REACTOR PRESSURE VESSELS AS TYPE B TRANSPORT CONTAINMENT BOUNDARIES

R. E. Nickell (1) and *T. J. Griesbach* (2)

(1) Applied Science & Technology, 16630 Sagewood Lane, Poway, CA 92064 U. S. A.
(2) ATI Consulting, Inc., 3860 Blackhawk Road, Suite 160, Danville, CA 94506 U. S. A.

SUMMARY

Transportation risk and personnel exposure, as well as the cost of decommissioning nuclear power plants, can all be reduced significantly through the one-time use of the reactor pressure vessel as a containment boundary for shipping the activated internal components from the reactor site to a burial site. In order to help provide the technical basis for this end-use application, the ASME Board on Nuclear Codes and Standards, through its Subcommittee XI, has prepared a draft nuclear code case that contains requirements for any modifications to the vessel, including materials, design, fabrication, and examination. In particular, the requirements for evaluation of potential brittle fracture as the result of potentially low ambient shipping temperatures combined with hypothetical transportation accident loading are addressed. Existing ASME Code Section XI rules for linear elastic fracture mechanics evaluation of irradiated reactor pressure vessels have been adapted and included in the code case.

INTRODUCTION

Decommissioning of nuclear power plants that have been shut down either prematurely or at the end of the planned service life involves the removal and disposal of major systems, structures, and components, including the reactor pressure vessel and internals, at burial sites that could be some distance from the plant location. The activated metal represented by the reactor internals components, in particular, is a radioactive material hazard to the public health and safety during the transport from the plant to the burial site, a hazard that can be reduced considerably by the one-time shipment of the immobilized, grouted-in-place internals inside the reactor pressure vessel. This procedure also reduces the personnel exposure related to cutting and packaging the internals for multiple shipments, and reduces the cost of decommissioning significantly.

However, even for one-time use of the reactor pressure vessel as a transportation package containment boundary, this application is governed in the United States by the requirements of Title 10, Part 71, of the Code of Federal Regulations (10 CFR 71), "Packaging and Transportation of Radioactive Materials," in particular the mechanical performance standards for Type B containers. 10 CFR 71 requires that packages used to transport radioactive material be designed with consideration of normal transport and hypothetical accident events that might occur at temperatures as low as -20°F (-29°C). At such temperatures, many ferritic steels, including those used for the construction of nuclear reactor pressure vessels, are at or near their nil-ductility transition temperature (T_{NDT}), as determined by standard drop weight testing methods (e.g., ASTM E208). Therefore, brittle fracture initiation and propagation is a failure mode of concern, especially potentially unstable crack propagation caused by severe hypothetical accident conditions that involve relatively high dynamic stresses in the material.

Therefore, the combination of the impact/puncture loading conditions for normal conditions of transport and hypothetical accident conditions, and the need to evaluate the potential for brittle fracture of the vessel material, including any embrittlement and reduction in fracture toughness due to prolonged exposure to neutron irradiation, is a major issue for such one-time shipments.

The low-alloy steels used for the construction or reactor pressure vessels are noted for their extremely high quality and resistance to brittle fracture over the range of reactor operating temperatures from room temperature to about 290°C (550°F); however, guidance provided by the U. S. Nuclear Regulatory Commission (NRC) in Regulatory Guide 7.12 for Type B packagings requires nil-ductility transition (NDT) temperatures for thick-walled containment boundaries that cannot be met by these high-quality steels even prior to exposure to neutron irradiation.

This paper discusses potential alternatives to the regulatory guidance, such as the application of brittle fracture evaluation rules contained in the ASME Code Section XI, Appendix A, that can be used to assure protection to the public for the one-time shipments of reactor pressure vessels acting as containment boundaries for activated internals, based upon the consensus ASME Nuclear Code Case development process. The particular example cited is that of the Portland General Electric Company's Trojan Nuclear Plant reactor pressure vessel and internals, which are to be shipped from the plant site near Rainier, Oregon, to the burial site near Richland, Washington, a shipping route that involves both barge and overland modes. In addition to the brittle fracture evaluation, the example includes a probabilistic fracture mechanics study to verify that the risk levels are less than those associated with spent fuel transport casks.

ASME SECTION XI SPECIAL TASK GROUP

In response to a request by Portland General Electric and other U. S. utilities interested in the modification of reactor pressure vessels to qualify as Type B transport packaging containment boundaries, the ASME Section XI Subcommittee (SC XI), under the jurisdiction of the ASME Board on Nuclear Codes and Standards (BNCS), created a special task group to develop appropriate rules in early 1997. The scope of the rules were intended to cover general requirements; material requirements, including those for existing materials such as the low-alloy steel vessel material itself and any additional gamma shielding needed to satisfy transportation criteria; design requirements, such as the evaluation of transportation loads not considered in the original vessel design;

fabrication requirements, such as those for welding integral attachments and closure devices to the vessel; and examination and testing requirements, if any.

The Task Group on Reactor Pressure Vessels as Shipping Containers, chaired by R. S. (Steve) Lewis of Entergy Operations, Inc., held its first meeting on May 5, 1997, in conjunction with ASME Code meetings in Kansas City, Missouri, with the decision to proceed with the fast-track development of a Section XI code case. The code case was to rely heavily on existing requirements in Section XI for materials, welding procedures, and examination.

The major technical issue for the Task Group concerned the criteria to be met when evaluating the vessel for potential brittle fracture. However, in addition to structural integrity evaluation concerns, the reactor vessel may require modification to accommodate transport configuration and conditions. Integral or non-integral lifting devices may be added, including the possibility of modifying the closure studs and flanges to provide a portion of the lifting requirements. Additional gamma shielding may be required to meet the transportation exposure limits. Nozzles, openings, and penetrations may be closed off to provide physical protection against any release of the radioactive contents (the detailed activation analysis for the Portland General Electric Trojan reactor pressure vessel and internals indicated 155 curies for the vessel internal surface contamination and 2.1 million curies for the activated metal of the internals). All of these issues had to be addressed by the proposed Section XI code case.

DESIGN SPECIFICATION

The code case places the responsibility on the Owner to prepare a modified Design Specification that includes the additional thermal and mechanical loadings required for the reactor pressure vessel in its one-time application as a containment boundary for the immobilized internals. The Owner may chose to review the proposed route of travel, covering all of the transport modes involved (e. g., rail, road, or barge), and evaluate the potential loading conditions based on risk assessment. Guidance for the classification of risks and associated load severity is provided by the thermal and mechanical performance test conditions for multiple-use Type B packages contained in 10 CFR 71. These performance test conditions are divided into two categories – normal conditions of transport and hypothetical accident conditions.

The Owner should modify the existing Design Specification to include a set of loading conditions that are expected during the one-time vessel transport sequence, called Level A/B Service Conditions; a set of unexpected loading conditions under which the vessel acting as a containment boundary would continue the transport sequence with only a verification type of inspection, called Level C Service Conditions; and a set of unexpected loading conditions under which the vessel acting as a containment boundary may sustain substantial damage, but still provide adequate protection to the public health and safety, called Level D Service Conditions. These loading conditions need not correspond exactly with the performance test conditions of 10 CFR 71 for multiple-use Type B packages, but are subject to approval by regulatory authorities.

The modified Design Specification should also include the appropriate elements of the Repair/Replacement Plan, in accordance with the provisions of IWA-4000. Conversion of the reactor pressure vessel to a containment boundary for one-time use may require: (1) installation of additional gamma shielding that enables the vessel and internals as a package to meet transport dose limits; (2) incorporation of lifting devices, such as trunnions, in the form of welded or bolted attachments; (3) welding of closure devices over vessel nozzles, openings, and penetrations; and (4) attachment of impact-limiting devices to protect the vessel and contents during design-basis impact events.

In addition, the modified Design Specification should define the jurisdictional boundaries between the vessel and any non-integral attachments, such as impact-limiting devices, recognizing that transmittal of thermal and mechanical loads through the impact-limiting devices is a major design consideration.

BRITTLE FRACTURE EVALUATION

The major technical issue addressed by the code case is the prevention of brittle fracture of the reactor pressure vessel. No specific design criteria are provided in 10 CFR 71 for protecting against potential brittle fracture initiation and unstable crack growth of Type B transport packages. However, theoretical evaluations carried out at the Lawrence Livermore National Laboratory (LLNL) in the early 1980s (Schwartz, 1984) led to the publication of Regulatory Guide 7.12 (NRC, 1991). This guidance identifies the acceptable T_{NDT} for ferritic steels at their lowest service temperature (LST), where the LST is assumed to be -20°F (-29°C) per the regulations. As an example, the acceptable T_{NDT} for an eight-inch-thick ferritic steel vessel at LST = -20°F is -135°F. Typical initial T_{NDT} values for the plates and welds of the reactor pressure vessels are substantially higher than this limit (e.g., between -20°F and +10°F). Although reactor pressure vessel steels are of extremely high quality and provide excellent resistance to brittle fracture over the range of reactor operating temperatures from room temperature to about 290°C (550°F), the guidance in Reg. Guide 7.12 cannot be met by typical low-alloy steel reactor pressure vessel materials even prior to exposure to neutron irradiation during service and subsequent upward shifts in T_{NDT}. For this reason, the application of existing rules contained in the ASME Code Section III, Division 3, for spent fuel and high-level waste transport packaging containment boundaries was deemed to be non-feasible, because of the conservatism of the nil-ductility transition (NDT) temperature requirements in that document that are referenced to Regulatory Guide 7.12.

The ASME Section XI Task Group examined alternatives to the regulatory guidance, such as the application of brittle fracture evaluation rules contained in the ASME Code Section XI, Appendix A, which would permit the use of linear elastic fracture mechanics to demonstrate adequate margin against crack initiation and unstable crack growth. The Task Group reasoned that this approach has been found to be acceptable by regulatory authorities for many safety-related applications, including demonstration of fitness for continued service of embrittled reactor pressure vessels, and should therefore be acceptable for assuring protection to the public for one-time shipments of the vessel and internals, provided that the loading conditions are properly defined. This same approach has also been included as one of the options for the revised IAEA safety series documents that provide requirements for radioactive material (RAM) transport packagings.

TROJAN REACTOR VESSEL BRITTLE FRACTURE EVALUATION

Technical support for the ASME Section XI, Appendix A approach is provided by the brittle fracture evaluation from the Portland General Electric Trojan Nuclear Plant Reactor Vessel and Internals Removal (RVAIR) Project. This project involves the transport of the Trojan reactor pressure vessel, with the activated internals grouted in place, from the plant site near Rainier, Oregon, by barge transport up the Columbia River and overland transport to the burial site near Richland, Washington. In order to protect the approximately 900-tonne (1000-ton) package from damage caused by hypothetical transport accidents, torus-shaped expanded polyurethane foam impact limiters were designed for attachment to the upper and lower ends of the vessel prior to shipment. Stresses and deformations from a variety of credible accident scenarios were then determined by finite element analysis, and fed into linear elastic fracture mechanics models at critical locations along the vessel length. The accident scenarios included hypothetical drop conditions onto essentially unyielding surfaces from heights up to about 6 meters (19 feet), and hypothetical drop conditions onto a regulatory mild steel puncture pin from a height of one meter (40 inches).

The fracture mechanics analysis procedure consisted of postulating a hypothetical circumferential surface flaw and predicting the potential for crack initiation under the peak applied dynamic stresses from the impact analysis. The depth of the postulated flaw was determined from ASME Section XI inservice inspection flaw acceptance criteria, assuming the maximum allowable depth (1.9 per cent) for a full-circumferential flaw, as confirmed by previous ten-year examination results for the Trojan vessel. Dynamic material toughness (K_{Ia}) values were derived from original vessel RT_{NDT} data supplemented by results from the Trojan vessel surveillance program. Values were determined for both the inside and outside surfaces.

The largest RT_{NDT} was found to be about 112°F at the inside surface in the vessel beltline region. Numerous locations were considered in order to evaluate the potential coincidence of high applied stress and low material toughness. The critical points evaluated were: (1) at the inside and outside surfaces near the severed inlet and outlet nozzles, where the local dynamic stresses were largest; (2) at the inside surface near the core mid-plane, where the RT_{NDT} shift due to neutron irradiation embrittlement is the greatest; and (3) at an intermediate location near the upper shell course and nozzle course circumferential weld. The procedures of Section XI Appendix A were used to calculate the applied stress intensity, compare with material toughness, and determine the margin against crack initiation.

The results of the evaluation showed adequate margin against crack initiation at all locations. For example, a factor of 2.0 on applied stress intensity was found for the outside surface location just below the severed outlet nozzle at -20°F ($K_{Iapplied}$ of about 18 ksi$\sqrt{}$in, compared to a K_{Ia} of about 36 ksi$\sqrt{}$in), which was the smallest margin at any location in the vessel for the side drop impact event. Margins were much greater in the reactor beltline region, even for the locations of maximum RT_{NDT} shift, because of the lower applied stresses.

BRITTLE FRACTURE RISK ASSSSMENT

In addition to this alternative evaluation of potential brittle fracture during transport, Portland General Electric also conducted a probabilistic fracture mechanics study to verify that the risk levels are less than those associated with multiple-use spent fuel transport casks, basing the evaluation on a previous risk-assessment study by Lawrence Livermore National Laboratory (Schwartz, 1984) that preceded Reg. Guide 7.12.

This risk assessment evaluated a variety of brittle fracture criteria, including those identified as "fracture arrest criteria" and those defined as "fracture initiation criteria." The "fracture arrest criteria" have no actual relationship with the demonstration of fracture arrest, but merely denote a set of dynamic tear test data that are more properly called the Pellini fracture toughness reference curve approach, whereas the "fracture initiation criteria" refer to standard linear elastic fracture mechanics procedures used in the ASME Code Sections III and XI (i.e., Appendix A) for both initial construction and for evaluation of flaws detected during inservice examinations. Of the various options evaluated in NUREG/CR-3826, only the Pellini approach was selected by the NRC staff for incorporation into Reg. Guide 7.12.

However, all of the criteria evaluated in NUREG/CR-3826 were compared to a common benchmark called the "limit state probability." More precisely, this limit state probability represents the conditional probability that an existing defect will either initiate or grow to an unstable size under hypothetical transportation accident loading.

For the Pellini fracture toughness reference curve approach, NUREG/CR-3826 based the conditional probability assumption on a pre-existing through-wall crack, with applied stresses normal to the crack surface equal to the material yield strength. In other words, the limit state probability calculated in NUREG/CR-3826 for the Pellini approach is the conditional probability that the material fracture toughness is below the Pellini reference fracture toughness curve, given that a through-wall crack exists at all locations in the vessel and that all locations are subjected to yield-level stresses. Typical values of the conditional probability of brittle failure for SA 508 Class 4 material at -20°F were found to be of the order of 10^{-2}, a relatively high conditional probability that is the result of the very conservative assumptions on crack size, stress level, and Pellini reference curve fracture toughness values. On the basis of this very conservative calculation, NUREG/CR-3826 established 10^{-2} as the target conditional failure probability not to be exceeded by other brittle fracture evaluation methods, such as those based on fracture initiation.

For the alternative fracture initiation criteria, NUREG/CR-3826 determined the conditional probability of crack initiation, given a surface-breaking flaw equal in size to the ASME Code Section XI acceptance limits and a probability of non-detection of a flaw twice that size. Again, to simplify the probabilistic calculation, the stresses normal to the surface of that flaw were assumed to be equal to the yield strength, irrespective of the location of the flaw. Typical values of the conditional probability of crack initiation under these assumptions are in the range 10^{-3} to 10^{-5}. Again, these conditional failure probabilities are relatively high, as the result of the given size of the flaw, the probability of non-detection of even larger flaws, and the level of stress normal to the surface of that flaw.

No calculations were provided in NUREG/CR-3826 for the case where the stress levels were determined from transport accident stress analyses, which would have required the incorporation of joint probability distributions for flaw existence, flaw size, probability of non-detection (and repair, if the flaw exceeded the size given in ASME Section XI for continued operation), and stress distribution. More realistic calculations (McConnell, et al., 1990) have shown that the nominal conditional probability of failure, as defined by crack initiation and through-wall crack propagation, is about 5×10^{-7} (based on a Gaussian ferritic material fracture toughness distribution with a mean of 120 ksi$\sqrt{\text{in}}$ and a standard deviation equal to 10 % of the mean), approximately two orders of magnitude lower than the 10^{-3} to 10^{-5} range cited in NUREG/CR-3826. This lower probability was calculated even though the accident data base that was used included events that generate stresses in the cask wall well above yield strength. However, such stresses do not exist everywhere in the cask wall and the joint probability of high stresses and large flaws coexisting with low material fracture toughness at the same point in the cask wall was found to be extremely low.

More recent calculations for Portland General Electric determined the conditional probability of brittle fracture for the Trojan vessel, given a severe transport accident, to be in the range of 10^{-4} to 10^{-5}, using the VISA code. The difference between this estimate and those obtained by McConnell, et al. is largely due to the penalties incurred from accounting conservatively for additional stresses at the base metal-clad interface in the Trojan vessel, and from the simplified definition of brittle failure (i.e., crack initiation) used by McConnell, et al. Both sets of realistic calculations agree when the effects of the cladding stresses are removed, but differ substantially from the limit state probabilities calculated in NUREG/CR-3826.

The differences can be explained by examining the assumed distributions of material fracture toughness, especially the low-fracture toughness "tail" of the distributions. McConnell, et al., found that, when the dynamic fracture toughness, K_{ID}, was assumed to be deterministic, with a value of either 80 ksi$\sqrt{\text{in}}$ or 40 ksi$\sqrt{\text{in}}$, the conditional failure probability increased from the nominal value of 5×10^{-7} to 4.7×10^{-6} or 3.3×10^{-4}, respectively, all other probabilistic parameters remaining the same. However, when the dynamic fracture toughness distribution was chosen to have the same mean K_{ID} of 120 ksi$\sqrt{\text{in}}$, but the standard deviation is of 33 % of the mean, rather than the 10 % standard deviation in the nominal case, the conditional failure probability increased to 2.7×10^{-3}. When this same calculation was carried out with a lower-bound cut-off of 20 ksi$\sqrt{\text{in}}$ on K_{ID}, the conditional failure probability decreased to 1.7×10^{-6}. This confirmed that essentially all of the contribution to failure probability was attributable to the low-toughness tail of the fracture toughness distribution.

The NUREG/CR-3826 fracture toughness distributions show that a typical mean value is approximately 90 ksi$\sqrt{\text{in}}$, with an approximate standard deviation of about 26.5 % of that value. Such a standard deviation would be expected to produce artificially high estimates of conditional failure probability, with the probability of brittle failure overestimated by a factor of about 10^3.

CONCLUSIONS

An ASME Section XI Appendix A fracture mechanics assessment of a radioactive material (RAM) transportation package consisting of the Trojan reactor pressure vessel with grouted-in-place internals has shown that adequate margin is available against crack initiation from hypothetical transportation accident loading conditions. Furthermore, probabilistic fracture mechanics risk assessments have shown the conditional probability of brittle failure, given such a hypothetical accident, to be in the range of 10^{-5} to 10^{-7}, in agreement with other severe accident studies. These two assessments provide supporting evidence that the approach being taken by the ASME Section XI Task Group for a Nuclear Code Case covering the one-time use of reactor pressure vessels as Type B shipping containers is technically valid.

REFERENCES

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SESSION 4.4

Safety Culture
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