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OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Project Number 694

July 19, 2010

OG-10-246

Secretary
Attn: Rulemakings and Adjudications Staff
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: PWR Owners Group
Transmittal of PWROG Comments on the NRC "Proposed Rule Changes to 10 CFR 50, Relating to Code Case N-770 and Nonmandatory Appendix E, Evaluation of Unanticipated Operating Events", Docket ID NRC-2008-0554, PA-MSC-0483

Enclosed are the Pressurized Water Reactor Owner's Group's (PWROG's) comments to the conditions that the NRC is planning to impose on ASME B&PV Code Case N-770 and Appendix E through 10CFR 50.55(a) (Enclosures 1 - 3).

We urge the NRC to adopt Code Case N-770-1 instead of N-770. Adoption of N-770-1 would address conditions 5 through 11 and 13 and 14 from 10CFR 50.55(a). We have included responses to all the NRC's conditions including those not explicitly addressed by N-770-1 and would be willing to discuss these further so that N-770-1 can be adopted by the NRC without conditions (Enclosure 1).

ASME, Section XI, Appendix E was developed 23 years ago by a group of experts experienced in reactor pressure vessel integrity evaluations, including NRC staff, NRC consultants from National Laboratories and industry participants. The technical basis for the appendix was peer reviewed and the computational results used to define the Appendix E criteria were verified by NRC, NRC consultants and industry Code members. Since then, Appendix E has been used to effectively and efficiently assess the integrity of reactor pressure vessels following unanticipated events that resulted in pressures and temperatures outside the limits established for normal operation.

We have carefully reviewed the NRC comments regarding ASME, Section XI, Appendix E. Our review included the original basis for the Appendix E criteria, the recent NRC comprehensive study to develop the alternate PTS rule, recent industry work to define an alternative risk-informed procedure for establishing limits for normal reactor pressure vessel operation, and just completed work performed by industry in response to ACRS questions regarding the consistency of Appendix E with the risk-informed basis used to define the limits in the alternate PTS rule, 10CFR50.61a.

Template = SECY-067

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Based on our review we conclude that the evaluation procedures in Appendix E, Paragraphs E-1200 and E-1300 provide adequate safety margins for evaluating reactor pressure vessel integrity following an unanticipated event that results in pressures and temperatures outside the limits established for normal operation. Additionally, we find that Appendix E is consistent with risk-informed acceptance criteria for normal operating and unanticipated events. Consequently, modifying Appendix E as suggested by the NRC is unnecessary and disallowing use of paragraph E-1200 of Appendix E will result in an undue hardship without any compensating increase in safety.

Enclosed are our responses to the NRC comments regarding Appendix E (Enclosure 2), and a draft Pressure Vessel and Piping (PVP) paper (Enclosure 3) that has been prepared to assess the consistency of Appendix E with the risk-informed basis used to define the limits in the alternate PTS rule.

For technical questions regarding the enclosed PWROG comments, please contact Gary Elder (Westinghouse) at (412) 374-4884 or Carol Heinecke (Westinghouse) at (412) 374-2758. If you have any additional questions or comments on the enclosed information, feel free to contact Jim Molkenthin in the PWROG office at (860) 731-6727 or me at (704) 382-8619.

Sincerely,

Melvin L. Arey, Jr.

Melvin L. Arey, Jr.
Chairman, PWR Owners Group

MLA:JPM:las

Enclosures: (1) – PWROG Comments to Code Case N-770
(2 and 3) – PWROG Comments to Appendix E and draft PVP paper.

cc:
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Response to NRC Conditions on Code Case N-770

Proposed Condition:
The NRC proposes to add a condition (§ 50.55a(g)(6)(ii)(F)(2)) to require that welds mitigated by inlays, cladding, or stress improvement by welding, be categorized as unmitigated welds pending plant-specific NRC review of the mitigation techniques and NRC authorization of an alternative ASME Code Case N–770 Inspection Item for the mitigated weld. ASME Code Case N–770 provides inspection methods and frequencies for welds mitigated by certain specified techniques. Inspections of mitigated welds are performed much less frequently than unmitigated welds. Requirements for most of the mitigation methods are contained in other ASME code cases under development. The NRC has typically approved the application of pressure boundary weld mitigation techniques on a case-by-case basis. This condition is necessary to ensure that appropriate mitigation techniques are applied to welds before they are categorized as mitigated under Code Case N–770.

Response:
All mitigation techniques, with the exception of Mechanical Stress Improvement Process (MSIP), discussed in Code Case N–770 are the subject of separate Code Cases which will be subject to approval by the NRC. MSIP meets the requirements of Appendix I of Code Case N–770 and has been separately approved by the NRC. If approved mitigation techniques are employed a separate review of the reclassification of the welds should not be required.

This proposed section, requiring that welds that have been mitigated by weld inlay or onlay of corrosion resistant cladding be categorized for ISI frequency as Inspection Item A-1, A-2, or B, is not consistent with other proposed requirements, or with later revisions of Code Case N-770. For example, (g)(6)(ii)(F)(6) requires that a weld that has been mitigated by inlay or corrosion resistant cladding, and then is found to be cracked, be reclassified as and inspected using the frequencies of Inspection Item A-1, A-2, or B. This indicates that an uncracked weld that has been mitigated by inlay or corrosion resistant cladding would NOT be categorized as inspection Items A-1, A-2 or B following an acceptable pre-service examination. Another example is proposed Section (g)(6)(ii)(F)(7), which requires that a weld mitigated by inlay or corrosion resistant cladding be examined each interval if at hot leg temperatures, and as part of a 25 percent sample plan on a 20 year frequency if at cold leg temperatures, which is not consistent with Inspection Item A-1, A-2, or B.

Proposed Condition:
The NRC proposes to add a condition (§ 50.55a(g)(6)(ii)(F)(3)) to require that the baseline examination of welds in Inspection Items A–1, A–2, and B (unmitigated welds) be completed at the next refueling outage after the effective date of the final rule. Paragraph –2200 of Code Case N–770 permits welds in Inspection Items A–1, A–2, and B (unmitigated welds) that have not received a baseline examination to be examined within the next two refueling outages from adoption of the Code Case. Welds in Inspection Items A–1, A–2, and B are the

welds most likely to experience PWSCC and some of these welds may not have received a baseline examination, even under the industry initiative, MRP–139. This condition is necessary to ensure the integrity of these welds by requiring that all welds in Inspection Items A–1, A–2 and B be inspected at the first opportunity to perform the inspections.

Response:
For some plants, the final rule approval timing may be such that there is not adequate time to plan and prepare for the required baseline inspection and prepare repair contingencies, e.g. approval of the rule in June and the next refueling outage for a plant is in September. By providing a window of the next two refueling outages, the required planning and preparation can be accommodated.

Proposed Condition:
50.55a(g)(6)(ii)(F)(4)) to require essentially 100 percent coverage for axial flaws. Paragraph –2500(c) of Code Case N–770 permits examination of axial flaws with inspection coverage limitations provided essentially 100 percent coverage for circumferential flaws is achieved and the maximum coverage practical is achieved for axial flaws. This requirement on inspection limitations is inconsistent with comparable inspection requirements of the ASME B&PV Code, Section XI. Axial flaws can lead to through wall cracks and leakage of reactor coolant, which is a safety concern. This condition is necessary for the NRC to ensure that, through NRC review of an authorization of alternative inspection coverage, appropriate actions are being taken to address potential inspection limitations for axial flaws.

Response:
The requirement was put in Code Case N–770 for those instances where essentially 100% coverage cannot be achieved due to interferences from other structures. In this case, if essentially 100% coverage for circumferential flaws (100% of the susceptible material volume) and the maximum coverage practical achieved for axial flaws, and limitations noted in the examination report, the coverage requirements were considered to be satisfied. This would assure that examinations necessary to prevent a "break before leak" were completed. The modifications required to obtain larger coverage for the axial flaws would result in increased dose to personnel which would not be justified for safety concerns.

It is not uncommon for the DM welds in the PWR plants to have a taper transition from one side of the weld to the other side of the weld. This taper transition typically will not meet the flatness requirements needed to achieve essentially 100% coverage of the exam volume for a PDI qualified examination when examining for axially oriented flaws. The taper transition cannot be removed by simply removing excess weld material in the weld crown. It would typically require a change to the design of the components and welded connection to obtain a surface geometry that would allow essentially 100% coverage of the exam volume when examining for axially oriented flaws. Because an axially oriented PWSCC flaw is limited to the PWSCC susceptible material, the axial flaw size would not be large enough to result in a safety concern. This has been documented in numerous MRP reports and PWROG evaluations. Because the axially oriented PWSCC flaw does not present a safety concern, it should not be necessary to achieve essentially 100% coverage of the exam volume when examining for axially oriented flaws.

If this condition is placed on Code Case N–770, does it negate taking credit for previous "baseline inspections" of butt welds that met the requirements of MRP-139 and Code Case N–770?

<u>Proposed Condition:</u>
The NRC proposes to add a condition (§ 50.55a(g)(6)(ii)(F)(*5*)) to reword Paragraph –3132.3(b) on determining flaw growth using wording consistent with that used in the ASME B&PV Code, Section XI. Paragraph –3132.3(b) contains the statement that a "flaw is not considered to have grown if the size difference (from a previous examination) is within the measurement accuracy of the nondestructive examination (NDE) technique employed." The "measurement accuracy of the NDE technique employed" is not defined in the code case or in the ASME B&PV Code. Use of this terminology may result in a departure from the past practice when applying ASME B&PV Code, Section XI. Under the requirements of Section XI, one concludes that flaw growth has not occurred when a "previously evaluated flaw has remained essentially unchanged." The proposed condition uses this wording. This condition is necessary to clarify the requirements for determining whether flaw growth has occurred and make the requirements consistent with ASME B&PV Code requirements endorsed by the NRC in 10 CFR 50.55a.

<u>Response:</u>
Code Case N-770-1, approved by the ASME on Dec. 25, 2009, Paragraph –3132.3(b) has been modified to read as follows:

> *Previously evaluated flaws that were mitigated by the techniques identified in Table 1 need not be reevaluated nor have additional or successive examinations performed if new planar flaws have not been identified or the previously evaluated flaws have remained essentially unchanged.*

Adoption of Code Case N-770-1 would remove this condition.

<u>Proposed Condition:</u>
The NRC proposes to add a condition (§ 50.55a(g)(6)(ii)(F)(6)) on welds that are determined through a volumetric examination to have cracking that penetrates beyond the thickness of the inlay or cladding. The condition would require such welds to be reclassified as Inspection Item A–1, A–2, or B, as appropriate, until

corrected by repair/ replacement activity in accordance with IWA–4000 or by corrective measures beyond the scope of Code Case N–770. Code Case N–770 would permit welds mitigated by inlay or cladding (i.e., onlay) in Inspection Items G, H, J, and K, to remain in those Inspection Items if cracking that penetrates through the thickness of the inlay or cladding occurs. The purpose of an inlay or cladding is to provide a corrosion resistant barrier between reactor coolant and the underlying Alloy 82/182 weld material that is susceptible to PWSCC. If cracking penetrates through the thickness of an inlay or cladding, the inspection frequencies of Inspection Items G, H, J, and K would no longer be appropriate even after satisfying the successive examination requirements of paragraph –2420. This condition is necessary because welds with cracking that penetrates beyond the thickness of the protective barrier of the inlay or cladding would no longer be mitigated and would need to be inspected under one of the Inspection Items for unmitigated welds.

Response:
Code Case N-770-1, approved by the ASME on Dec. 25, 2009, added the following to the end of Note 16(c):

> *If cracking penetrates beyond the thickness of the inlay or onlay, the weld shall be reclassified as Inspection Item A-1, A-2, or B, as appropriate, until corrected by repair/replacement activity in accordance with IWA-4000 or by corrective measures beyond the scope of this Case (e.g., stress improvement).*

Adoption of Code Case N-770-1 would remove this condition.

Proposed Condition:
The NRC proposes to add a condition (§ 50.55a(g)(6)(ii)(F)(7)) on welds in Inspection Items G, H, J, and K, (welds mitigated by inlay or cladding) that the ISI surface examination requirements of Table 1 should apply whether the inservice volumetric examinations are performed from the weld outside diameter or the weld inside diameter. Code Case N–770 only requires a surface examination for welds in Inspection Items G, H, J, and K if a volumetric examination is performed from the weld inside diameter surface. A volumetric examination performed from the weld outside diameter surface would not be capable of detecting flaws in an inlay or cladding. This condition is necessary to ensure that weld inlays or cladding are still performing their intended function of providing a protective barrier between the reactor coolant and the underlying Alloy 82/182 weld that is susceptible to PWSCC.

Response:
Code Case N-770-1, approved by the ASME on Dec. 25, 2009, modified the "Extent and Frequency of Examination" column in Table 1 to state:

> *".....Twenty-five percent of this population shall receive surface examination (17) performed from the weld inside surface and a volumetric examination (16) performed from either the inside or outside surface......."*

This same modification was applied to Inspection Item G, H, J, and K. Adoption of Code Case N-770-1 would remove this condition.

Proposed Condition:
The NRC also proposes, as part of a new condition as § 50.55a(g)(6)(ii)(F)(7), to require that all hot-leg operating temperature welds in Inspection Items G, H, J, and K (welds mitigated by inlay or cladding) be inspected each interval and that a 25 percent sample of cold leg operating temperature welds in Inspection Items G, H, J, and K be inspected whenever the core barrel is removed (unless it has already been inspected within the past 10 years) or 20 years, whichever is less. Code Case N–770 permits welds in Inspection Items G, H, J, and K to be placed in a 25 percent sample inspection program under certain conditions after the required initial inspection. The NRC has performed analyses of crack growth in welds mitigated by Alloy 52/152 inlay or cladding using experimentally derived crack growth data for this weld material. The results of those analyses show that welds in Inspection Items G, H, J, and K at hot leg temperature have to be examined once per interval and welds at cold leg temperature have to be inspected under a sample inspection program to detect potentially significant crack growth. This condition is being proposed to ensure that ASME Code allowable limits would not be exceeded and PWSCC would not lead to leaks or ruptures.

Response:
Code Case N–770 requires that a pre-service inspection and at least one inservice inspection be performed before a weld mitigated by inlay or onlay can be put in the 25% population. This would provide early crack detection and the detection of any fabrication induced cracks. Thereafter, the leading indicator approach is taken in that the hottest, most susceptible, welds are inspected each interval. If these show indications of new cracking or growth of existing cracks, then the additional and successive examination paragraphs of the Case would apply to expand the examination. This is consistent with the philosophy applied to all the other mitigation techniques employed in the Case.

The analysis performed by Battelle assumed that a crack was present and then grown. However, no experimental data has been produced that shows that a PWSCC crack can be initiated in alloy 690 material. The performance of steam generator tubes made from alloy 690 would also support the absence of PWSCC initiated cracks in this material. Hence, with two inspections performed prior to placing the hot leg inlays and onlays in the 25% population, and the inspection of the most susceptible welds each interval, this provides defense in depth for future cracking. Even with the extremely conservative assumptions employed in the Battelle analysis, cold leg inspection is not justified unless flaws are discovered in the hot leg welds which is the approach taken in this Case.

Response to NRC Conditions on Code Case N-770

Proposed Condition:
The NRC proposes to add a condition (§ 50.55a(g)(6)(ii)(F)(*8*)) to prohibit the first examination following weld inlay, cladding, or stress improvement for Inspection Items D, G, and H from being deferred to the end of the interval. Code Case N–770 provides requirements on the timing of the first examination following weld inlay, cladding, or stress improvement. Inspection Items D, G, and H pertain to mitigation of cracked welds and the timing of the initial examinations in the code case has been specified in the code case so that the welds are not in service for an extended time period prior to the initial examination. However, the code case does not explicitly preclude deferral of these examinations to the end of the interval. Therefore, this NRC condition is needed to ensure that the initial examinations of welds in Inspection Items D, G, and H take place on an appropriate schedule to verify the effectiveness of the mitigation process.

Response:
Code Case N-770-1, approved by the ASME on Dec. 25, 2009, modified Notes 11(b)(1) and (2) as follows:

> *11(b) Examinations of welds originally classified Table IWB-2500-1, Category B-F welds, Item Numbers B5.10, and B5.20 prior to mitigation, may be deferred following weld inlay, onlay, overlay, or stress improvement, as follows:*
> *(1) Examination for Inspection Item C may be deferred to the end of the interval and performed coincident with the vessel nozzle examinations required by Category B-D.*
> *(2) The first examinations following weld inlay, onlay, weld overlay, or stress improvement for Inspection Items E through K shall be performed as specified. For Inspection Item D, the first examinations following stress improvement may be performed any time within 10 years following mitigation. Subsequent examinations for Inspection Items D through K may be performed coincident with the vessel nozzle examinations required by Category B-D.*

Adoption of Code Case N-770-1 would remove this condition.

Proposed Condition:
The NRC proposes to add a condition (§ 50.55a(g)(6)(ii)(F)(*9*)) on Measurement or Quantification Criterion I–1.1 of Appendix I to require the assumption in the weld residual stress (WRS) analysis of a construction weld repair from the inside diameter to a depth of 50 percent of the weld thickness extending 360° around the weld. Measurement or Quantification Criterion I–1.1 does not specify the circumferential extent of the repair that must be assumed. This condition is necessary to clarify the size of the repair to be assumed in the weld residual stress analysis which would ensure that appropriate criteria for the WRS analysis are used for mitigation by stress improvement.

Response to NRC Conditions on Code Case N-770

Response:
Code Case N-770-1, approved by the ASME on Dec. 25, 2009, modified paragraph I-1.1 to read as follows:

> *".....A pre-stress improvement residual stress condition resulting froma construction weld repair from the inside surface to a depth of 50% of the weld thickness and extending for 360 deg. Shall be assumed."*

Adoption of Code Case N-770-1 would remove this condition.

Proposed Condition:
50.55a(g)(6)(ii)(F)(*10*)) on Measurement or Quantification Criterion I–2.1 of Appendix I to require that the last sentence be replaced. This criterion was inappropriately worded since this criterion pertains to the permanence of a mitigation process by stress improvement and plastic "shakedown" rather than "ratcheting" is the phenomenon that could lead to stress relaxation. This condition is necessary to clarify the type of analysis necessary to ensure that the mitigation process is permanent and that the inspection frequencies associated with the process continue to be correct.

Response:
Code Case N-770-1, approved by the ASME on Dec. 25, 2009, modified paragraph I-2.1 to read as follows:

> *"....The analysis or demonstration test shall account for (a) load combinations that could relieve stress due to shakedown and (b) any material properties related to stress relaxation over time."*

Adoption of Code Case N-770-1 would remove this condition.

Proposed Condition:
The NRC proposes to add a condition (§ 50.55a(g)(6)(ii)(F)(*11*)) to require that in applying Measurement or Quantification Criterion I–7.1 of Appendix I, an analysis be performed using IWB–3600 evaluation methods and acceptance criteria to verify that the mitigation process will not cause any existing flaws to grow. Measurement or Quantification Criterion I–7.1 permits the growth of existing flaws in welds mitigated by stress improvement. This is an inappropriate provision since the process of mitigating by stress improvement is intended to prevent growth of existing flaws which could lead to leakage or rupture of the weld. This condition is necessary to ensure that stress improvement of welds with existing flaws is an effective mitigation technique consistent with the inspection frequency in the code case.

Response:
Code Case N-770-1, approved by the ASME on Dec. 25, 2009, modified paragraph I-7.1 to read as follows:

> *An analysis shall be performed using IWB-3600 evaluation methods and acceptance criteria to verify that the mitigation process will not result in*

any existing flaws to become unacceptable over the life of the weld, or before the next scheduled examination.

This wording will assure that stress improvement of welds with existing flaws is an effective mitigation technique consistent with the inspection frequency in the code case. It is also consistent with the Code Case methodology. If we were to require that flaws do not grow, than why would subsequent examinations need to be performed? Adoption of Code Case N-770-1 would remove this condition.

Proposed Condition:

The NRC proposes to add a condition (§ 50.55a(g)(6)(ii)(F)(*12*)) to require that the NRC be provided with a report if the volumetric examination of any mitigated weld detects new flaws or growth of existing flaws that exceed the acceptance standards of IWB–3514 and are found to be acceptable for continued service through an analytical evaluation or a repair or the alternative requirements of an ASME code case. The report would summarize the evaluation, along with inputs, methodologies, assumptions, and cause of the new flaw or flaw growth and would be provided to the NRC prior to the weld being placed in service. Welds that are mitigated have been modified by a technique, such as weld inlays, cladding, or stress improvement. Mitigation techniques are designed to prevent new flaws from occurring and prevent the growth of any existing flaws. If volumetric examination detects new flaws or growth of existing flaws in the required examination volume, the mitigation will not be performing as designed and the NRC will need to evaluate the licensee's actions to address the problem. Therefore, this condition is needed to verify the acceptability of the weld prior to being placed back in service.

Response:

Submittal of this report to the NRC is appropriate.

Proposed Condition:

The NRC proposes to add a condition (§ 50.55a(g)(6)(ii)(F)(*13*)) to require that the last sentence of the Extent and Frequency of Examination for Inspection Items C and F be revised. Inspection Items C and F apply to butt welds mitigated by full structural weld overlays of Alloy 52/152 material. Note 10 of the Code Case requires that welds in Inspection Items C and F that are not included in the 25 percent sample be examined prior to the end of the mitigation evaluation period if the plant is to be operated beyond that time. This condition would ensure that welds in the 25 percent sample are also examined prior to the end of the mitigation evaluation period; that is, prior to the end of life of the overlay predicted by the mitigation evaluation. Inspection prior to the end of the mitigation evaluation period is necessary to ensure that appropriate information has been obtained to verify the condition of the weld overlay and update the analysis for the predicted life of the weld overlay.

Response to NRC Conditions on Code Case N-770

Response:
Code Case N-770-1, approved by the ASME on Dec. 25, 2009, added the following sentence to the Extent and Frequency of Examination for Inspection Items C and F:

> *For each overlay in the 25% sample that has a design life of less than 10 yr., at least one inservice inspection shall be performed prior to exceeding the life of the overlay.*

Adoption of Code Case N-770-1 would remove this condition.

Proposed Condition:
50.55a(g)(6)(ii)(F)(*14*)) on the 1/2-inch (13 mm) dimension shown in Figures 2(b) and 5(b) of Code Case N–770. The condition would require that a dimension "b" be used instead of c inch, where "b" is equivalent to the nominal thickness of the nozzle or pipe being overlaid, as appropriate. The code case contains information on component thicknesses to be used in application of the acceptance standards of ASME B&PV Code, Section XI, IWB–3514, to evaluate flaws detected during preservice inspection of weld overlays. The 1/2-inch (13 mm) dimension shown in Figures 2(b) and 5(b) is nonconservative. The appropriate dimension is a function of the nominal thickness of the nozzle or pipe being overlaid and not a single specified value for all pipes and nozzles. This condition is necessary to ensure that acceptance standards used for evaluation of any flaws detected during preservice inspection of weld overlays assure an appropriate level of safety.

Response:
Code Case N-770-1, approved by the ASME on Dec. 25, 2009, removed the 1/2-inch (13 mm) dimension shown in Figures 2(b) and 5(b) of Code Case N–770 and replaced them with dimensions "X" and "Y". The notes beneath each figure define dimensions "X" and "Y" as follows:

> *Dimension "x" or "y" is equivalent to the nominal thickness of the nozzle end preparation or the pipe, respectively, being overlaid.*

Adoption of Code Case N-770-1 would remove this condition.

Proposed Condition:
The NRC proposes to add a condition (§ 50.55a(g)(6)(ii)(F)(*15*)) on the use of the acceptance standards of ASME B&PV Code, Section XI, IWB–3514, for evaluating indications in inlays or onlays. The proposed condition specifies that the thickness "t" in IWB–3514 is the thickness of the inlay or onlay. The code case requires that the preservice examination for inlays or onlays consist of a surface examination, which does not allow planar flaws, and a volumetric examination. The volumetric examination allows the use of the acceptance standards of IWB–3514 provided the surface examination acceptance standards

are satisfied. That is, it would allow the acceptance of some subsurface indications, but IWB–3514 acceptance standards would only allow very small flaws. However, the code case does not specify the value "t" to be used in the application of IWB–3514. The appropriate value "t" when applying IWB–3514 to inlays or onlays is the thickness of the inlay or onlay, since the acceptance standards in this case only apply to accepting flaws within the inlay or onlay. This condition is necessary to preclude the misapplication of the acceptance standards of IWB–3514 and potential acceptance of flaws that could compromise the integrity and function of the inlay or onlay as a protective barrier.

Response:
Note 15(e) does not explicitly define the value of "t". However, the wording implies that when you are evaluating flaws in the inlay'onlay, the thickness of the inlay/onlay is the "t" to be used and when evaluating flaws in the base material, the base material thickness is "t". In a future revision to N-770, these definitions can be added to note 15(e) to remove this condition.

Proposed Condition:
The NRC proposes to add a condition (§ 50.55a(g)(6)(ii)(F)(*16*)) on welds mitigated by stress improvement by welding in Inspection Items D and E to not permit them to be placed into a population to be examined on a sample basis after the initial examination. Stress improvement by welding is also called an optimized weld overlay. Code Case N–770 permits welds mitigated by this technique to be placed in a 25 percent inspection sample after the initial examination. Sample inspections could result in three-quarters of the welds never being examined after the initial examination. Although full structural weld overlays have been used extensively in the nuclear industry for many years, the industry does not have experience with optimized weld overlays. Optimized weld overlays are designed to rely on the outer 25 percent of the original Alloy 82/182 material to satisfy the design margins and would not satisfy design margins if significant cracking were to occur. If significant cracking were to occur in the Alloy 82/182 material, the optimized weld overlay material would prevent the weld from leaking and could potentially rupture without prior evidence of leakage under design basis conditions. The proposed condition is necessary to ensure that all optimized weld overlays are periodically inspected for potential degradation.

Response:
Code Case N–770 requires that a pre-service inspection and at least one inservice inspection be performed before a weld mitigated by an optimized overlay can be put in the 25% population. This would provide early crack detection and the detection of any fabrication induced cracks. Thereafter, the leading indicator approach is taken in that the hottest, most susceptible, welds are inspected each interval. If these show indications of new cracking or growth of existing cracks, then the additional and successive examination paragraphs of

the Case would apply to expand the examination. This is consistent with the philosophy applied to all the other mitigation techniques employed in the Case.

Response to NRC Comments Concerning ASME B&PV Code, Section XI, Nonmandatory Appendix E, "Evaluation of Unanticipated Operating Events."

Introduction and Background

In the mid to late 1970s there were a number of low temperature overpressure (LTOP) and pressurized thermal shock (PTS) events that resulted in pressures and temperatures that were outside the limits established for normal operation of pressurized water reactor (PWR) reactor pressure vessels (RPVs). At the time these events occurred there were no NRC guidelines or industry standards available to uniformly assess the effect of these events, and each event was assessed on an ad-hoc basis by the NRC and industry. This procedure led to extended reactor start-up and shutdown times and inefficient use of industry and NRC resources.

As a result of these events, NRC and industry took action on several fronts. First, NRC required installation of protective measures to preclude LTOP type transients that were observed to occur at approximately 200°F during reactor start-up. In addition, NRC issued 10 CFR 50.61 and developed Regulatory Guide 1.154 for assessing PTS events.

At the same time, industry and NRC, through participation in ASME Code committees, initiated action to develop a procedure that would allow NRC staff and utility engineers to quickly and efficiently assess the condition of a PWR reactor pressure vessel following an event that resulted in pressure and temperature conditions outside the limits established for normal plant operation. The Code committee consisted of a group of experts experienced in RPV integrity evaluations, including NRC staff, NRC consultants from National Laboratories and industry engineers.

The Code committee defined two major paragraphs in Appendix E. Paragraph E-1200 provided conservative screening criteria for evaluation of RPV integrity following an unanticipated isothermal pressure transient or an unanticipated pressurized thermal transient. The purpose of Paragraph E-1200 was to provide an evaluation procedure that could be used in a relatively short time to assess RPV integrity following an unanticipated event. For example, for pressurized thermal transients, Paragraph E-1200 states that the structural integrity of the RPV is assured if the pressure does not exceed the design pressure and the coolant temperature is not less than RT_{NDT} + 55°F at any time during the event, where RT_{NDT} is the highest adjusted reference temperature (for weld or base material) at the inside surface of the reactor vessel as determined in Regulatory Guide 1.99 Rev. 2. Paragraph E-1300 defined a procedure for performing a detailed transient-specific analysis in the event the criteria in Paragraph E-1200 could not be satisfied. In the event that the criteria in Paragraph E-1300 could not be satisfied, additional analyses or other actions were to be taken to assure that acceptable margins of safety would be maintained during subsequent operation

The technical basis for the appendix was peer reviewed and the computational results used to define the Appendix E criteria were verified by NRC staff, NRC consultants and industry Code members. The technical basis for the appendix is provided in EPRI report NP-5151, "Evaluation of Reactor Vessel Beltline Integrity Following Unanticipated Operating Events," April 1987. As a result of the ASME Code effort Appendix E was approved by the ASME Code.

Following general NRC practice, no regulatory document comparable to Appendix E was developed by NRC because Appendix E was considered by the NRC and industry to be the standard by which unanticipated events would be evaluated. This practice is used by the NRC to preclude the necessity of using their resources to develop redundant NRC guidelines or regulations when acceptable industry standards are available. When the NRC believes an industry standard is, by itself inadequate, that standard is augmented by the NRC. An example of this is ASME, Section XI, Appendix G, where Appendix G of 10CFR50 adds several additional requirements to ASME, Section XI, Appendix G.

Recently, during the development of the alternate PTS rule, members of the Advisory Committee on Reactor Safety (ACRS) asked if there was a procedure that could be used to evaluate RPV integrity if a PTS type event actually occurred. The response from the NRC staff was that Appendix E would be used. The ACRS then asked if Appendix E was consistent with the risk-informed basis used to define the limits in the alternate PTS rule. Following that discussion, industry initiated an analysis to assess if the current evaluation procedure in Appendix E, Paragraph E-1200 is consistent with risk-informed criteria and adequate for evaluating LTOP or PTS type events that may occur. Industry has just completed this work and has written a draft PVP paper to document the results. This paper is included as part of this response. The results from this work show that the current evaluation procedure in Appendix E, paragraph E-1200 is consistent with risk-informed criteria and is a conservative procedure for evaluating RPV integrity following an unanticipated operational event.

Response to NRC Comments

Responses to the NRC comments are presented in the following paragraphs. The responses demonstrate that the assumptions and the overall evaluation procedure in ASME, Section XI, Appendix E is conservative and is an acceptable method to evaluate RPV integrity following an unanticipated operational event.

A brief summary of the results from the just completed industry assessment of the safety margins corresponding to application of the Appendix E criteria and the consistency of Appendix E with risk-informed criteria follows the responses to the NRC comments.

NRC Comment 1

The justification for selecting the 1-inch deep flaw is given in the EPRI report as follows: The crack size range has an upper limit of one inch. Experience shows that the fabrication practice and inspection requirements for nuclear pressure vessels generally preclude the undetected presence of larger flaws.

The above qualitative justification for selecting the 1-inch depth for the postulated flaw is not sufficient. The ASME B&PV Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure," analysis, which can be considered as the first "screening" criterion for safe operation of an RPV, is based on a postulated flaw of one-quarter of the RPV wall thickness (1/4T). The Section XI, Appendix E analysis is employed when the ASME B&PV Code, Appendix G requirements are exceeded due to an out-of-limit condition. Hence, it is considered as the second "screening" criterion, *i.e.,* once satisfied, a refined analysis or a special

RPV inspection is not needed. As the second screening tool, the Section XI, Appendix E analysis has to be conservative.

Response to Comment 1

The original selection of the 1-inch deep axial surface flaw was based on several factors, including the performance of preservice surface examinations that can detect surface or near-surface flaws less than a millimeter in length and preservice and inservice volumetric examinations that indicated no large flaws were present in the vessel base metal.

A review of recent service experience indicates almost all of the operating reactor vessels have completed their first ten-year volumetric inspection of the vessel with no indication of any significant flaws in the base metal and no indication of cladding flaws extending into the vessel base metal. These inspections were performed per ASME Section XI, Appendix VIII, or to the supplemental requirement of NRC Regulatory Guide 1.150, both of which were especially concerned with flaws near the interface between the stainless steel vessel cladding and ferritic base metal.

The original selection of the 1-inch deep axial surface flaw has proved to be conservative based on continued service experience and has been verified by the results from the comprehensive flaw evaluation performed by the NRC during their recent work to revise the PTS rule.

The results from the flaw evaluation in the NRC PTS study are presented in Section 7.5 of NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)", August 2007, and portions are briefly summarized here.

- "No surface breaking flaws were identified in all of the weld material examined, nor was a credible physical mechanism for surface flaw generation identified."

- "Virtually all non-volumetric flaws found in welds were lack of side-wall fusion defects that exist on the fusion line between the deposited weld metal and the plate or forging being joined. Additionally, this observation implies that axial welds contain *only* axially oriented flaws whereas circumferential welds contain *only* circumferentially oriented flaws."

- "The entire inner-diameter of a nuclear RPV is clad with a thin layer of stainless steel to prevent corrosion of the underlying ferritic steel. Lack of inter-run fusion (LOF) can occur between adjacent weld beads, resulting in circumferentially oriented cracks."

- "While the data in [***Simonen***] shows a high probability (1 to 10 flaws per meter of deposited cladding weld bead) of obtaining very shallow LOF defects (1% of the clad layer thickness), only two deep LOF defects, having depths of ~50% and ~63% of the clad layer thickness, were found in all of the cladding inspected. Simonen found no evidence of LOF defects that completely compromised the clad layer".

- "We assumed that these surface breaking defects exist only in single layer cladding. Multi-layer cladding was assumed to have no surface breaking flaws because the likelihood of two LOF defects aligning in two different weld layers is quite remote."

- "It should also be noted that the empirical data used as the primary evidence to establish the distribution of embedded weld flaws do not, and cannot, provide any information about the maximum size a flaw can be. For this reason, it was decided to truncate the non-repair flaw distribution at 1-in. (2.54-cm) and the repair flaw distribution at 2-in. (5.08-cm). In both cases, the selected truncation limit exceeds the maximum observed flaw size by a factor of 2. We performed a sensitivity study with FAVOR and ascertained that, within reasonable bounds on truncation limit dimension, the estimated through-wall cracking frequency is not influenced in any significant way by the truncation limit."

The results from the NRC work demonstrate that flaws in welds are embedded flaws, and the only flaws that have potential to be surface flaws are circumferential flaws that originate in the cladding. Based on the results from the NRC comprehensive study and inservice inspection results to date it can be concluded that use of a 1-inch deep axial surface flaw provides a credible conservative assumption for evaluating unanticipated events, especially in light of the NRC's conclusion that "no surface breaking flaws were identified in all of the weld material examined, nor was a credible physical mechanism for surface flaw generation identified", that a 2-inch flaw is twice as large as any flaw found in the NRC study and is much larger than any flaw detected in operating nuclear pressure vessels, and the estimated through-wall cracking frequency is not influenced in any significant way by the truncation limit.

NRC Comment 2

In the probabilistic fracture mechanics (PFM) analyses supporting the proposed PTS rule, the truncated flaw depth for a repair weld flaw is 2 inches. For a deterministic analysis, the possibility of having a repair weld flaw line up with a clad flaw to become a surface flaw cannot be ruled out.

Response to Comment 2

The results from the flaw evaluation in the NRC alternate PTS rule are presented in Section 7.5 of NUREG-1806 and state in part:

- "It should also be noted that the empirical data used as the primary evidence to establish the distribution of embedded weld flaws do not, and cannot, provide any information about the maximum size a flaw can be. For this reason, it was decided to truncate the non-repair flaw distribution at 1-in. (2.54-cm) and the repair flaw distribution at 2-in. (5.08-cm). In both cases, the selected truncation limit exceeds the maximum observed flaw size by a factor of 2. We performed a sensitivity study with FAVOR and ascertained that, within reasonable bounds on truncation limit dimension, the estimated through-wall cracking frequency is not influenced in any significant way by the truncation limit."

This statement indicates that the truncated value of 2-inches is, in fact, a factor of two larger (in the limiting case of a repair weld) than any flaw ever seen in the study or found in service after several thousand years of reactor operation. In addition, the real flaws are embedded flaws, while the postulated flaw is a surface flaw, which adds additional conservatism. Finally, the NRC demonstrated that "within reasonable bounds on truncation limit dimension, the estimated through-wall cracking frequency is not influenced in any significant way by the truncation limit."

Moreover, in Section 3.3.3.4 of NUREG-1806, the NRC concludes:

- "In FAVOR, flaws simulated to exist further than ⅜·*twall* from the inner diameter surface are eliminated, a priori, from further analysis. This screening criterion is justified based on deterministic fracture mechanics analyses, which demonstrate that for the embrittlement and loading conditions characteristic of PTS, such flaws have zero probability of crack initiation. As illustrated in Figure 3.5, in practice, crack initiation almost always occurs from flaws that having their inner crack tip located within 0.125·*twall* of the inner diameter, further substantiating the appropriateness of eliminating cracks deeper than ⅜·*twall* from further analysis."

The results presented in NUREG-1806, Section 3.3.3.4 correspond to transient pressure and temperature stresses with a margin of 1. In this instance, the results show that a flaw larger than 1-inch (approximately 0.125 *twall*) has almost no contribution to failure. Coupling this fact with the application of a safety margin of 1.6 on both the pressure and thermal K_I values used to develop the criteria in Paragraph E-1200 ensures that any transients that may contribute to RPV failure will be screened out by application of the Paragraph E-1200 criteria.

In addition, the NRC stated:

- "Multi-layer cladding was assumed to have no surface breaking flaws because the likelihood of two LOF defects aligning in two different weld layers is quite remote."

If the likelihood of two LOF defects aligning in two different weld layers is quite remote, then the likelihood that a flaw in the cladding would line up with an embedded flaw in the weld also is quite remote. Moreover, the flaw in the cladding is circumferential while the Appendix E analysis uses the more conservative axial flaw orientation. Consequently, there is no meaningful flaw alignment effect.

NRC Comment 3

The Pressure Vessel Research User's Facility (PVRUF) and Shoreham RPV flaw data, used to develop generic flaw distributions for the proposed PTS rule, identified flaws that were consistently smaller than the proposed bounding flaw. However, the PVRUF and Shoreham data represent only a limited sampling of all RPV welds and may not directly provide an adequate bounding flaw size for a deterministic analysis like that of ASME B&PV Code, Section XI, Appendix E.

Response to Comment 3

The results from the flaw evaluation in the NRC alternate PTS rule are presented in Section 7.5 of NUREG-1806 and state in part:

- "Consequently, it is not possible to ensure on an empirical basis alone that the flaw distributions developed based on these data apply to all PWRs in general. However, the flaw distributions proposed in [***Simonen***] rely on the experimental evidence gained from inspections of the materials summarized in Table 7.1 do not rest solely on this empirical evidence. Along with these data Simonen et al. used both physical models and expert opinions when developing their recommended flaw distributions. Additionally, where detailed information was lacking Simonen et al. made conservative judgments (for example, all NDE indications were modeled as cracks and, therefore, potentially deleterious to RPV integrity). This combined use of empirical evidence, physical models, expert opinions, and conservative judgments allowed Simonen et al.to propose flaw distributions for use in FAVOR that are believed to be appropriate/conservative representations of the flaw population existing in PWRs in general."

Based on this NRC conclusion the 1-inch deep axial surface flaw used as the basis for defining the Appendix E evaluation criteria can be considered conservative for application to PWRs generally.

NRC Comment 4

The use of a 1/4T flaw assumption also provides additional assurance that any service-induced growth of current fabrication flaws will be bounded for any RPVs having experienced severe transients over the course of their operating lifetimes.

Response to Comment 4

An evaluation of sub-critical crack growth is included in NUREG-1806, and the NRC concluded:

- "Growth of initial fabrication defects attributable to sub-critical cracking mechanisms does not need to be considered."

The details of the NRC evaluation are presented in NUREG-1806, Section 3.3.3.2.

Consequently, based on the NRC work in NUREG-1806, Section 3.3.3.2 there is no need to increase the evaluation flaw depth from 1-inch to 1/4T to accommodate subcritical flaw growth of fabrication defects.

NRC Comment 5

Requiring that a 1/4T flaw be used in the LEFM evaluation with a margin of 1.4 applying to K_{Im} in the two LEFM criteria establishes a consistent approach regarding the postulated flaw size in the two deterministic LEFM analyses in ASME B&PV Code, Section XI, Appendices E and G. Applying the margin of 1.4 only to K_{Im} is consistent with the ASME B&PV Code, Section XI, Appendix G approach, making the decreased margin between the two appendices traceable. The proposed use of a smaller margin of 1.4 in the ASME B&PV Code, Section XI, Appendix E analysis is justified because all significant stress intensity factors resulting from an actual transient are considered. Further, using a 1/4T flaw is also consistent with prior NRC approaches for evaluation of RPV structural integrity after out-of-limit events. The EPRI NP-5151 report mentioned that reference toughness K_{IR} has been used in the LEFM evaluation in the prior NRC evaluation of RPV structural integrity after out-of-limit events. Consistent with the evolution of the ASME B&PV Code, Section XI, Appendix G analysis, the NRC now proposes to use K_{Ic} instead of K_{IR} in the ASME B&PV Code, Section XI, Appendix E analysis.

Response to Comment 5

At this time, there have been several risk-informed assessments made for various reactor pressure vessel conditions. These include the alternate PTS rule, alternate risk-informed ASME, Section XI, Appendix G procedures to define limits for normal startup and shutdown of the reactor pressure vessel, and the criteria in ASME, Section XI, Appendix E. What has become clear from these studies is that what is important are the limits established for pressure, temperature, or material RT_{NDT}, not the specific deterministic variables, such as flaw size and margin that may be used to define these limits. For example, the pressure temperature limits in ASME, Section XI, Appendix G could easily be defined with a smaller reference flaw size and higher margins on load, or with a larger reference flaw size and decreased margin on RT_{NDT}, or any combinations of these. This means that changing from a 1-inch flaw to a T/4 flaw for purposes of consistency is not an adequate technical justification for the change without an assessment and understanding of the underlying overall safety margin provided by the change.

In addition, Section 3.3.3.5 of NUREG-1806, states in part:

- "When running a plant-specific analysis using FAVOR, we only calculated the CPTWC for TH transients that reach a minimum temperature at or below 400°F (204°C). This *a priori* elimination of transients is justified based on experience and deterministic calculations, both of which demonstrate that such transients lack adequate severity to have non-zero values of CPTWC, even for very large flaws and very large degrees of embrittlement. Additionally, the results of our plant-specific analyses (reported in Chapter 8) show that a minimum transient temperature of 352°F (178°C) must be reached before CPTWC will rise above zero, validating that our elimination of transients with minimum temperatures above 400°F (204°C) does not influence our results in any way."

The criteria in Appendix E, Paragraph E-1200 for thermal transients states that the coolant temperature of any transient cannot fall below RT_{NDT} + 55°F at pressures up to design pressure, where RT_{NDT} is the highest adjusted reference temperature (for weld or base material) at the inside surface of the reactor vessel and includes the margin term defined in Regulatory Guide 1.99 Rev. 2. For older plants with high radiation sensitive materials, such as those considered in the industry Appendix E assessment and the NRC plant specific PTS assessments, the limiting mean RT_{NDT} at the vessel inner surface is approximately 270°F and the RG 1.99 margin typically is approximately 60°F. This means that minimum transient temperature corresponding to Appendix E criterion is approximately 385°F. The NRC results described in NUREG-1806, Section 3.3.3.5 provide further confirmation that the criteria and evaluation procedure in Appendix E, provide adequate safety margins since the NRC plant specific analyses for the limiting plants show that a minimum transient temperature of 352°F (178°C) must be reached before the conditional probability of through-wall cracking (CPTWC) will rise above zero even for very large flaws and very large degrees of embrittlement.

In summary, because all evidence indicates that use of a 1-inch deep axial surface flaw provides a credible conservative assumption for evaluating unanticipated events, and because the criteria in Appendix E, Paragraph E-1200 provide adequate levels of safety there is no safety benefit to changing from a 1-inch flaw to a T/4 flaw and disallowing the use of Paragraph E-1200 for the purpose of having consistency with Section XI, Appendix G.

Risk-Informed Assessment of Appendix E

The results from industry's just completed evaluation of Appendix E demonstrate that the criteria in Appendix E, Paragraph E-1200 provide adequate margins relative to normal operation and unanticipated events, such as PTS and LTOP. The details of this evaluation are summarized in this section and are contained in the draft PVP paper, which has been included with this response.

Introduction and Background

The development of the alternate PTS rule and, recently, a proposed risk-informed procedure for ASME, Section XI, Appendix G, led to initiation of this study to determine if the Appendix E evaluation criteria are consistent with risk-informed acceptance criteria.

This study considered a range of event conditions that cover a wide range of unanticipated operational events. The event characteristics were selected based on service experience and previous probabilistic risk assessment (PRA) analyses. For PWR vessels, these events include isothermal overpressure events and pressurized rapid cool-down events.

Risk-Informed Acceptance Criteria

To assess the consistency of Appendix E with risk-informed evaluation criteria an allowable risk-informed acceptance criterion was defined for unanticipated events that actually occur. The bases for determining the risk-informed allowable are the safety goals defined by the U.S. NRC for core damage frequency (CDF) and large early release frequency (LERF) of fission products at operating nuclear plants. These goals set limits on CDF and LERF from all events that may lead to core damage and subsequent potential for large early fission product release. In addition, no single event should constitute a large portion of the CDF and LERF.

The approach used to assess the consistency of Appendix E with risk-informed criteria is based on the through-wall cracking frequency, TWCF, or

$$TWCF = F \times CPF,$$

where

F is the event frequency (events/ reactor operating year), and

CPF is the conditional probability of failure during the event (dimensionless).

For purposes of this study the frequency of an unanticipated event that has actually occurred at a plant is set equal to 1.0/yr., and the required maximum CPF is set equal to 1E-6. The magnitudes of the frequencies and CPFs for PTS transient that contribute to failure vary over a wide range from approximately 1E-2 to 1E-4, so that in general the transients that contribute to failure for PTS and other events have CPF values significantly greater than the 1E-6 value used to assess the consistency with risk-informed applications. Consequently, the Appendix E risk-informed assessment criterion of CPF = 1E-6 will screen out any transient with even a very small contribution to failure.

Analysis Assumptions

The CPF was computed using the same FAVOR Computer Code software that was used for the PTS Risk Study by NRC and the Risk-Informed Appendix G Study by EPRI. The beltline regions of the vessel having the maximum values of RT_{NDT} were modeled and used for the FAVOR analyses. The models include variation in fluence and material chemistries for the beltline region plates and welds. The axial welds are the limiting material. The analyses were performed using the same distribution of embedded flaws in the vessel wall that were used in PTS and Appendix G risk-informed analyses.

Evaluation Procedure

The following procedure was used to define the pressure and temperature combinations that satisfy the risk-informed criteria for isothermal, overpressure events. First, a constant pressure and temperature were specified. These values were used in the FAVOR software to compute CPF. If the initial selections did not correspond to CPF = 1E-6, then a new temperature was selected and CPF was recalculated. This procedure was repeated until the temperature was found that produced CPF = 1E-6 at the original specified pressure. This process was completed for a range of constant pressures and corresponding temperatures to generate an isothermal pressure temperature curve where each point on the curve corresponds to CPF = 1E-6.

A similar procedure was used for the evaluation of PWR vessel cool-down events. In this instance, the vessel cooled down at a specified rate from the temperature at normal full power operation to a final temperature, while holding pressure constant.

Results

The results are presented in Figure 1 for isothermal pressure transients, such as LTOP, and in Figure 2 for pressurized thermal transients, such as PTS, where pressure is plotted as a function of $T\text{-}RT_{NDT}$ for conditions corresponding to CPF = 1E-6. Included in the figures are the Appendix E limits from Paragraph E-1200.

The results in the Figure 1 show that risk-informed curve corresponding to CPF = 1E-6 lies to the left of the limit curve from Appendix E, Paragraph E-1200 for isothermal pressure transients. This result demonstrates that Appendix E is consistent with risk-informed evaluation criteria, corresponds to a CPF of less than 1E-6 and provides an effective and conservative methodology to screen out unanticipated isothermal pressure transients that may contribute to RPV failure.

The results in the Figure 2 for pressurized thermal transients show that risk-informed curve corresponding to CPF = 1E-6 is essentially coincident with the Appendix E criteria at pressures above 2,400 psi and cool-down rates greater than 400°F/hr and lies to the left of the limit curve from Appendix E, Paragraph E-1200 for pressures less than 2,400 psi and cool-down rates less than 400°F. This result demonstrates that Appendix E is consistent with risk-informed evaluation criteria, corresponds to a CPF of less than 1E-6 and provides an effective and conservative methodology to screen out unanticipated pressurized thermal transients that may contribute to RPV failure.

Conclusions

The results presented in Figure 1 and Figure 2 show that the current criteria in Appendix E:

- are consistent with a risk-informed evaluation approach,
- are conservative compared to the CPF = 1E-6 risk-informed curves,
- will screen out unanticipated isothermal pressure transients and pressurized thermal transients that may contribute to RPV failure, and
- provide an appropriately conservative methodology for evaluating RPV integrity following an unanticipated isothermal pressure transient or pressurized thermal transient that exceeds the operational limits in PWR plant operating procedures.

Consequently, modifying Appendix E as suggested by the NRC is unnecessary and disallowing use of paragraph E-1200 of Appendix E will result in an undue hardship without any compensating increase in safety.

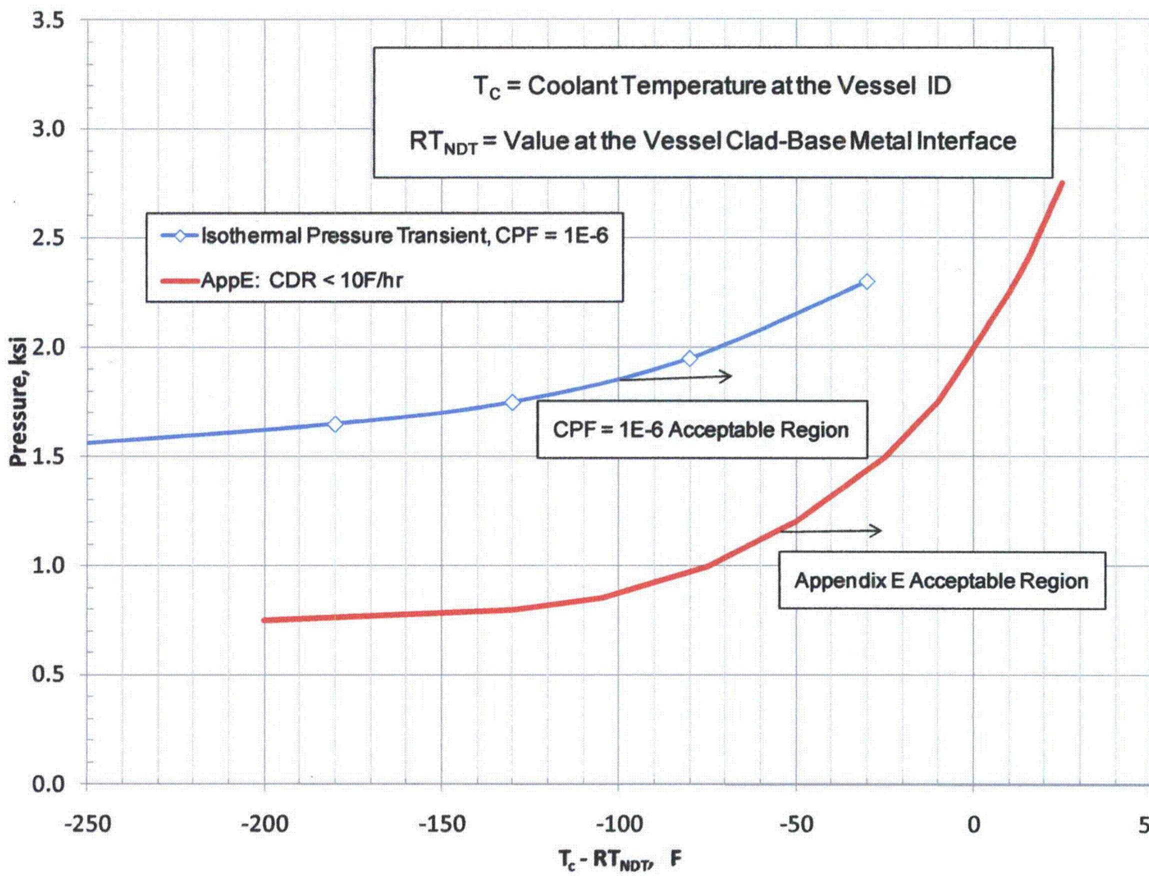

Figure 1: Comparison of the Limit in Appendix E, Paragraph E-1200 with the CPF = 1E-6 Pressure versus Temperature Line for PWR Unanticipated Isothermal Pressure Events.

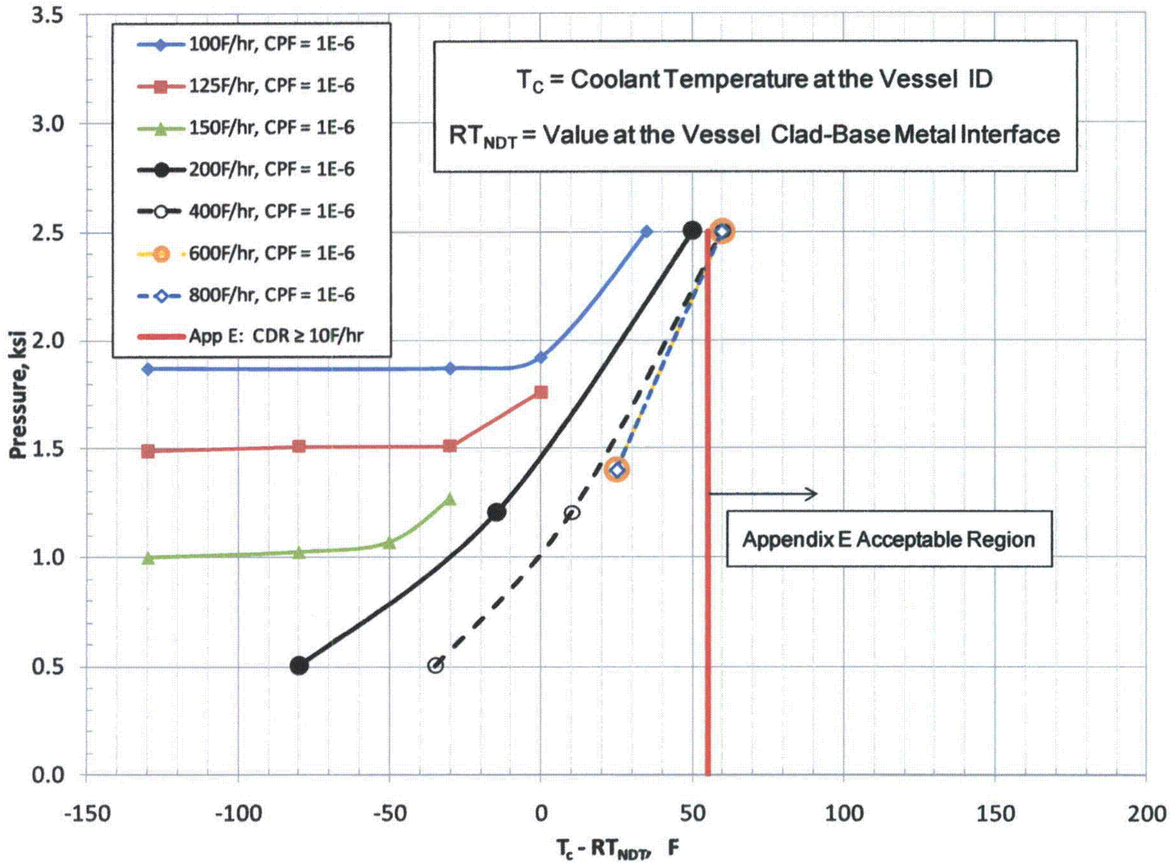

Figure 2: Comparison of the Limit in Appendix E, Paragraph E-1200 with the CPF = 1E-6 Pressure Versus Temperature Lines for PWR Unanticipated Pressurized Cool-down Events.

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A RISK-INFORMED ASSESSMENT OF ASME SECTION XI, APPENDIX E

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ABSTRACT

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code [1], Section XI, Non mandatory Appendix E, "Evaluation of Unanticipated Operating Events", provides a deterministic procedure for evaluating reactor pressure vessel (RPV) integrity following an unanticipated event that exceeds the operational limits defined in plant operating procedures.

The recently developed risk-informed procedure for Appendix G to Section XI of the ASME Code [2, 3], and the development by the U.S. Nuclear Regulatory Commission (NRC) of the alternate Pressurized Thermal Shock (PTS) rule [4, 5, 6] led to initiation of this study to determine if the Appendix E evaluation criteria are consistent with risk-informed acceptance criteria.

The results of the work presented in this paper demonstrate that Appendix E is consistent with risk-informed criteria developed for PTS and Appendix G and ensures that evaluation of RPV integrity following an unanticipated event would not violate material or operational limits recently defined using risk-informed criteria.

Currently, Appendix E does not have evaluation criteria for BWR vessels; however, as part of this study, risk-informed analyses were performed for unanticipated heat-up events and isothermal, overpressure events in BWR plant designs.

INTRODUCTION

Appendix E of Section XI of the ASME Code provides a methodology to justify continued operation of the RPV following an unanticipated event that results in temperatures and pressures outside the limits defined in the plant operating procedures.

Two major paragraphs are defined in Appendix E. Paragraph E-1200 provides conservative screening criteria for evaluation of RPV integrity following an unanticipated isothermal pressure transient or an unanticipated pressurized thermal transient. The purpose of Paragraph E-1200 is to provide an evaluation procedure that could be used in a relatively short time to assess RPV integrity following an unanticipated event.

Paragraph E-1300 defines a procedure for performing a detailed transient analysis in the event the criteria in Paragraph E-1200 cannot be satisfied. If the criteria in Paragraph E-1300 cannot be satisfied additional analyses or other actions are to be taken to ensure that acceptable margins of safety would be maintained during subsequent operation.

Appendix E was developed for PWR plant designs using deterministic fracture mechanics analyses and acceptance criteria. The recently developed risk-informed procedure for Appendix G to Section XI of the ASME Code [2,3], and the development by the NRC of the alternate PTS rule [4, 5, 6] led to initiation of this study to determine if the Appendix E evaluation criteria are consistent with risk-informed acceptance criteria.

This work evaluated selected plants where the vessel materials surrounding the core were most sensitive to neutron irradiation. The evaluation included determining appropriate material properties, reviewing operating history, and performing probabilistic fracture mechanics analyses.

This study considered a range of event conditions that would envelop a wide range of unanticipated operational events. The event characteristics were selected based on service experience and previous PRA analyses. For PWR vessels, these events include isothermal overpressure events and pressurized rapid cool-down events.

Currently, Appendix E does not have evaluation criteria for BWR vessels; however, as part of this study, risk-informed analyses were performed for unanticipated heat-up events and isothermal, overpressure events in BWR plant designs.

This paper documents a risk-informed methodology used determine if Section XI, Appendix E is consistent with the risk-informed criteria employed to define operating and material limits for the recently developed risk-informed procedures for Appendix G to Section XI of the ASME Code and the alternate PTS rule.

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APPROACH

The approach used in this study defined bounding conditions for RPV vessel materials irradiation sensitivity and for isothermal pressure transients and pressurized thermal transients. These material and transient conditions were evaluated using the FAVOR Software [7, 8] to determine those combinations of pressure and coolant temperature, T_C, where the calculated conditional probability of failure (CPF) for the RPV was 1E-6. The CPF value of 1E-6 was defined to be consistent with safety goals established by the NRC [9] and the criteria used in the PTS [5, 6] and Section XI, Appendix G risk-informed evaluations [2, 3].

The pressure and T_C conditions corresponding to a CPF = 1E-6 were then compared to the deterministic Appendix E limits. The deterministic Appendix E limits were considered to be consistent with a risk-informed approach if they were found to be reasonably conservative compared to the pressure and T_C conditions corresponding to a CPF = 1E-6.

This approach was applied to the screening criteria in Appendix E, Paragraph E-1200. Paragraph E-1300 defines a detailed transient-specific fracture mechanics analysis procedure that can be used in the event that the screening criteria in Paragraph E-1200 cannot be satisfied. A risk-informed evaluation of the procedure in Paragraph E-1300 was not necessary because the underlying procedure in Paragraphs E-1200 and E-1300 are essentially the same [10], and if the results from the risk-informed evaluation of the criteria in Paragraph E-1200 indicate the procedure is consistent with a risk-informed approach then the criteria in Paragraph E-1300 also would be consistent with a risk-informed approach.

APPENDIX E "EVALUATION OF UNANTICIPATED OPERATING EVENTS",

Appendix E, Paragraph E-1200 provides conservative screening criteria for evaluation of RPV integrity following an unanticipated isothermal pressure transient or an unanticipated pressurized thermal transient. For isothermal (cool-down rate less than 10°F/hr) pressure transients RPV integrity is assured if the conditions in Table 1 are maintained throughout the event.

Table 1: Maximum Allowable Pressure as a Function of (T_C – RT_{NDT}) for Isothermal (cool-down rate < 10°F/hr) Pressure Transients: Design pressures greater than 2400 psig (16.5 MPa)

$T_C - RT_{NDT}$ °F (°C)	Maximum Allowable Pressure psig (MPa)
+25 (14) and greater	1.1 x Design
+15 (8)	2400 (16.5)
+10 (5.5)	2250 (15.5)
0 (0)	2000 (13.8)
−10 (−5.5)	1750 (12.1)
−25 (−14)	1500 (10.3)
−50 (−28)	1200 (8.3)
−75 (−42)	1000 (6.9)
−105 (−58)	850 (5.9)
−130 (−72)	800 (5.5)
−200 (−111)	750 (5.2)

RT_{NDT} is the highest adjusted reference temperature (for weld or base material) at the inside surface of the reactor vessel as determined by Regulatory Guide 1.99 Rev. 2 [11].

For pressurized thermal transients (cool-down rate ≥ 10°F/hr) RPV integrity is assured if the maximum pressure does not exceed the design pressure and (T_C – RT_{NDT}) is not less than 55°F (31°C).

If the criteria in Paragraph E-1200 cannot be satisfied then adequate structural integrity of the RPV beltline region is assured if it can be shown by analysis, using the input of Table 2, that the following criterion is met throughout the event:

$$1.4\,(K_{Im} + K_{It}) + K_{Ir} \leq K_{Ic}$$

K_{Im} is the stress intensity factor due to membrane stress, K_{It} is stress intensity factor due to thermal stress, K_{Ir} is stress intensity factor due to residual stress, and K_{Ic} is the fracture toughness per Section XI, Appendix A, Article A-4000.

Table 2. Evaluation Input for Plant and Event Specific Linear Elastic Fracture Mechanics Analysis

Variable	Value
Pressure	Event pressure time history
Temperature	Event temperature time history
Heat transfer	Event/plant specific flow/mixing conditions
Crack type	Semi-elliptical surface flaw
Minimum initiation crack size	$0.0 < a \leq 1.0$ in. (25 mm) [Note 1]
Crack orientation	Longitudinal
K_{Ic}/K_I location	Surface and maximum depth
Clad effects	Clad to be considered in the thermal stress and fracture mechanics analyses [Note 2]
Transition toughness	K_{Ic} per Article A-4000
Upper shelf toughness	(In course of preparation)
Fluence	Fluence at the time of the transient
Shift curve	Regulatory Guide 1.99 Rev. 2
Residual stress	Appropriate distribution for the fabrication process or linear distribution with +10 ksi (+69 Mpa) at the inside surface and -10 ksi (-69 Mpa) at the outside surface

If the criteria in Paragraph E-1300 cannot be satisfied additional analyses or other actions are to be taken to ensure that acceptable margins of safety would be maintained during subsequent operation.

RISK-INFORMED ACCEPTANCE CRITERIA

To assess the consistency of Appendix E with risk-informed evaluation criteria an allowable risk-informed acceptance criterion was defined for unanticipated events that actually occur. The bases for determining the risk-informed allowable are the safety goals defined by the U.S. NRC for core damage frequency (CDF) and large early release frequency (LERF) of fission products at operating nuclear plants [9]. These goals set limits on CDF and LERF from all events that may lead to core damage and subsequent potential for large early fission product release. In addition, no single event should constitute a large portion of the CDF and LERF.

The applicable percentage of total CDF and LERF for any event should be selected on a case-by-case basis to provide an effective balance between efficient plant operation and safety, and to ensure that the total safety goal is not absorbed by relatively few events, which may then preclude later application of risk-informed evaluations for other events.

For both the PTS and risk-informed Appendix G studies, the conservative assumption was made that CDF and LERF are equal to the frequency at which flaws grow through the vessel wall. The same assumption was made for this Appendix E risk-informed study. The allowable through-wall cracking frequency (TWCF) for PTS is 1E-6 (events per operating year) [5, 6] and the allowable used to define the risk-informed Appendix G procedure is 2E-7 (events per operating year) [2, 3].

To maintain consistency with the criteria used in the PTS rule and the risk-informed Appendix G, an allowable TWCF equal to 1E-6 events per year was used to assess the consistency with Appendix E. This value was selected because it should be greater than the value used for anticipated events in a risk-informed Appendix G, but no greater than that used for the alternate PTS rule. In this instance, the risk-informed criterion then can be expressed as

$$\text{TWCF} = \text{CPF} \times \text{F} \leq 1\text{E-6 events per operating year,} \quad (1)$$

CPF is the conditional probability that a flaw grows through the vessel wall during an unanticipated event (dimensionless), and F is the frequency of the unanticipated event (events per operating year).

Although, on average, unanticipated events are relatively low frequency, the fact that the event did occur at a specific plant requires that the frequency, F, for that event and specific plant be set equal to one. Consequently, from Equation 1, CPF $\leq$ 1E-6 can be used to assess the risk consistency of Appendix E for unanticipated events that actually occur at a plant.

The magnitudes of the frequencies and CPFs for PTS transients that contribute to failure vary over a wide range from approximately 1E-2 to 1E-4, so that, in general, the transients that contribute to failure for PTS and other events have CPF values significantly greater than the 1E-6 value used to assess the consistency with risk-informed applications. Consequently, the Appendix E risk-informed assessment criterion of CPF = 1E-6 will screen out any transient with even a very small contribution to failure and is conservative.

PFM ANALYSIS PROCEDURE AND INPUT

Probabilistic fracture mechanics (PFM) analyses were performed for a wide range of RPV temperatures, pressures and cool-down rates to determine the conditions that would provide CPF equal to approximately 1E-6. The procedure used to perform the calculations involved the following steps:

1. Identify the PWR and BWR vessels having the maximum values of RT_{NDT} in the RPV beltline.
2. Define the PWR and BWR vessel beltline geometries and associated material locations for each of the vessels identified in Step 1.
3. For each material in the vessel beltline model define the fluence, applicable element contents, and unirradiated value of RT_{NDT}, $RT_{NDT\,(u)}$ [12].
4. Define a matrix of isothermal pressure transients and pressurized thermal transients that encompass the range of interest for potential unanticipated events.
5. The following procedure was used to compute CPF for evaluation of isothermal pressure transients in BWR and PWR vessels. First, a constant pressure and temperature were specified. These values were used in the FAVOR software to compute CPF. If the initial selections did not correspond to CPF = 1E-6, then a new temperature was selected and CPF was recalculated. This procedure was repeated until the temperature was found that produced CPF = 1E-6 at the original specified pressure. This process then was repeated for various combinations of final temperature and pressure to get enough points to construct a curve over the pressure range of interest, where each point on the curve corresponds to CPF = 1E-6.
6. The following procedure was used to compute CPF for evaluation of pressurized cool-down thermal transients in PWR vessels. In this instance, the vessel cooled down at a specified rate from the temperature at normal full power operation to a final temperature, while holding pressure constant. Figure 1 illustrates the type pressure and temperature time histories that were used for this evaluation. These conditions were used in the FAVOR software to compute CPF. If CPF was not 1E-6 then the final temperature was changed and the process was repeated until the final temperature was found that produced 1E-6 at the original specified pressure and cool-down rate. This process was repeated for the originally specified cool-down rate for various combinations of final temperature and pressure to get enough points to construct a curve over the pressure range of interest, i.e., 500 psi to 2500 psi, where each point on the curve corresponds to CPF = 1E-6. Individual curves were developed for cool-down rates ranging for 100°F/hr to 800°F/hr.
7. The following procedure was used to compute CPF for evaluation of pressurized thermal transients in BWR vessels. Previous work to develop the risk-informed procedure for Appendix G [2, 3] indicated that heat-up rather than cool-down was more limiting for BWR vessels, and calculations were performed to generate a heat-up pressure and temperature curve, where each point on the curve corresponds to CPF = 1E-6. Unlike cool-down, where flaws near the inner wall would be major contributors to failure, flaws in both the inner and outer wall locations can be significant contributors to failure during heat-up. Consequently, both flaw locations were evaluated for heat-up. The CPF = 1E-6 curve was developed for a heat-up rate of 60°F/hr. Higher rates were not evaluated because BWR systems' designs preclude any higher heat-up rates. In this instance, an initial temperature at start-up was selected and heat-up proceeded at a specified rate and constant pressure to the temperature corresponding to full power operation. Figure 2 illustrates the type pressure and temperature time histories used for this evaluation. These conditions were entered into FAVOR to compute CPF. If CPF was not 1E-6 then the initial start-up temperature was changed and the process was repeated until the temperature was found that produced 1E-6 at the original specified pressure and heat-up rate. This process was repeated for various combinations of initial start-up temperature and pressure to get enough points to construct a curve over the pressure range of interest, i.e., 500 psi to 1500 psi, where each point on the curve corresponds to CPF = 1E-6.

Software for Computing CPF

The CPF values were computed with software that uses Monte Carlo sampling to compute the conditional probability of reactor pressure vessel failure for any specified pressure and temperature time history at the vessel inner surface. This software, FAVOR, was developed by Oak Ridge National Laboratory (ORNL) under sponsorship of the U.S. NRC, and has been used extensively by the NRC to evaluate PTS transients in PWR reactor pressure vessels.

Two versions of FAVOR were used; namely, FAVOR [7], which evaluates flaws in the inner portions of the vessel wall, and FAVOR-HT [8], which evaluates flaws

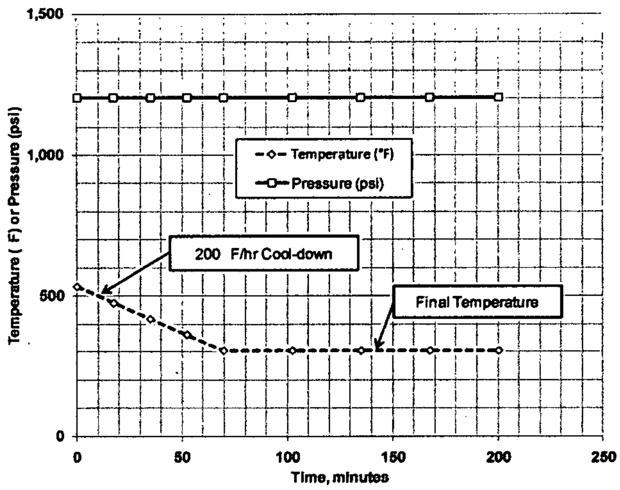

Figure 1. Illustration of PWR Cool-down Pressure and Temperature Time Histories for FAVOR Analysis

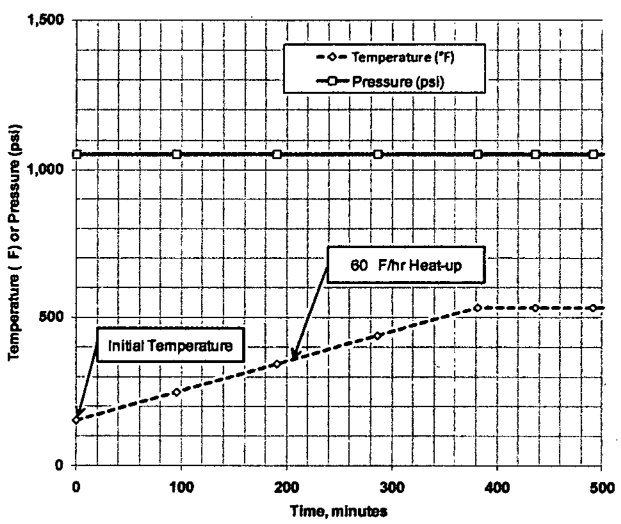

Figure 2. Illustration of BWR Heat-up Pressure and Temperature Time Histories for FAVOR Analysis

in the outer portion of the vessel wall. FAVOR is comprised of a load module, which contains the load and temperature time histories for the event, and a PFM module, which contains the irradiation degradation related variables and associated sampling parameters. The FAVOR software received extensive verification and validation [13, 14, 15, 16] as part of the PTS study.

FAVOR uses fracture mechanics models to predict extension and growth of flaws located in the vessel wall. Non-ductile fracture models, based on linear elastic fracture mechanics (LEFM) computational procedures, are used to predict initial extension of existing flaws and through-wall cleavage fracture at temperatures where the toughness ranges from low to relatively high values (the lower shelf and transition temperature regions). Ductile fracture models, based on elastic-plastic fracture mechanics (EPFM) computational procedures, are used to predict ductile through-wall crack growth at temperatures where the toughness is high (the upper shelf temperature region).

The irradiation degradation, flaw size, and toughness related variables used by the software are sampled from statistical distributions. The computations include consideration of uncertainty in the distributions and values sampled from the distributions. Monte Carlo sampling techniques are used to select variable values from their respective distributions. Pressure, stress, temperature, and geometry related variables are not sampled during the analyses. Variation in the pressure and temperature are accounted for in operation by consideration of gauge measurement uncertainty. The software models the temperature dependence of the appropriate material physical and mechanical properties.

The weld and base metal volumes in the vessel beltline are assumed to contain distributions of embedded flaws. The probability of flaw extension (i.e., the probability that the material resistance to flaw extension is less than applied potential for flaw extension) is computed for each flaw and sampled toughness in a vessel wall segment. The individual probability values for each flaw are summed to obtain the calculated conditional probability of flaw extension for the vessel wall segment. This computation is repeated for a large number of trials for the vessel segment to obtain a distribution of conditional probability of flaw extension for the vessel segment.

Flaws that have a non-zero probability of extending under the applied loading conditions then are evaluated to compute the probability that the flaw will grow through the vessel wall. In this study, vessel failure is predicted to occur when a flaw grows 90% through the vessel wall. The individual values for each flaw are summed to obtain the calculated conditional probability of failure for the vessel wall segment. This computation is repeated for a large number of trials for the vessel segment to obtain a distribution of conditional probability of failure for the vessel wall segment. The mean values of these distributions are used to estimate the conditional probabilities of flaw extension and vessel segment failure in this study.

The FAVOR software includes an option to take advantage of the crack blunting phenomenon known as warm prestress [7]. This phenomenon retards the extension of flaws under certain loading conditions where the total applied stress intensity factor ($K_{Im} + K_{It}$) values are decreasing or have passed their peak value during the loading event. The warm prestress option was used in this study.

FAVOR Irradiation Degradation Model

The irradiation degradation model used in FAVOR to compute the mean value of the change in RT_{NDT} from neutron irradiation, ΔRT_{NDT}, is described by the relationship [7]:

$$\Delta RT_{NDT} = MF + CRP$$

$$MF = A(1-0.001718T_i)(1+6.13PMn^{2.47})\sqrt{(\phi t_e)}$$

where A = $1.140\text{x}10^{-7}$ for forgings
$1.561\text{x}10^{-7}$ for plates
$1.417\text{x}10^{-7}$ for welds
T_i = irradiation temperature
P = bulk P (wt%)
Mn = bulk Mn (wt%)

$$(\phi t)_e = \begin{cases} \phi t \text{ for } \phi \geq 4.39\text{x}10^{10} \text{ n/cm}^2\text{ - s} \\ \phi t\left(\dfrac{4.39\text{x}10^{10}}{\phi}\right)^{0.259} \quad \text{for } \phi < 4.39\text{x}10^{10} \text{ n/cm}^2 - \text{s} \end{cases}$$

where $(\phi t)_e$ = effective (flux-corrected) fluence
ϕ = flux ($\text{n/cm}^2\text{-s}$)
t = irradiation time (s)

$$CRP = B(1+3.77Ni^{1.191})f(Cu_e, P)g(Cu_e, Ni, \phi t_e)$$

where B = 102.3 for forgings
135.2 for plates in vessels manufactured by Combustion Engineering (CE)
102.5 for non-CE plates
155.0 for welds
Ni = bulk Ni (wt%)

$$Cu_e = \begin{cases} 0 \text{ for Cu} < 0.072 \text{ wt.\%} \\ \min[\text{Cu}_{\text{actual}}, Cu_{\max}] \text{for Cu} > 0.072\text{wt.\%} \end{cases}$$

where Cu_e = effective Cu level

Cu_{actual} = bulk Cu level (wt%)
Cu_{max} 0.243 for typical (Ni > 0.5) Linde 80 welds
0.301 for all other materials.

$$f(Cu_e, P) = \begin{cases} 0 \text{ for Cu} \leq 0.072 \\ [\text{Cu}_e - 0.072]^{0.668} \text{ for Cu} > 0.072 \text{ and P} \leq 0.008 \\ [\text{Cu - 0.072} + 1.359(\text{P - 0.008})]^{0.668} \text{ for Cu} > 0.072 \text{ and P} > 0.008 \end{cases}$$

$$g(Cu_e, Ni, \phi t_e) = \frac{1}{2} + \frac{1}{2}\tanh\left[\frac{\log_{10}(\phi t)_e + 1.139Cu_e - 0.448Ni - 18.120}{0.629}\right]$$

Materials

The vessel material conditions evaluated in this study are summarized in Table 3. The table lists the plant and plant design, the material form having the highest RT_{NDT} in the vessel beltline, the maximum fluence at the ID surface of the beltline, $RT_{NDT(u)}$, and the mean RT_{NDT} at the vessel cladding-base metal inner face. The first two plants listed in the table are the PWR and BWR having the highest RT_{NDT} in the beltline region of the vessels. The remaining plants were used in sensitivity studies to determine the effect of RT_{NDT} on the calculated CPF.

Table 3. List of Plants and Material and Irradiation Conditions Included in the CPF Analyses.

Plant ID	Material Form with the Highest RTNDT	Neutron Fluence at the Vessel ID (E >1 MeV) n/cm²	$RT_{NDT(u)}$ °F	Mean RT_{NDT} at the Vessel ID °F
PWR-F	Axial Weld	4.83E19	-56	269
BWR-C	Axial Weld	1.13E18	-20	160
PWR-B	Axial Weld	2.96E19	-12	225
PWR-C	Plate	1.74E19	74	253
PWR-H	Plate	2.86E19	40	86

Flaw Distribution

An extensive study was performed previously by the NRC to define the distribution of flaws in vessels fabricated in accordance with ASME Code and U.S. regulatory requirements [5]. This flaw distribution was used by the NRC for the probabilistic fracture mechanics evaluation of vessel failure during postulated PTS events [5, 6]. This distribution described the number and size of embedded flaws distributed within the vessel wall as a function of vessel wall thickness. This flaw distribution was used in this study with modification for the thickness of the vessels listed in Table 1. All flaws used in the evaluation are embedded flaws. Table 4 presents a flaw distribution generated by the FAVOR software for 10,000 vessel trials for PWR-F.

Selection of embedded flaws is based on several factors, including preservice surface examinations that can detect surface or near surface flaws less than a millimeter in length, and preservice and inservice volumetric examinations that indicated no large flaws were present in the vessel base metal. In addition, the NRC in their study [5] found no surface breaking flaws in their volumetric and destructive examinations of several welds and concluded that there was no credible physical mechanism for surface flaw generation.

Table 4. Embedded Flaw Distribution for Welds in Plant PWR-F.

Flaw Through-wall Extent inch	Number of Flaws in Ten Thousand Vessel Simulations
0.085	5,120,182
0.170	1,014,049
0.255	73,315
0.340	17,977
0.425	6,667
0.510	2,635
0.595	1,191
0.680	591
0.765	337
0.850	180
0.935	122
1.020	66
1.105	48
1.190	36
1.275	26
1.360	23
1.445	12
1.530	3
1.615	5
1.700	3
1.785	4
1.870	4
Total Flaws	6,237,476

Moreover, review of recent service experience indicates almost all of the operating reactor vessels in the U. S. have completed their first ten-year volumetric inspection of the vessel with no indication of any significant flaws in the base metal and no indication of cladding flaws extending into the vessel base metal. These inspections were performed per ASME Section XI, Appendix VIII, or to the supplemental requirement of NRC Regulatory Guide 1.150, both of which were especially concerned with flaws near the interface between the stainless steel vessel cladding and ferritic base metal.

Typical Beltline Model

The input to the FAVOR software PFM module is a mapping of fluence at the vessel ID surface, element (Cu, Ni, P, Mn) contents, $RT_{NDT(u)}$, relative position of the welds and plate, and dimensions for each of the materials in the vessel beltline. The beltline model and subregions used in the PFM analyses for plant PWR-F are shown schematically in Figure 3, where the section is cut along the center line of subregions 1 and 4 and rolled out 360°. The shaded portions of the model represent welds, and the unshaded portions represent plates. Each of the 33 subregions is assigned a fluence, element content, $RT_{NDT(u)}$, and position and dimensions representative of the vessel beltline condition for the plant.

During the sampling process each subregion is assigned a flaw from a flaw distribution similar to that presented in Table 4. The beltline model was constructed so that the fluence at all the circumferential weld subregions was the maximum anywhere in the circumferential weld, and the plate subregions adjacent to the axial welds had the same fluence as the axial welds. The fluence in the each of the axial weld and remaining plate subregions was the maximum value anywhere in that subregion.

RESULTS

PWR Isothermal and Thermal Transients

The CPF computational results for the limiting PWR vessel are presented in Figure 4, where pressure is plotted as a function of bulk coolant temperature, T_C. The figure shows the isothermal curve and curves for the various cool-down rates. The results indicate that the position of the pressure temperature curve is relatively insensitive to cool-down rate at cool-down rates greater than approximately 400°F/hr. The difference between the isothermal (ISO) case and the case for a cool-down rate (CDR) of 100 °F/hour is due to the way these events were modeled in the FAVOR input. For the ISO case model, there would never be a benefit due to warm pre-stressing; however, there likely would be a benefit for the CDR models.

To allow application of the methodology to the population of operating PWRs the results in Figure 4 were normalized by plotting pressure as a function of the difference between the bulk coolant temperature T_C and the RT_{NDT} at the base metal-clad interface. The results for the isothermal pressure transients are presented in Figure 5, and the results for the pressurized thermal transients are presented in Figure 6. Figures 5 and 6 include a comparison with the criterion in the current Appendix E.

Because Appendix E uses RT_{NDT} defined by Regulatory Guide 1.99, Rev. 2 [11], it includes the margin term defined in the Regulatory Guide. To have comparable conditions for the calculated results and the Appendix E criterion the margin term was added to the mean value for Plant F shown in Table 3. The margin term accounts for uncertainty and is added to obtain conservative upper bound values of RT_{NDT} [4]. The margin term typically ranges from 56 to 65°F for highly irradiated welds, similar to those in Plant F, and a value of 60°F is used here for illustration.

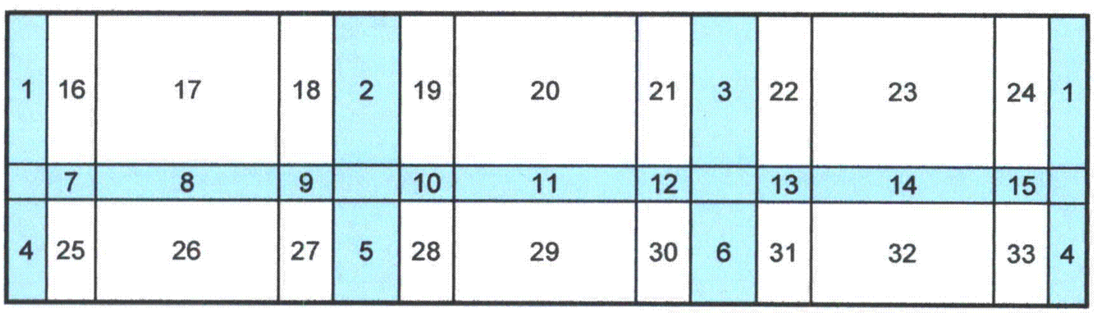

Figure 3. Schematic Representation of a Vessel Beltline Model Used in the PFM Analysis

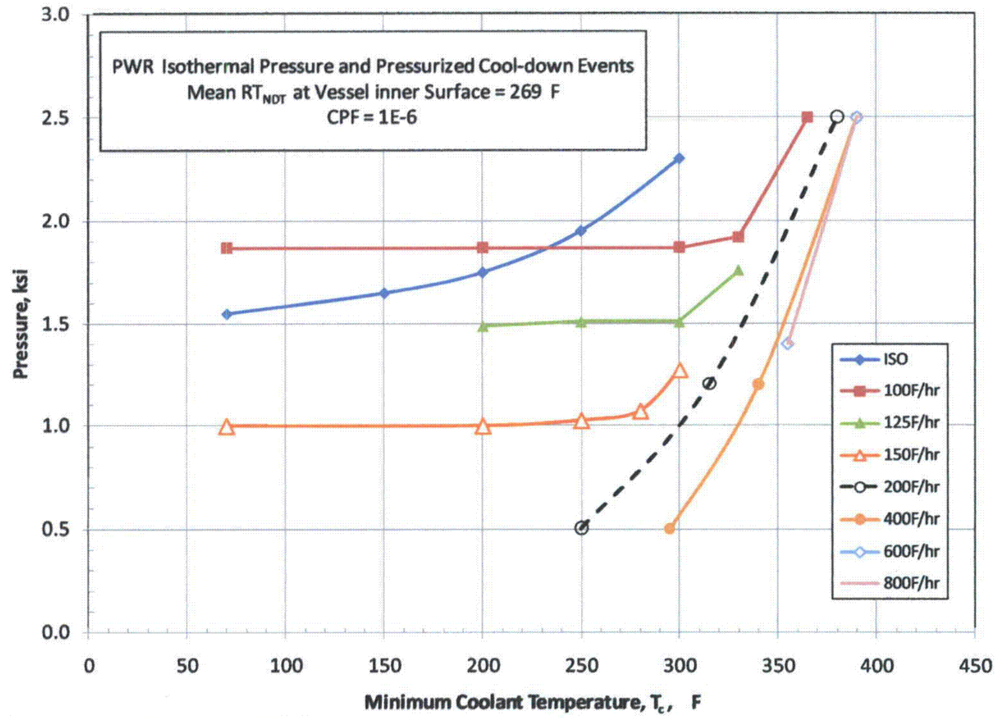

Figure 4. CPF Computational Results For PWR Isothermal Pressure Transients and Pressurized Thermal Transients

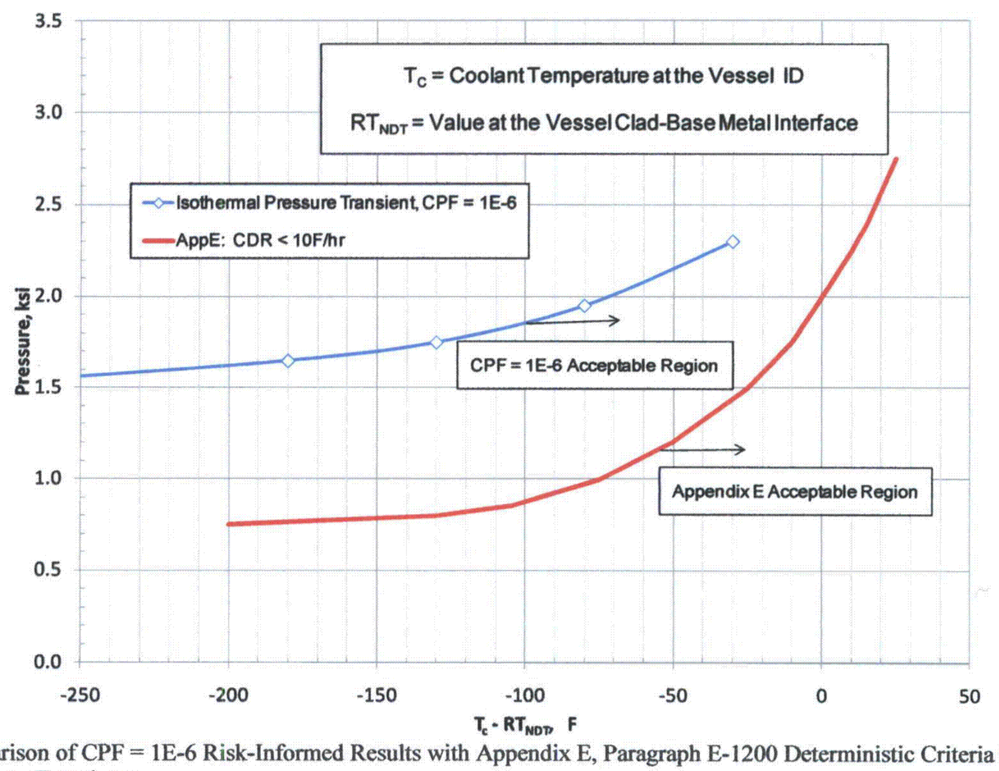

Figure 5. Comparison of CPF = 1E-6 Risk-Informed Results with Appendix E, Paragraph E-1200 Deterministic Criteria for PWR Isothermal Pressure Transients.

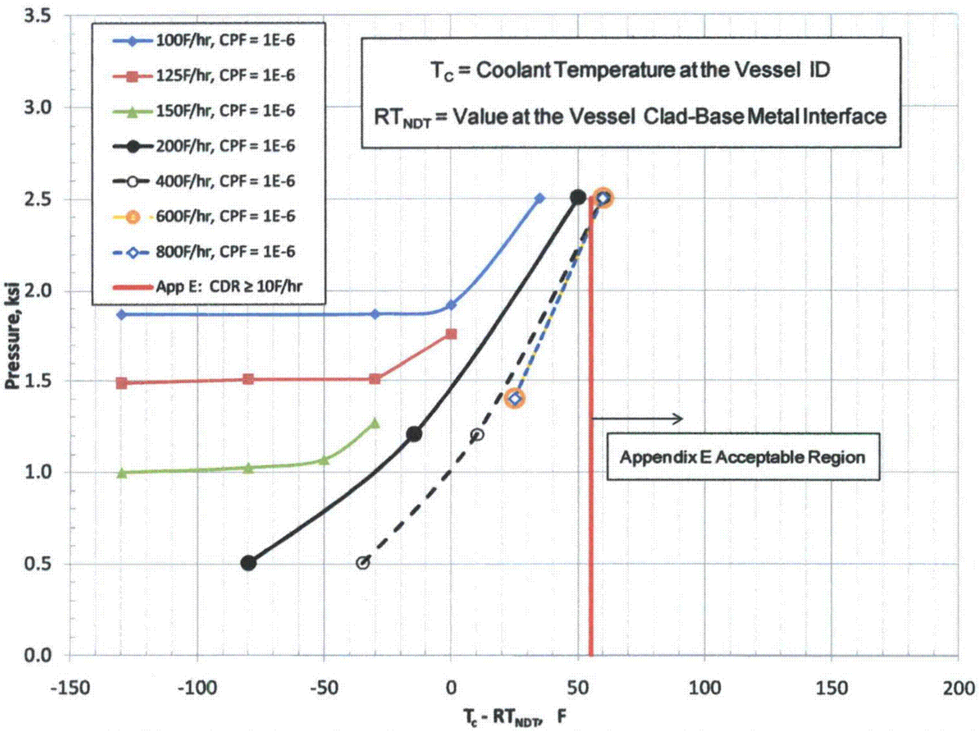

Figure 6. Comparison of CPF = 1E-6 Risk-Informed Results with Appendix E, Paragraph E-1200 Deterministic Criteria for PWR Pressurized Thermal Transients.

The results in Figures 5 and 6 show that the CPF = 1E-6 lines are to the left of the current Appendix E limits. Consequently, the Appendix E criteria are consistent with a risk-informed approach, provide a conservative bound to the CPF = 1E-6 risk-informed lines, and will screen out any unanticipated events that may contribute to RPV failure. Appendix E, appears to be especially conservative for pressures less than 1,500 psi and cool-down rates less than about 200°F/hr.

The results shown for cool-down rates from 200°F/hr to 800°F/hr are very similar to those obtained from the original deterministic Appendix E analysis [10]. In the original analysis a vertical line was drawn at T_C - RT_{NDT} = 55°F, where the allowable pressure was equal to the design pressure without regard to cool-down rate, as shown in Figure 6. This conservatism was added so the initial evaluation in Paragraph E-1200 could be performed quickly without detailed assessment of transient cool-down rates.

PWR Computation Verification and Sensitivity Study

Independent calculations were performed to verify the computational results and ensure that the calculated CPF computed for plants with the highest RT_{NDT} were conservative compared to plants with lower RT_{NDT}. This verification compared selective results from Figure 4 with independently performed calculations. The results from the verification calculations are presented in Table 5, and provide verification of the computational results shown in both Figures 5 and 6. As can be seen, the maximum differences are less than 30 percent, which is well within the computational accuracy of the FAVOR Computer Code for 10,000 vessel simulations. In addition, the CPF computed for the plants with lower RT_{NDT} are less than the plants with the limiting RT_{NDT} so the results shown in Figures 4, 5, and 6 are bounds for PWR plants generally.

BWR Isothermal Transients

Figure 7 presents the risk-informed results obtained for the limiting BWR vessel, where pressure is plotted as a function of minimum bulk coolant temperature. The results show there essentially is no difference between the isothermal curve and the curve corresponding to a 60° F/hr heat-up.

To allow application of the methodology to the population of operating BWRs the results in Figure 7 were normalized by plotting pressure as a function of (T_C – RT_{NDT}), where RT_{NDT} is computed at the clad base metal interface, and includes a margin term equal to 60°F. Figure 8 shows the results with pressure plotted as a function of T_C - RT_{NDT}.

Table 5. Results from the PWR Sensitivity and Computational Verification Studies

Plant/RT_{NDT}[a] (°F)	CDR (°F/Hr)	Pressure, P (Ksi)	T_C-RT_{NDT}[a] (°F)	T_C (°F)	Calculated CPF	Percent Difference
F / 270	800	2.5	120	390	1.26E-06	26.1
F / 270	600	1.4	85	355	9.96E-07	-0.4
F / 270	400	1.2	70	340	1.01E-06	0.6
F / 270	200	0.5	-20	250	9.34E-07	-6.6
F / 270	150	1.27	30	300	1.01E-06	1.2
F / 270	125	1.49	-70	200	1.03E-06	2.7
F / 270	100	2.5	95	365	1.06E-06	5.7
F / 270	100	1.87	-36	234	9.74E-07	-2.6
F / 270	100	1.87	-200	70	9.74E-07	-2.6
F / 270	0 (ISO)	1.55	-200	70	1.12E-06	12.0
F / 270	0 (ISO)	1.87	-36	234	7.08E-07	-29.2
F / 270	0 (ISO)	2.3	30	300	1.04E-06	4.0
C / 253	200	0.5	-20	233	7.70E-07	b
B / 225	800	2.5	120	345	<1.00E-10	b
H / 86	200	0.5	-20	66	<1.00E-10	b
D / 99	800	2.5	120	219	<1.00E-10	b
a. The RT_{NDT} value listed here is the mean value at the clad base metal interface.						
b The calculated value of CPF should be $\leq 10^{-6}$ since RT_{NDTS} is less than 270 °F.						

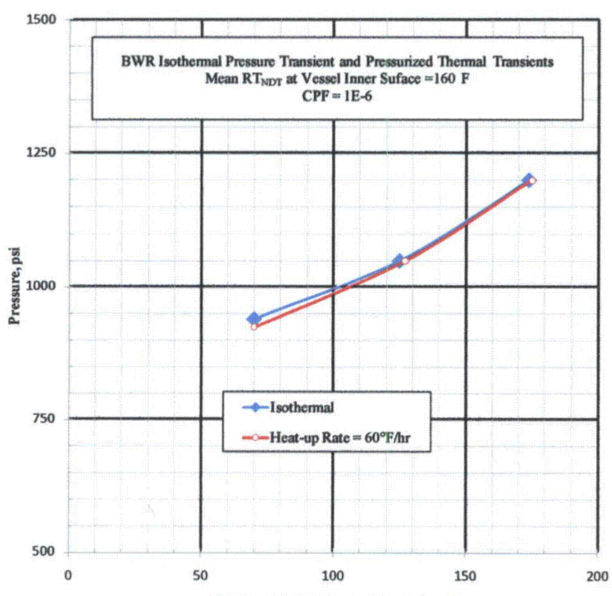

Figure 7: CPF = 1E-6 Computational Results for BWR Isothermal Pressure Transients and Pressurized Thermal Transients

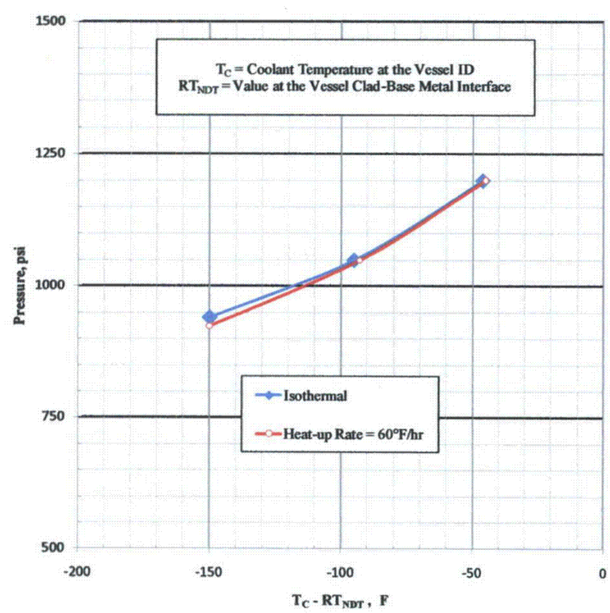

Figure 8. CPF = 1E-6 Computational Results for BWR Isothermal Pressure Transients and Pressurized Thermal Transients

Currently, Appendix E does not include evaluation criteria for unanticipated events in BWRs so no comparisons can be made between the risk-informed results and Appendix E deterministic evaluation criteria.

CONCLUSIONS

- The results in this study indicate that the current Appendix E evaluation criteria are consistent with acceptable risk-informed criteria, are conservative compared to the risk-informed curves, and will effectively screen out unanticipated events that may contribute to RPV failure.

- The current criteria in Appendix E is conservative and can be applied for evaluation of isothermal pressure transients and pressurized thermal transients that exceed the operational limits in PWR plant operating procedures.
- The results presented in Figures 5 and 6 provide a technical basis for relaxation of the current criteria in Appendix E for PWR plant vessels, especially for pressures less than approximately 1,500 psi.
- This study provides risk-informed criteria that can be used as a basis to define Appendix E criteria to evaluate unanticipated isothermal pressure transients and pressurized thermal transients in BWR vessels.

RECOMMENDATIONS

Appendix E should be modified to incorporate the risk-informed results from this study. This modification could provide some relaxation of the current requirements for PWRs, and provide evaluation procedures for unanticipated events that may occur in BWRs.

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Rulemaking Comments

From:	Scata, Linda A [scatala@westinghouse.com]
Sent:	Monday, July 19, 2010 4:30 PM
To:	Rulemaking Comments
Cc:	Molkenthin, James P
Subject:	Transmittal of PWROG Comments on the NRC "Proposed Rule Changes to 10 CFR 50, Relating to Code Case N-770 and Nonmandatory Appendix E, Evaluation of Unanticipated Operating Events", Docket ID NRC-2008-0554, PA-MSC-0483
Attachments:	OG-10-246.pdf; NRC Conditions on N-770 rev 1.doc; NRC Appendix E Enclosure.doc; NRC Appendix E Enclosure Number 2.doc

To Whom It May Concern:

Attached are the PWROG comments on the NRC "Proposed Rule Changes to 10 CFR 50, Relating to Code Case N-770 and Nonmandatory Appendix E, Evaluation of Unanticipated Operating Events", Docket ID NRC-2008-0554, PA-MSC-0483.

For technical questions regarding the enclosed PWROG comments, please contact Gary Elder (Westinghouse) at (412) 374-4884 or Carol Heinecke (Westinghouse) at (412) 374-2758. If you have any additional questions or comments on the enclosed information, feel free to contact Jim Molkenthin in the PWROG office at (860) 731-6727 or Melvin L. Arey at (704) 382-8619.

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