainty can be decreased when more reliable model or method is applied. This paper shows two examples of calculationng of the load factor when the aleatory uncertainty and the epistemic uncertainty can be separated.

Results

When aleatory uncertainty and epistemic uncertainty are separated, load factor is shown as followings;

a. Case when both load and resistance have log-normal distribution

When both the load and the resistance have log-normal distribution, the load factor γ is described as $\gamma = \gamma_r \cdot \kappa_u$. Where, γ_r is the load factor considering aleatory uncertainty only, κ_u is coefficient for by the epistemic uncertainty, and then, both are shown as below;

$$\gamma_r = \frac{\mu_S}{S_n} \exp\left\{\alpha_S \cdot \beta_T \cdot \zeta_{S,r} \left(1 - \frac{\zeta_{S,r}}{2}\right)\right\}, \quad \kappa_u = \exp\left[\alpha_S \cdot \beta_T \cdot \zeta_{S,u} \left\{a - \frac{\zeta_{S,u}}{2}\right\}\right],$$

respectively.

b. Case when the uncertainty is described by logic tree

When we evaluate the seismic hazard curves, we often use the logic tree analysis to cope with the for epistemic uncertainty. The logic tree analysis gives the fractile hazard. Then, we tried to calculate the load factors from the fractile hazard. The load factors considering the aleatory uncertainty only is close to the 50 percentile hazard, while the load factors considering both the aleatory uncertainty and the epistemic uncertainty is close to about 70 percentile hazard.

Conclusions

There are several problems in separating the epistemic uncertainty from the aleatory uncertainty. However, if eachboth uncertaintyies can be separated in load factor, the total amount of the load factor can be decreased by using precise model and/or advanced analysis method make load factor decrease and. It is very useful for seismic design of nuclear power plant facilities.

Research associated with the July 2007 NCO earthquake at the Kashiwazaki-Kariwa nuclear power plant (7-2064)

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The Tokyo Electric Power Company's (TEPCO) Kashiwazaki-Kariwa (KK) plant is the largest nuclear power plant in the world, with a total output of 8,212 MW. The KK plant is 16 kilometers away from the epicenter of the 2007 Magnitude 6.6 Niigataken-Chuetsu-Oki (NCO) offshore earthquake, which took place at 10:13 a.m. on July 16, 2007. Ground motion recordings at the basemat of the seven boiling water reactors at the site revealed that the S2 seismic design level had been significantly exceeded during the event.

Restarting a nuclear power plant following an earthquake that exceeds the plant seismic design basis entails a number of tasks to verify that damage has been identified and evaluated and that the plant is in a safe condition to resume operation. TEPCO has been undertaking these efforts continuously since the earthquake occurred. EPRI has conducted several research projects in support of TEPCO's efforts related to technical issues associated with the assessment of the plant seismic response, the performance of an independent peer review of the TEPCO seismic walkdown/evaluation program and a review of the seismic margin for selected KK plant critical components.

Based on EPRI's independent visual review performed, KK safety-related structures, systems and components (SSCs) performed very well in response to the NCO earthquake. No visible damage to the representative safety-related components reviewed could be detected. This was attributed, among other factors, to the rugged seismic design practice for the KK plants, particularly for the supports and anchorage.

Instances of damage were identified associated with some non-safety related (NSR) SSCs. While the results of these NSR failures and damage may not have had a critical safety related ramification, some key upgrades have been recommended which could prevent issues that occurred following the earthquake related to communications, fire protection, and available services.

The focus of this paper is to summarize the numerous technical efforts that are being conducted related to the seismic response and the seismic margins

associated with the Kashiwazaki nuclear plant's safety related components which survived the massive earthquake. The engineering community can learn significant lessons from this recent earthquake's effects to the world's largest nuclear plant, particularly in light of the fact that many nuclear plants (world-wide) are currently facing increases in the seismic hazard to the plant as well as being required to assess the seismic risks to the plant using either seismic margin methods or seismic probabilistic risk methods.

Seismic risk analysis utilizing the PGA and PGV simultaneously as ground motion measures (7-2389)

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Objective

In seismic probability safety analysis, ground motion intensity is usually expressed by a single index such as peak ground acceleration (PGA), spectral acceleration for a specified period, or peak ground velocity (PGV). Limiting the number of indices, however, gives greater uncertainty in the estimation of annual failure probability that is given by convolving seismic hazard curve and seismic fragility curve, since information except for ground motion intensity is missed. Authors proposed seismic hazard analysis utilizing PGA and PGV simultaneously as ground motion input measures (Fukushima et al. 2007). In this study, seismic fragility analysis utilizing PGA and PGV is conducted and advantage of vector-valued risk analysis is illustrated by comparing it with single-valued risk analysis.

Methodology and results

Seismic fragility of the structure is expressed by the seismic fragility plane that gives the failure probability under the given combination of PGA and PGV. In this study, the Monte-Carlo simulation was employed in order to obtain the conditional failure probability. Random variables were some characteristics that control the shape of the target response spectrum of input ground motion.

Using generated ground motions, model structure, 7-storey RC building, was analyzed so that the median and log-normal standard deviation of response for each combination of PGA and PGV were obtained. Moreover, failure probability at each PGA-PGV bin was calculated from the stochastic response and capacity. Relationship between PGA-PGV bin and failure probability was smoothed to express in the form of fragility plane.

From the fragility planes, it can be seen that both PGA and PGV are adequate as ground motion measure for limit state of slight damage, since the inclinations

of hazard plane in both direction are identical. On the contrary, for limit states of moderate damage, severe damage and collapse, PGV can be better measure than PGA. This is observed in inclination of fragility plane. The severer the limit state is, the stronger this tendency is.

Conclusions

In this paper, the seismic fragility analysis using PGA and PGV as ground motion measure is proposed. In constructing the procedure, the effects of response spectral shape on the ratio of PGA to PGV is examined so that the numerous input ground motions for Monte-Carlo simulation can be generated. By applying the method to model structure that is 7-story RC frame building, followings are obtained.

By expressing probability characteristic value s as a function of PGA and PGV, seismic fragility plane can be obtained.

Though both PGA and PGV are adequate as ground motion measure for slight damage, PGV is preferable for severer damage in which the natural period of structure gets larger due to inelastic behaviour.

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Seismic isolation of the IRIS NSSS building (7-2399)

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The safety-by-design™ approach adopted for the design of the International Reactor Innovative and Secure (IRIS) resulted in the elimination by design of some of the main accident scenarios classically applicable to Pressurized Water Reactors (PWR) and to the reduction of either consequences or frequency of the remaining classical at-power accidents initiators. As a result of such strategy the Core Damage Frequency (CDF) from at-power internal initiating events was reduced to the 10^{-8} /ry order of magnitude, thus elevating CDF from external events (seismic above all) to an even more significant contributor than what currently experienced in the existing PWR fleet.

The same safety-by-design™ approach was then exported from the design of the IRIS reactor and of its safety systems to the design of the IRIS Nuclear Steam Supply System (NSSS) building, with the goal of reducing the impact of seismically induced scenarios. The small footprint of the IRIS NSSS building, which includes all Engineered Safety Features (ESF), all the emergency heat sink and all the required support systems makes the idea of seismic isolation of the entire nuclear island a relatively easy and economically competitive solution. The seismically isolated IRIS NSSS building dramatically reduces the seismic excitation perceived by the reactor vessel, the containment structure and all the main IRIS ESF components, thus virtually eliminating the seismic-induced CDF. This solution is also contributing to the standardization of the IRIS plant, with a single design compatible with a variety of sites covering a wide spectrum of seismic hazards.

The conceptual IRIS seismic isolation system is herein presented, along with a selection of the preliminary seismic analyses confirming the drastic reduction of the seismic excitation to the IRIS NSSS building. Along with the adoption of the seismic isolation system, a more refined approach to the computation of the fragility analysis of the components is also being developed, in order to reduce the undue conservatism historically affecting seismic analysis. The new fragility analysis methodology will be particularly focused on the analysis of the isolators themselves, which will now be the limiting components in the evaluation of the overall seismic induced CDF.

Insights gained from the Beznau Seismic PSA (7-2405)

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Keywords: seismic PSA, seismic LERF, importance of seismic to risk

PSA studies performed for Light Water Reactors (LWRs) have shown dominating risk contributions from seismic events. There are several reasons for this finding. New Probabilistic Seismic Hazard Analyses (PSHAs) calculate a higher seismic hazard than perceived in the past. In addition, earthquakes represent a common-mode attack on all safety systems including the containment. On the other side, refined methods are currently available in the area of seismic PSA that enable assignment of higher seismic capacities to structures, systems and components.

In this paper, the most recent results and insights gained from the Beznau Seismic Level 2 PSA study are shown. Beznau nuclear power plant is the oldest operating pressurized water reactor (PWR) worldwide. The plant was backfitted extensively during the last two decades by the construction of additional and seismically more robust safety systems.

The paper first presents the most important characteristics and methods applied in the actual Beznau Seismic PSA study. The study represents a Level 2 study that fully considers containment integrity and that quantifies the Large Early Release Frequency (LERF) for seismic events. The paper also shows the important risk contributors with respect to the Core Damage Frequency (CDF) as well as with respect to LERF. One main conclusion of the study is that the seismic capacity of the containment represents a key role with respect to the seismic Large Early Release Frequency LERF. In addition, the calculated results indicate that a seismic design of the reactor building of Advanced Light Water Reactors (ALWRs) of 0.5 g HCLPF may be too low even in areas of low to moderate seismicity.

Finally, conclusions are drawn with respect to the risk contribution from seismic events.

Thermal-hydraulic analysis for accidents in OPR1000 and evaluation of uncertainty for PSA (7-2478)

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Probabilistic Safety Assessment (PSA) is a mathematical tool to evaluate numerical estimates of risk for nuclear power plants (NPPs) and can be used to calculate the probability of damage to the core as a result of sequences of accidents identified. After the first comprehensive application of the method, reactor safety study, WASH-1400, PSA has become a standard tool in safety evaluation of not only NPPs but also industrial installation. When PSA is performed, thermal hydraulic analysis is necessary to obtain the basic data, from which system success criteria for construction of event tree and the allowable time for human reliability analysis are determined. Up to now, this analysis has been undertaken with various system codes such as RELAP, RETRAN, MELCOR and MAAP4. However, it is well known that deterministic assumptions and input values have been often applied to the analysis even if most of them are best-estimate codes. To acquire more realistic result, the analysis with nominal value and realistic assumptions needs to be carried out and the uncertainties from the result of the analysis are needed to be essentially quantified.

The aim of the present study is to develop a best-estimate thermal hydraulic analysis methodology applicable to PSA as well as to quantify uncertainty. In the present study, Optimized Power Reactor 1000 (OPR1000), which is the standard nuclear power plant in Korea, was selected as the objective power plant. MARS code was chosen, which is best-estimate code and has been developed at Korea Atomic Energy Research Institute (KAERI) by consolidating and restructuring the RELAP5/MOD3.2 code and COBRA-TF code. Korea Hydro and Nuclear Power (KHNP) Cooperation has already performed PSA of OPR1000 and made the accident sequence table in which the accidents have ranked along to the frequency of occurrence. Thus the accidents in the table as mentioned are required to be analyzed with the input made based on the realistic assumptions. Moreover the uncertainties from the results of analyses should be quantified. To do what are aforesaid, Phenomena Identification and Ranking Table (PIRT) of each accident is necessary since we cannot consider all parameter for reason of calculation time and cost every analysis. PIRT for each accident has been already made by the group of expert and is desirable to be

reconstructed if needed. With its own range and distribution, each candidate parameter in the PIRT was simulated in MARS code. On this occasion, it was assumed that the range had 95% confidential interval and acceptable assumption is applied only when the information about the distribution of parameter does not exist. A number of calculations by MARS code were performed repetitively with varying the input value of certain parameter within its uncertainty range. The peak cladding temperatures (PCTs) from the calculation results, with which it was determined if the core was damaged, were used to construct the response surface and quantify the uncertainties. Conventionally, there are several methods to quantify the uncertainties; Monte Carlo Method (MCM), Latin Hypercube Sampling Method (LHSM), Response Surface Method (RSM), etc. In the present, MCMs are the most widely used means for uncertainty analysis. However, after full consideration of time and cost for the present study, RSM is most suitable to perform the study since there are more than ten parameters for each accident and it takes too much time to use MCM to carry out uncertainty analysis. The regression equation for PCT was obtained by RSM and the randomly sampled values from the range of each parameter were substituted for the equation. As a result, the distribution of PCT of each accident was gained and it was used to assess the core damage frequency (CDF) from the PSA which is already performed.

Estimation of leak and break frequencies for probabilistic safety analyses of piping systems (7-2529)

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In the framework of the BfS-project SR 2608 the details on the estimation of leak and break frequencies in piping systems contained in the report on methods and data for the Probabilistic Safety Analysis (PSA) has been updated and extended by GRS. Based on the hitherto existing regulation new methodical aspects were introduced. The statistical method based on the evaluation of the German operational experience for piping systems with different diameters was updated by the inclusion of structure reliability models based on fracture mechanics calculation procedures. The example of application of the statistical estimation method for leak and break frequencies of piping systems with a diameter of 50 mm out of the volume control system of a German pressurized water reactor contained in the PSA data volume was updated. To this end the operational experience considered so far (up to 1995) was extended with respect to cracks, leaks and breaks in the volume control system of German PWR up to the year 2006 and the operating time included (191 years of operation) was accordingly extended to 341 years. Under these conditions new calculations of leak and break frequencies have been performed and the results have been compared with the previous values.

A temperature characteristic diagnosis algorithm of the abnormal signal simulation analysis module by using probabilistic techniques (7-2546)

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The circuit simulation analysis and diagnosis methods are used to assess instruments in detail when they give apparently abnormal readings. The simulations can be useful for investigating what the signal and circuit characteristics would look like for a variety of symptoms that can result from very high temperature environment conditions. Instrument circuits are first modeled and tested using specific circuit simulation program. Then degraded temperature conditions are introduced by modifying the instrument circuit models. The response characteristics of the simulated instrument circuit to degraded temperature conditions provide the basis for diagnostic information. The checklists list the steps necessary to obtain information from instrument loops which may be degraded, but for which the detector should still be providing valid signals. The role of the circuit simulation is to determine the diagnostic steps that can be taken to evaluation if the temperature condition is real or a result of instrument loop degradation. In this paper, a new simulator, ASSA module, through an analysis of the important circuits modeling under temperature accident conditions has been designed. It has a special function that means probabilistic techniques for a diagnosis algorithm of circuit-component degradations including a temperature characteristic data of the basis for diagnostic method. We present probabilistic techniques that make synergy use of available process information for diagnosis and detection of component fault in a circuit-component system. We begin by describing the motivation for using probabilistic techniques for systems diagnostics and then define probabilistic expressions that embody the diagnostics knowledge of interest. We show that a combination algorithm of a Bayesian expression with the solution to the Chapman-Kolmogoroff equation contains the diagnostic information of interest while explicitly making use of available process information obtaining the probability density function corresponding to feasible circuit-component transitions by an adaptive Kalman filtering.

Fragility functions for seismic performance assessment of safety-related reinforced concrete nuclear structures (7-2557)

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Squat (shear-critical) reinforced concrete walls are widely used in nuclear power plants and other safety-related nuclear structures to provide resistance to extreme earthquake loadings. Performance assessment of such structures utilize fragility functions that relate the probability of exceeding one or more damage thresholds to either a ground-motion or response (demand) parameter such as peak ground acceleration, spectral acceleration at a selected period, story drift or component plastic deformation.

Fragility functions are developed for squat reinforced concrete walls with aspect ratio (height-to-length or (h_w/l_w) of 2 or less) by review and statistical evaluation of experimental data in the literature. The experimental data includes tests of three cross-section types: rectangular, barbell and flanged. Per modern practice, a demand parameter is used to construct the curves. Experimental damage data is characterized using damage states and methods of repairs. Documents that provide guidelines for repair of reinforced concrete walls, observations from experimental programs, previous research on retrofit of squat walls and expert opinion are used to identify the most appropriate damage states and their corresponding methods of repair. Damage states are characterized generally by direct indicators of damage such as initiation of cracking, maximum concrete crack width, extent of concrete crushing, sliding shear displacement, and reinforcement yielding, buckling, and fracture. Each of these damage states is linked with one of four methods of repair, namely, cosmetic repair, epoxy injection, partial wall replacement, and wall replacement.

Different families of fragility functions are required for each cross-section type but the data do not support the development of fragility surfaces to accommodate axial force, rebar ratio and aspect ratio as input variables. Story drift is used as the demand parameter. Scopes of repair are provided elsewhere.

Experience from a seismic probabilistic safety assessment of a German PWR (7-2566)

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Introduction

Nuclear power plants in Germany have been subject to several deterministic seismic safety assessments. After each safety assessment, components of the plant were upgraded. Additional deterministic assessments will not essentially improve the seismic resistance of the plant. As a new demand from authority, seismic probabilistic safety assessments (seismic PSA) have to be conducted for German nuclear power plants [BfS-37/05].

In this contribution experience from a seismic PSA of one of the latest erected German PWR will be presented. Its operation started in the year 1984. A full scale probabilistic analysis by evaluating safety products for structures, plant systems and components was performed in order to estimate the core damage probability caused by earthquakes. Furthermore, limitations and benefits of a seismic PSA will be discussed.

Method

Stress calculations use the horizontal peak ground acceleration as a measure for the strength of an earthquake. The objective is to estimate the horizontal peak ground motion acceleration A for which the seismic response of a component exceeds the component capacity resulting in its failure. By assumption, A is a log-normally distributed random variable.

To perform a seismic safety analysis of a nuclear power plant, safety systems, components and structures needed for plant shutdown and for long term heat removal must be examined. For components which are accessible for determining a seismic safety product, fragility curves were evaluated. A fragility curve provides a conditional frequency of failure in dependence from horizontal peak ground motion acceleration at the plant site. Fragility curves were determined for pipes, core internals, buildings, switchgears, heat exchanger supports, and pump supports. For each component analyzed, fragility curves for two non-exceedance probability levels Q were evaluated: The fragility curve for $Q = 0.5$ provides a frequency of failure determined without any conservatism.

The fragility curve for $Q = 0.95$ delivers a failure frequency with 95% probability of non-exceedance. Figure 1 shows fragility curves for a low pressure pipe with DN 600.

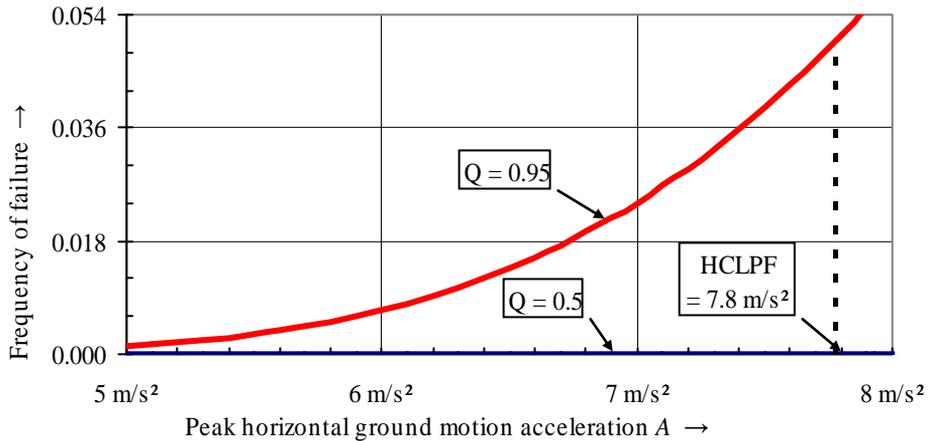


Figure 1. Fragility curves for $Q = 0.5$ and $Q = 0.95$, low pressure pipe with DN 600.

At non-exceedance probability level $Q = 0.5$, the frequency of failure by an earthquake with horizontal peak ground acceleration $A = 7.8 \text{ m/s}^2$ is less than 0.01% for the pipe analyzed. At probability level $Q = 0.95$, the frequency of non-failure by this earthquake acceleration is $(1 - 0.05) = 0.95$. Pipes are very earthquake resistant, as extensive studies in the USA demonstrated too [NUREG-4334]. The HCLPF peak ground acceleration is generally considered to be approximately 95% confidence of less than 5% frequency of failure.

Not for all components a seismic safety product can be determined by analytical tools. When a component is not accessible to an analytical calculation of its strength, seismic tests or experience from earthquakes in fossil power plants and in industrial plants can be applied. Moreover, seismic ruggedness of a component can be deduced from its use in a vibrational environment. For assessing electrical equipment and mechanical active components, generic HCLPF accelerations or GERS (Generic Equipment Ruggedness Spectra) are used. Generic HCLPF accelerations for different components are given for instance in [NUREG-4334]. A GERS expresses a level of seismic acceleration to have a sufficient ruggedness to perform as required [EPRI-5223].

For the seismic PSA an updated seismic hazard curve was used. In addition a site specific ground response spectrum was used and compared to the original ground response spectrum from the design phase of the plant.

Result

A calculated annual probability of core damage of 10^{-7} emphasizes a good seismic plant design. This result includes conservative assumptions for limiting the extent of the analysis. Even at low earthquake intensities a loss of offsite power was assumed. Additionally an unavailability of non-safety related systems was postulated. At a strong earthquake building failure is dominant.

Limitations and benefits of seismic probabilistic safety assessment

Safety factors of buildings and structures turned out to be very difficult to derive from the calculations at the plant design phase. The fragility curves for buildings and structures may include some conservatism.

Small bore pipes are not periodically examined by x-ray testing. During an earthquake there is a risk of pipe breaking starting from an undetected crack. An earthquake can initiate a small loss of coolant.

Electrical cable connections can oxidize by time. Electrical insulation can brittle by time. There is a small database for seismic resistivity of aged electrical components. The probability of failure for aged electrical components was estimated by using generic Hclpf accelerations. Aged electrical components also contribute to the seismic risk.

A seismic PSA must consider earthquake intensities lower and higher than the design base earthquake. A deterministic seismic risk assessment tends to review only the code compliance of components. The seismic PSA must look beyond code requirements.

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8. Issues Related to Operations, Inspection and Maintenance

Life cycle management including inspection, online monitoring, maintenance, repair and replacement. Life prediction, reliability, availability, deterministic and probabilistic integrity risk assessment. Utility vs. regulatory perspectives on periodic safety reviews, license renewal, safety and economic issues. Non-destructive examination (NDE) methods and techniques for structural integrity assessments, reliability and validation of NDE methodologies, and NDE training. Lessons learned.

Fretting wear resistance nuclear fuel design & operating experience (8-1619)

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The Fretting Wear is emerging issue of PWR Nuclear Fuels in U.S., Europe and Worldwide. Many nuclear stations still suffered from significant fuel failure caused by the grid-to-rod fretting wear failures [1]. The grid-to-rod induced fuel rod fretting failures occurred at various PWR fuel assembly designs. These fretting wear may be caused by external and internal vibration sources. The extent of the wear volume and wear depth are depend on grid-to-rod contact configurations.

In this paper it will be proposed the fretting wear resistance nuclear fuel design with wide contact area as Fig 1. to reduced fretting wear failure even if same amount of flow vibration sources. And this paper will be presented various fretting wear test results for suggested wide contact area fuel design to contrast narrow contact area fuel design it was currently used in PWR in worldwide. Also this paper present test method and the test apparatus of grid to rod fretting wear as shown Fig. 2. And will be presented the analysis and test results of wide contact vs. narrow contact geometry wear depth against load as shown Fig. 3. And will be presented the wear rate of wide vs. narrow design as shown Fig. 4. We irradiated 4 Lead Test Assembly (LTA) in commercial PWR reactor to verify in reactor fuel performance. So in this paper it will be presented the results of pool side examination (PSE) and post irradiation examination (PIE).

According to irradiation results of PWR coolant activity analysis the proposed wide contact fuel design experienced very good results of fretting wear performance.

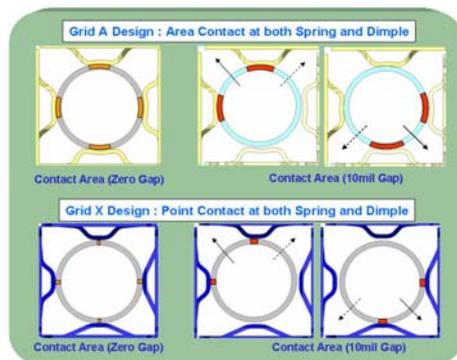


Figure 1. Grid design concept with Wide contact vs. Narrow contact.

8. Issues Related to Operations, Inspection and Maintenance

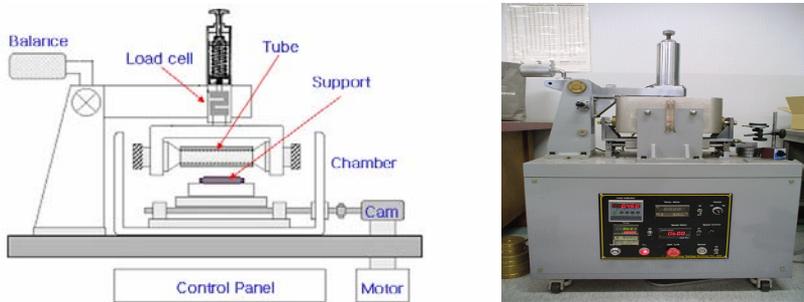


Figure 2. Grid-to-rod fretting wear tester.

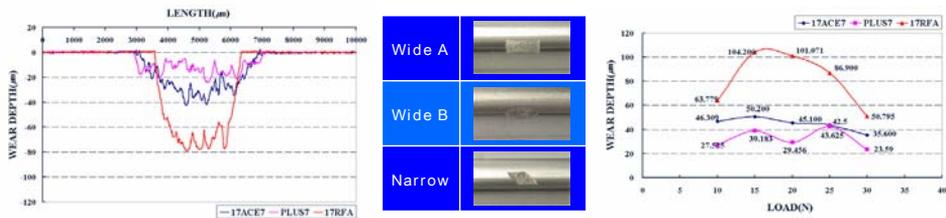


Figure 3. Grid-to-rod fretting wear test results.

Wear Rate $K(\times 10^{-13}Pa^{-1})$	Material A	Material B
Wide B	1.721	2.344
Wide A	1.868	1.896
Narrow A	6.437	6.264

Figure 4. Wear rate of Wide vs. Narrow design.

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Aging problems and residual life time evaluation of the WWER-1000 MW containment shell structure (8-1622)

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Introduction

There are two units of type WWER-1000 MW which are in operation more than 20 years in NPP Kozloduy, Bulgaria. Some specific aging problems appeared during that time and different technical solutions are applied. It is useful to share this experience with the engineering community and to discuss the proper measures for the future exploitation.

Aim of the work

The main goal of this paper is to generalize the specific aging problems and the residual life time evaluation of the WWER-1000 MW containment shell reinforced concrete structures of Unit 5 and 6 in Kozloduy NPP, Bulgaria. The main problems of the containment shell structure are connected with the original prestressing system. After a comprehensive analysis of the behaviour of the system for a long time a new prestressing system is proposed and successfully implemented.

Essential results

First of all the different factors and degradation mechanisms are investigated following the procedures given in [1]. Several types of in situ and laboratory tests are performed for specific elements of the civil structures. Based on these results and on the available technical information an evaluation for the condition of the reinforced concrete structure is done. The containment shell structure is a prestressed structure and the condition of the prestressing system is very important. Some problems with the original system and several new technical solutions for the new system are discussed. The residual life time evaluation is done on the base of the complex analysis of the all available information from the construction time until now.

Conclusions

Several techniques for investigation of the concrete and steel are recommended. Prescriptions are given for the periodical inspections of the important parts and details of the containment shell structures.

Some specific issues are considered for the instrumental monitoring and the control of the aging mechanisms. Special attention is paid to the monitoring of the structure – geodetic monitoring, monitoring of the stress and strain state of the concrete and the monitoring of the prestressing system. A concept is recommended for the future development and modernization of the monitoring systems. A proper measures are suggested for reducing the aging effects which are the basis of the maintenance program for these structures.

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Monitoring relative humidity and temperature for life-time assessment of sandwich-type concrete structures (8-1647)

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Introduction

Deterioration of concrete is one of the basic questions in the life time management of buildings and structures in nuclear power plants. Two of the most important factors in building deterioration subjected to outdoor conditions are moisture and temperature. Moisture is a major factor in physical deterioration processes that are typically caused by restrained moisture movements and freezing or they can be connected to chemical or biological attacks. Nowadays a severe climate may be a reason behind a chemical load on building. In addition, moisture will increase the heat flow through a structure and thus increase the consumption of heating energy. The continuous monitoring of temperature and relative humidity provides not only important information for life-time management of sandwich-type concrete structures but also introduces the possibilities of systematic condition monitoring in developing the predictive maintenance of power plant facilities.

Methodology

A new thermal and moisture monitoring method was developed at the laboratory of structural engineering and building physics. The new method does not require large investment of time and facilities for collecting service life information. The method was tested in repaired buildings. For that purpose, a network based monitoring system was established by using modern communication techniques to gather large amounts of data with little effort. The monitoring network system was found to be useful for assessing the repaired building façade performance and giving knowledge about the physical functioning of building envelopes. The system developed and tested is easily adapted to various types of structures where the parameters traced may also vary.

The development of the thermal and moisture monitoring method included laboratory work and field measurements. The laboratory work focused on designing and testing the RHT-monitoring network system (Figure 1) including the calibration of the temperature and relative humidity devices. The field measurement was carried out to monitor the temperature and relative humidity of three facades that were repaired with different methods. The thermal and moisture condition was monitored at regular intervals of 15 minutes for more than two years.

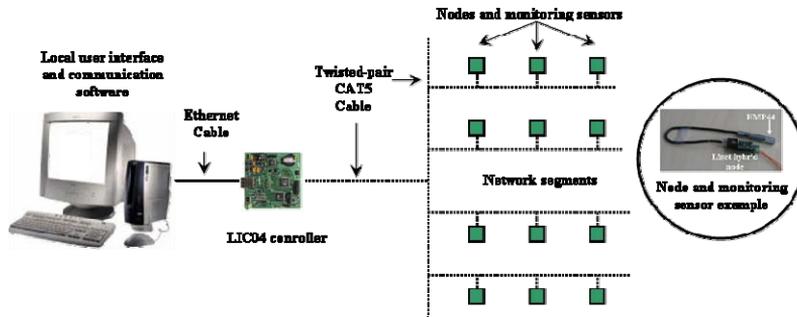


Figure 1. Schematic diagram of the RHT monitoring network system.

Essential results

The results of relative humidity and temperature monitoring provide an opportunity to have a close look at hygrothermal performance of the wall assembly and assessing the performance of the repaired facades.

The long-term moisture response indicators RHTT1 and RHTT2 indices were calculated [3]. The RHTT1 index is to examine the potential for biological growth, RHTT2 index to examine the potential for corrosion.

The freezing thawing index (FT) for examining the potential for frost damage is defined as the number of cycles when temperatures oscillate between the freezing and thawing point for the facade components that are almost at the critical moisture saturation level.

Table 1. Example of the RHTT1, the RHTT2, and the freezing thawing (FT) indices for the rendering coat and the original outer concrete panel of the repaired façade.

Repaired facade			June 2005 to June 2006			June 2006 to February 2007		
			RHTT1	RHTT2	FT	RHTT1	RHTT2	FT
Rendering coat	North-east	First floor	742	1386	116	431	1049	68
		Sixth floor	448	883	78	340	856	77
	South-west	First floor	266	515	97	203	577	63
		Sixth floor	422	797	101	225	648	86
Original outer concrete panel	North-east	First floor	1438	1833	0	0	0	0
		Sixth floor	264	332	0	0	0	0
	South-west	First floor	239	302	0	0	0	0
		Sixth floor	1165	1475	0	0	0	0

According to Mukhopadhyaya (2005), the safe values for RHTT and FT indices for various building materials are not available yet, but they are useful to indicate the deterioration potential in different repaired materials and methods.

Conclusion

The RHT monitoring network system is very useful for gathering a large amount of data about the thermal and moisture performance of repaired facades, which provides a much better understanding of how the environment and the building interact and can complement visual inspections. By measuring the temperature and relative humidity of building components systematically, we can determine the potential for deterioration, wetting and drying patterns in building components, and changes in moisture content. Documented monitoring and analysis are especially important when applied to maintenance and repairing in safety classified structures such as nuclear power plants. These create a basis to maintain also the reliability of civil engineering parts of the plants by predictive maintenance and corrective actions that are carried out in a timely manner.

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German nuclear power plants utility ageing management – long term fatigue evaluation of safety relevant components (8-1652)

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Integral Plant Life Management comprises the technological and administrative measurements of relevant systems and components during the lifetime of a Nuclear Power Plant (NPP) to safeguard the safety requirements.

In this context aspects on Ageing Management (AM) cover all safety relevant Systems Structures and Components (SSCs). In more detail, within the German understanding a clear differentiation between Plant Life Management (PLIM) and Ageing Management is done. PLIM refers to the entire NPP (primarily under utility responsibility) in order to guarantee safe operation and additionally to increase plant availability while minimizing unplanned outages and planned downtimes. In contrast to this, AM refers to safety relevant SSCs only and is done under the supervision of the responsible safety authority. Within AM operational ageing mechanisms like physical and material ageing phenomena are covered due to the fact that a sufficient knowledge of stresses and strains are indispensable for a precise prediction of the lifetime and subsequently for safe and reliable operation. Suitable and applicable assessment tools like comprehensive temperature surveillance measures are indispensable.

Based on the demand of a systematic procedure, an explicit classification of safety and availability significant SSCs into three different groups is feasible. The differentiation of mechanical, I&C and building components is based on diverse safety requirements.

Passive components of the reactor coolant pressure boundary account for highest safety requirements. Especially for these components, thermal fatigue is one significant long-term degradation mechanism, due to the fact that thermal loadings lead to most fatigue relevant stresses and strains in piping systems.

Therefore temperature-measuring facilities were applied to various positions on primary circuit piping where most significant thermal loads were expected. During recent years of operation these measuring positions are being reviewed and adapted, leading to comprehensive, representative and global information of existing thermal loadings.

Based on this extensive amount of measured data a detailed analysis of all relevant thermal-loading events is feasible.

Measured values are evaluated on a per-component-basis with regard to relevant temperature loads (temperature alterations, thermal gradients, stratifications) and their corresponding number of occurrences. This way bookkeeping of load cycles is performed to determine actual fatigue usage factors. In load cycle counting lists for each registered event a partial fatigue usage factor is stored. To avoid too much conservatism, those load cycle counting lists are also prepared on a per-component-basis by analyzing the temperature and stress distributions for reference transients. Thermal boundary conditions (e.g. heat transfer coefficients) of reference transients are determined, based on measured temperatures respecting thermal inertia of the measuring equipment. This procedure guarantees both adequate realism and necessary conservatism.

At NPP construction time predicted fatigue usage factors were defined based on calculations of specified transients. So various uncertainties, resulting due to the fact that precise operation conditions were not known in detail at plant design time, are part of this predicted end-of-life fatigue usage factors. In comparison to detailed analysis being based on real measured values, it is obvious that predictions adapted on specified loads will lead to more conservative results mostly. But nevertheless, in a small number of cases the registered events show new effects, not accounted for at design state, so that analyzing these data leads to substantial progress in component integrity assessment knowledge. Additionally these technical experiences on measured thermal loads will give a direct input in optimizing operation mode of relevant SSCs and subsequently minimizing fatigue relevant stresses.

So the goal of all applied surveillance, measuring, and analysing measures is a detailed, reliable and exact knowledge of the system's status. Herein the surveillance of thermal loads as one important root cause for ageing relevant material degradation mechanisms is the basis for every further action. A detailed knowledge of the loading history based on measured values will lead to the reduction of conservative results. Additionally the assessment of new and possibly unspecified loading events including a statement of the influence to a SSC specific fatigue usage factor is feasible.

This publication will give a brief introduction to the basics of Ageing Management (AM) in German NPPs while clarifying the delineation to Plant Life Management (PLIM). Surveillance measures on primary circuit piping components are essential for comprehensive analysis of thermal fatigue being one significant long-term degradation mechanism. This component specific data is the basis for further detailed analysis in order to determine a data-based actual fatigue usage factor and finally to deduct required measures proactively.

Considerations related to long-term operation for CANDU 6 NPP (8-1663)

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Long term operation (LTO) of a nuclear power plant may be defined as operation beyond an established timeframe set forth by, for example, licence term, design, standards, licence, and/or regulations, which has been justified by safety assessment, with consideration given to life limiting processes and features of SSCs. Managing physical or material ageing of SSCs important to safety aims to maintain their design safety margins above SSC specific requirements (see Fig. 1) and thus to minimize risk to public health, environment and safety, [1].

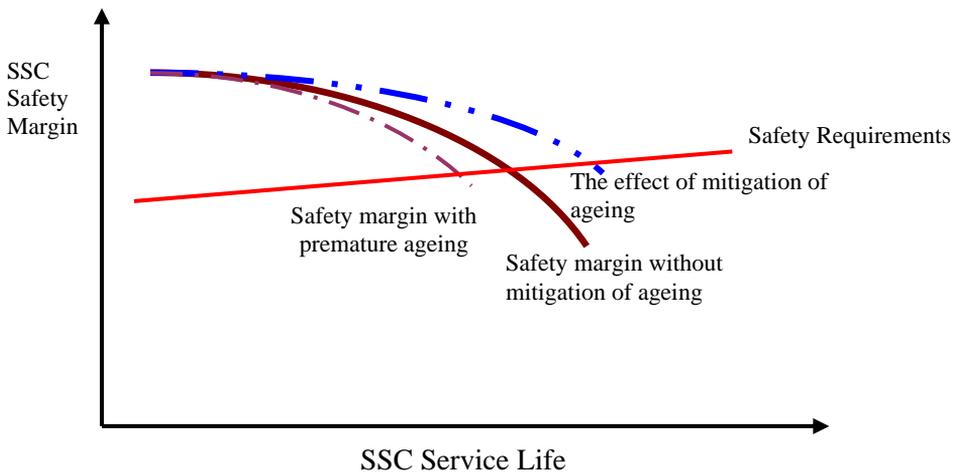


Figure 1. Relation between SSC safety margin and service life.

Approximately 25% of the NPPs in the world have been operated for more than 30 years, and about 70% for more than 20 years. The first commercial CANDU® unit Douglas Point was put into commercial operation in 1968 (40 years ago) and the first CANDU 6 units operate for 25 years. There are 11 CANDU 6 units in operation. As in Canada, China and Korea the LTO Program is already implemented in Romania where Cernavoda NPP Unit 1 was put into commercial operation on the 2nd of December 1996 and the Cernavoda NPP Unit 2 was put into commercial operation on the 5th of October 2007. The following actions have been performed [2]:

- ◆ Fuel channel inspections using Ultrasound (US) & Eddy Current (EC) detectors were performed for Unit 1 into 1999, 2003 and 2008.
- ◆ For Steam Generators the periodic inspection program was started in 1998. The tubes (EC), the nozzles (US), internal supports and the walls (US) were inspected periodically.
- ◆ Feeder inspection program was started in 2003 for wall thinning measurement; elbows extrados cracks identification and welds monitoring.
- ◆ PSR was started in 2008 to be performed for Cernavoda NPP Unit 1, after 10 years of commercial operation.

The LTO program, proposed to be applied at Cernavoda NPP, is supported both by the experience of CANDU 6 owners and by the results of research conducted within INR Pitesti. Institute for Nuclear Research Pitesti (INR) is the main responsible RTD organization for development of national technical support of nuclear power in Romania. The activity of the Institute is oriented with priority towards applied and engineering research within RTD programs, connected to present and future specific issues of NPP, especially those using CANDU 6 type reactors.

Figure 2 illustrates a number of mechanisms which have been experienced in CANDU 6 reactor systems as they are currently perceived in terms of ageing predictability.

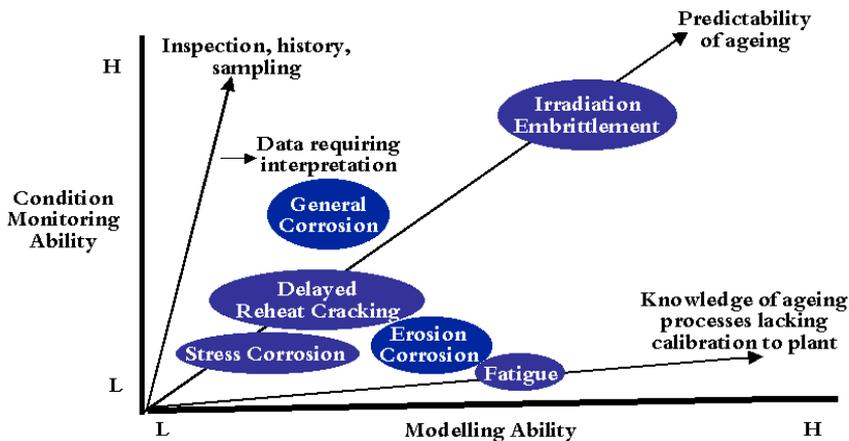


Figure 2. Illustration of modelling ability and condition monitoring ability for typical ageing mechanisms.

Such estimations indicate where R&D investment is required for predictability to be improved. It is clear that significant R&D expenditure is needed if modelling and condition monitoring technology is to achieve the required level for an effective ageing management based on predictability. This should lead to

an increase in the use of mitigation measures that, if applied early enough, can prevent crack initiation and/ or further deterioration. Over the past 7 years, INR Pitesti has been working on R&D Programs to support a comprehensive and integrated Cernavoda NPP Life Management Program, [3]. A comprehensive R&D support to LTO program applicable to CANDU 6 NPP has been proposed.

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Service life management system of concrete structures in nuclear power plants (8-1685)

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The objective of the project is to develop a predictive service life management system (SLMS) for concrete structures in nuclear power plants. The management system includes prediction of service life, guarding of safety and serviceability limits, prediction of maintenance and repair actions, calculation of life cycle costs and environmental impacts, evaluation of risks and inspection of structures. By the SLMS the safety, accepted structural performance and uninterrupted service of concrete structures are ensured during the planned service life of a nuclear power plant.

The SLMS is connected with the in-service inspection system of NPP. Thus the observed condition of structures is brought to the process of service life prediction and decision making on maintenance and repair actions. The system is provided also with qualitative and quantitative risk analyses, financial and ecological life cycle analyses and detailed structural condition analyses.

The methodological ground of the service life management system was developed during the EC FP5 project LIFECON (2001–2003). The structures are divided into smaller structural parts which can be treated as homogenous with respect to materials, structural features and environmental conditions. These structural parts are called “modules” and they serve as basic structural units in the analysis and planning processes of the system. The structural databases which serve as initial data sources in the calculation processes are consistent with the modular breakdown of structures. A specific nuclear feature in partitioning is that safety classification of structural parts and their impact on PSA analyses can be considered, which improves the possibilities for concentrating plant life management actions to structural parts according to their hierarchical order of importance.

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The core of the management system consists of a combined condition, cost and environmental impact analysis. The condition analysis is produced based on degradation models, predefined limit states of condition and the Markov Chain method. The condition analysis is capable of predicting the probability of the modules to be at any of the condition states at any year during the treated design period. The analysis contains also an automatic condition guarding system which triggers maintenance, repair and rehabilitation (MR&R) actions whenever the predefined limit state of condition is exceeded with a maximum allowable probability.

The actual service life management system is supplemented by structural analyses and risk analyses as not all degradation can be predicted by simple degradation models. The structural analyses are used to evaluate the cracking behavior of concrete structures in ultimate and serviceability limit state. The risk analyses are conducted for corrosion of the steel liner and prestressing tendons in the containment building.

Statistical assessment method for the optimization of the inspection need for nuclear steam generators based on existing inspection data (8-1731)

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The cost of steam generator inspections in nuclear power plants is high. Therefore, it is important to optimize the inspections, both with respect to timing as well as coverage. Presently the inspection requirements vary from plant to plant, without any clear physical or statistical basis. Because each steam generator is an individual, it is not possible to derive common requirements for all. However, based on a systematic statistical analysis of existing inspection data for each steam generator it is possible to optimize the inspection need, without reducing the reliability of the operation of the steam generator tubes.

A new quantitative assessment methodology for the accumulation of flaws due to stochastic causes like fretting has been developed for cases where limited inspection data is available. Additionally, a new quantitative assessment methodology for the accumulation of environment related flaws, caused e.g. by corrosion in steam generator tubes, has been developed. The method that combines deterministic information regarding flaw initiation and growth with stochastic elements connected to environmental aspects requires only knowledge of the experimental flaw accumulation history. The method, combining both types of flaw types, provides a complete description of the flaw accumulation and there are several possible uses of the method. The method can be used to evaluate the total life expectancy of the steam generator and simple statistically defined plugging criteria can be established based on flaw behavior. This way the inspection interval and inspection coverage can be optimized with respect to allowable flaws and the method can recognize flaw type subsets requiring more frequent inspection

8. Issues Related to Operations, Inspection and Maintenance

intervals. The method can also be used to develop statistically realistic safety factors accounting for uncertainties in inspection flaw sizing and detection.

Examples of the application of the assessment method are provided for real steam generators, both in the case of stochastic damage as well as environment related flaws.

Development of RI-ISI at STUK (8-1794)

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Introduction/background

Several risk-informed applications have been introduced since the beginning of 1990's in Finland. At the end of 1990's, STUK completed a pilot project that included, for example, testing of RI-ISI methods for Loviisa 1&2 and Olkiluoto 1&2. The licensees provided qualified input data for the pilot study.

Aim of the work

Together with international development of risk-informed methods, regulatory oversight was gradually targeted towards systems, components and structures posing significant risk. Pilot projects gave experience and insight for development of risk-informed regulatory requirements.

Essential results

In 2003, STUK issued updates for regulatory guides YVL 2.8 and 3.8, which set forth regulatory requirements on risk-informed in-service inspection (RI-ISI) applications. RI-ISI scope covers all plant systems and safety classes, including non-nuclear. The scope includes bellows, seals, hoses, small instrument pipes, etc.

STUK requires that RI-ISI applications include expert panel which turned out valuable already in pilot project.

In 2007, Loviisa finished full-scope RI-ISI study, which included all systems in the plant. The risk-informed in-service inspection program was introduced for Loviisa 1, in 2008. Significant differences were implemented compared to old inspection program. TVO will finish RI-ISI project for Olkiluoto 1, in 2009.

Olkiluoto 3 will be the first NPP unit implementing risk-informed pre-service inspection program. No detailed international standards are yet available for pre-service inspection of NPPs.

Methods used in Finland for RI-ISI applications follow ASME Code, Section XI Appendix R. Also viewpoints of ENIQ Report nr. 23, "European Framework Document for Risk-informed In-service Inspection" are included.

According to STUK experience, risk related to change of operating state may be significant and has to be taken into account if pipe rupture causes shutdown or if repair of the rupture requires shutdown. In these cases the plant will shut

down with impaired systems. The full spectrum of operating states has to be taken into account. In Loviisa unit 1, there are pipe segments where the significant risk comes from rupture during shutdown states.

Summary/conclusions

In Finland, oversight and operation of nuclear facilities is becoming increasingly risk-informed. As part of this, plant-wide risk-informed inspection programs are used in Finnish nuclear power plants. Full-scope RI-ISI programs require that the underlying PRA covers all operating states and initiating events and includes the risks of possible changes between operating states with impaired systems.

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Monitoring pipe thinning using two accelerometers (8-1801)

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There are many pipes in secondary cooling system of nuclear power plants and coal-fired power plants. In these pipes, high pressure and high temperature fluids are moving at very high velocity, and it causes pipe thinning through flow accelerated corrosion (FAC). In 2004, it was reported that because of pipe thinning, there was a leakage of coolants in Mihama nuclear power plant in Japan, and several men were killed. As we can see in this case, pipe thinning in power plants not only does financial and time damage, but also causes damage of people's lives. So it is very important to monitor and supervise pipe thinning. As of now, the most widely used monitoring method uses ultrasonic waves to estimate the thickness of pipe wall. This method measures the thickness of dozens of check points in pipes by ultrasonic-type sensor one by one, and estimates the degree of pipe thinning. So, as the number of check points in pipes increases, it requires more and more time and manpower to install the sensors. Furthermore, if the pipes are surrounded by heat insulator, it has to be removed before the sensor is installed, and this causes additional expenses. And in case the number of pipes under monitoring be too high, it is impossible to judge the degree of pipe thinning quickly, so this method's applicability falters. Therefore a method to inspect a large area of piping systems quickly and accurately is needed. In this paper, we proposed the method for monitoring pipe thinning by using two accelerometers. Our basic idea come from that a group velocity of impact wave is different as wall thickness. If we install two vibration sensors outside of the pipes and measures traveling velocity of flexural waves regularly, we can estimate and monitor the degree of pipe thinning quickly. To obtain the group velocity, time-frequency analysis is used. This is because an arrival time difference can be measured easily in time-frequency domain rather than time domain. In order to test the method we experimented with pipes, and get the result that group velocity varies according to the degree of pipe thinning. It verified this method can be used to monitor the pipe thinning.

Activities of OECD/NEA in the fields of integrity and ageing of components and structures (8-1804)

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The Integrity and Ageing of Components and Structures Working Group (IAGE) of the Organisation for Economic Cooperation and Development (OECD)/ Nuclear Energy Agency (NEA) was established, under the Committee on the Safety of Nuclear Installations (CSNI), to advance the current understanding of those aspects relevant to ensuring the integrity of structures, systems and components, to provide for guidance in choosing the optimal ways of dealing with challenges to the integrity of operating as well as new nuclear power plants, and to make use of an integrated approach to design, safety and plant life management.

The working group operates through three subgroups dealing with a) integrity and ageing of metal structures and components, b) integrity and ageing of concrete structures and c) seismic behaviour of components and structures.

The group operates through annual plenary meetings and technical workshops and by issuing state-of-the-art reports and topical opinion papers. Among other items, the recent and planned activities of the group include the following:

- updating of the IAGE integrated plan
- conducting a meeting of specialists on seismic hazard assessment in April, 2008 in Lyon, France, with planned publication of the proceedings
- conducting a specialist meeting on risk informed piping integrity management in June, 2008, Madrid, Spain, with the planned publication of the proceedings
- publishing reports on: 1) summarising the main findings and conclusions of a series of OECD/NEA workshops and extracting the seismic information most relevant to current nuclear practices; 2) A Decade of CSNI Activities in the Area of Ageing of Nuclear Power Plant Concrete Structures
- discussing the worldwide implications on nuclear facilities of the July 16, 2007 Niigata-ken Chuetsu-oki earthquake and its effects at the Kashiwazaki-Kariwa Nuclear power Station,

- supporting a benchmark, SMART 2008, being conducted in Saclay, France, on seismic design and assessment analysis for multi-story reinforced concrete buildings subjected to torsion and nonlinear effects
- supporting the IAEA extra-budgetary programme on seismic safety of existing NPP's
- conducting a specialist meeting on ageing management of thick walled concrete structures in October 2008, Prague, Czech Republic, with the planned publication of the proceedings
- improving robustness assessment methodologies for structures impacted by missiles
- thorough exchange of information on PTS Rules /Fitness for service criteria on different member countries for LTO of RPV; fatigue; plant ageing; LB LOCA redefinition / LBB break exclusion for operating and new plants
- joint IAEA/NEA catalogue on nuclear facilities that have experience an earthquake.

This paper will detail some of the recent activities and products of the IAGE group with special emphasis on the metal and concrete activities, since another SMIRT 20 paper will detail the activities of the seismic group.

Effects of concrete creep and shrinkage on the stress conditions of a post-tensioned containment structure for steam generator replacement project (8-1812)

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Introduction

Over the last 20 years, there has been an increasing number of Steam Generator Replacement (SGR) projects in the United States and worldwide. In most cases, SGR projects involve the creation of a temporary construction opening in the containment structure to facilitate the movement of the old/new steam generators out of/into the containment structure. Prior to the concrete removal, the post-tensioning tendons passing through the planned opening will be detensioned and removed. Additionally, vertical and hoop tendons in the immediate vicinity of the opening will be detensioned to minimize the prestress level within the opening. After completion of the steam generator replacement operations, the construction opening will be restored by placing new concrete. Once the new concrete achieves its target design strength, the removed/replaced and detensioned tendons will be retensioned.

Ideally, the prestress levels in the containment wall will be restored to its design basis condition prior to the SGR outage. However, the state of compressive stresses will not be fully recovered due to the following conditions:

- 1) Redistribution of dead and remaining prestress load (after removal/detensioning of tendons) from the opening area to the surrounding concrete due to concrete removal.
- 2) Redistribution of prestress load after the concrete is restored and tendons are retensioned due to the difference in creep and shrinkage values of the existing and replacement concrete.

Structural analyses are required to account for both effects above and verify that the interim and restored configurations of the containment structure will continue to meet the plant's licensing commitments (e.g., accident pressure/temperature,

safe shutdown earthquake loads, etc.). The problem of redistribution of stresses resulting from the creation of an opening in a prestressed containment wall is well understood, and References 3 and 4 provide details on how to properly capture such effects. On the other hand, incorporation of the time-dependent effects of creep and shrinkage in the new concrete and potential transfer of prestress load from the new to the old concrete becomes more involved and the analyst usually relies on simplified models and assumptions to include this complex phenomenon.

Aim of the work

The purpose of this paper is to investigate the effects of concrete creep and shrinkage on the prestress load of a post-tensioned containment structure with two dissimilar materials (i.e., existing and new concrete for the restored area). ANSYS general finite element program (Reference 2) is used to analyze the containment structure during the various construction stages. The concrete creep behavior is implemented using ANSYS defined material creep models and step-by-step nonlinear calculations are performed to follow the evolution of the prestress load. Shrinkage behavior is introduced in the model as temperature increments using the ANSYS time-dependent material table feature. Based on the results of numerical simulations, a simplified methodology is developed for general design/analysis purposes. This approach is based on the recommendations provided in Reference 1 combined with the principle of superposition presented in Reference 5. The validity of the proposed simplified method is examined by comparing stress conditions of a containment structure during/after SGR project with the results obtained from the nonlinear finite element analysis.

Summary/conclusions

The analysis results show that the difference in creep and shrinkage values of the existing and restored concrete has an impact on the redistribution of prestress load and needs to be properly accounted when performing structural adequacy evaluation of a post-tensioned containment structure for SGR project. Parametric analyses are performed and results presented highlighting the influence of the various variables. A simplified methodology is developed that accurately estimates the transfer of prestress load in the vicinity of the restored opening, and provides the practitioner a simple tool to account for this complex phenomenon.

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Microbially influenced corrosion in cooling water systems – development of a new protection concept for system components conveying brackish water (8-1815)

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Background and aim of the work

In recent years, corrosive findings ascribed to Microbially Influenced Corrosion (MIC), have been increasingly observed on cooling water systems in Northern German Nuclear Power Plants. Despite counter measures being applied in the form of the application of more corrosion-resistant materials, there is however, an obvious increase in the amount of corrosion detected. The protection of components by selected materials is thus to be considered effective only to a limited extent so far. [1–10]

Research programme and results

In order to improve the understanding of causal connections regarding the occurrence and progression, corrosion experiments were carried out at NPP Brokdorf.

By means of this research program (field tests), high-alloyed materials with different Pitting Resistance Equivalent Numbers (PREN), various surfaces, and various coatings were evaluated with respect to their corrosion behaviour in natural brackish water.

Within the scope of this research it has been possible to identify the actual causes for the damage on pumps in contact with brackish water. This is especially the case in combination with the colonization by micro-organisms, resulting crevice conditions beneath the biofilms and an attack takes place on especially vulnerable areas on the metal surfaces by chlorides from the brackish water. The research program at the Brokdorf NPP showed that the use of materials with high nominal pitting resistances as a countermeasure against such damage is not suitable. Particularly with regard to larger components, it is not

possible to ensure during production that the actual alloy composition meets the requirement in each area of the component.

Conclusions

Subsequent material evaluations, in combination with other measures, provided a new standard of knowledge for the development of a protection concept for components conveying brackish water.

Components can be protected in a reliable way with sufficiently dimensioned, impervious and undamaged coatings. It was also proven that a cathodic protection will protect the components especially well against corrosive attacks when exposed to brackish water.

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PAMS – piping and component analysis and monitoring system application and visualisation (8-1835)

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Introduction/background

The PAMS system contains up to date information enabling analysis, studies and monitoring of an operating plant. Although developed being an “as build” system it is even used in the design phase and for parameter studies.

The system consists of separate and stand-alone programs-modules and inter-related Microsoft® Access database tables. All modules can be independently updated and used for their own purpose as well as used together.

When “ready” the system will consist of the following modules:

- A document database containing documents associated with the information contained in the database.
- Several interconnected databases containing information like geometry, boundary conditions, materials and material properties, loading, pipe contents, material chemical composition, detected cracks etc.
- Pre-processing, editing, visualization and animation modules for the above mentioned items inclusive several visual and logical checks on the soundness and validity of the data. One of the latest achievements is the animation of transient loads.
- Interface modules to make indata files for application programs and to read data back from the result files into the result database.
- Tailor-made analysis modules to perform post processing of previously obtained results, like fatigue monitoring, crack growth monitoring, optimization of inspection intervals (RIISI) etc.

One of the main rules is that all data is accompanied by the associated source references. Typically these are material standards, isometric & detail drawings etc. In the future also a validity status will be added as well as the period that the pipe or component is installed in the plant. If the reference report of a load is invalid then the system will “know” that the subsequent analyses and results are not valid. This is very important as subsequent analysis, like fatigue and fracture mechanical analysis that are also performed directly from the same database shall use up-to-date input.

At present the complete geometry, materials and loads of the main piping systems in two nuclear power plants have been entered into the system.

Aim of the work

The aim of the work is to perform analyses using indata files that are generated from the databases. Commercially available analysis programs will be used as much as possible (FEM, CFD, fatigue, crack growth etc.). For special purposes programs are developed separately.

A special and unique feature is that the source reference list is automatically incorporate into the indata files and input data refers to the associated source references.

As results are read back into the database, they can be used for documentation or subsequent analysis. Later on visualization will be added to the result database. Then it will be possible to visualize results obtained by application programs.

Essential results

Using the animation module the system has been used to visually check the transient piping loading applied in a large project. Indata was made for stand-alone piping analysis and combined piping and building analysis and analysis of geometrical details. A large RI-ISI project was performed with a purpose made integrated software module. Furthermore, as all information in the databases is accompanied by its source reference the system is used as a well organized archive where documentation associated to piping systems can be found through the piping system model itself.

Summary/conclusions

From the development stage, the PAMS system has now been taken into productive use with both commercial software and special purpose tailor made programs.

As the same data is used over and over for different projects the reliability of this data increases continuously. Associated source references are automatically integrated into indata files thus reducing the need for separate documentation.

A high reliability of the information in the system is achieved through automatic data soundness check, reduced input possibilities through pull-down menus and different possibilities to do alphanumeric and/or visual checks of the information entered in the system.

The use of the system will definitely increase when other parties get acquainted to it.

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Update on Canadian regulatory oversight of ageing management for nuclear power plants (8-1842)

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This paper provides an update on Canadian Nuclear Safety Commission (CNSC) staff perspectives on managing the safety aspects of ageing of structures, systems, and components (SSC) of nuclear power plants (NPP). Managing the safety aspects of NPP ageing requires a proactive, systematic, and integrated ageing management approach for the coordination of all activities relating to the understanding, control, monitoring, and mitigation of ageing degradation of SSC through the lifecycle of a NPP. A CNSC regulatory document on ageing management based on modern international guidelines is described. In addition, the development of CNSC staff review guidelines for New Builds is included. Finally, CNSC participation in a number of ageing-management and structural integrity initiatives with industry and other national regulatory agencies both, within Canada and at international level are presented.

Performance surveillance of Gentilly-1 reactor building GFRP repair using fiber optic sensors and strain gauges (8-1848)

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At approximately 30 years old, Gentilly-1, a permanently shutdown CANDU™ 250 MWe Nuclear Power Plant that is currently in the Storage With Surveillance (SWS) phase underwent repair of the prestressed containment concrete ring beam. The fill concrete that protects anchorages of the prestressing cables had deteriorated primarily due to Alkali Aggregate Reaction (AAR) and a repair program was carried out. The repair was conducted in order to protect the prestressing anchorages from corrosion and to ensure the continued integrity of the structure during the time required for decay of radioactive material before final decommissioning.

The repair strategy involved removal of all deteriorated concrete in zones categorized as either shallow (<50 mm) or deep (>50 mm). The voids were filled with repair mortar or repair concrete and then covered with sheets of Glass Fiber Reinforced Polymer (GFRP). This was the first field application of GFRP material on a CANDU¹ reactor containment structure. In order to meet SWS phase requirements of maintaining the structure, and to assess the long term performance of the first GFRP repaired CANDU structure, a performance surveillance program was implemented.

Instrumentation was embedded both within the concrete repair material and also within the GFRP sheets. The program included the usage of Fiber Optic Sensors (FOS), Vibrating Wire Strain Gauges (VWSG) and thermocouples. Fifteen VWSGs were embedded in the repaired concrete. Measurements captured strains induced by early age concrete hydration processes and also hardened concrete strains and temperature effects. To compensate for temperature effect on the gauge body and to estimate concrete strain due to temperature

¹ CANDU is a trademark of Atomic Energy of Canada Limited.

variations, temperature is measured by the thermistors provided within the body of the VWSG.

Twelve Fabry-Perot FOS were installed. Eight of them were installed on the GFRP sheets with two layers of protection, while the other four sensors were installed in smart patches made of strips of Carbon Fiber Reinforced Polymer (CFRP). To estimate concrete strain due to temperature variations, four thermocouples were bonded to the GFRP sheets in close proximity of the FOS in four locations (North, South, East and West).

The primary purpose of this paper is to present experience gained in order to promote confidence in new experimental technologies adopted in the Gentilly-1 containment ring beam repair. This paper presents the 8-year performance of the repaired concrete and the sensor technologies used. Four key aspects of the Gentilly-1 repair surveillance program are discussed:

- i) strains and temperature variations of the repaired concrete and the GFRP
- ii) effectiveness and performance of the GFRP for concrete repair
- iii) testing and validation of fiber optic sensor technology, and
- iv) effectiveness of the remote monitoring system.

Recent advances in seismic non-destructive testing, and associated finite element based evaluation, utilized on a pre-stressed concrete reactor containment at a NPP in operation (8-1882)

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The majority of the nuclear power plants in the world have been in operation for a long period of time. The NPP industry is facing a period of huge investments related to large-scale modernization and power upgrade projects of the existing plants, and the building of new ones. One major financial condition for the pay-off of these investments is to be able to operate the plants for a longer period, thus raising questions related to residual lifetime estimations of containment structures.

Within the scope of the Euratom FP5 program, a consortium consisting of Force Technology, Denmark, Scanscot Technology, Sweden, Electricité de France, France and Barsebäck Kraft, Sweden carried out the project “Concrete containment management using the Finite Element technique combined with in-situ Non-Destructive Testing of conformity with respect to design and construction quality (CONMOD)” (see [1], [2] and [3]). The main conclusion from the CONMOD project is that a new approach combining Non Destructive Examination with Finite Element Analysis methods is both workable and necessary in order to be able to accurately determine and predict the condition of nuclear power containment structures.

The CONMOD project has been a pioneering study and has shown the way to a new approach regarding condition assessment and ageing management. It was however, largely a feasibility study. The proposed technology needs further research and development along with site specific procedures to be implemented and validated in practice.

At the moment, an implementation project regarding non-destructive examination of a pre-stressed concrete reactor containment in operation, and associated finite element technology, is carried out at a BWR plant in operation. The project is based on the outcome of the CONMOD project described above. The purpose of the project is to

- implement recent advances in non-destructive testing inspection strategies and planning
- validate and refine results from non-destructive testing using finite element analyses
- study and development of logistics for application to concrete containments in service.

The thickness, dynamic Young's modulus, depth to delaminations, voids, and cracks in concrete structures can be evaluated non-destructively by using seismic wave based techniques. The speed and attenuation of seismic waves reflect the dynamic elastic properties of the material. Seismic methods are therefore well suited to solve problems related to ageing management of thick walled concrete structures.

The Multichannel Analysis of Surface Waves method (MASW) is a non-destructive seismic method to evaluate the stiffness (wave speed) variation with depth. The method makes it possible to analyse all type of seismic waves propagating within the structure. The Impact Echo (IE) method is a technique to estimate the thickness of concrete structures. The method relies on a good estimate of seismic velocities in the structure and a precisely measured resonant frequency. These inherent limitations and advantages with each method support a combined analysis. We present recent advances on these methods where both resonant frequencies (IE) and seismic velocities (MASW) are evaluated in a combined manner from the same multichannel data set.

Typically each traditional seismic non-destructive testing (NDT) technique is based on a simplified model and measurement set-up, tuned for a specific application, e.g. Impact Echo for thickness, surface waves for stiffness. These simplified models are 1D models based on homogeneous layers with finite thickness extending to infinity in the other directions. Further more, these traditional methods cannot be used to accurately predict the effect of the finite size source and receivers, near field effects, 2D and 3D effects from the actual geometry and variable material properties, scattering from cracks and voids etc. For the future research and development of NDT techniques for concrete containment walls it is important to move towards more realistic 2D and 3D theoretical models. It can be argued that the true potential of seismic wave based techniques has not been fully explored due to the simplified theoretical models used for the evaluation of measured data. Recent studies have demonstrated the benefits of a joint evaluation (inversion) of different type of waves and resonant phenomena. This research trend emphasizes the need to move from simplified 1D models to more sophisticated 2D and 3D full waveform models for the evaluation of seismic measurements.

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Study on the boric acid corrosion behavior of disk/seat materials in SI check valves (8-1889)

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The disk/seat materials of check valves in nuclear power station's SI system have been repeatedly damaged. From the analysis, several causes have been deduced.

The causes were classified as three categories: 1) cavitation from the valve leakage, 2) cavitation from the valve disk chattering, 3) corrosion from the boric acid evaporation and concentration.

The 3rd cause have been assumed that corrosion was occurred from the concentrated boric acid which had been formed by boric acid particles blended again with leaked water of the valve. These boric acid particles seemed to be formed from the evaporation of boric acid that had been remained in the seat end part of the valve which had minute leakages. These leakages could be occurred in the condition that 2 phase fluid existed when the pressure of foregoing part in the check valves decreased under the saturated pressure. This corrosion phenomenon was occurred in the boundary region between water and steam. Therefore this condition can be satisfied only when 2 phase fluids in front of the check valves are formed. The fact that the upper part of the disk/seat had no corrosion implies that boric acid existed in the form of particles above 185 that is the temperature of boric acid dissolution. The previous experience also showed that corrosion from evaporation and concentration was inevitable when the foregoing part of the check valves in SI system were in the condition of 2 phase fluid. This conclusion was based on the observation that the backside of the disk also had corrosion from evaporation and concentration.

In this study, the accelerated condition was formed in the environment of high pressure and temperature and corrosion rates were estimated varying with concentration of boric acid as a parameter. The corrosion rates were measured from satellite that was the weld part of the currently used and A59 that was the candidate weld materials for replacement. These materials were also used in the calculation modeling the geometries of check valves using BEM (Boundary Element Method).

The complex approach to the determination of NPP Steam Generators Heat-Exchange Tubes Plugging Criterion (NPP SG HET PC) on the basis of analyses of the processes of the tubes damaging during NPP operation (8-1910)

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As a result of eddy-current control (ECC) methods development appears the opportunity of early revealing of defects, which incipient in a walls of heat-exchange tubes (HET) of steam generators (SG) during operation.

Preventive plugging of a defective HET, as a result of ECC, has essentially increased the reliability and the safety of reactor facility operation, but thus volumes of HET plugging have simultaneously sharply increased. In some cases HET are unreasonably muffled as a result of imperfect ECC equipment application, or a mistakes of the control measurements, or imperfection of criterion of HET plugging. Unreasonable HET plugging essentially reduces SG service life. In this connection the scientifically proved determining of SG HET Plugging Criterion gets a problem a great value. This Criterion should be to provide a necessary level of NPP safety, but thus simultaneously to minimize the number of HET plugging.

The approach suggested in the given work is based on the developed Phasic Model of Damage (PMD) of NPP WWER-1000 type SG HET. In this model it has been shown, that by key parameters determining a dynamics of HET damages, are the thickness of deposits on a surface of SG HET and kinetics of an oxidizer reduction in a local electrochemical cell around of growing defect.

In this paper represented the results of a HET plugging criterion substantiation by the example of one of NPP Unit with steam generators PGV-1000. The corrosion defects of HET have been found out during the prestarting period of that Unit after ECC of SG HET. By nature that defects constitutes as pitting and the corrosion cracks developing from pitting. Pitting and cracks have arisen in HET metal under the influence of chlorides and the oxygen, the which have got in steam generators volume during installation of NPP equipment.

The substantiation of criterion of SG HET plugging was based on:

- Defect depth in a HET wall limiting values estimated definition from the conditions of the crosspiece breakdown in front a crack tip due to

instability of plastic deformation of metal. Limiting values depends on the relation of depth of crack (**a**) to its length on a forming line of tube (**l**);

- Experimental confirmation of estimated value of limiting depth of defect;
- An estimation of the admitted size of defect in view of an opportunity of its maximal growth during the period between periodical ECC;
- An estimation of the maximal error at definition of depth of a crack on data ECC;
- Definition of plugging criterion, as the limiting value of a defect depth determined by ECC, at which is created a danger of a tube destruction excess during reactor facility operation at the period between periodical ECC.

Seismic qualification and upgrade of safety important pipelines support systems in reactor building of units 5 and 6, Kozloduy NPP (8-1934)

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The aim of the task was to qualify seismically the existing safety important piping installed in the reactor building. The need for seismic qualification came from reassessment of review level earthquake (RLE) established for Kozloduy NPP site. The pipelines should be properly qualified according to the new higher seismic level.

In the previous steps of units 5 and 6 Modernization Program execution, the list of pipelines was determined and the corresponding pipelines were modeled using computer codes PepS and PipePlus. The pipes, which did not satisfy the criteria, were identified. For this group of pipelines, the supports' modifications in the computer models were introduced, in order to achieve acceptable demand to capacity D/C ratio for the most critical pipe sections. For each pipeline no less than two different modifications of pipe support were proposed and compared, considering different aspects as, cost effectiveness, number of additional supports, ease of implementation on site, maintenance requirements, etc. Based on the proposed modifications the decision for appropriate pipe supports modification was taken and the corresponding detailed design was developed. The support system modifications include modification of existing supports as well as addition of new supports.

The pipelines to which the upgrade of support system was performed belong to safety important systems such as: fire extinguishing system, spent fuel cooling system, essential service water system, core cooling system and steamgenerator emergency feedwater system. All pipelines are situated in the reactor building.

The capacity of the supports was calculated according to ASME BPVC Section III Subsection NF "Supports" and according to design data from the support maker. For the check of the linear supports, the criteria from ASME NF-3320 and ASME NF-3623 was used, with coefficient from Table NF-3623 (b)-1 according to the level of the analysis and the type of the load case.

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For the calculation of the additional and the modified supports, design during the detailed design phase, the same code was used. For the design of the necessary steel structure and the connection with the existing ones, the Bulgarian standards for design of steel structures was used. For the design of the anchoring in the concrete structures, Bulgarian codes for design of concrete and reinforced concrete were used, along with technical data from the anchors producers.

The seismic upgrade of safety systems pipelines, vulnerable to seismic event were successfully implemented on both units 5 and 6 during units outage in 2006.

Proactive Management of Materials Degradation (PMMD) and enhanced structural reliability (8-1954)

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The Nuclear Regulatory Commission (NRC) has undertaken a program to lay the technical foundation for defining proactive actions so that future degradation of materials in light water reactors (LWRs) is limited and, thereby, does not diminish either the integrity of important LWR components or the safety of operating plants. The United States is currently implementing license extensions of 20 years on many plants, and consideration is now being given to the concept of “life-beyond-60”, license extension from 60 to 80 years and potentially longer. In support of NPP license renewal over the past decade, various national and international programs have been initiated. This paper discusses efforts by the U.S. Nuclear Regulatory Commissions Proactive Management of Materials Degradation (PMMD) to determine the effectiveness of emerging NDE techniques.

All parts of a nuclear power plant (NPP) are subject to the continuous time-dependent degradation of materials due to normal service conditions, which include normal operation and transient conditions. The PMMD program is investigating the many materials and components in NPPs and the materials degradation phenomena that affect them in an attempt to predict and prevent future problems. As some forms of degradation, such as stress corrosion cracking, are characterized by a long initiation time followed by a rapid growth phase, new inspection or monitoring technologies may be required. New NDE techniques that may be needed include techniques to find SCC precursors, on-line monitoring techniques to detect cracks as they initiate and grow, and improved current NDE technologies. As the reactors operate well beyond their originally-planned lifetimes many reactor components may have insufficient NDE programs in place to prevent failures, as the NDE programs were not designed for 60 or more years of operation.

The need for new NDE technologies and techniques necessitates a methodology for the evaluation of new NDE techniques to determine if the new NDE technologies have a strong technical basis in a timely fashion. Previous efforts in evaluating NDE technologies and techniques have lacked a structured methodology and have been very time consuming. The methodology is based on previous evaluations of NDE technologies. The methodology consists of four possible distinct phases. The first phase of the evaluation is an expert review of the new technique that reviews the physics of how the technique works and the applicability of the technique to the affected components. The second phase of the methodology is a series of laboratory tests to determine the effectiveness of

the technique under laboratory conditions and to explore the essential variables for the technique. In the third phase, after the laboratory testing, a round-robin test would be conducted using commercial vendors to determine if the field-deployed systems can operate under realistic conditions. The final phase would be the implementation of performance-based testing to assure that the inspectors, technology, and technique are all able to provide an adequate probability of detection for degradation in NPP components.

Additionally, a comprehensive review of reactor components will be needed to determine if new inspection regimes may be required to deal with new degradation mechanisms that may emerge over time. As reactors lifetimes are expanded, degradation mechanisms previously considered too long-term to be of consequence (such as concrete and wiring insulation degradation) may become more important.

The paper describes the methodology developed to evaluate new NDE technologies and techniques. A model BWR component is scrutinized to determine the efficacy of the current NDE requirements for the component when viewed at extending the lifetime of the component beyond 60 years is also presented.

Ageing management of steam generator internals (8-1959)

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Keywords: nuclear, power plant, steam generators, ageing, ageing management, integrity, monitoring, preventive maintenance, failure orientated maintenance

Safe operation as well as economical aspects of nuclear power plants depend on *integrity and function* of the components and systems in use. More general, integrity and function can be expressed in terms of *quality*. Component quality is established during the *state of design* and the *manufacturing process*. In *operation* this quality (safety, remaining life) can be affected by ageing phenomena.

Therefore, the EnBW nuclear power plants in Neckarwestheim, Germany, have introduced a comprehensive ageing management process. Within ageing management the quality of components and systems in operation has to be assessed, regularly. Depending on the demands on the quality, three groups of systems, structures and components have been defined:

- group 1: Integrity Concept (guarantee of integrity – control ageing phenomena)
- group 2: Preventive maintenance (maintain initial quality – prevent systematic failures)
- group 3: Failure orientated maintenance (re-establish initial quality).

Components and systems that are relevant to the safety of a plant are classified into groups 1 and 2. These groups usually are target of ageing management programs.

In the paper, the entire ageing management process is discussed using the example of the steam generator internals.

The steam generator tubes (heat exchanger) are categorized into group 2 within the ageing management procedure, i.e. the quality in operation is maintained by use of inspection and testing. Results of damage mechanisms in operation like reduction of wall thickness, cracks etc. are controlled by non-destructive testing. On the basis of the surveillance results the quality status is reviewed, regularly (design aspects, actual structure, operation experience and actual knowledge). There were some problems with wastage in the first years of operation. Since a change in water chemistry these problems have been reduced, drastically. Loose parts fretting still occurs, from time to time. This effect is

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controlled by non-destructive testing and by a very sensitive leakage (activity) monitoring on the secondary side of the steam generators. In total, the number of closed pipes due to wastage and fretting signs is low.

Based on the results of review discussions, other steam generator internals have been introduced into the ageing management process. If the feedwater distribution and support structure shows failures, for example, falling parts can affect the integrity of the steam generator tubes. Consequently, these structures have been included in the ageing management process. As with other systems, structures and components of group 2, extensive measures of preventive maintenance concentrate on the prevention of systematic failures.

Assessment of gaseous pollution from hot cutting processes in NPP disassembling (8-1960)

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The gaseous pollution in NPP decommissioning is a main concern, and it should be appropriately estimated. It is important, consequently, to generate a methodology for the preventive evaluation of gaseous emissions occurring from cutting process. Said assessment can be also referred to various other cases.

Cutting processes have been carried out in an experimental campaign at Caorso NPP – Boiling Water Reactor (BWR), 870 MWe – built in the 70s, completely operating in the years 1981–1986, shut down on 1987 and at present under dismantlement.

In the present work, the calculated values of the pollution arisen from plasma torch and oxyacetylene arc cutting processes are shown, related to the volatilized material, the volume of the powders per hour, and the concentration in the environment and at the inhalation. The cut components – carbon steel with different surface treatments – are no activated plates and pipes exposed to the reactor coolant steam.

The evaluation of the dilution between emissions and air at the inhalation is discussed, introducing also suggestions to perform the said cutting procedures in a safe way.

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Repair and strengthening of damaged reinforced concrete slabs with CFRP (8-2047)

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In the past few years, several major international research programs were launched to investigate the feasibility of using technologies of polymer composite for repair and retrofitting structural systems. One of these successful applications is the external strengthening for repair and upgrade the structural capacity and rigidity of axially loaded concrete members in particular beams or columns. In these cases, several laminates of composite are bonded to the finished concrete surface in the hoop or longitudinal direction for enhancing the flexural capacity of these members.

The object of the present study supported by EDF (Electricité de France) focuses on the investigation of the behaviors of large-scale RC slabs retrofitted by CFRP for the purpose of the application of this reinforcement process to the RC shell of hyperbolic cooling towers. The objectives were to observe the phenomenological behavior of the reinforced and unreinforced slab subjected to combined membrane and bending loads and to provide useful data in checking the accurateness of non-linear finite element simulation.

A specific experimental device built at INSA Lyon allows the application to concrete slabs ($2000 \times 2000 \times 100$ mm), of varied combinations of membrane and flexural loads in two directions. Several tests are conducted to gauge the pertinence and the limit of the CFRP reinforcement on the non damaged or pre-damaged concrete slab. The initial damage is obtained thanks to an initial loading where membrane and/or flexural loads are combined; the structure is then unloaded and repaired with carbon fibre reinforced polymer strips. Test results indicate that the composite system restores not only the original capacity of the damaged slab but also an appreciable enhancement of the bearing capacity.

The finite element software CAST3M is used to analyze the unreinforced and reinforced concrete slabs. For this purpose, the multilayered DKT element (Discrete Kirchhoff Triangle) is used to simulate the concrete, the steel reinforcement and the CFRP layer. Comparison between experimental and numerical results shows good agreement for both the control slab, and the non-damaged reinforced or the damaged reinforced concrete slabs.

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Development of integrity evaluation program for pipe wall thinning (8-2071)

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Evaluation of pipe integrity against local wall thinning due to erosion and corrosion is increasingly important in maintenance of carbon steel pipes in nuclear power plants. Though a few programs for assessment of the thinned pipe integrity have been developed in domestic nuclear engineering, they are limited to straight pipes using methodology proposed in ASME Section XI Code Case N-597.

In this study, an engineering program for evaluation of the integrity of all kinds of pipes such as straight, elbow, reducer and branch pipes was successfully developed, which was designated as PiTEP[®] (Pipe Thinning Evaluation Program). The developed program includes four sequential steps to evaluate the integrity such as, first evaluation step on design code (ASME Section XI Code Case N-597), next its engineering method and then a couple of evaluation methods of our own. As PiTEP[®] is constructed in user friendly GUI (graphic user interface) environment, we expect plant engineer can easily operate it only with measured thickness data, basic operation conditions and pipe data. Some experimental tests were also conducted with elbows to verify that the results of the program for the assessment of thinned pipe integrity are appropriate.

From the development of evaluation program for pipe integrity against local wall thinning, we had some conclusions as follows; first, it was possible to have optimum criteria for thickness of thinned pipe for its integrity based on the recent technological methods in the program. Secondly, it was found that the results of the program have sufficient conservative margin comparing to the verification test results. Thirdly, PiTEP[®] can be easily applied to nuclear plants because the program is constructed in user friendly GUI (graphic user interface) environment.

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Fabrication flaw density and distribution in piping weldments¹ (8-2476)

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The U. S. Nuclear Regulatory Commission supported the Pacific Northwest National Laboratory (PNNL) to develop empirical data on the density and distribution of fabrication flaws in nuclear reactor components. These data are needed to support probabilistic fracture mechanics calculations and studies on component structural integrity. PNNL performed NDE inspections and destructive testing on archived piping welds to determine the fabrication flaw size and distribution characteristic of the flaws in nuclear power plant piping weldments. Eight different processes and product forms in piping weldments were studied including wrought stainless steel and dissimilar weldments. Parametric analysis using an exponential fit was performed on the data. Results were created as a function of the through wall size of the fabrication flaws as well as the length distribution. The results are compared and contrasted with those developed for reactor pressure vessel processes and product forms. The most significant trend was that the density of fabrication flaws versus through wall size was higher in piping weldments than that for the reactor pressure vessel weldments but smaller than that found for reactor pressure vessel weld repairs. Curves showing these distributions are presented.

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Fabrication flaw density and distribution in weld repairs¹ (8-2477)

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The Pacific Northwest National Laboratory (PNNL) is developing a generalized flaw distribution for the population of nuclear reactor pressure vessels and for piping welds in the U. S. operating reactors. The purpose of the generalized flaw distribution is to predict component-specific flaw densities. The estimates of fabrication flaws are intended for use in fracture mechanics structural integrity assessments. Structural integrity assessments, such as estimating the frequency of loss-of-coolant accidents, are performed by computer codes that require, as input, accurate estimates of flaw densities. Welds from four different cancelled reactor pressure vessels and a collection of archived pipes have been studied to develop empirical estimates of fabrication flaw densities.

This paper describes the fabrication flaw distribution and characterization in the repairs weld metal of vessels and piping. This work indicates that large flaws occur in these repairs which are complex in composition and sometimes include cracks on the ends of the repair cavities. Parametric analysis using an exponential fit is performed on the data.

Construction records where available were reviewed. It is difficult to make conclusions due to the limited number of construction records reviewed. However, the records reviewed to date show a significant change in repair frequency over the years when the components in this study were fabricated. A description of repair flaw morphology is provided with a discussion of fracture mechanics significance.

¹ The work was sponsored by the U.S. Nuclear Regulatory Commission under U. S. Department of Energy Contract DE-AC05-76RL01830; NRC JCN Y6398; Mr. Wallace Norris, Program Monitor.

Graphite blocks reloading consideration in HTR-PM (8-2486)

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After the experimental reactor HTR-10 was built and succeeded in operation, INET is designing another reactor, HTR-PM. The service life was 40 years and the graphite blocks reloading was not considered. But we design the graphite reloading scheme for the future use in HTR.

There are some common requirements for graphite block reloading in high temperature reactor:

1. Strict control leakage of radiation.
2. Shield requirement.
3. High reliability, simple structure of reloading equipment.
4. Remote control of the reloading equipment.

Compare to the HTGR and HTTR, there are some differences for HTR-PM:

1. Only the graphite structure was reloaded in HTR-PM, other metal equipments were not reloaded.
2. The surrounding condition was air, not the Helium. For HTGR and HTTR, the gas in the reactor is helium when reloading graphite.
3. The reloaded material has less radiation for HTR-PM. The estimated radiation was 1Ci.
4. The graphite block reloaded in HTR-PM was more complicated in shape. There were 3 kinds of graphite: top reflector, central graphite cylinder and side reflector.

Special process was designed for the graphite block reloading. The preparation process includes 6 steps. For the reloading process, exactly 90 steps were contrived to reloading 3 kinds of reflectors. We don't give detail here in the abstract, but will give in our paper.

8 kinds of assembly products were designed for the graphite reloading:

1. Top reflector reloading machine, shown in Fig. 1.
2. Central graphite cylinder reloading machine, shown in Fig. 2.
3. Side reflector reloading machine, shown in Fig. 3.
4. Hoisting barrel.
5. Sealing gate
6. Transfer channel
7. Transfer trolley
8. Crane.

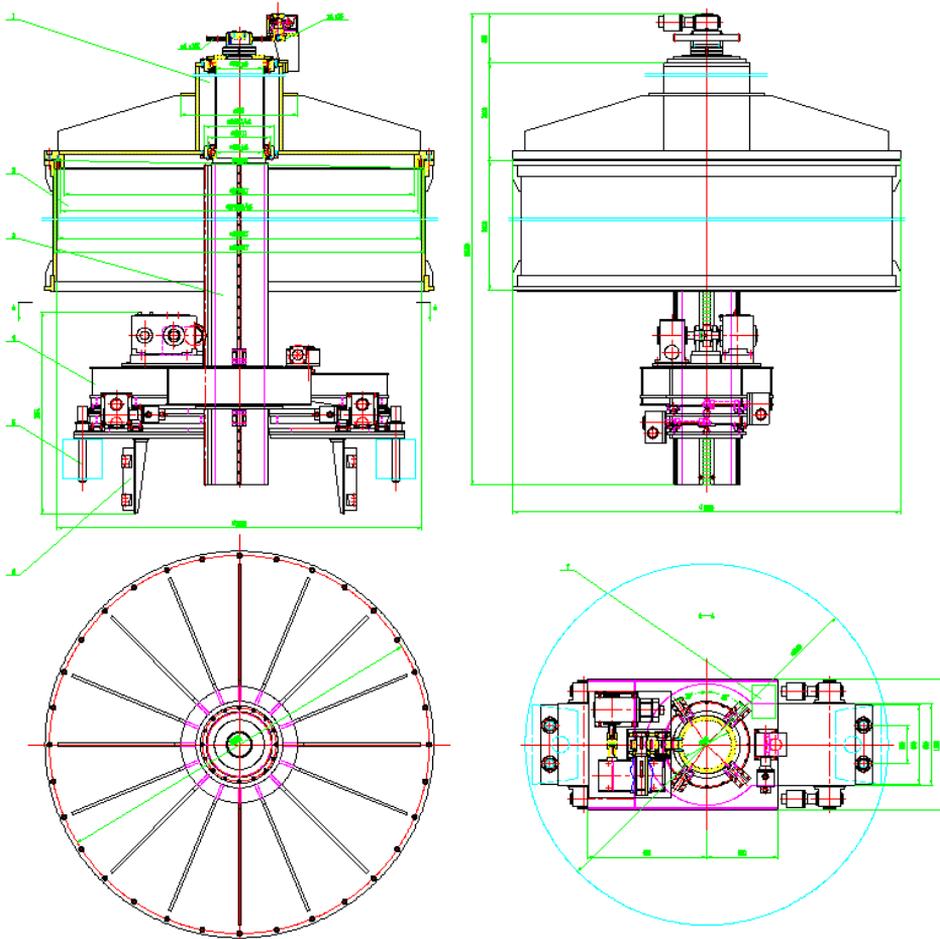


Figure 1. Top reflector reloading machine.

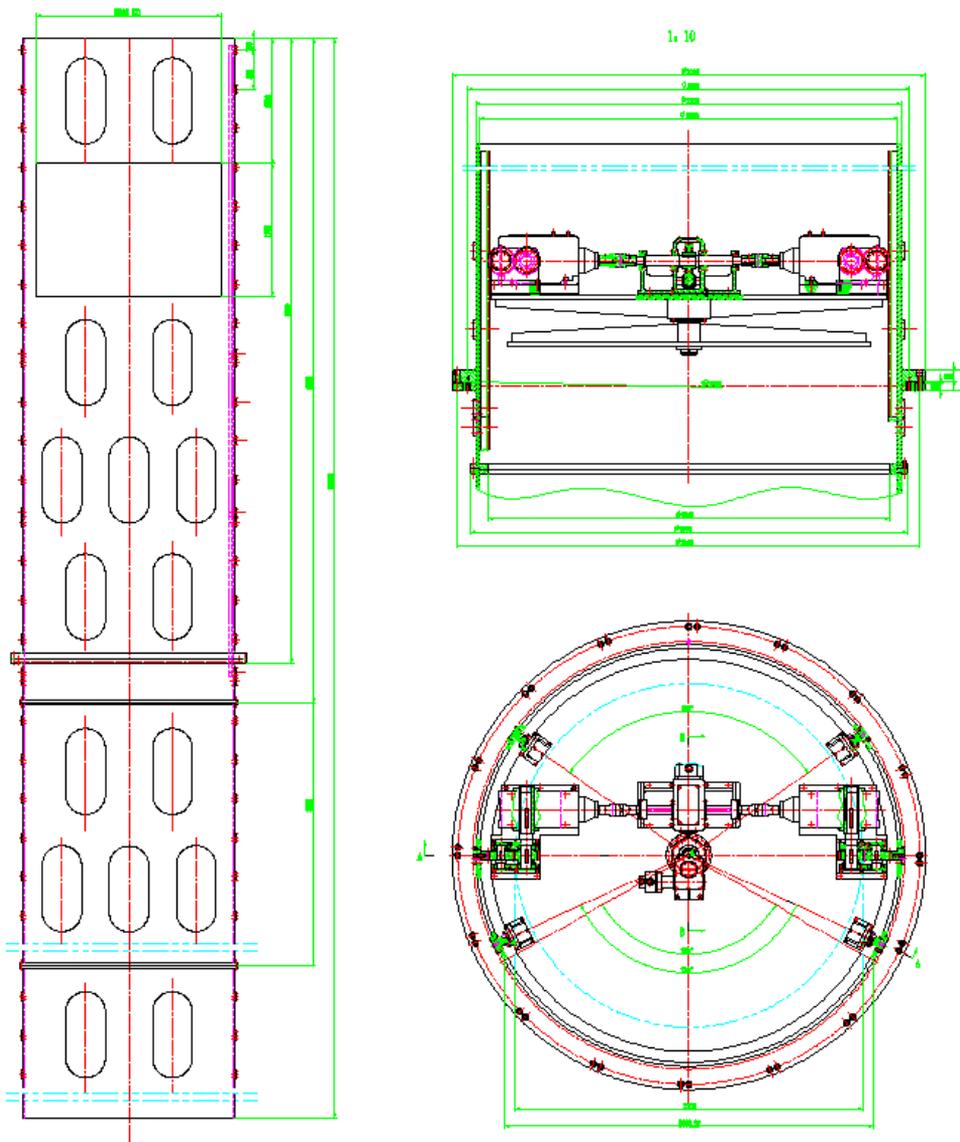


Figure 2. Central graphite cylinder reloading machine.

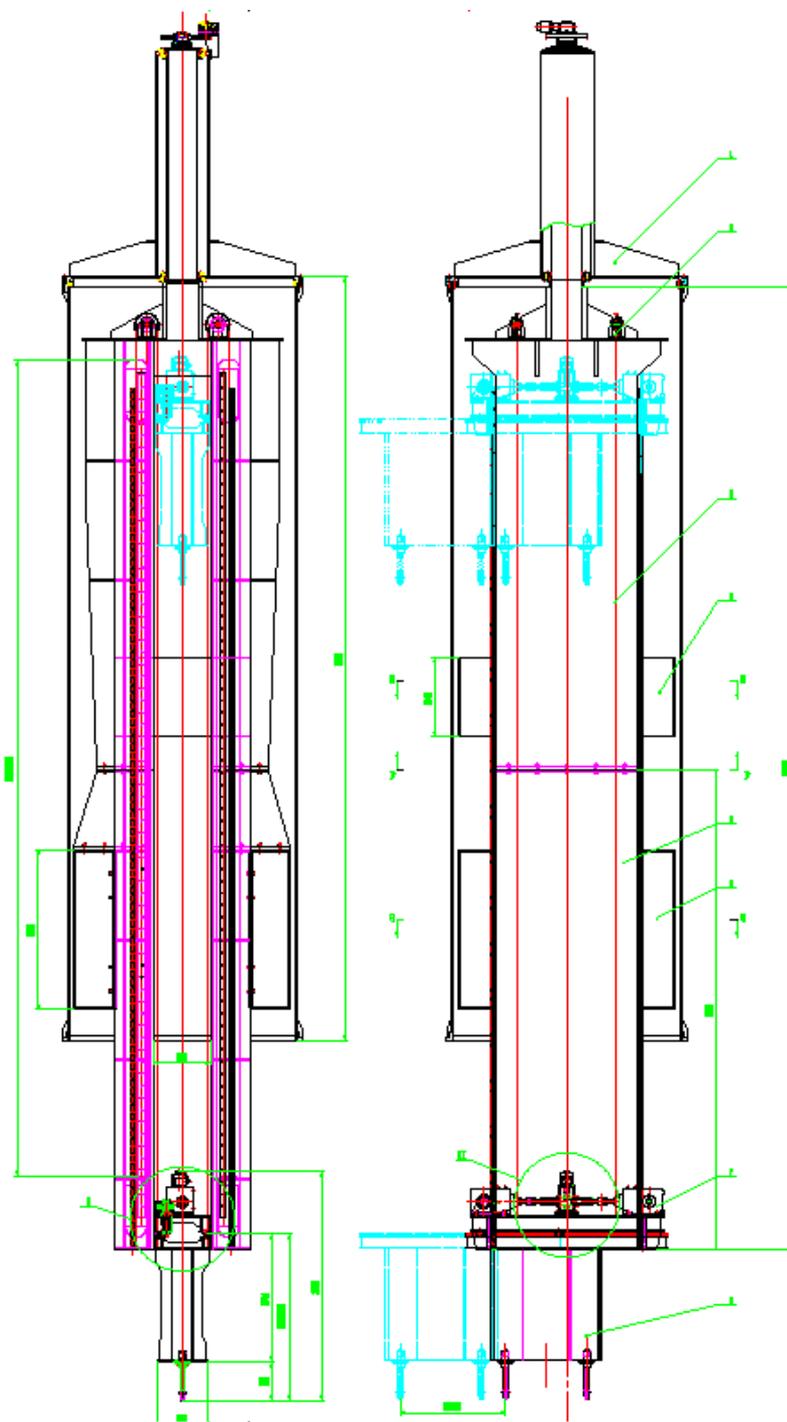


Figure 3. Side reflector reloading machine.

Korean experience in aging management for long term operation of NPP (8-2533)

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Recently long-term operation of nuclear power plant beyond its licensed term has become a worldwide trend in so called nuclear renaissance era. To obtain license approval of the long-term operation, aging management strategies and programs of systems, structures, and components should be established and implemented [1].

Aging management programs (AMP) were established for two leading nuclear power plants in Korea, that is, Kori Unit 1 for PWR and Wolsong Unit 1 for PHWR. A total of 39 AMPs were prepared for Kori Unit 1 according to NUREG-1801[2]. These included an in-service inspection of safety class 1, 2, 3 components, nickel-alloy nozzles and penetrations, reactor vessel surveillance, loose part monitoring, electrical cables and connections not subject to environmental qualification requirements, etc. Alloy 600 AMP was prepared to ensure that integrity of nickel-alloy nozzles and penetrations is maintained against the primary water stress corrosion cracking (PWSCC), which recently has become an important issue worldwide. This includes a survey of all Alloy 600 dissimilar metal welds in Kori Unit 1 and an assessment of the integrity of components highly susceptible to PWSCC. Residual stress analysis and measurement in dissimilar metal weldment were also studied.

Time-limited aging analysis (TLAA) of the critical SSC related to safety should be also done and the report should be submitted for the long term operation of a NPP according to MEST Notice 2008-17 [3]. A total of 10 TLAA items for Kori Unit 1 were identified and reviewed. These included reactor vessel neutron embrittlement analysis, metal fatigue analysis, environmental qualification of equipment, wear of the neutron flux detector tube, the reactor coolant pump flywheel, the spent fuel pool liner, thermal aging embrittlement of the cast austenitic stainless steel, etc.

It was found that most of the existing AMPs can properly and effectively manage the aging phenomena shown in the critical SSC. However, for some of them, e.g. the nickel-alloy nozzles and penetrations, the AMP revision was necessary. 13 AMPs in total including boric acid corrosion program and Alloy 600 program were revised or were newly prepared. In the TLAA of the metal fatigue of the main components (reactor vessel, control rod drive mechanism, reactor internals, reactor coolant pump, steam generator, reactor coolant system piping, safety injection tank), the cumulative usage factor was found to be less than

8. Issues Related to Operations, Inspection and Maintenance

1.0 (acceptance criteria), even with the occurrence of transients conservatively predicted at the end of the extended operation of 40 years. A fatigue monitoring system was additionally installed in Kori Unit 1 to ensure operational safety for another 10 years of operation. It was subsequently confirmed that the integrity of the main components would be maintained until the end of continued operation of Kori Unit 1[4].

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Numerical and analytical framework for analysis of crack initiation and propagation under thermal fatigue loading (8-2581)

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Material degradation at thermal fatigue loading is an important problem arising in the life-time management of piping components of nuclear power plants (NPP). In particular, cracking phenomenon in tee connections due to mixing of the hot and cold fluids has been extensively investigated in a past decade within national and European research projects, e.g. THERFAT, NULIFE.

This paper considers several aspects related to the numerical and analytical modelling of crack initiation and propagation under thermal fatigue loading. These include the assessment of thermal-hydraulic boundary conditions, stress analysis, fatigue and fracture mechanics calculations. Appropriate analytical models and tools are presented demonstrating possibilities for a simulation of surface crack growth under random thermal-mechanical loading. The significance of the data scatter is additionally discussed and examples of a probabilistic analysis are presented.

A model to monitoring real-time fracture of concrete subjected to the load from tendons by AE technique (8-2603)

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Using acoustic emission technique (AE) to monitor the load test on RC structures can reveal more details for the material integrality. Many of relevant studies [1, 2] have verified that Kaiser Effect exists in reinforced concrete structures, which could be applied in an aged pre-stressed concrete structure that the tendons need to be replaced. This application may be useful on the safety-related or high-strength demanded structures, such as the post-tension (tendon) system of the pre-stressed reinforced concrete containment (PRC). The tendon system results in a distributive pressured state within the concrete of the entire containment wall to prevent fracture [3] due to the inside peak pressure; however, some parts of this system like the anchor trumplate or the bearing plate may cause noticeable concentration load on the concrete. It is noted that most non-destructive testing (NDT) techniques are not possible to inspect the defects within these areas. This research is addressed on the development of AE technique to monitor the routine in-service inspection (ISI) program with loosening and reloading procedures. Through the monitoring results, the integrality situation at these critical areas may be evaluated. The issue is significant for those containments which need to be extended in their service lives.

This research provides a preliminary study for a mock-up test later to model the concrete around and under the bearing plate. The AE technique using a self-developed instrument is applied to record/monitor the emitted ultrasonic wave in a small-scale specimen during load test being performed in a MTS system. The obtained hit-count history profiles (vs. time or load), the Kaiser Effect (or the crack incubation) and the crack-control mechanism of re-bars in concrete will be observed and discussed in this paper.

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Methodology research on prediction for operating lifetime of PWR RPV (8-2619)

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The residual lifetime of the operating PWR NPP is always concerned by the owners and the RPV plays a critical role in the lifetime prediction of PWR NPP due to irradiation induced embrittlement of the active core beltline and its non-replaceability. The PWR NPP in question have been operating over half of the initial design lifetime in China and overall survey and investigation for the RPV have been carried out focusing on the Time Limited Aging Analysis (TLAA). Based on the assessment of the pressure - temperature limits on maintaining the reactor coolant pressure boundary, integrity of RPV under the potential pressurized thermal shock (PTS) and the upper shelf energy (USE) of irradiated beltline, the residual lifetime of RPV is predicted in this paper.

The effectiveness of chemical cleaning in reducing the risk of leakage in steam generator tubing: a Bayesian approach (8-2621)

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Steam generators (SG) experience a variety of corrosion related degradation mechanisms, such as pitting, wall thinning and stress corrosion cracking. Chemical cleaning and water lancing are two effective maintenance methods that remove the sludge from the steam generator and minimize the potential for under deposit degradation of SG tubing. In a risk-based life cycle management of SG fleet, it is important to quantify the benefit of the maintenance (chemical cleaning or water lancing) in terms of reduction in the risk of tube leakage that leads to the forced outage of the reactor.

The aim of the paper is to present a probabilistic model to evaluate the risk of tube leakage in steam generators and calibrate the parameters using realistic data from a nuclear station in Canada. A Bayesian methodology is presented to evaluate the effectiveness of chemical cleaning based on the data regarding the occurrence of leakage before and after chemical cleaning on SGs in nuclear plant in Canada.

The practical examples presented in the paper show that chemical cleaning is partially effective in reducing the risk of tube leakage.

A proposal for a unified model on nuclear power plant life management including maintenance optimisation (8-3173)

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In recent years many electric utilities and nuclear power plants adopted policies for improved coordination of both safety and non-safety programs, called plant life management (PLIM), also in view on plant life extension programs, but mainly for an optimisation of operating costs. The implementation of PLIM programs has followed many different approaches, being intrinsically dependent on the national regulatory framework and technical traditions.

In Countries with some experience, the PLIM program proved very convenient, especially when coupled with Maintenance, Surveillance and Inspection (MS&I) optimization: average savings are reported in the range of 20–30% of total (maintenance) costs.

Moreover, in terms of safety, the control of equipment reliability, significantly improved with PLIM models for example through Ageing Management Program (AMP) and Reliability Centred Maintenance (RCM), made a long term asset management of the overall plant possible and the overall safety indicators significantly improved in many cases.

This is why R&D tasks are needed in this phase, not only in the long term extrapolation of the component integrity and behaviour, but also in new management strategies at the plant (PLIM), able to address organisational issues, spare part management, staff ageing, component obsolescence, etc, which are typical components of the PLIM.

The Framework Programme 7 of the EU in the area dedicated to the reactor systems calls for a research effort “to underpin the continued safe operation of all relevant types of existing reactor systems (including fuel cycle facilities), taking into account new challenges such as life-time extension and development of new advanced safety assessment methodologies (both the technical and human element)”.

A unified European model for PLIM was developed at the JRC with the support of a network of stakeholders (SENUF), and validated at some EU nuclear plants.

¹ The views expressed in this paper are those of the author and do not necessarily reflect those of the EC.

8. Issues Related to Operations, Inspection and Maintenance

This paper provides a summary of the model features, the result of its validation at some plants and summarises the perceived scientific/technological challenges for the FP7 on which JRC proposes to focus, based upon its competencies and skills, having in mind both the European and world-wide context and its potential evolution.

9. Waste Management, Fuel Cycle Facilities and Decommissioning

Evaluation of existing facilities. Risk-informed and performance-based considerations. Probabilistic safety assessment. Siting, design and construction of fuel cycle facilities. Issues related to long-term and short-term waste storage.

Development of neutron shielding materials for nuclear fuel storage facilities (9-1707)

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In the context of optimisation of nuclear power plants, spent fuel discharged from reactors have higher burn up and consequently have more residual heat. Installations and storage casks have to withstand the severe temperatures resulting from increasing residual heat of the spent nuclear fuel.

A family of new neutron shielding materials was developed by AREVA TN International for spent nuclear fuel in storage facilities and transport/storage casks showing high shielding capabilities for a range of temperature corresponding to the needs.

These materials are composed of a thermosetting resin (vinylester or polyester resins) and mineral fillers (alumine hydrate and zinc borate). The cross linking of the polymer leads to a rigid three-dimensional lattice, solid and resisting to transport conditions, especially the temperatures.

Tests of the neutron shielding materials have been performed to establish:

- Shielding/attenuation efficiency in compliance with nuclear site specification, in case of transport compliance with IAEA safety standards TS-R-1.
- Good behavior in accident conditions.
- Radiation and heat resistance during the period considered: storage (40 years) or transport (depending of transport frequency).
- Quality and homogeneity of the shielding material.
- Fire resistance, with self extinction property.

Its shielding ability for neutron radiation is related to a high hydrogen content (for slowing down neutrons) and a high boron content (for absorbing neutrons). Source of hydrogen is organic matrix (resin) and alumine hydrate; source of boron is zinc borate. Atomic concentrations are equal to 5.10^{22} at/cm³ for hydrogen and 9.10^{20} at/cm³ for boron.

The manufacturing process of these materials is easy: it consists in mixing the fillers and pouring at ambient temperature. It allows to obtain any geometry.

Temperature resistance of these materials was evaluated by performing accelerated tests of samples at different temperatures (120°C to 170°C). According to tests results, the range of maximal temperature of use can reach 160°C.

Several shielding materials from TN International are now qualified for use in Germany, France, Belgium and other countries.

Modelling the aging of concrete as a technical barrier in nuclear waste disposal facilities (9-1834)

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Introduction

The low- and intermediate level wastes accumulating during the operation of nuclear power plants may be disposed of in an underground repository in the bedrock. The safety of the repository is ensured by multiple engineered barriers wherein the concrete-made containers usually play an important role. It is required that the barriers must be serviceable at least 500 years after sealing the repository. However, there exists neither experience nor any historical evidence of reinforced concrete structures, whose service life is even close to that demanded. The reinforced concrete has been used as a construction material about 150 years. Therefore, the designing and justification of structures of that kind have to be based on knowledge of fundamental degradation mechanisms of reinforced concrete under such conditions. This also emphasized the importance to develop a mathematical model by which the influence and interaction of recognized degradation mechanisms can be estimated.

Scope of the work

The primary object of the study is to develop a generalized numerical model for estimating concrete degradation in final disposal conditions in Finnish bedrock. Specific requirements are that the numerical implementation of the model has to be made by using the finite element method and a computer program that is commonly available. The mechanisms recognized and studied were: aerial carbonation of concrete, moisture ingress, chloride penetration, concrete corrosion caused by both sulfate and magnesium intrusion and leaching of cement paste compounds into ground water. In addition, the effects of concrete admixtures (herein silica fume and blast furnace slag) were included into the FEM-model. The main objectives of the FEM-based modeling are to make it possible to consider different geometrical shapes and possible variation in material properties of concrete structures with a reasonable accuracy.

Essential results

In this phase a conceptual structure of the model has been created and the model has been tested by two-dimensional numerical analyses made for some details of nuclear waste disposal facilities under consideration in Finland. In the analyses, the coupling between all the relevant deterioration mechanisms was considered. The results have been compared with values obtained from experiments and conventional calculation methods.

The results clearly demonstrate the effects of different types of concrete mixes on deterioration of reinforced concrete structures in studied environment. A low water-to-binder ratio retards the ingress of harmful compounds and delay corrosion initiation. Using of silica fume or blast furnace slag as a replacement of cement has beneficial effects on the durability of reinforced concrete as well.

The results received have already confirmed the initiating hypothesis of the study that the long-term deterioration of reinforced concrete cannot be estimated with a sufficient accuracy by conventional single phenomenon models, but the interaction between different deterioration phenomena shall be considered. The ingress of harmful compounds into concrete depends on compound specific diffusivities. The results received indicate that the interaction have a significant and time-depended effect on the diffusivities that also affect the estimated penetration depths of compounds in long term. Therefore, the benefits of the method developed and its numerical implementation are obvious for safety analyses of nuclear waste disposal facilities.

Numerical and experimental structural assessment of a half scale model of a nuclear spent fuel elements transportation package under 9 m drop tests (9-1927)

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The applied qualification requirements for the packages used in the transportation of nuclear spent fuel elements are very severe due the nature of the radioactive content. They include the so-called normal conditions of transport and the hypothetical accident conditions.

The 9 m drop tests are the most critical hypothetical accident conditions. The package qualification under these conditions shall be conducted using full scale models (prototypes), small scale models, numerical simulations and a combination of physical tests and numerical simulations. The choice of the qualification approach depends on economical and safety aspects.

To comply with the nuclear safety functions, as the containment of the internal products and biological shielding, the package itself has several components connected to each other in different ways (welded parts, flanged connections, surface contacts, etc.).

This paper presents a discussion on the combination approach with tests and numerical simulations for the structural assessment of a half scale model of a package for transportation of nuclear research reactor spent fuel elements under 9 m drop tests.

The numerical simulations of the 9 m free drops over a rigid surface of half scale model of the transportation package under different orientations were conducted using a finite element explicit code considering several nonlinear aspects as the nonlinear materials models and properties, the different package materials stiffness, and the different types of the contacts between the package components and between the package and the rigid surface, including the friction in the contacts.

The tests were also conducted for different drop orientations. The half scale model under test was instrumented to assess the deceleration levels and all tests were recorded with pictures, high speed camera movies, size measurements and localized damage characterization.

The numerical and experimental results are compared and comments and conclusions are addressed based on the comparison. Also, some recommendations are issued on the use of the numerical simulations for the full scale tests of the package.

Technical challenges related to the spent nuclear fuel dry cask storage/transportation analysis and design (9-1936)

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Keywords: spent nuclear fuel, dry cask storage, storage and transportation, technical challenges

With the introduction of new, and highly sophisticated spent fuel dry cask storage and transportation designs involving three-dimensional non-linear dynamic finite element analyses, using codes such as: ANSYS, LS-DYNA, ABAQUS and others, industry as well as the U. S. Nuclear Regulatory Commission (NRC) is faced with unique technical challenges for which there are few precedents.

Licensees, spent nuclear fuel dry cask storage and transportation vendors, and, in turn, the NRC are managing the potential application of: 1) exotic and unconventional non-code approved materials; 2) increased mass and fissile material loads; 3) increased number of fuel assemblies for both, boiling water reactor (BWR) and pressurized water reactor (PWR) fuels; 4) burn-up credit; 5) fracture mechanics analytical approach; 6) loading high burn-up fuels; and 7) analysis vs. testing, etc. Addressing these design considerations has created new technical challenges to demonstrate that the integrity of the fuel assemblies is maintained during the storage and/or transportation phase, and to demonstrate compliance with the applicable requirements of Title 10 of the Code of Federal Regulations (10 CFR) Part 72, and 10 CFR Part 71 respectively.

This paper will discuss, in general, these and other issues from an overall perspective. This paper will also elaborate, in detail, some of the issues related to analysis vs. testing, and the merits and pitfalls of full-scale vs. scaled model testing used to demonstrate the structural integrity of the cask and its contents. I will also briefly discuss pre-test and post-test calculations, validation and benchmarking of analyses based on drop tests, test positions, test sequences and other state-of-the-art measurement techniques to measure the decelerations, and other relevant issues.

Optical strain measurement of plastic strain localization in nuclear waste copper canisters (9-2012)

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Keywords: copper, weld, plastic strain, localization

In Finland and Sweden the spent nuclear fuel will be stored in a deep repository in copper corrosion barrier canisters surrounding cast iron inserts. The 50 mm thick copper canisters will be sealed using either electron beam welding (EB) or friction stir welding (FSW) to join the tubes and the lids/bottoms. The canisters will deform in the repository conditions e.g. due to hydrostatic pressure. The deformation will localize to different discontinuities, such as defects and microstructural heterogeneity.

This study compared the localization of plastic deformation in EB and FSW welds as well as in the base materials (both forged and extruded) using optical strain measurement methods. The results show that in the base materials the deformation occurs very uniformly over the entire gauge length. In FSW welds the deformation localizes in the middle of the weld, however, the tensile strength is similar to that of the base materials. In EB welds the deformation localizes to the large grains in the middle of the weld and to the steep grain size gradient between the weld and the base material. Tensile strength is lowest in the EB welds (175 MPa as compared to 200 MPa or higher for the other samples).

Planning of one-piece removal of BWR reactor pressure vessels at Barsebäck Nuclear Power Plant, Unit 1 & 2 (9-2571)

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The decommissioned Barsebäck Nuclear Power Plant (NPP) with its two units Barsebäck 1 and 2 is situated on the Swedish west coast in the rural district of Kävlinge. The building process of Barsebäck 1 was started in 1970 and the plant was in commercial service in 1975. Barsebäck 2 was in service two years later. The two reactors are in principal identical and of BWR-type (Boiling Water Reactor) delivered by ASEA-Atom with the capacity of 615 MW each. The Barsebäck NPP units 1 and 2 were decommissioned through governmental decisions November 30 1999 and May 31 2005 respectively.

Even if the shutdown dates for the two units differ, it is probable that the dismantling of Barsebäck 1 and 2 will be conducted as a joint project. This, according to the present time-table, will mean that a pilot project will be initiated around 2015 and dismantling of Barsebäck would start sometime around 2020. At the moment, extensive planning activities are carried out. One major part of the demolition of the plant is the dismantling of the reactor pressure vessels (RPVs) and its internals. There exist two major optional strategies for dismantling of a RPV and its internals;

- Segmentation and cutup of the RPV and the RPV internals at plant.
- One-piece removal of the whole RPV, with or without the RPV internals.

The study presented here covers the examination and planning of a one-piece removal of the RPVs at Barsebäck Unit 1 & 2, including the RPV internals.

In Sweden, it is the responsibility of the plant owner to plan and carry out the dismantling of the decommissioned NPP, and to restore the site. Management, transportation, intermediate storage and disposal of spent fuel and radioactive waste from the Swedish nuclear power plants is, on the other hand, the responsibility of the Swedish Nuclear Fuel and Waste Management Co. There is at the moment two ongoing investigations in Sweden regarding the one-piece removal of RPVs. The first one is initiated by Barsebäck Kraft AB, and carried out by Scanscot Technology AB, regarding the detailed planning of the one-piece removal of the two RPVs at the decommissioned Barsebäck NPP, as a basis

for a future management decision regarding what strategy to apply for dismantling of the RPVs and its internals. The scope of this project covers the removal of the RPV from the reactor containment, and its transportation to the quay at site, for further transportation by boat. The second project is managed by the Swedish Nuclear Fuel and Waste Management Co to from a more general perspective investigate the possibility and cost for one-piece removal of RPVs in Sweden, including the transportation and storage of the RPVs.

In a pilot study [1], six different alternatives for one-piece removal were studied, and the pros and cons listed for each of them. The outcome of the pilot study was an identification of three methods as the most promising ones (see Figure 1);

- a) Lifting with a crane.
- b) Removal in horizontal position using tower gantry.
- c) Lowering inside the reactor containment.

These techniques has now been studied in detail, including technical lifting aspects, necessary building modifications and demolition, and radiological consequences, all ending up in cost estimations for each of the three alternatives ([2], [3]). Based on the outcome of this study, the preferred one-piece removal technique will be identified.

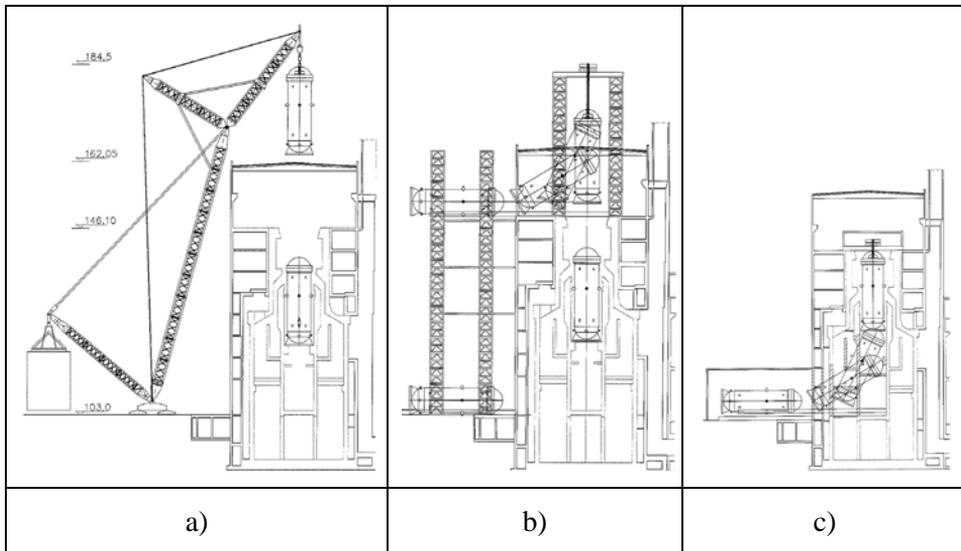


Figure 1. Decommissioned Barsebäck NPP Unit 1 and 2. Investigated reactor pressure vessel one-piece removal techniques (taken from [2]); a) Lifting with crane, b) Removal in horizontal position using tower gantry, c) Lowering inside the reactor containment.

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Stability analysis of storage of spent fuel in stack of trays in pool (9-2604)

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The spent fuel bundles from PHWRs are stored under water in spent fuel storage facilities in stacks of 20 trays inside the water depth of 6–8 meters.

All the safety systems, structures and components of storage facility are required to meet OBE level of earthquake (0.1 g). A stability analysis of stack of trays of fuel bundles in seismic condition has been carried out for 20, 25 & 30 trays using transient FEM analysis based on time history of pool and discussed in this report. These results are being verified with shake table tests using dummy fuel bundles.

A finite element model of stack of spent fuel trays has been developed to represent the behavior under seismic condition. The models are created for each individual tray and spent fuel bundles are modeled as a lumped mass in tray FE model. The interlock between trays has been simulated using combination (i.e. spring + Gap) element. The contact element has been used in FE model to capture the lifting and sliding motion between trays and the same has been used for contact between resting surface and bottom most tray of stack. A time history analysis was performed for stack of 20, 25, and 30 trays with different friction of coefficient (0.1, 0.2 & 0.3) to verify its overall stability against turning and sliding under seismic event. These stacks of trays are found stable under the designed level of earthquake.

10. Challenges of New Reactors

Design and construction programs for advanced, generation IV and fusion reactors. Performance and behavior of metallic and non-metallic materials and components in high temperature environment. Thermal hydromechanics. Regulatory framework, safety research and development of codes and standards. Passive safety features. Proliferation-resistant design.

The challenge of nuclear reactor structural materials for Generation IV Nuclear Energy Systems (10-1586)

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Generation IV Nuclear Energy Systems have been studied in the framework of the “Generation IV” International Forum, there can be little doubt that nuclear reactor structural materials technology, including: Selection, development and qualification, is one of the key issues to success of Gen IV Nuclear Energy Systems. By striving to meet this challenge, which is beyond the experience of the current nuclear power plants, the belt zone structural materials are required good resistance to irradiation damage, high thermal stress capacity, excellent resistance encompassing stress-corrosion cracking, and highly predictable responses to extreme levels, compatibility with Heat-Transfer media and other materials, very long-term stability enhanced in the system, adequate resources and easy fabrication as well as weld-ability, etc, to guarantee the security and integrity of the pressure boundary. This paper analyses and compares several materials under active consideration performance requirements, and also discusses candidate materials for use in different reactor components, which include various ferritic-martensitic steels, advanced austenitic stainless steels, nickel-base super-alloys, oxide-dispersion strengthened alloys, refractory alloys and etc. It is demonstrated that new materials, such as metals, carbides, nitrides, oxide, novel alloy, solid solutions or composites focuses on higher stability and better mechanical performances. In addition, the emergence of new international structural materials initiatives and their potential roles will be described.

Design considerations for developing a steam generator for integral modular reactor SMART (10-1589)

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SMART (System-integrated Modular Advanced Reactor) is a small sized integral type PWR with a sensible mixture of new innovative design features and proven technologies aimed at highly enhanced safety and improved economics. SMART is applicable to an energy source for electricity generation and seawater desalination, promising a new era of nuclear energy utilization only with small-sized reactors.

The steam generator is one of the key and unique components to attain the arrangement of integral feature. SMART has eight identical SG cassettes located in the annular space between the reactor pressure vessel and the core support barrel. Each steam generator cassette is of once-through design with a number of helically coiled tubes where the feed-water flows upward producing superheated steam, and the primary reactor coolant flows downward in the shell side. The steam generator is also used as the heat exchanger for the passive residual heat removal system.

The design of the steam generator cassette shall meet the requirements for thermal hydraulic performance, arrangement and size, water chemistry and materials, structural integrity including in-service inspection, operation and maintenance, and flow induced vibration.

In this paper, the design requirements developed for SMART steam generator cassette are then introduced, and the various designs of the SMART steam generator along the development stage are summarized. Also the design details, rationale, and some technically unresolved issues with their remedial ideas are given and reviewed. The unresolved issues given in this paper seem to be common to the integral reactor, thus their remedial ideas can be useful for the development of a modular type steam generator cassette for the integral reactor.

Dynamic analysis methodology for stacked graphite fuel blocks of a VHTR using a commercial structural analysis code (10-1678)

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The gas-cooled reactor systems which are currently considered as new candidates for a very high temperature heat source for industries and hydrogen production plants have many columns of stacked graphite blocks such as reflector blocks, fuel blocks, and core support blocks in its reactor internal structures. An earthquake loading on the stacked block columns causes rocking responses of the blocks and solid impacts between them, and may lead to structural integrity problems, because the blocks are not fully constrained and have gaps between neighboring blocks.

The dynamic analysis of block structures has a long history. In the historically early stages of the structural and dynamic analysis of the stacked graphite fuel blocks, the special computers of high computing power with the dedicated computer programs were needed for the analyses to make short the computational time and reduce the cost. In 1975, T.H. Lee presented a methodology for analyzing the nonlinear response of a column of stacked prismatic fuel blocks for GT-MHR [1]. In 1979 T. Ikushima and T. Nakazawa presented their work results on a seismic analysis of a column of stacked prismatic fuel blocks for HTTR [2]. They made and used the dedicated computer programs for the analyses. Figure 1 shows the schematics of the models they developed.

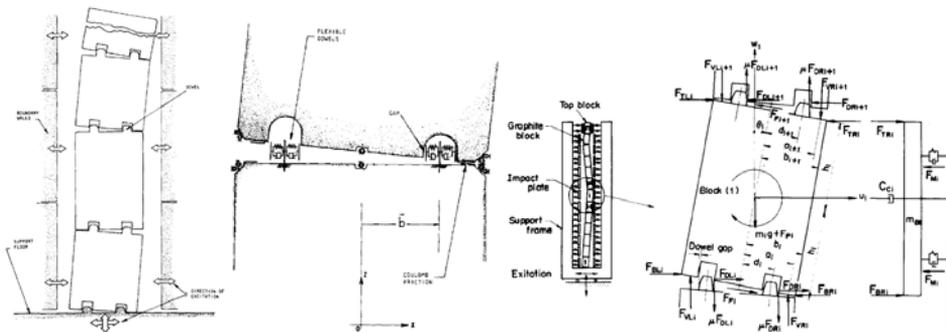


Figure 1. Schematics of the classical models for the stacked graphite fuel blocks.

At the present days, the computing power of personal computers has been remarkably increased and the commercial codes for structural analyses provide

many useful modeling procedures and analysis options. The purpose of this paper is to introduce the finite element models and the dynamic analysis examples of the stacked graphite fuel blocks of a prismatic type gas-cooled reactor performed on a personal computer using the commercial structural analysis code, ABAQUS. Figure 2 and Figure 2 show examples for the proposed finite element models and the seismic response results of the single and multi stacked graphite blocks.

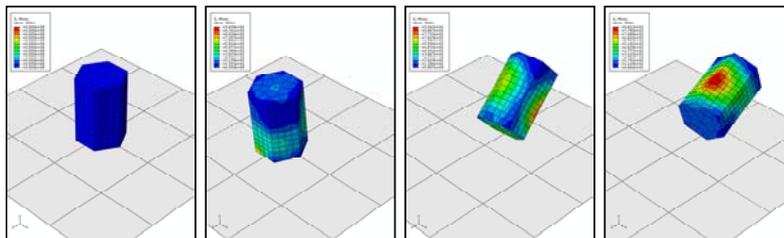


Figure 2. A FE model and the dynamic response of a single simple block on a moving floor.

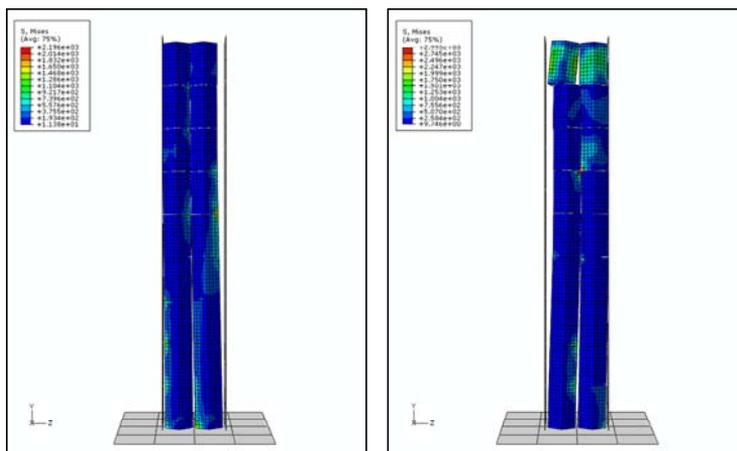


Figure 3. A FE model and the dynamic response of two columns of multi stacked blocks on a moving floor with vertical walls.

Few attentions have been paid to the dynamic analysis methodology for the stacked block structures using commercial codes and useful example studies reported are rare. This study shows that the commercial code can be very useful for the dynamic analysis of a reactor core internal structure consisting of the stacked graphite blocks.

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RCC-MR 07 code: specificities and recent developments (10-1686)

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The fourth edition of RCC-MR code has been issued on October 2007 by AFCEN (Association Française pour les règles de Conception et de Construction des Matériels des Chaudières Electro-nucléaires) and results of an important work by AREVA NP and CEA to develop and improve design and construction rules of the previous edition.

The improvements and new developments added in the new version of the RCC-MR differ from the last 2002 edition by an enlargement of the scope of the code not only applicable to mechanical equipments in fast breeder reactors working at high temperatures but also to the ITER vacuum vessel, and other nuclear components.

The last evolutions of the code are summed up as follow:

1. In the field of design rules, improvement of defect assessment rules and the creep-fatigue rules for shells and pipes, and extension of the scope of the code concerning bolts,
2. In relation with the development program of the ITER vacuum vessel, introduction of a new quality class for the box type structures and a specific appendix dealing with fabrication requirements of the ITER vacuum vessel,
3. Modification or addition of requirements in accordance with the European Pressure Equipment Directive with in particular the replacement of French standards by European ones and as far as possible by harmonized European standards.

FE analysis of ITER 40° vacuum vessel sector and stress assessment according to French nuclear code RCC-MR (10-1705)

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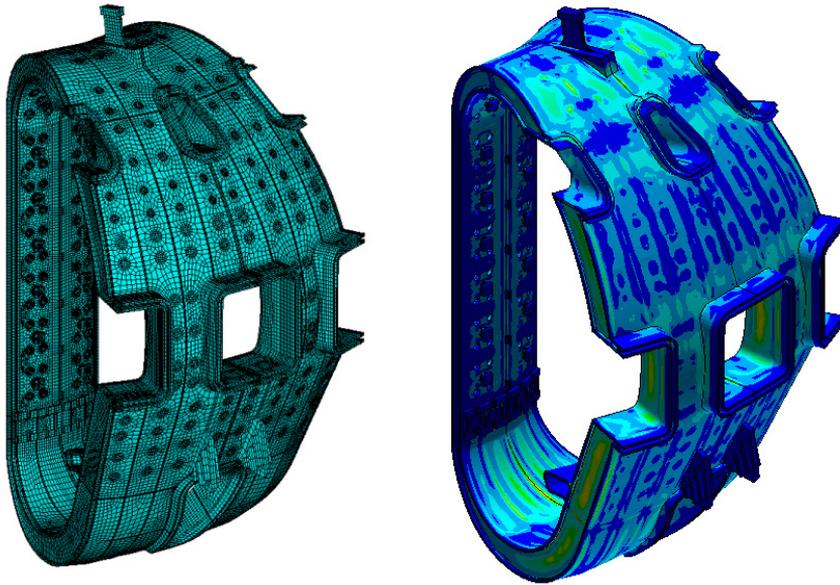
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The progressive increase in computational resources and the recent extensive Finite Element Method (FEM) development has enabled the use of these codes in the mechanical calculation of pressure vessel and equipment (DBA, Design By Analysis) and in nuclear calculation codes due to the high level of accuracy and reliability. The analysis procedures used for the checking and treatment of stresses are very similar for the most widely used Nuclear codes such as are described in ASME Sec. III and RCC-MR, version 2007, which is the code used in the ITER Vacuum Vessel. A linear analysis of the model is used in the FE stress assessment technique and a further stress check is carried out based on stress categorization, which is required for a further assessment by comparison with allowable limits. This is a complex and non-intuitive process, especially so in the current analysis since this procedure is more complex compared to standard pressure equipment due to the unusual geometry, which includes many gross structural discontinuities and zones with potentially high stress concentration effects. To assist in the process of separation of the membrane, bending and peak stresses, “ad-hoc” tools are programmed in the FEM software (ANSYS). In order to overcome the difficulties caused by the stress categorization of secondary stresses, which in RCC-MR, RB-3224.35 are defined as “the fraction of the total stress which can disappear as a result of small scale permanent deformation minus the peak stress”, sub-modelling techniques with elastoplastic material behaviour laws are used to accurately quantify when the zones overstressed are small enough to disappear as a consequence of localized plastic permanent deformations so that the stresses may be considered as secondary. This paper presents an approach based on the construction of a detailed 3D

model of the ITER Vacuum Vessel and its subsequent FEM analysis, utilising the RCC-MR Ed. 2007 code.

Considering that new reactors have a complex geometry which is in some cases very different to conventional reactors (pressurized double shell, housings, vacuum inside...), the new analysis strategies allow facing the mechanical design problems derived from these complex geometries.



A high temperature gas loop to simulate VHTR and nuclear hydrogen production system (10-1870)

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Introduction

Very high temperature gas cooled technology and nuclear hydrogen production technology are being developed in KAERI for the nuclear hydrogen production system. The outlet temperature of the high temperature gas cooled reactor developed so far ranges 750 to 900°C. However, to produce hydrogen with economical efficiency, the coolant outlet temperature of VHTR should exceed 950°C. In this paper, a development of small scale gas loop to investigate coupling of the VHTR and hydrogen production system is introduced. A process heat exchanger developed which connects VHTR and hydrogen production system is tested in the gas loop. The whole process of gas loop development from the design to the construction is introduced. Also, some of the test result of heat exchanger structural integrity analysis and test is discussed.

Small nitrogen gas loop

Primary loop

The primary loop simulates VHTR heating system. The design pressure and the design temperature of the primary loop are 6 MPa and 1000°C respectively. The process gas of the primary loop is nitrogen gas. Primary loop consists of pre-heater, main heater, hot gas duct, gas circulator, and isolation valves. Most of the pipe is thermally insulated to prevent over heating at the outer pipe structure.

Secondary loop

The secondary loop simulates the sulfuric acid hydrogen production process. A sulfuric acid (H₂SO₄) loop is an open loop and consists of a H₂SO₄ storage tank, a H₂SO₄ feed pump, a pre-heater, a heat exchanger (evaporator), a PHE, a separator, a SO₂ trap, and a H₂SO₄ collector (Figure 1). Cold 98% H₂SO₄ is

superheated to 500°C. In a superheating process, H_2SO_4 decomposes into H_2SO_4 and SO_3 . In the PHE, some fraction of the SO_3 is dissolved into SO_3 and O_2 . The toxic SO_3 is separated in the separator and the sulfur dioxide in the mixture gas is removed in the NaOH trapping system.

Process heat exchanger

Process heat exchanger is a coupling components between VHTR system and nuclear hydrogen system. A concept has been developed and structural analysis utilizing finite element analysis has been done before the experiment. The finite element analysis results have shown that the process heat exchanger can withstand a considerable amount of pressure difference between loops. This heat exchanger was tested by circulating sulfur-acid in the secondary flow channel.

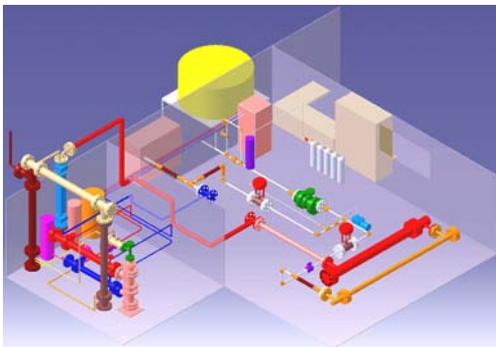


Figure 1. 3D Gas Loop Model.

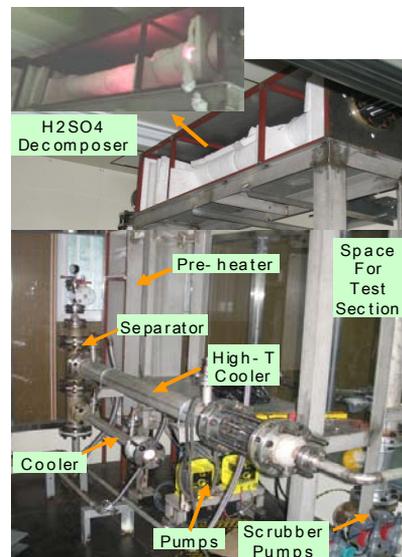


Figure 2. Gas Loop of secondary side.

Summary

Small scale nitrogen gas loop was designed and constructed for the test of process heat exchanger at the elevated temperature. The development process including the design and the fabrication of the test loop is presented. Process heat exchanger was tested in sulfuric acid environment. As a next step, middle size He loop is being constructed in KAERI from 2009.

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Impact of engineered safety features on AHWR containment (10-1881)

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The proposed Advanced Heavy Water Reactor (AHWR) employs a double containment envelope viz, a primary and secondary containment, which together act as last physical barrier to the release of radioactivity following an accident. The primary containment is completely surrounded by the secondary containment.

As a part of the Probabilistic Safety Assessment (PSA), it was required to perform studies pertaining to effect of Engineered Safety Features (ESFs) on containment pressure, temperature and flows and leakages from the AHWR containment.

The containment building of AHWR is proposed to provide with Engineered Safety Features (ESFs) such as Containment coolers, Primary containment controlled discharge (PCCD), Primary containment filtration & pump back system (PCFPB) and Secondary containment filtration, recirculation and purge system (SCFRP). ESFs are provided for post-accident management such as containment clean up, containment depressurization etc. to minimize the amount of activity release to the environment during an accident. Activity release consists of ground and stack release and it depends upon long term containment pressure, temperature transients and activity concentration in various volumes of the containment. The containment pressure and temperature depends upon the discharge rate of high enthalpy fluid into the containment. Containment pressure is also influenced by leakage rate from the containment, steam condensation on containment walls and status of ESFs as well. To study the effect of ESFs on flows and leakages from the AHWR containment, an analysis was carried out based on availability and non availability of various ESFs.

Thermal hydraulic transient analysis was carried out using in-house containment thermal hydraulics code 'CONTRAN'. Modules for simulating the engineered safety features were incorporated with this code. Deposition of aerosols in containment volumes were calculated using NAUA code. For the purpose of analysis, a three volume configuration of entire AHWR containment with Gravity Driven Water Pool (GDWP) is considered. The primary containment is divided into two accident based volumes viz. V1 volume and V2 volume housing the high enthalpy and low enthalpy systems respectively. The region between primary and secondary containment is considered as third volume.

This paper presents the results of analysis of a postulated LOCA case, 200% RIH break with failure of shutdown systems (1 & 2). Blowdown mass and

energy discharged into containment were given as input to CONTRAN code for estimating the pressure, temperature, junction flow rates etc. in all volumes for 72 hrs, which were then supplied, along with activity release rates, as inputs to NAUA code for calculating deposition rate in the containment volumes. Finally, the blowdown mass, energy data and activity deposition rate (calculated using NAUA code) were then given as inputs to CONTRAN code and recalculated the pressure, temperature, activity concentration in all volumes and leakages from containment volumes.

Analysis was carried out for a number of cases, postulated based on availability/unavailability of ESFs. Activity released out of containment were obtained for all the cases in terms of ground level, stack level and total activity release from containment for 72 hours from the initiation of the accident.

This paper highlights the importance of operation of ESF in reducing the activity release to the environment.

Preliminary analysis of the structural effects due to dynamic loads of the isolated next generation lead cooled reactor (10-1887)

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The main purpose of this preliminary study deals with the evaluation of the structural effects due to the dynamic loads exerted and propagated through the lead coolant during a safety shut down earthquake with reference, as an example, to the isolated ELSY system configuration (CEE-7 Framework Project).

Seismic base isolation is increasingly used to protect structures and their contents against dangerous ground motions as well as mitigate the structural effects, on the internals walls and reactor components of the induced dynamic load and of the coupling between coolant and vessel.

An adequate predictive numerical modelling, by means a 3-D finite element model, was set up and a non-linear approach was used for the foreseen structural preliminary analyses and simulations of the plant and internals behaviours, in order to describe the interactions among the different subsystems.

Moreover the fluid-structure interaction problem, due to the high density of the retained primary coolant, has received a particular attention in relation to the possible hydrodynamic interaction, between lead and the surrounding internals, as well as the sloshing wave motion (the lead may be accelerated and can impact on the structures walls) that may significantly influence the stress level in the reactor pressure vessel (RPV). As for the seismic analysis, the isolation systems may influence the seismic capacity of as-built structure to reduce the intensity of the propagated seismic loads.

Numerical results are presented and discussed highlighting the importance of the fluid-structure interaction effects as well as the isolation technique effectiveness, which is expected to be effective in raising the reliability of internals and vessel structures, during an earthquake event.

Radiotoxicity perspectives for different ELSY working hypotheses: towards a sustainable fuel cycle (10-1905)

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The acceptability of nuclear energy in part arises from the capability of handling its own wastes until a final safe disposal. In particular, the spent fuel represents the most problematic waste to be managed: the presence of Long Lived Radioisotopes (i.e. Minor Actinides) imposes binding restrictions regarding the waste management and its disposal in geological repositories.

Therefore, the radiologic load of the spent fuel is one of the main drawbacks of nuclear energy production.

In the present climate of nuclear renaissance new solutions have to be pursued in order to guarantee the full sustainability of the nuclear fuel cycle. Within such context, the formalization of the Generation-IV prescriptions is intended to guide the design of new concept reactors capable to overcome the drawbacks of the present generation systems, mainly by implementing the fast spectrum features.

According to this, the present study aims at demonstrating the possibility to reduce the spent fuel radiotoxicity, by analyzing different fuel cycle hypotheses for the European Lead-cooled System (ELSY), the 600 MWe Generation-IV Lead cooled Fast Reactor under investigation in Europe within the 6th EURATOM Framework Programme.

The up to date ELSY reference configuration (characterized by square, wrapper-less Fuel Assemblies) has been modeled and some realistic operating conditions have been then simulated using the MCNPX transport code.

A standard operation for a core designed to achieve unitary Breeding Ratio was studied at first. Two different desirable operative conditions are then analyzed, concerning the extension of the burn up to 100 GWd/t and the adoption of the “adiabatic” core concept. The latter configuration refers to a system where the burn up of Minor Actinides equals their build up by loading the respective equilibrium concentrations in the fresh fuel.

For each of the conditions indicated the wastes radiologic load has been calculated by assuming a typical reprocessing upon a 2 years aged spent fuel with 0.1% losses. The evolution of the resulting waste radiotoxicity (evaluated using ICRP72 coefficients) has been then analyzed and compared with the one of present Generation-II reactors spent nuclear fuel (i.e. typical 1000 MWe PWR).

A preliminary sensitivity analysis of the results has been also performed by adopting different burn up codes (CINDER90 in MCNPX v.2.6.0, ORIGEN v.2.2 with MCNP5 and ERANOS) and cross section libraries (JEFF3.1 and ENDF/B-VII).

A comparative analysis upon the collected data showed a significant reduction in the long term activity and the corresponding radiotoxicity, suggesting the viability of the adiabatic core configuration as the most interesting solution for an actual reduction of both the High Level Waste volumes and the associated radiotoxicity.

Material challenges of the new advanced gas cooled systems (10-2038)

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Introduction/background

For new Generation IV gas cooled systems such as the very high temperature reactor (VHTR) and gas cooled fast reactor (GCFR) industrially established materials are at or near their limits in certain applications and new materials and/or an understanding of the behaviour of existing materials under more severe environmental reactor conditions will be required. For metals, ODS alloys, ferritic-martensitic steels, and additional super-alloys offer some solutions but require, significant R & D in terms of their properties and behaviour under component conditions. Gas Cooled Generation IV systems may also require the deployment of non-metallic materials (e.g. high-temperature fibrous insulation in the reactor or power conversion systems, composite materials as alternatives to metals for in-core components, ceramic materials for use at high temperatures (950–1000°C) for components such as the Heat exchanger. Such materials are difficult to bond, pressure form, and machine. The codification and understanding of the behaviour of these materials with respect to design and operation is very much in its infancy.

Purpose and results

Previous programme details and intermediate results have been presented for the HTR system [1]. This paper reviews the present understanding and the progress made in interpreting and establishing property and behavioural information on metallic and carbon based materials for the VHTR and GCFR systems in the RAPHAEL and EXTREMAT EU 6th Framework projects. The work will summarize the main findings of the experimental programmes undertaken, including tests to develop Mod 9Cr 1Mo steel as a vessel steel, irradiation tests on graphite for the VHTR core, progress in understanding composites and ceramics for key component applications. Results will be presented for joints and welds, irradiation and corrosion and microstructural modeling. Cross cut information from similar situations in the Fusion reactor will also be included.

Summary/conclusions

The work will conclude on the main achievement so far and on the needs in terms of future R & D activities in Europe for these systems. The recommendations will take into account the expected input of information from the Generation IV International programmes and from ongoing national programmes.

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The stress assessment of reflector graphite bricks in high-temperature gas-cooled reactor (10-2044)

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This article briefly describes the structural arrangement and features of the graphite internals of the 10 MW high-temperature gas-cooled reactor-test module (HTR-10) before the stress and deformation of the side reflector graphite bricks are analyzed under normal and accidental conditions for the whole lifetime. The maximal stress and deformation of the graphite bricks are caused by the fast neutron irradiation and the high temperature. The safety assessment of the stresses in the graphite bricks has carried out and the deformation can not affect the graphite internals function, it can be drawn that the stress and deformation of the graphite bricks will not affect the reactor safety during the 20 years lifetime.

Study on steam generator helical tube integrity assessment of HTR (10-2051)

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In a nuclear power plant, the steam generator (SG) heat transfer tubes account for almost 80% of the primary loop pressure retaining boundary. However, they are also the weakest link in the loop.

In pressurized water reactor (PWR), heat transfer tube rupture can result in the bypass of the containment, which will provide radioactive fission materials in the primary loop coolant a direct pathway to communicate with atmosphere, resulting in a LOCA (loss of coolant accident). Multiple tube ruptures could lead to a meltdown of the nuclear reactor.

In high temperature gas-cooled reactor (HTR) with steam cycle, steam generator connects and isolates the primary radioactive helium loop from the secondary non-radioactive water and steam loop. The pressure of the secondary side is higher than the helium pressure of the primary side. Heat transfer tube rupture will cause water and steam in the SG to flow into the primary loop, mixing with the helium flow in the reactor core. The increase of steam concentration in the pebble bed enhances the reactor core moderation, which introduces positive reactivity and increases the reactor power and temperature. When a large amount of steam leaks into the primary loop, the iodine deposit on the steam generator is washed into helium coolant and the incoming steam reacts with the fuel elements and the graphite components at high temperature, which produces water gas. This directly affects the integrity of the graphite components, which release radioactivity to the environment. The expanding gas in the superheated reactor core will increase the primary loop pressure. When the pressure in the reactor exceeds the set pressure of the safety valve in the pressure relief system of the primary loop, the safety valve will open and release radioactive products to the reactor building and the atmosphere. In reference [1], the analysis of a double-ended guillotine break (DEGB) of a HTR-10 SG tube showed that the safety valves in the pressure relief system will open and discharge a gas volume equal to 1/4 of the total helium inventory in the primary loop. Therefore, the integrity of SG tube is also very important to the safety of HTR.

In nuclear industry, mainly two kinds of steam generator are used, that is U-tube steam generator and helical coil tube steam generator. The former is used in PWR with loop-type layout reactor coolant system, such as AP1000 and EPR, while the later finds its application in HTR, such as HTR-10, and PWR with integral reactor coolant system, such as SMART and IRIS. The HTR-10 SG heat

transfer tube is a typical helical coil tube. This paper takes the helical tube of HTR-10 steam generator as an example to present some research results on the tube integrity assessment of the helical coil tube steam generator.

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Researches for development of regional energy reactor, REX-10 (10-2070)

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Recently, today's global pattern of energy supply is not sustainable. The provision of affordable energy services is a fundamental prerequisite for economic growth and development. In addition, the environmental problems such as energy crisis, global warming and acid rain are issued and more demand for reliable electricity supply increases. As one of the realistic solutions, the extension of the peaceful uses of nuclear energy has been suggested. Small and medium nuclear reactors with non-electric applications arise as an alternative energy source. The main non-electric applications are defined as district heating, desalination (of sea, brackish and waste water), industrial heat supply, ship propulsion and the energy supply for spacecraft. The small and medium reactors are under development in several advanced countries with goals of low capital costs, short construction periods, high performance and enhanced safety. RERI (Regional Energy Research Institute for Next Generation) introduces a new paradigm for energy supply system. REX-10 which is stable small nuclear reactor with thorium fuel for power generation and nuclear district heating has been developed. The design objectives of REX-10 are inherent safety, non-proliferation and economical efficiency. The newly designed REX-10 has been developed to maintain system safety in order to be placed in a densely populated region, island, etc. For high safety, natural circulation, pool-type vessel and low operation pressure are introduced. In addition, the thorium fuel cycle with 20 year lifetime without exchanging fuel is considered for the sake of the non-proliferation. Moreover, the economical efficiency is ensured by the unmanned automatic control. The system pressure and capacity are determined properly for regional energy reactor. The operation pressure is 2.0 MPa and the thermal power is 10 MWth. The major research activities for REX-10 design are natural circulation, steam-gas pressurizer and thorium fuel cycle.

Summary of SMIRT20 preconference topical workshop – Identifying structural issues in advanced reactors (10-2504)

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The Idaho National Laboratory (INL, USA) and IASMiRT sponsored an international forum Nov 5–6, 2008 in Porvoo, Finland for nuclear industry, academic, and regulatory representatives to identify structural issues in current and future advanced reactor design, especially for extreme conditions and external threats. The purpose of this Topical Workshop was to articulate research, engineering, and regulatory Code development needs. The topics addressed by the Workshop were selected to address critical industry needs specific to advanced reactor structures that have long lead times and can be the subject of future SMiRT technical sessions. The topics were; 1) structural/materials needs for extreme conditions and external threats in contemporary (Gen. III) and future (Gen. IV and NGNP) advanced reactors and 2) calibrating simulation software and methods that address topic 1. The workshop discussions and research needs identified are presented.

The Workshop successfully produced interactive discussion on the two topics resulting in a list of research and technology needs. It is recommended that IASMiRT communicate the results of the discussion to industry and researchers to encourage new ideas and projects. In addition, opportunities exist to retrieve research reports and information that currently exists, and encourage more international cooperation and collaboration. It is recommended that IASMiRT continue with an off-year workshop series on select topics.

Development of high efficiency and high capacity gas/gas heat exchanger for gas-cooled reactors (10-2535)

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Helium gas cooled reactors are candidates for future main power supply from viewpoints of economical efficiency, safety, operability and multi-purpose heat utilization. For the practical application of gas cooled reactor, high efficiency and high capacity (600 ~ 1000 MW) gas/gas heat exchanger is one of the most important equipments. The compactness and the performance upgrading of the equipment determine the feasibility (success or failure) of the whole plant. The high surface area to unit volume ratios of a plate-fin type heat exchanger mean that it is about twenty times smaller than the equivalent tubular heat exchanger. Problems of severe service condition such as high temperature (650 ~ 950°C) and high pressure (~ 7 MPa), however, have not been solved. In order to overcome these problems, a development program of the high capacity compact gas/gas heat exchanger has been carried out.

High heat exchange volume density is necessary for economical low-cost of the heat exchanger. Narrow pitch fin is expected to give a plate-fin type heat exchanger high heat exchange volume density and high structural strength.

The result of our study is summarized as follows.

- 1) The trial manufactures of the plate-fin type heat exchanger of the large scale structure body with narrow 1.2 mm pitch fin from Hastelloy-X, which is Ni-based alloy and can be used at 850°C and above, and from stainless steel 316 were carried out for the first time in the world.
- 2) The strength characteristics of the ultra plate-fins became clear through the various element examinations and structure examinations. Moreover a practical stress analysis method was investigated on the basis of the homogenization approach. A data-base for the strength design including the fundamental deformation of the ultra offset plate-fins structure was developed.

- 3) Heat transfer characteristics and pressure loss were evaluated experimentally, and heat transfer and flow analysis was practiced by simulating the structure of actual equipment. It was confirmed that the plate-fin type heat exchangers had the volume density more than 25 MW/m^3 by these results.
- 4) By studying and organizing design requirements of recuperators and intermediate heat exchangers of the planned HTGR plant, numerical evaluation of both the thermal performance and the structural strength was carried out and recuperators and intermediate heat exchangers for HTGR plant were designed.

Acknowledgments

This work shows the R&D results in 2004 to 2007 by Mitsubishi Heavy Industries in the program of “The Technology Development of High Efficiency and High Capacity Gas/Gas Heat Exchanger Necessary to the Practical Application of Gas-cooled Reactor”, which is entrusted from Ministry of Education, Culture, Sports, Science and Technology of Japan.

Computational assessment of the reactor vessel cooling options in a prismatic core VHTR (10-3139)

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The Very High Temperature Reactor (VHTR) has been selected for the Nuclear Hydrogen Development and Demonstration (NHDD) project. For the NHDD plant, the primary coolant inlet and outlet temperatures are considered to be 490 and 950°C, respectively. Due to its high operating temperature, the design of the reactor pressure vessel (RPV) is one of the important issues in the NHDD design. Both the SA508/533 steel and high-Cr steels (e.g. 9Cr-1Mo-V steels) are expected to be candidate materials for the VHTR pressure vessel. Because of its extensive experience base as an ASME Section III code-approved material for Light Water Reactor, the SA508/533 steel has emerged as a strong candidate for the RPV. In order to use this material, however, a design is needed to maintain the RPV temperature below the ASME code limit, which is 371°C during normal operation and 538°C for up to 1000 h during accident conditions.

In this study, three types of vessel cooling options for a prismatic core VHTR to keep the RPV temperature below the normal operating limit are suggested. In the first option, the coolant inlet flow is routed through riser channels in the permanent side reflector (PSR), which is a base configuration of all three designs. A vessel cooling system (VCS) supplying cold helium flow between the RPV and the core barrel is added to cool down the RPV in case that the RPV temperature is still higher than its limit. In the second option, external vessel cooling is introduced with the modified inlet flow configuration. The cooling fluid is air in the reactor cavity, outside of the RPV. Air blowers should be installed around the bottom side of the RPV. The last option is to use insulation material instead of the direct cooling of the RPV by internal cold helium flow or external air flow. The location of insulator can be either on the inner surface of the RPV or at the interface surface between the PSR and the core barrel.

The performances of the vessel cooling options are evaluated by using a system thermo-fluid analysis code, GAMMA+, and a commercial computational fluid dynamics code, CFX, during normal operation and accidents. The GAMMA+ model includes the reactor coolant system, the reactor cavity, the passive Reactor Cavity Cooling System (RCCS), and the VCS. The CFX code was used to model in more detail a 1/54 sector corresponding to the region associated with a single PSR riser channel, extending in the radial direction from

the PSR to the RCCS downcomer wall. According to the results, the modified inlet flow configuration with the VCS flow provides the most viable results. The external cooling option does not ensure an effective cooling of the RPV. The insulation option provides an effective temperature reduction of the RPV but needs careful consideration in a view of the fuel safety margin during accidents.

Generation IV material issues – case SCWR (10-3143)

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Generation IV nuclear power concepts have become an active research topic all over the world during the last 5 - 10 years. There are six concepts accepted by the GenIV international forum (GIF) with the common aims to promote both efficiency and safety of the technology. New concepts will offer attractive features but at the same time they also bring new and demanding challenges e.g. for the materials technology due to increased operating temperatures and irradiation doses as well as more aggressive coolants and/or longer life time expectations than of GenII and GenIII plants. In this paper an overview of the material issues is given with special emphasis on supercritical light water reactor concept (SCWR).

This paper reviews the performance of commercial candidate materials for SCWR in-core applications focusing on the corrosion and stress corrosion cracking issues (SCC). Within the FP6 program “HPLWR Phase 2” -project (High Performance Light Water Reactor) general corrosion tests (i.e., oxidation rate tests) and SCC susceptibility have been performed on selected iron and nickel based alloys at 500°C and 650°C in supercritical water under the pressure of 25 MPa. The oxygen concentration of the inlet water was 0–150 ppb in all tests. The oxidation behavior was studied using weight gain measurements, scanning electron microscopy in connection with energy dispersive spectroscopy (SEM and EDS, respectively) and X-ray diffractometry (XRD). The SCC tests were slow strain rate tests (SSRT) performed using a step motor controlled loading device. The samples were strained with a nominal rate of $3 \times 10^{-7} \text{ s}^{-1}$.

Ferritic–martensitic (F/M) steels containing chromium have generally good resistance to stress corrosion cracking. However, they suffer from fast oxidation in the SCW conditions. Austenitic stainless steels and Ni-based alloys have better oxidation resistance but, on the other hand, are more susceptible to stress corrosion cracking. SSRT test showed that 316 NG, 1.4970, 347 H and PM2000 are not susceptible to SCC at 500°C based on fracture surface examination, but the experimental steel BGA4 showed a considerable susceptibility to intergranular SCC. The austenitic stainless steels were generally observed to be SCC susceptible at 650°C, which corresponds well with the data reported in literature. The high chromium ODS (Oxide Dispersion Strengthened) steel PM2000 was SCC resistant at both test temperatures.

10. Challenges of New Reactors

Alloys with high nickel content were not considered for the SCC studies because Ni has a strong negative effect on neutronics of the reactor core. Therefore, the present candidate materials for the core internals are austenitic stainless steels and high chromium ODS alloys.

The Finnish Sustainable Energy (SusEn) project on New Type Nuclear Reactors (NETNUC) (10-3163)

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Increased international attention has recently been devoted to reactor concepts that differ essentially from the existing light water reactors. Basic processes of these new concepts, known as Gen IV reactors, are fundamentally different from those used today. Some new features may create new type safety challenges. They also aim to push nuclear reactor technology to completely new regimes of performance parameters, thus raising engineering challenges.

In order to allow Finland to benefit from these new technologies and influence their development, it is necessary to join relevant international projects, develop domestic expertise on critical technologies involved, and participate in ongoing international efforts to develop safety requirements for them. The NETNUC project is a multidisciplinary consortium to carry out basic research to generate scientific knowledge needed for Gen IV reactors and to educate a new generation of research scientists in the field. Five targeted tasks are carried out in LUT, TKK and VTT.

Main research areas in LUT are the Super Critical Water Reactor (SCWR) and gas-cooled reactors, the latter being a new field in LUT. Linking of the strong thermal hydraulic background of LUT to the safety research of reactors operating in supercritical pressure conditions has been studied. CFD calculations have been carried out to simulate the BWR pool tests of LUT, which are relevant also for SCWR. The research of gas-cooled reactors in LUT has focused on the numerical modelling of coolant flow and heat transfer in the reactor core of a pebble-bed type high-temperature gas-cooled reactor. There has been co-operation with a research group in LUT that has experience in the modelling of combustion processes and gas-phase.

Main research areas for TKK are thermal-hydraulics of SCWR and Monte Carlo calculations on reactors utilizing thorium fuel. In addition to this the different aspects of the fuel cycle have been analyzed. SCWR thermal hydraulics has been investigated by simulations with APROS process simulation software. Also co-operation with Kurchatov Institute, Russia has been initiated. Thorium breeding in nuclear reactors has been investigated with MCNP simulations. Several auxiliary programs for thorium fuel performance calculations with FEMAXI-6 have been developed.

Main research areas in VTT are fast reactors and SCWR concept. In 2008, the first calculation system capable of analysing fast reactor concepts was taken into use at VTT. The validation of this system was started as well. The work included development of calculation methods by creating a new burnup calculation method based on calculating matrix exponentials. In the field of supercritical water cooled reactors, the participation in the EC HPLWR2 project was continued with development of thermal hydraulics models and applying them to the safety studies of the High Performance Light Water Reactor.

Understanding of corrosion phenomena of candidate materials under SCWR (Supercritical Water Reactor) conditions necessitates a reliable experimental testing of materials and therefore also a development of monitoring techniques for the relevant conditions. The long term objective of this project is to perform a state-of-art study that would serve as a guide for the selection of in-core materials for the SCWR. One of the key performance indicators for the material selection is the corrosion and oxidation behaviour of materials in SCWR conditions. Therefore, in situ studies of the oxide films forming on the internal component candidate materials in contact with supercritical water are needed. At VTT, the Contact Electric Impedance (CEI) technique has been successfully used for in situ characterization of the electrical and transport properties of the corrosion layers both in LWR and SCWR conditions. In 2008, first step towards employing a pneumatic servo-controlled bellows system for the oxidation studies under SCWR conditions were performed. Calibrations and installations of the double-bellows system into the supercritical autoclave have been performed. Usability of the double-bellow system was verified up to 650°C and 25 MPa. The materials to be studied by CEI technique have been chosen based on their general oxidation rates at SCW conditions.

In the future, it is likely, that process heat and steam from new type high-temperature nuclear power plants is used in the integrated industry, such as bio-refineries. It is important that also simulation tools support processes with multi-phase chemistry. The feasibility of combining rigorous multi-phase chemistry with dynamic process simulator, APROS, was evaluated at VTT. Three different case processes were studied: fibre suspension and bleaching chemistry in pulp and paper mills and boiler water chemistry. The multi-phase chemistry and process simulation were successfully combined.

The national Finnish network GEN4FIN has continued its operation; the results of NETNUC are disseminated through GEN4FIN to the Finnish industry. VTT has also been active in the preparation of Strategic Research Agenda (SRA) of SNETP. Also Nordic co-operation on GEN IV field has been started – the NOMAGE 4 network funded by NKS is participated by industrial companies and other partners and co-ordinated by Studsvik company from Sweden. The work will be linked with NETNUC.

Thermal hydraulic transient analysis of the high performance light water reactor using APROS and SMABRE (10-3164)

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Introduction

High Performance Light Water Reactor (HPLWR) is a European supercritical light water-cooled reactor concept that is currently studied in the “HPLWR2” project. VTT Technical Research Centre of Finland has participated in performing safety analyses for the HPLWR using two thermal hydraulics system codes: APROS [1] and SMABRE [2, 3]. The functionality of these codes was extended to supercritical pressures by means of introducing a pseudo two-phase region, which is located along the pseudo-critical line, in the supercritical pressure region [4, 5]. This approach ensures that second order phase transitions between liquid and gaseous states, through the supercritical pressure region, are calculated in a sound manner, effectively treating the phase transition as a first-order transition. The material properties of water and steam have also been extended to cover the supercritical region. In addition, various heat transfer and friction correlations recommended in literature for the supercritical region [6–8] were implemented in APROS.

Simulation models of the HPLWR were created for APROS and SMABRE. The aim of these models is to be as mutually similar as possible, in order to enable comparison between the codes, and to represent the current work-in-progress design of HPLWR with high accuracy. Both models include a detailed description of the HPLWR reactor pressure vessel (RPV) with the current three pass core design [9]. The SMABRE model includes also a rough model of the steam cycle. Core neutronic solution is not included in either model.

Aim of the work

The capability of APROS and SMABRE for safety analyses of HPLWR is illustrated by comparison of various transient scenarios including power decrease and the transition from supercritical to subcritical pressures. These calculations are a test to the codes' functionality as well as the HPLWR design

and models. The comparisons give promising results about the capability of the codes for supercritical region calculations.

APROS is also used for comparing the differences induced by the use of different heat transfer and friction correlations for sub- and supercritical pressures in supercritical region calculations. The comparisons are performed in steady-state calculation and in one transient scenario. The comparison results are presented and their relevance for transient analyses discussed.

Summary

The technical feasibility of the European design for supercritical water-cooled reactor is studied in the “HPLWR2” project. VTT participates in the project by performing safety analysis using two system codes, APROS and SMABRE, which were modified for functioning at supercritical pressures. Thermal hydraulic simulation models of the HPLWR have been created for the codes according to the current work-in-progress design.

In this paper, a few transient analyses simulations for the HPLWR are presented. These results prove that APROS and SMABRE can be applied to simulations at supercritical pressures as well as to transitions from supercritical to subcritical pressures,. These results are also used for assessing the functionality of the current design of the HPLWR. Finally, the effect of the heat transfer and friction correlations on the simulation results is examined with APROS, and their relevance for safety analyses calculations discussed.

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