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Demonstrating Structural Adequacy of Nuclear Power Plant Containment Structures for Beyond Design-Basis Pressure Loadings

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DEMONSTRATING STRUCTURAL ADEQUACY OF NUCLEAR POWER PLANT CONTAINMENT STRUCTURES FOR BEYOND DESIGN-BASIS PRESSURE LOADINGS

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ABSTRACT

Demonstrating the structural integrity of U.S. nuclear power plant (NPP) containment structures, for beyond design-basis internal pressure loadings, is necessary to satisfy Nuclear Regulatory Commission (NRC) requirements and performance goals. This paper discusses methods for demonstrating the structural adequacy of the containment for beyond design-basis pressure loadings. Three distinct evaluations are addressed: (1) estimating the ultimate pressure capacity of the containment structure (10 CFR 50 [1] and US NRC Standard Review Plan, Section 3.8) [2]; (2) demonstrating the structural adequacy of the containment subjected to pressure loadings associated with combustible gas generation (10 CFR 52 [3] and 10 CFR 50 [1]); and (3) demonstrating the containment structural integrity for severe accidents (10 CFR 52 [3] as well as SECY 90-016 [4], SECY 93-087 [5], and related NRC staff requirements memoranda (SRMs)). The paper describes the technical basis for specific aspects of the methods presented. It also presents examples of past issues identified in licensing activities related to these evaluations.

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INTRODUCTION

The containment structure is the most safety-significant structure at US NPPs because it houses the primary nuclear steam supply system components, and must provide a leak-tight boundary around the reactor system to prevent the release of radioactive material to the surrounding environment, in the event of an accident. Because of its safety significance, there are a number of evaluations of containment that need to be performed to demonstrate the containment structural integrity under internal pressurization from the design-basis accident and from beyond design-basis accidents.

For design basis loadings, the design requirements for containments are given in 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants" and in 10 CFR 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." These regulations dictate that the containment structure must be designed and evaluated for various loadings, including dead load, live loads, thermal, pressure due to pipe breaks, earthquake, and wind, in specific load combinations with corresponding acceptance limits. The design-basis loading for internal pressure is the maximum pressure resulting from the design-basis loss of coolant accident (LOCA), also referred to as the design basis accident. However, to ensure the safety performance of the containment structure, additional structural evaluations are needed to meet the regulatory requirements and performance goals that pertain to beyond design-basis pressure loadings. The three evaluations of the containment to demonstrate structural integrity for beyond design-basis pressure loadings are discussed in the following sections.

Containment Internal Pressure Capacity Above Design-Basis Pressure

The accident at Three Mile Island, Unit 2 NPP, on March 28, 1979, raised questions about the pressure retaining capability of the containment structure for beyond design basis pressure loadings. The staff updated the review criteria in US NRC Standard Review Plan (SRP) Sections 3.8.1 and 3.8.2 for concrete and steel containments, respectively, to include a demonstration that the ultimate internal pressure capacity of containment is substantially higher than the design-basis accident pressure.

10 CFR 50, General Design Criteria (GDC) 50, "Containment Design Basis," requires that the reactor containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions caused by a LOCA. In addition, 10 CFR 52.47(a)(9) for design certification (DC) applications, and 10 CFR 52.79(a)(41), for combined license (COL) applications identify that applications for light water-cooled nuclear power plants shall include an evaluation of the facility against the SRP, where in Sections 3.8.1 and 3.8.2, the need for evaluation of the containment margin is described. This paper addresses methods for estimating the margin by predicting the internal pressure capacity for containment structures above the internal pressure for the design-basis LOCA. The internal pressure capacity in this estimation is an internal pressure capacity at which the structural integrity is retained and a failure leading to a significant release of fission products does not occur.

In SRP 3.8.1 and 3.8.2, Revision 2 (March 2007), the staff updated the review guidance on ultimate pressure capacity, and identified simplified methods to estimate the ultimate pressure capacity of containments. This paper expands on the guidance in SRP 3.8.1 and 3.8.2, in order to promote a consistent evaluation of ultimate pressure capacity by licensees and applicants.

Combustible Gas Control Inside Containment

According to 10 CFR 52.47(a)(12) for DC applications, and 10 CFR 52.79(a)(8) for COL applications, applications for light-water-cooled nuclear power plants must include an analysis and description of the equipment and systems for combustible gas control, as required by 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors." Specifically, 10 CFR 50.44(c)(5) provides requirements for demonstrating containment structural integrity. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment" [6], provides acceptance criteria for meeting the requirements of 10 CFR 50.44, and demonstrating containment structural integrity. Regulatory Guide 1.7 already identifies specific acceptance criteria for steel and concrete containments based on the ASME Code. It also states that an acceptable approach should consider as a minimum, a combination of dead load and an internal pressure of 0.31 MPa [45 psig]. However, the regulatory guide does not describe acceptable containment structural models, analysis methods, and the accident sequences to evaluate. This paper describes methods that complement the regulatory position in Regulatory Guide 1.7.

Commission's Severe Accident Performance Goal

According to 10 CFR 52.47(a)(23) for DC applications, and 10 CFR 52.79(a)(38) for COL applications, applications for light-water reactor (LWR) designs shall include a description and analysis of design features for the prevention and mitigation of severe accidents. Section C.I.19.8 of Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," issued June 2007 [7], provides guidance on implementing these requirements. According to Section C.I.19.8, this analysis and description should specifically address the issues and performance goals identified in SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, and SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993, which the Commission approved in staff requirement memoranda (SRMs) dated June 26, 1990, and July 21, 1993, respectively. In those two SECY papers and related SRMs, the NRC established performance goals for containment structures in nuclear power plants under severe accident conditions. This paper recommends methods that can be used to meet those performance goals.

RECENT LICENSING EXPERIENCE

During recent licensing application reviews for advanced reactors, a number of issues have been identified relating to the implementation of the regulations, performances goals, and regulatory guidance for demonstrating the structural integrity of containments for beyond design-basis pressure loads.

In estimating the ultimate pressure capacity of the containment, one applicant utilized a probabilistic approach which was also intended for use in probabilistic risk assessments and severe accident analyses. The pressure capacity for various structural elements was based on the median pressure capacity. As highlighted in SRP Sections 3.8.1 and 3.8.2, the intention is to use a deterministic approach and Code-specified minimum properties rather than a probabilistic approach.

In two cases, the applicant applied an internal pressure loading for the combustible gas generation inside containment equal to 0.31 MPa [45 psig], without consideration of pressures generated by a 100% fuel cladding-water reaction. As discussed in Regulatory Guide 1.7, the intention is that 0.31 MPa [45 psig] be used only if it is higher than the pressures associated with the fuel cladding-water reaction.

Questions have also arisen regarding what severe accidents and acceptable structural integrity criteria should be considered in satisfying the NRC's deterministic containment performance goals for advanced LWRs, as presented in SECY-90-16, SECY-93-087, and corresponding SRMs. As an example, the Commission's severe accident performance goals identify that after the initial 24 hour period, the "containment should continue to provide a barrier against the uncontrolled release of fission products." Since this criterion is not defined, the implementation of this is subject to considerable interpretation.

To ensure appropriate and consistent implementation of the regulations, performance goals, and regulatory guidance, this paper presents methods to demonstrate the structural integrity of the containment for beyond design-basis pressure loadings.

SCOPE

This paper addresses structural integrity evaluations for containment structures and pressure-retaining structural barriers constructed of steel, reinforced concrete, and prestressed concrete.

For metal containments, the scope includes the components designed and constructed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III, "Rules for Construction of Nuclear Facility Components" [8], Division 1, Subsection NE.

For reinforced and prestressed concrete containments, the scope includes those components designed and constructed in accordance with ASME Code, Section III, Division 2, Subsection CC.

Containment Pressure Capacity above Design Pressure

NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories—An Overview," issued July 2006 [9], reports the results from a series of tests conducted on reinforced and prestressed concrete and free-standing steel containment vessel models. The containment model tests showed that global free-field strains on the order of 2.0–3.0 percent for steel containments and 1.5–2.0 percent for reinforced concrete containments can be achieved before failure occurs. For prestressed concrete containments, NUREG/CR-6810, "Overpressurization Test of 1:4-Scale Prestressed Concrete Containment Vessel Model," March 2003 [10] and NUREG/CR-6809, "Posttest Analysis of the NUPEC/NRC 1:4-Scale Prestressed Concrete Containment Vessel Model," March 2003 [11] provided additional results. Analysis of the results presented in NUREG/CR-6810 shows that global free-field hoop strains in the containment wall of 0.5 and 1.0 percent can be achieved before the onset of unrestrained wall deformations or rupture, respectively. Analysis of the results in NUREG/CR-6810 also shows that free-field average strains of 0.9 and 1.4 percent (including strains from the initial pre-stressing and strains from the internal pressurization) can be achieved for the hoop tendons before the onset of unrestrained wall deformations or rupture, respectively.

The results from the above referenced studies, along with other considerations such as the potential for strain risers more severe than those in the containment models tested, have led to the simplified global strain limits presented in the March 2007 revisions to SRP Sections 3.8.1 and 3.8.2 for use in predicting the internal pressure capacity of a containment.

Global strain limits for the containment structures are not applicable to the assessment of large bolted closures (e.g., boiling-water reactor (BWR) steel closure heads, equipment hatches, personnel airlocks). A separate evaluation of internal pressure capacity for these containment components should be performed to determine the controlling structural element.

Combustible Gas Control Inside Containment

As required by 10 CFR 50.44(c)(2), containments for new water-cooled reactors must have an inerted atmosphere, or the hydrogen concentrations in the containment (during and following an accident that releases an amount of hydrogen equivalent to that generated by a 100-percent fuel clad-coolant reaction, uniformly distributed) must be limited to less than 10 percent (by volume), while maintaining containment structural integrity and appropriate mitigating features.

For new water-cooled reactor containments that do not rely upon an inerted atmosphere to control combustible gases, 10 CFR 50.44(c)(3) requires that they have the capability to control combustible gas generated from a metal-water reaction involving 100 percent of the fuel cladding surrounding the active fuel region, so that there is no loss of containment structural integrity. These containments for water-cooled reactors must be able to establish and maintain safe shutdown and containment integrity with systems and components capable of performing their intended functions during and after exposure to the environmental conditions created by hydrogen burning.

As required by 10 CFR 50.44(c)(5), an applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is acceptable to the NRC, and the analysis must address an accident that releases hydrogen generated from a 100-percent fuel clad-coolant reaction, accompanied by hydrogen burning. Applicants must also demonstrate that systems necessary to ensure containment integrity are able to perform their functions under these conditions. Regulatory Guide 1.7 specifically addresses requirements in 10 CFR 50.44(c)(5) to demonstrate containment structural integrity.

The March 2007 revisions to SRP Sections 3.8.1 and 3.8.2 specifically identify accident load combinations that include pressures resulting from the generation of hydrogen inside containment. The load combinations specified are consistent with the regulatory position in Regulatory Guide 1.7.

Commission's Severe Accident Performance Goal

The NRC's deterministic containment performance goal for advanced LWRs, as presented in SECY-93-087, states that

> "The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containment stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products."

There is no guidance in the March 2007 revisions to SRP Sections 3.8.1 and 3.8.2 that addresses the deterministic severe accident performance goal.

RECOMMENDED METHODS

Prediction of Containment Internal Pressure Capacity above Design Pressure

This evaluation provides a measure of the safety margin above the design-basis accident pressure, for the containment structure by assessing the pressure capacity of the containment at which the structural integrity is retained, and a failure leading to a significant release of fission products does not occur. The intention of this analysis is to address the containment structural capacity and not to evaluate the potential effects of containment structural response on other connected components, such as attached piping and other equipment. To estimate the ultimate pressure capacity, a nonlinear finite element analysis of the containment structure will typically be needed to examine the overall response. Large penetrations are usually included in the finite element model; smaller penetrations and penetration closure components are typically analyzed using a separate finite element model, test data, or both.

For cylindrical containment structures and axisymmetric components of the containment, it may be feasible to use closed-form solutions or semiempirical methods to estimate the ultimate pressure capacity. In such cases, the applicant should document an adequate description of and technical justification for all simplifications. Test results may also be used; however, the applicant should provide sufficient information to demonstrate the applicability of the test results to the particular containment design and loading condition.

The documentation submitted to the staff should be sufficient for the staff to make an independent determination of the safety margin above the design-basis accident pressure.

In determining the containment internal pressure capacity, and also in the interpretation and evaluation of results, the applicant should consider the following:

a. The use of a three-dimensional finite element model is recommended. Axisymmetric or partial (e.g., half model or wedge) finite element models can be used, if a sufficient technical basis is provided.

b. For the purpose of estimating the safety margin above the design-basis accident pressure, a static analysis is usually sufficient. However, if dynamic response effects are important, the static pressure capacity may need to be reduced to account for such effects. One approach to determine the pressure capacity is a nonlinear dynamic analysis. Another approach, which should be subject to detailed review, is the use of an appropriate dynamic amplification factor applied to the static results.

c. The initial condition for the nonlinear analysis of the containment structure should be the linear elastic response caused by dead load and design-basis accident pressure, at the design-basis accident temperature. The internal pressure is incrementally increased until a specified failure criterion is reached (e.g., deflection limit; strain limit; solution divergence). When performing this analysis, the applicant should document the pressure(s) corresponding to initial yielding of the liner, reinforcing steel, prestressing tendon (if applicable), and steel components not backed by concrete (e.g., closure head, hatch) for concrete containments. For steel containments, the applicant should document the pressure(s) corresponding to initial yielding of the steel shell, and initial yielding of other steel components (e.g., closure head, hatch).

d. The nonlinear stress-strain curve for steel materials (steel liner, reinforcing steel, prestressing tendons, steel components, steel shell) should be based on the code-specified minimum yield strength for the specific grade of steel and a stress-strain relationship beyond yield that is representative of the specific grade of steel. The stress-strain curve used in the analysis should correspond to the design-basis accident temperature.

e. For concrete containments, the tensile strength of concrete should be neglected, and the analysis should include the nonlinear stress-strain curve in compression, corresponding to the design-basis accident temperature.

f. The following are recommended simplified methods for determining the pressure capacity of cylindrical containments:

(1) For cylindrical reinforced concrete containments, the pressure capacity analysis may be based on attaining a maximum global membrane strain away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 1 percent. The specific location of interest is the steel reinforcement in the hoop direction, closest to the inside surface of the concrete. The inside radius of the concrete wall should be used in calculating the strain in the hoop reinforcing steel.

(2) For cylindrical prestressed concrete containments, the pressure capacity may be estimated based on satisfying both of the following strain limits: (1) a total tensile average strain in tendons away from discontinuities (e.g., hoop tendons in a cylinder) of 0.8 percent, which includes the strains in the tendons before pressurization and the additional straining from pressurization; and (2) a global free-field strain for the other materials that contribute to resist the internal pressure (i.e., liner, if considered, and rebars) of 0.4 percent. The pressure capacity is to be based on the contribution from each element considered in the analysis, using the stress-strain curve for each material and the strain level in each material, as determined based on overall strain compatibility between all of the credited structural elements (liner, if considered, tendons and rebars). If a nonlinear finite element analysis is performed, the estimated pressure capacity should be based on the first realization of either 0.8 percent strain in the tendons or 0.4 percent strain in any of the other steel pressure-resisting elements (i.e., liner, if considered, and rebars) or the simultaneous realization of both strain limits.

If closed form solutions are applicable and are utilized to estimate the pressure capacity, the pressure capacity should be based on the contribution from each element considered in the analysis, using the corresponding nonlinear stress-strain curves, at the design-basis accident temperature.

(3) For concrete containment structures, the evaluation should also consider concrete failure modes, such as concrete shear and concrete crushing, to determine the controlling containment failure mode. The concrete failure modes may occur at pressures lower than those corresponding to the above steel strain limits.

(4) For cylindrical steel containments, the pressure capacity analysis may be based on attaining a maximum global membrane strain away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 1.5 percent.

g. The analysis methods described above apply to the overall containment structure. A complete evaluation of the internal pressure capacity should also address major containment penetrations, such as the removable drywell head and vent lines for BWR designs, equipment hatches, personnel airlocks, and major piping penetrations.

h. Under internal pressure, a potential failure mode of ellipsoidal and torispherical steel heads is buckling resulting from a hoop compression zone in the knuckle region. The analysis should evaluate this failure mode to determine if it is the limiting condition for the pressure capacity of the containment. The analysis should consider nonlinear material and geometric behavior and address the effect of initial geometric imperfections, either explicitly (direct modeling) or implicitly (through the use of appropriate imperfection sensitivity knockdown factors). If supported by test data, residual postbuckling strength can be considered in determining the pressure capacity.

i. Appendix A to NUREG/CR-6906 provides more detailed guidance on developing finite element models and performing analyses for pressures beyond the design-basis accident pressure.

j. The evaluation should also consider the potential for containment leakage at pressure levels below the calculated structural capacity. The applicant should perform analyses to demonstrate that leakage from containment components, such as penetrations, bolted connections, seals, hatches, or bellows, is sufficiently small for the calculated pressure and temperature capacity conditions. Otherwise, the pressure capacity should be based on a defined total leakage limit from these components. It should be noted that, at elevated temperature levels, seals and gaskets at penetrations and connections may not be sufficiently effective in preventing leakage. The criteria and technical basis for acceptable leakage from the containment, typically given in terms of the percentage of containment volume flow per day at the given pressure, or in terms of the leakage area, should be documented and submitted for review by the staff.

k. Details of the analysis and the results obtained should be documented in report form and submitted to the staff for review. The report should include:

(1) design internal pressure, as defined in Subarticle NE-3100, "General Design", and in Subarticle CC-3200, "Load Criteria" [8];

(2) calculated static pressure capacity;

(3) dynamic pressure capacity, if applicable (static pressure capacity reduced to account for dynamic amplification effects);

(4) associated failure modes;

(5) for concrete containments, the stress-strain relation of the liner steel and reinforcing or prestressing steel and the behavior of the liner under the postulated loading conditions in relation to that of the reinforcing or prestressing steel;

(6) criteria governing the original design and criteria used to establish failure;

(7) analysis details and general results, which include (1) modeling details, (2) description of computer code(s), (3) material properties and material modeling, (4) loading and loading sequences, (5) failure modes, and (6) interpretation of results, with all assumptions made in the analysis and test data (if relied upon) clearly stated and technically justified; and,

(8) appropriate engineering drawings adequate to allow verification of modeling and evaluation of analyses employed for the containment structure.

If the evaluation of the containment pressure capacity relies on an analysis that is intended to address more than one regulatory requirement or performance goal, the applicability of the analysis to address each specific requirement or performance goal should be clearly explained.

Combustible Gas Control Inside Containment

The purpose of this evaluation is to demonstrate the structural integrity of the containment in accordance with the requirements in 10 CFR 50.44, for pressure loadings associated with hydrogen generation caused by the reaction between the fuel cladding and the water coolant.

Regulatory Position 5 of Regulatory Guide 1.7 provides acceptance criteria to meet the structural requirements of steel and concrete containments in accordance with 10 CFR 50.44. For the required pressure load and dead load, steel containments should meet the Service Level C requirements of ASME Code, Section III, Division 1, Subsection NE 3220, and concrete containments should meet the Factored Load Category requirements of ASME Code, Section III, Division 2, Subarticle CC 3720. The applicable load combinations are delineated in SRP Section 3.8.1 for concrete containments and SRP Section 3.8.2 for steel containments.

In accordance with the regulatory guidance described above, the containment should be evaluated for (1) specified combinations of the pressure arising from the fuel cladding-water reaction, hydrogen burning, and postaccident inerting (if applicable); or (2) 0.31 MPa [45 psig], whichever is higher.

The analysis to demonstrate the structural integrity of the containment for these pressures should consider the following:

a. The development of a finite element model of the containment using the approach described for the containment capacity evaluation is recommended, subject to the limitations discussed below.

   (1) ASME Code-specified material properties should be used. These should correspond to the metal temperature(s) resulting from the hydrogen-generation event.
   (2) For steel elements, linear elastic material properties may be used. For concrete, the nonlinear stress-strain relationship or an equivalent linear elastic curve may be used.
   (3) The potential structural dynamic amplification effects caused by the pressure transient loading associated with hydrogen gas generation or the burning of hydrogen, if significant, should be included in calculating the response of the containment.

b. The accident sequence used to determine the pressure load should address the hydrogen mass and energy releases generated from a 100-percent fuel clad-coolant reaction, accompanied by the burning of hydrogen, and postaccident inerting (if applicable). For inerted containments, burning does not need to be considered.

c. Regulatory Position 5 of Regulatory Guide 1.7 provides the acceptance criteria for the resulting stresses. As noted in Regulatory Guide 1.7, an instability (buckling) calculation is not required for steel containments. For concrete containments, it is only necessary to demonstrate that the liner strains satisfy the limits presented in ASME Code, Section III, Division 2, Subarticle CC-3720.

d. The evaluation of the structural integrity of the containment for pressure loads resulting from 100% fuel cladding-water interaction should be documented and submitted to the staff for review.

If the evaluation to demonstrate the containment pressure integrity for the hydrogen-generated pressure loads relies on an analysis that is intended to address more than one regulatory requirement or performance goal, the applicability of the analysis to address each specific requirement or performance goal should be clearly explained.

Commission's Severe Accident Performance Goal

The purpose of this evaluation is to address the Commission's deterministic containment performance goals in accordance with SECY-90-16 and SECY-93-087, and the corresponding SRM, dated June 26, 1990, and July 21, 1993, respectively. As specified in SECY-93-087, the containment should maintain its role as a reliable, leak-tight barrier for approximately 24 hours following the onset of core damage, under the more likely severe accident challenges. Following this initial 24-hour period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

Initial 24-Hour Period Following the Onset of Core Damage:

a. The applicant should provide the technical basis for identifying the more likely severe accident challenges to be reviewed by the staff on a case-by-case basis. An example of a recommended way to identify the more likely severe accident challenges is to consider the sequences or plant damage states that, when ordered by percentage contribution, represent 90 percent or more of the core damage frequency.

b. From the set of pressure/temperature transients determined from item a above, identify one or a group of pressure/temperature transients that envelop the entire set of pressure/temperature transients. The reduced set of pressure/temperature transients are used to analyze the containment structure. For concrete containments, the pressure transient controls the peak response. Concrete has significant thermal inertia and its response to the associated temperature transient is relatively small. Therefore, it is recommended to analyze concrete containments for the sequence or damage state with the highest peak pressure and its corresponding temperature transient. For thin-walled steel containments, the potential effect of the temperature transient on the steel material properties at the time of peak pressure should be considered.

c. The development of the global and localized finite element models of the containment, using the approach described for the containment capacity evaluation is recommended, subject to the limitations discussed below.

   (1) For steel containment elements, linear elastic material properties may be used. For concrete, the nonlinear stress-strain relationship or an equivalent linear elastic curve may be used. The material properties should be based on the estimated material average temperature at the time of peak pressure, for each severe accident considered. In this regard, see Appendix A to NUREG/CR-6906 for information on material properties for concrete containments.

   (2) The potential structural dynamic amplification effects caused by the pressure transients for the severe accident events, if significant, should be included in calculating the response of the containment.

d. The use of the ASME Code Service Level C limits for metal containments or the Factored Load Category for concrete containments is recommended to demonstrate the deterministic performance goal for the first 24 hours. This includes the evaluation of the containment for stability or buckling, in accordance with the ASME Code.

Period More Than 24 Hours After the Onset of Core Damage:

a. Recommended ways to meet the performance goal that "…the containment should continue to provide a barrier against the uncontrolled release of fission products" (for the more likely severe accident challenges, after the initial 24-hour period), include the following:

   (1) The maximum pressure and temperature following the initial 24-hour period are enveloped by the maximum pressure and temperature during the initial 24-hour period; or

   (2) The maximum pressure and temperature following the initial 24 hour period meet the applicable Level C or Factored Load acceptance criteria; or

   (3) The calculated release for the more likely severe accident challenges, following the initial 24-hour period, meets site-specific design criteria for fission product released from the containment, in accordance with the requirements of 10 CFR 100.21 [12], "Non-Seismic Siting Criteria," and 10 CFR 50.34, "Contents of Applications; Technical Information."

   If an applicant utilizes alternative methods to meet this performance goal, sufficient justification acceptable to the staff should be provided.

b. The development of a finite element model of the containment using the approach described for the containment capacity evaluation is recommended, subject to the following limitations:

   (1) The stress-strain curve for steel and concrete materials should correspond to the temperature associated with the more likely severe accident events. The effect of elevated temperature on the elastic modulus for all materials should be considered.

   (2) The potential structural dynamic amplification effects caused by the pressure transients for severe accident

events, if significant, should be included in calculating the response of the containment.

c. If the approach described above in (a.)(3) is used, then the applicant should provide sufficient information to enable the staff to review how the calculated release of fission products was determined and how it compares to the site-specific design criteria for fission product release from containment, in accordance with the requirements of 10 CFR 100.21 and 10 CFR 50.34. The analysis to determine the fission product release from the containment should consider all possible pathways, including components such as penetrations, bolted connections, seals, hatches, and bellows.

SUMMARY

This paper presents methods to demonstrate the structural integrity of nuclear power plant containment structures for beyond design-basis internal pressure loadings. These methods address three distinct evaluations, as follows: (1) estimating the ultimate pressure capacity of the containment structure, (2) demonstrating the structural adequacy of the containment subjected to pressure loadings associated with combustible gas generation, and (3) demonstrating the containment structural integrity for severe accidents. For each of these three evaluations, recommendations are provided for the loadings to be considered, the development of the model(s), material properties, analysis methods, and acceptance criteria.

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DISCLAIMER

The findings and opinions expressed in this paper are those of the authors, and do not necessarily reflect the views of the U.S. Nuclear Regulatory Commission or Brookhaven National Laboratory.