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Ref. # 10CFR50.55a(z)(1)

April 20, 2015

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT
DOCKET NO. 50-445
RELIEF REQUEST 1B3-3 FOR UNIT 1 INSERVICE INSPECTION FOR APPLICATION OF AN ALTERNATIVE TO THE ASME BOILER AND PRESSURE VESSEL CODE SECTION XI EXAMINATION REQUIREMENTS FOR REACTOR PRESSURE VESSEL COLD LEG WELD INSPECTION FREQUENCY
(2007 EDITION OF ASME CODE, SECTION XI, 2008 ADDENDA
THIRD INTERVAL START DATE: AUGUST 13, 2010
THIRD INTERVAL END DATE: AUGUST 12, 2020)

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(z)(1i), Luminant Generation Company, LLC (Luminant Power) is submitting Relief Request 1B3-3 (see attachment) for Comanche Peak Unit 1 for the third ten year inservice inspection interval. Luminant Power is requesting an alternative for the reactor pressure vessel cold leg weld inspection frequency as specified in Code Case N-770-1 from a period of not to exceed 7 years to a period not to exceed 9 years. The alternative process provides an acceptable level of quality and safety.

Luminant Power requests approval of this relief request by December 15, 2015, to support the upcoming CPNPP Unit 1 refueling outage.

This communication contains no new licensing basis commitments regarding Comanche Peak Unit 1. Should you have any questions, please contact Mr. Jack Hicks at (254) 897-6725.

Sincerely,

Luminant Generation Company LLC

Rafael Flores

By: Fred W. Madden
Fred W. Madden
Director, External Affairs

A047
NRR

Attachment – Relief Request 1B3-3 for Code Case N-770-1 Reactor Pressure Vessel Cold Leg Weld Inspection Frequency Extension

c - Marc L. Dapas, Region IV
Balwant K. Singal, NRR
Resident Inspectors, Comanche Peak
Robert Free, TDLR
Jack Ballard, ANII, Comanche Peak

COMANCHE PEAK NUCLEAR POWER PLANT UNIT 1
Relief Request Number 1B3-3
Code Case N-770-1 RPV Cold Leg Weld Inspection Frequency Extension
(Third 10-Year ISI Interval Start Date: August 13, 2010)

1. ASME Code Component Affected:

The affected components are the Comanche Peak Nuclear Power Plant Unit 1 (CPNPP1) reactor vessel cold leg nozzle-to-safe-end welds (TBX-1-4100-14, TBX-1-4200-14, TBX-1-4300-14 and TBX-1-4400-14), which are Alloy 600 welds covered by Code Case N-770-1, Table 1, Inspection Item B. [Reference 1]

Examination Category	Inspection Item	Description
CC N-770-1	B	TBX-1-4100-14, Loop 1 cold leg nozzle-to-safe-end weld
CC N-770-1	B	TBX-1-4200-14, Loop 2 cold leg nozzle-to-safe-end weld
CC N-770-1	B	TBX-1-4300-14, Loop 3 cold leg nozzle-to-safe-end weld
CC N-770-1	B	TBX-1-4400-14, Loop 4 cold leg nozzle-to-safe-end weld

CPNPP1 reactor vessel cold legs operate at an average temperature of 555.74°F

2. Applicable Code Edition and Addenda:

CPNPP1 is currently using the 2007 Edition through 2008 Addenda of the ASME Section XI Boiler and Pressure Vessel Code. However, Code Case N-770-1, as referenced in 10CFR50.55a(g)(6)(ii)(F), is the applicable code document for this Relief Request.

3. Applicable Code Requirement:

Table 1 of Code Case N-770-1 requires volumetric examination of essentially 100% of Inspection Item B pressure retaining welds once every second inspection period not to exceed 7 years.

4. Reason for Request: Acceptable level of quality and safety (10CFR50.55a(z)(1)).

Relief is being requested at this time due to the NRC imposition of Code Case N-770-1 through rulemaking and the scheduling aspects of the new requirement conflicting with the current plans at CPNPP1. Due to this conflict, Luminant is requesting an alternative that provides an acceptable level of quality and safety as compared to the requirements of Code Case N-770-1, as conditioned by 10CFR50.55a.

Examination of Code Case Item A-2 (hot leg) and Code Case Item B (cold leg) welds are performed from the inside diameter (ID) of the pipe at CPNPP1 due to extremely limited access provisions from the outside surface of the pipe. The CPNPP1 Item A-2 and Item B welds are located inside a "sandbox," which was installed during original plant construction after all welding was completed. The inspection of the Item A-2 (hot leg) welds from the ID does not require removal of the reactor vessel (RV) lower internals (core barrel), while the inspection of the Item B (cold leg) welds from the ID requires that the core barrel be removed for access. The cold leg weld examination, under ASME Section XI inspection requirements, occurs once per interval, which normally is scheduled to coincide with the inspection of the RV shell welds, thus minimizing core barrel removal evolutions. Inspection of these RV cold leg nozzle welds on a six- or seven-year interval requires removal of the core barrel solely for the purpose of performing these dissimilar metal (DM) weld nozzle

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inspections. Removal of the core barrel should be minimized for a variety of reasons as explained in Section 5.

Baseline inspections of Code Case N-770-1 Inspection Item B welds, TBX-1-4100-14, TBX-1-4200-14, TBX-1-4300-14 and TBX-1-4400-14 were performed in the Spring of 2010 (1RF14). The ultrasonic examinations performed in 2010 met Section XI, Appendix VIII requirements, including examination volume of essentially 100%. Table 1 of Code Case N-770-1 requires the successive examination of these welds to be performed by the Spring of 2017. Therefore, inspection of these welds would require removal of the core barrel during the Spring of 2016 (1RF18) refueling outage.

Since inspection of these welds requires that the core barrel be removed from the reactor vessel, these inspections had previously been planned to be performed concurrently with the reactor vessel shell weld inspections. Rescheduling the Code Case N-770-1 Inspection Item B weld inspections from the Spring of 2016 (1RF18) refueling outage to the Spring of 2019 (1RF20) refueling outage would allow the Code Case N-770-1 inspections and the vessel shell weld inspections to be performed during the same refueling outage. This would eliminate the need to remove the core barrel during the Spring of 2016 (1RF18) refueling outage resulting in the elimination of an additional core barrel removal, reduce radiation exposure and elimination of a critical lift in containment

5. <u>Proposed Alternative and Basis for Use:</u>

10CFR50.55a(z) states:

> "Alternatives to the requirements of paragraphs (b) through (h) of this section or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation, or Director, Office of New Reactors, as appropriate. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that:
>
> (1) The proposed alternative would provide an acceptable level of quality and safety; or
>
> (2) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

Luminant believes that the proposed alternatives of this request provide an acceptable level of quality and safety.

CPNPP1 proposes a one-time extension to the Code Case N-770-1, Table 1, Inspection Item B, volumetric examinations from a period of not to exceed 7 years to a period of not to exceed 9 years. The inspections which are currently required to be performed by the Spring of 2017 will be performed in the Spring of 2019 (1RF20) refueling outage. The basis for this alternative is provided below.

Review of the industry service experience shows that cracking has only been observed in the hot leg piping locations with DM welds, and the cold leg locations continue to exhibit very reliable service.

<u>Core Barrel Removal</u>: Due to the limited access to the RV nozzles preventing the automated inspection from the outside diameter (OD), the RV cold leg nozzles are inspected remotely from the ID. This requires that the core barrel be removed for access. However, due to the design of the

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reactor vessel, core barrel removal is only required for cold leg examination. Previously, these examinations occurred under ASME Section XI inspection requirements once per interval, which coincided with the inspection of the RV shell welds, thus core barrel removal evolutions were minimized. Inspection of these RV cold leg nozzles on a six- or seven-year interval requires removal of the core barrel solely for the purpose of performing these DM weld nozzle inspections. Removal of the core barrel should be minimized for a variety of reasons. As with any heavy lift operation, there are inherent risks to the personnel involved in the lift activities. Experience has shown that there are also risks associated with equipment damage including damage to the lift rig, guide studs, or the lower internals and reactor vessel itself. Damage to these items has the potential to put plant personnel in further adverse situations along with significantly increasing outage time and dose. Removal of the reactor vessel lower internals assembly is considered to be a high risk, infrequently performed, critical lift due to the weight of the component, the tight clearances involved, and the radiation emitted by the assembly. For these reasons, only the personnel directly involved with the movement of the internals are allowed in containment during the evolution. Remote cameras, lasers, pulleys, and ropes are utilized to allow the personnel involved with the lift to be outside of the refueling cavity area to minimize personnel radiation exposure to the extent achievable. Most of the lower internals lifts are performed solely by viewing cameras. The Polar Crane operator(s) is instructed to remain behind shielding for ALARA purposes. Communications are via portable radios. Prior to lifting the lower internals, a "dry run" is typically performed where the crane is attached to the lifting rig and placed onto the guide studs in the reactor cavity. Temporary markings are then made to provide alignment references for the reactor vessel. These markings are used by the crane operator and the crew to align the crane to the vessel. The lifting rig is then moved to the storage location and a second set of markings made. Following completion of the "dry run," the lifting rig is installed onto the guide studs and the lower internals are latched onto the rig. The internals are then lifted until full load is achieved. This position is maintained for 10 minutes. Following the 10-minute hold, the internals are lifted out of the reactor vessel and moved onto their storage stand in the refueling cavity. For CPNPP removing the core barrel requires that it be partially raised approximately 10.5 feet above the refueling cavity water level in order to obtain a minimal clearance over the reactor vessel flange during transfer from the reactor vessel to the storage stand location. As can be expected, the radiation exposure levels for this activity are high and necessitate unrelated work to stop for evacuation of personnel from containment and a reliance on shielding for the polar crane operator. The dose received during the last CPNPP Unit 1 core barrel removal and re-installation was approximately 60 millirem with the implementation of the dose saving actions described above.

Flaw Tolerance: Westinghouse has performed a generic flaw tolerance evaluation to determine the maximum flaw sizes in the reactor vessel cold leg DM welds that would support continued operation for a period of 10 years. This evaluation was performed consistent with the evaluations performed for the Reactor Coolant Pump (RCP) nozzles, which were performed in accordance with the ASME Section XI guidelines for flaw tolerance as contained in paragraph IWB-3640. Along with the normal operating steady state piping loads, the impact of welding residual stresses under different safe end lengths and the various extent of inside surface weld repairs during the initial weld fabrication process were considered in the evaluation. These residual stresses were also calculated using finite element analysis techniques that are consistent with recent industry guidance as seen in MRP-287 [Reference 3]. A parametric study was performed to evaluate the residual stresses for the different weld and safe-end configurations present in the Westinghouse fleet. Based on a comparison of the various residual stress distributions from the parametric study, it was concluded that a long (Length > 4.5") safe end with either a 25% or 50% inside surface weld repair would produce limiting PWSCC crack growth results. A high and a low cold leg operating

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temperature were also considered in the evaluation to represent the range of operating temperatures in the fleet. Based on the circumferential crack growth results, even for the most conservative case (high temperature with a 25% weld repair), a flaw with a depth of 15% of the wall thickness would not grow to the maximum allowable ASME flaw size in less than 10 years of continued operation. It should be noted that the results are not representative of a single plant. These results were based on the limiting thickness in the Westinghouse PWR fleet combined with the limiting piping loads from another plant in the Westinghouse PWR fleet and, therefore, these results are conservative. All of the flaw tolerance analyses performed to date have shown that the critical crack sizes in large diameter butt welds operating at cold leg temperatures are very large. Assuming that a flaw initiates, the time required to grow to through wall is in excess of 20 years in most cases analyzed. The time to grow from a through wall leak to a crack equal to the critical crack size can be in excess of 40 years. Furthermore, the chances of a flaw initiating in a colder location are very low. [Reference 2]

Probability of Cracking: Probabilistic fracture mechanics (PFM) evaluations were performed by Westinghouse to address the identified degradation mechanisms of Primary Water Stress Corrosion Cracking (PWSCC) and Fatigue Crack Growth (FCG) on alloy 82/182 dissimilar metal butt welds. The evaluations performed considered the limiting butt welds in large diameter pipes and smaller diameter pipes based on the deterministic evaluations for the Westinghouse, CE, and B&W NSSS designs. The RV inlet nozzle and RCP welds were not specifically evaluated because they were not the limiting locations in the deterministic evaluations. Evaluations for each of the limiting locations considered the small axial leak and small circumferential leak failure modes. The circumferential leak probabilities at 40-years are small. It must be noted that all of these probabilities are for cases evaluated at hot leg or pressurizer operating temperatures. Though not explicitly evaluated, the probabilities for locations at cold leg temperatures would be less. A statistical analysis was performed by Westinghouse to assess the susceptibility of the RCP nozzle welds to PWSCC. The analysis considered available industry experience data for the locations of Alloy 82/182 DM welds. More specifically, the data analyzed included Alloy 82/182 DM welds in large diameter pipes. The collected service experience data was fit to a Weibull distribution, which was then used to calculate the probability of cracking as a function of Effective Full Power Years (EFPY). This was done for three different temperatures with the intent of covering the range of temperatures on the cold leg nozzle DM weld locations (548°F to 556°F), as well as a representative hot leg nozzle DM weld location (615°F). Three different cases were evaluated based on the data to which the Weibull distribution was fit. Case 1 is based on all the available inspection results, for reactor vessel nozzles, steam generator nozzles, pump nozzles, and pressurizer surge nozzles. Case 2 includes all the nozzles, except the pressurizer nozzles, and Case 3 includes only the reactor vessel and RCP nozzles. The results show there is no discernable difference between the cases at the cold leg temperatures. Furthermore, the predicted probability of cracking for the pump nozzle DM welds, operating at cold leg temperatures, is extremely low, even at 60 EFPY. The results of the Weibull fitting for the three cases indicate that even though DM welds have had flaws at hot temperature locations, none have been found at cold temperature butt weld locations, and this gives a very low probability of flaws existing in cold temperature locations. Results show the highest probability of an indication at cold leg temperatures was only 1.42%, at 60 EFPY (Case 1 at 556°F). In comparison, the probability (60 EFPY) at hot leg temperatures is 23.71% (Case 1 at 615°F). Analyses have been performed to calculate the probability of failure for Alloy 82/182 welds using both PFM and statistical methods. Both approaches, statistical and PFM have shown that the likelihood of either cracking or through-wall leaks, in large diameter cold leg welds, is very small. Furthermore, sensitivity studies performed using PFM have shown that even for the more limiting high temperature locations, an inspection frequency greater than that required by Section XI, such as that

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in MRP-139 [Reference 4] or Code Case N-770-1 has only a small benefit in terms of risk. Though past service experience may not be an absolute indicator of the likelihood of future cracking, the experience does give an indication of the relative likelihood of cracking in cold leg temperature locations versus hot leg temperature locations. While there is a significant amount of PWSCC service experience in hot leg locations, the number of indications is still small relative to the number of potential locations. Also, all indications have been detected before they were a safety concern. Therefore, if hot leg PWSCC is a leading indicator for cold leg PWSCC, and the higher frequency of inspections will be maintained for the hot leg locations, it is reasonable to conclude that a moderately less rigorous inspection schedule would be capable of detecting any cold leg indications before they became large enough to be a concern. [Reference 2]

Operating experience on PWSCC of Alloy 82/182 welds show that weld repairs performed during original plant construction are a significant contributor in the initiation and propagation of cracking. A review of the construction records and a weld repair search performed for the CPNPP1 Reactor Vessel nozzle Alloy 82/182 welds did not identify any weld repairs performed on these welds during original plant construction. Additionally, CPNPP1 began elevated pH operation of 7.4 at temperature to aid in source term and dose reduction. A statistical evaluation of PWSCC tests in 2005 (*Materials Reliability Program: Effects of Hydrogen, pH, Lithium, and Boron on Primary Water Stress Corrosion Crack Initiation in Alloy 600 for Temperatures in the Range 320-330°C (MRP-147)* [Reference 5] revealed that elevated pH also has a beneficial effect on mitigating PWSCC.

Examination of Code Case Item A-2 (hot leg) and Code Case Item B (cold leg) welds are performed from the ID at CPNPP1 due to extremely limited access provisions from the outside surface of the pipe. The CPNPP1 Item A-2 and Item B welds are located inside a "sandbox" which was installed during original plant construction after all welding was completed. The inspection of the Item A-2 (hot leg) welds from the ID does not require removal of the reactor vessel core barrel, while the inspection of the Item B (cold leg) welds from the ID does require removal of the reactor vessel core barrel.

In the Fall of 2008 (1RF13), ultrasonic (volumetric) and eddy current (surface) exams were performed on the Code Case N-770-1 Inspection Item A-2 (hot leg) welds, with no indications identified. Also, in the Spring of 2013 (1RF16), ultrasonic (volumetric) and eddy current (surface) exams were performed on the Code Case N-770-1 Inspection Item A-2 (hot leg) welds, with no indications identified. In the Fall of 2017, ultrasonic (volumetric) and eddy current (surface) exams are scheduled to be performed on the Code Case N-770-1 Inspection Item A-2 (hot leg) welds. Since PWSCC is temperature dependent, it would be expected that Inspection Item A-2 (hot leg) welds would show evidence of crack initiation before Inspection Item B (cold leg) welds. Therefore, the lack of any indications in the Inspection Item A-2 (hot leg) welds provides added assurance that the one-time extension of the inspection of the Inspection Item B (cold leg) welds by three years provides an acceptable level of quality and safety.

The baseline inspection of the Code Case N-770-1 Inspection Item B (cold leg) welds, as required by Code Case N-770-1, was performed in the Spring of 2010. At that time, in addition to the ultrasonic (volumetric) examination, an additional surface examination utilizing an eddy current technique was performed. Both the ultrasonic (volumetric) and eddy current (surface) examinations were performed from the ID surface and confirmed the absence of any indications after approximately 20 years of operation. The ultrasonic examinations performed in 2010 met Section XI, Appendix VIII requirements, including examination volume of essentially 100%. Since PWSCC initiates from the inside wetted surface of the pipe and propagates radially, an internal surface examination is the

preferred inspection technique for this failure mechanism. The use of eddy current examination in addition to the Code Case N-770-1 required ultrasonic examination provides a higher probability of detection of smaller flaws than an ultrasonic examination alone. Since the Code Case N-770-1 inspection frequency is based on flaw sizes associated with ultrasonic examination, the proposed alternative provides an equivalent protection against unacceptable PWSCC as does the Code Case N-770-1 exam schedule.

Conclusion: While there has been a large amount of service experience with PWSCC of Alloy 82/182 welds, this experience has been limited to those welds operating at hot leg temperatures or higher. There have been no incidents of cracking in welds operating at cold leg temperatures that can be attributed to PWSCC. Though the MRP-139 and Code Case N-770-1 requirements for more frequent inspections were taken as proactive measures, the accumulation of more positive service experience indicates that this increased inspection frequency for cold legs, in particular, is not necessary to maintain an acceptable level of safety and quality. Furthermore, it has been realized that accessing these cold leg weld locations for inspection may present an increased risk due to the complications associated with removal of the reactor vessel core barrel. There have been numerous studies performed to evaluate the likelihood of through-wall cracking and flaw tolerance in cold leg Alloy 82/182 welds. The analyses performed as the original basis for MRP-139 showed that the large diameter cold leg welds had high flaw tolerance and a very low probability of failure. More recent analyses, which considered design specific residual stress distributions, have confirmed the original conclusions that the flaw tolerance is high. Furthermore, the more recent analyses have shown that even large circumferential flaws, with a high likelihood of being detected during inservice inspection, will not grow to the maximum depth allowed by ASME Section XI in 10 years. These analyses have been performed based on the assumption that a flaw has initiated, which as shown by more recent probabilistic analyses based on service data is unlikely at the present time. It is therefore concluded that an interval of 10 years for re-examination of large diameter cold leg Alloy 82/182 locations will provide a more than adequate level of safety and quality. Furthermore, this will reduce the risks associated with movement of the reactor vessel core barrel. In summary, no weld repairs were documented on these welds during plant construction, elevated pH operation which decreases the probability of PWSCC crack initiation has been implemented at CPNPP1 since 2005, the hot leg DM examinations including both ultrasonic and eddy current inspections were performed in 2008 and 2013 with no indications identified, and the cold leg examinations including both ultrasonic and eddy current inspections were performed from the ID in 2010 with no indications identified. Based on the above facts, the one-time alternative inspection frequency of every 9 years instead of every 7 years provides an acceptable level of quality and safety. Thus eliminating the need to remove the core barrel during the Spring of 2016 (1RF18) refueling outage.

6. Duration of Proposed Alternative:

This request is applicable to Luminant's Inservice Inspection program for the third interval for Comanche Peak Unit 1.

7. Precedents:

1. Indian Point Unit 2 Fourth Inspection Interval Relief Request IP2-ISI-RR-14. "Code Case N-770-1 Reactor Coolant System Cold Leg Nozzle Weld Inspection Frequency Extension", as approved by the NRC in a letter dated February 2, 2012 (ADAMS Accession No. ML120260090)

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2. Indian Point Unit 3 Fourth Inspection Interval Relief Request IP3-ISI-RR-07. "Reactor Vessel Cold Leg Nozzle to Safe-end Weld Examinations", as approved by the NRC in a letter dated August 4, 2014 (ADAMS Accession No. ML14199A444)

8. <u>**Reference:**</u>

1. *Code Case N-770-1*, Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of listed Mitigation Activities Section XI, Division 1.

2. *PVP2011-57829*, Changing the Frequency of Inspections for PWSCC Susceptible Welds at Cold Leg Temperatures

3. *MRP-287*, PWSCC Flaw Evaluation Guidance

4. *MRP-139*, Primary System Piping Butt Welds Inspection and Evaluation Guideline

5. *MRP-147*, Materials Reliability Program: Effects of Hydrogen, pH, Lithium, and Boron on Primary Water Stress Corrosion Crack Initiation in Alloy 600 for Temperatures in the Range 320-330°C

9. <u>**Attachment:**</u>

1. *PVP2011-57829*, Changing the Frequency of Inspections for PWSCC Susceptible Welds at Cold Leg Temperatures

Proceedings of PVP2011
2011 ASME Pressure Vessels and Piping Conference
July 17-21, 2011, Baltimore, Maryland, USA

PVP2011-57829

CHANGING THE FREQUENCY OF INSPECTIONS FOR PWSCC SUSCEPTIBLE WELDS AT COLD LEG TEMPERATURES

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ABSTRACT

A project has been completed under the sponsorship of the EPRI Materials Reliability Program to evaluate the acceptability of returning to an inservice inspection (ISI) frequency of ten years for the large diameter cold leg pipes (525 to 580F), with Alloy 82/182 dissimilar metal (DM) welds. This effort addresses alternative inspection requirements with a frequency of 7 years that have recently been imposed in order to address the potential for service induced Primary Water Stress Corrosion Cracking (PWSCC) of these welds.

Careful review of the service experience shows that cracking has only been observed in the hot leg piping locations with DM welds, and the cold leg locations continue to exhibit very reliable service. There are a number of technical and practical arguments in favor of making this change, even beyond the excellent service experience, and these arguments are summarized in this paper.

- Pulling the reactor vessel (RV) core barrel is a serious activity which can entail many risks, so additional pulls should be avoided. Inspection at a frequency of less than 10 years involves additional core barrel pulls.
- The flaw tolerance of these large diameter cold leg pipes is very good, and example calculations show that reasonably large flaws are acceptable for ten years.
- The probability of cracks initiating in cold leg piping is significantly lower than that for piping at hotter temperatures, and a detailed model has been developed to demonstrate this.

Actions are underway to revise the relevant inspection requirements, back to a more typical Section XI ten-year interval, using this technical work as a basis.

INTRODUCTION

ASME Section XI [1] has, since its inception for piping inspections in 1974, specified a 10-year interval for inservice inspection of pressure-retaining welds. As a result of Alloy 600 and Alloy 82/182 cracking incidents, MRP-139 [2] and ASME Code Case N-770 [3] both require a more proactive periodic volumetric re-examination of cold leg Alloy 82/182 dissimilar metal (DM) butt welds, essentially every six or seven years. This population includes various branch connections, reactor coolant pump (RCP) inlet and outlet nozzles, steam generator

(SG) outlet nozzles, and the RV cold leg nozzles. The branch nozzles typical of the Babcock and Wilcox (B&W) and Combustion Engineering (CE) designs, are generally inspected from the outside diameter (OD) and have varying accessibility and personnel radiation exposure issues depending on plant design and environmental conditions. Only a limited number of US plants (one) have DM welds in the SG nozzles that require inspection within the scope of MRP-139 and Code Case N-770, but these welds will also typically be examined from the OD with plant-specific access and dose implications.

The RV cold leg nozzles are typically inspected from the inside diameter (ID) which requires that the core barrel be removed for access (See Figure 1). This exam, under ASME inspection requirements, occurs once per interval (10 years typically) which coincides with the inspection of the RV shell welds, thus minimizing core barrel removal evolutions. Inspection of these nozzles on a six- or seven-year interval requires removal of the core barrel solely for the purpose of performing these DM weld nozzle inspections. Removal of the core barrel should be minimized for a variety of reasons. As with any heavy lift operation, there are inherent risks to the personnel involved in the lift activities. Experience has shown that there are also risks associated with equipment damage including damage to the lift rig, guide studs, or the lower internals and reactor vessel itself. Damage to these items has the potential to put plant personnel in further adverse situations along with significantly increasing outage time and dose.

THE PERILS OF CORE BARREL REMOVAL

Removal of the reactor vessel lower internals assembly (core barrel) is considered to be a critical lift due to the weight of the component, the tight clearances involved, and the radiation emitted by the assembly. For these reasons, only the personnel directly involved with the movement of the internals are allowed in containment during the evolution. Remote cameras are utilized to allow most of the personnel involved with the lift to be outside of the refueling cavity area to minimize personnel radiation exposure. Most of the lower internals lifts are performed solely by viewing cameras. The Polar Crane operator(s) is instructed to sit on the floor of the cab or behind shielding and not to raise his head above the cab area of the crane for ALARA purposes. Communications are via portable radios. Prior to lifting the lower internals, a "dry run" is typically performed where the crane is attached to the lifting rig and placed onto the guide studs in the reactor cavity. Temporary markings are then made to provide alignment references for the reactor vessel. These markings are used by the crane operator and the crew to align the crane to the vessel. The lifting rig is then moved to the storage location and a second set of markings made. Following completion of the "dry run," the lifting rig is installed onto the guide studs and the lower internals are latched onto the rig. The internals are then lifted until full load is achieved. This position is maintained for 10 minutes. Following the 10-minute hold, the internals are lifted out of the reactor vessel and moved onto their storage stand in the refueling cavity.

For many plants removing the core barrel requires that it be raised well above the refueling cavity water level during transfer from the reactor vessel to the storage stand location. As can be expected, the radiation exposure levels for this activity are very high and necessitate unrelated work to stop for evacuation of personnel from containment and installation of shielding for the polar crane operator(s). Additionally some plants are configured such that the core barrel upper portion remains exposed above the refueling cavity water level during storage, often requiring installation of temporary shielding walls. These walls severely limit the ability to perform other outage cavity maintenance activities and involve significant time and dose for their handling.

The design of the internals lift rig is susceptible to operational and alignment problems. Multiple events involving issues such as crane misalignment or only having two of the three lifting legs engaged/disengaged during polar crane lift have resulted in significant damage to the lifting rig and reactor components. These events occurred after all fuel was removed from the core and thus did not pose a threat to nuclear, industrial, or environmental safety. However, ALARA principles under radiological safety were challenged. Additional worker dose was accumulated during the recovery operations.

Inspection of the reactor vessel nozzles from the ID is done remotely, and is much less dose-intensive than OD inspections, while it also avoids common OD access obstructions. However, removing the core barrel is only necessary for cold leg inspections, and is not justified at a frequency less than ten years, as will be shown in the work to follow.

SERVICE EXPERIENCE FOR COLD LEG ALLOY 82/182 BUTT WELDS

Originally, all dissimilar metal (DM) welds in pipes 4" NPS and greater, including those containing Alloy 82/182, in categories B-F and B-J, were volumetrically examined every 10 years, following the requirements of ASME Section XI. In some cases, these examinations were eliminated as part of a risk-informed ISI program while in other cases they were supplemented by visual inspections for boric acid leakage. Since 2005, a more aggressive volumetric examination schedule has been in place for A82/182 DM welds in the U.S., self-imposed by industry through the requirements of MRP-139. All such welds have now been examined at least once employing qualified examination methods. Similar accelerated inspections have also been performed at PWR plants worldwide. The majority of incidents of cracking in Alloy 82/182 weld materials or Alloy 600 base metal have occurred in the reactor vessel head penetrations, head penetration welds, or the pressurizer nozzle butt welds. These locations operate at hot leg temperatures or higher. A summary

of service experience for other reactor coolant piping welds is provided herein.

The location of large diameter (≥ 14 NPS) Alloy 82/182 welds operating at cold leg temperatures in the Westinghouse, Combustion Engineering, and Babcock and Wilcox plant designs are discussed in Section 3 of MRP-113 [4] and are summarized in Table 1. A summary of service experience [5,6] for Alloy 82/182 butt welds is provided in the following sections. Though there have been numerous incidents of PWSCC identified in the pressurizer nozzle welds, this service experience is not included since these cracking incidents have occurred at temperatures significantly higher than typical cold leg temperatures.

Reactor Vessel Nozzles

The only known incidents of PWSCC in the reactor vessel inlet and outlet nozzles have occurred in the Alloy 82/182 nozzle-to-safe end weld region of the outlet nozzle, which typically operates at hot leg temperatures (>600F). No cracking has been observed in the reactor vessel inlet nozzles, which operate at cold leg temperatures (<580F), after over 2500 reactor years of service. There are six incidents of cracking which have been observed to date. The first incidents occurred in the outlet nozzles of Ringhals 3 and 4, and Virgil C. Summer in the year 2000. Since that time, over 100 automated UT examinations of these welds in operating plants in the U.S. and internationally have been completed, typically coincident with the inspection of the reactor vessel shell welds. No additional surface indications had been found until 2008, when indications were identified in the outlet nozzles of two different reactor vessels. The first was at OHI-3 in Japan. This indication was detected prior to the application of water-jet peening which was being applied to mitigate PWSCC. The indication was measured by UT as being 10 mm in length and 5 mm in depth. When the indication was actually removed by progressive grinding, it was measured to have a depth of 20.3 mm and a length of 13.5mm. The cavity has been left in place. The second indication was detected at Salem Unit 1, prior to the application of the mechanical stress improvement process (MSIP). This indication was determined to be ~24% through-wall (~15 mm). Finally, in fall of 2009, an indication was found in the Seabrook reactor vessel outlet nozzle. These results are summarized in Table 2.

Steam Generator Primary Nozzles

Cracking in the steam generator nozzles has only been observed in the Alloy 82/132 inlet nozzle-to-safe-end weld region of steam generators in Japan. For plants in the U.S. that have stainless steel reactor coolant system main loop piping, steam generators were typically fabricated with stainless steel nozzle-to-safe-end welds. Many plants have replaced their steam generators and in doing so have installed steam generators with either stainless steel welds, or welds fabricated with Alloys 52 and 152 which are not considered to be susceptible to PWSCC. In some cases Alloy 82/182 welds were used with a layer of Alloy 52/152 to seal the Alloy 82/182 material from the primary coolant water. One plant in the U.S. does have Alloy 82/182 welds in the steam generator inlet nozzle-to-safe-end welds; however, this plant is currently considering options for mitigation of these welds.

In Japan, most steam generators were originally fabricated with Alloy 132 nozzle-to-safe-end welds. Alloy 132 is similar to Alloy 182 and is equally susceptible to PWSCC. Therefore, the Japanese PWRs with susceptible welds are implementing peening as mitigation for these welds. In preparation for peening, the inside surface of the welds must be inspected. While these inspections (and subsequent peening) had been successfully applied at five plants, during the inspections of Mihama 2 and Tsuruga 2 in the fall of 2007, indications were detected. In November of 2007, NISA, the Japanese regulatory, authority issued a guideline for each susceptible unit to inspect the nozzle-to-safe-end weld region at their earliest convenience. As a result, five additional plants have detected cracking in this region. All indications have been detected in the inlet nozzle-to-safe-end weld region, which operates at hot leg temperatures. No cracking has been observed in the colder SG outlet nozzle. A detailed summary of these findings is found in Table 3.

Other Piping Weld Locations

In Combustion Engineering (CE) and Babcock and Wilcox (B&W) plants, there are a number of Alloy 182 or Alloy 82 butt welds used to join stainless steel lines (instrumentation lines, drain lines, surge lines, etc.) to the main loop piping, which is carbon steel. There have been numerous incidents of cracking in these locations. Again, the cracking has been found predominantly in the high temperature lines, with very few incidents of cracking in the colder locations [5]. However, these few incidents in the colder locations have occurred in welds with diameters of significantly less than 14 NPS, which is the reason this size restriction was considered..

Conclusions

It can be concluded that all known incidents of cracking in the U.S. in large diameter Alloy 82/182 piping welds have occurred in locations operating at hot leg temperatures or higher.

FLAW TOLERANCE OF COLD LEG WELD REGIONS

In response to the early cracking incidents discussed above, a number of analyses were performed to assess the stability of piping with PWSCC flaws and determine the likelihood of through wall crack propagation. These analyses were documented in industry reports [7,8,9] and served as the basis for the inspection and evaluation guidelines identified in MRP-139. Now that the first round of inspections required by

MRP-139 is complete, it is appropriate to review the findings, to decide if the new inspection interval is appropriate.

CE Design RCP Suction and Discharge Nozzle DM Welds

The Alloy 82/182 dissimilar metal butt welds located at the safe-end regions of the CE designed reactor coolant pump suction and discharge nozzles present inspection coverage challenges, which hinder the likelihood of obtaining the required inspection coverage (i.e. > 90%). An extensive series of evaluations have been performed recently to address this challenge and are documented in WCAP-17128-NP, Revision 1 [10].

In reference [10], a series of flaw tolerance calculations were carried out to determine the time required for a postulated surface flaw to reach the ASME Section XI allowable flaw size. These calculations were performed in accordance with the ASME Section XI guidelines for flaw tolerance as contained in paragraph IWB-3640. Both fatigue crack growth (FCG) and PWSCC were considered, and the results were presented in terms of the allowable service time for a range of flaw sizes and shapes. The calculations determined the range of flaws which are acceptable for service periods from two to four years. These calculations include the required Section XI flaw evaluation margins and were presented for both axial and circumferentially oriented flaws. Residual stresses were calculated using finite element analysis techniques [4], assuming two cases, first no weld repairs, and second, weld repairs to a through-wall depth of 15% from the ID. The results for the circumferential flaws show that very large flaws can be tolerated in this region as the residual stress effects were found to retard flaw growth for circumferential flaws. While the results for the axial flaws do not exhibit as much tolerance as for circumferential flaws, the limited length of the flaw causes the aspect ratios to also be limited. Though not included in reference [10], additional analyses consistent with those described above were performed for circumferential flaws for a service period of 10 years. The results of these evaluations, with and without residual stresses due to weld repairs, are shown in Figures 2 and 3, respectively. These results show that flaws with an aspect ratio as large as 10 and a through-wall depth of 20% will be acceptable for at least 10 years.

Reactor Vessel Inlet Nozzle Flaw Tolerance Evaluations

Westinghouse has performed a generic flaw tolerance evaluation to determine the maximum flaw sizes in the reactor vessel inlet DM welds that would support continued operation for a period of 10 years. This evaluation was performed consistent with the evaluations performed for the RCP nozzles which were performed in accordance with the ASME Section XI guidelines for flaw tolerance as contained in paragraph IWB-3640. Along with the normal operating steady state piping loads, the impact of welding residual stresses under different safe end lengths and the various extent of inside surface weld repairs during the initial weld fabrication process were considered in the evaluation. These residual stresses were also calculated using finite element analysis techniques that are consistent with recent industry guidance [11]. A parametric study was performed to evaluate the residual stresses for the different weld and safe-end configurations present in the Westinghouse fleet. Based on a comparison of the various residual stress distributions from the parametric study, it was concluded that a long (Length > 4.5") safe end with either a 25% or 50% inside surface weld repair would produce limiting PWSCC crack growth results. A high and a low cold leg operating temperature were also considered in the evaluation to represent the range of operating temperatures in the fleet.

Based on the circumferential crack growth results shown in Figure 4, even for the most conservative case (high temperature with a 25% weld repair, as determined in reference [12]) a flaw with a depth of 15% of the wall thickness would not grow to the maximum allowable ASME flaw size in less than 10 years of continued operation. It should be noted that the results presented in Figure 4 are not representative of a single plant. These results are based on the limiting thickness in the Westinghouse PWR fleet combined with the limiting piping loads from another plant in the Westinghouse PWR fleet and therefore, these results are conservative [12].

Conclusions

All of the flaw tolerance analyses performed to date have shown that the critical crack sizes in large diameter butt welds operating at cold leg temperatures are very large. Assuming that a flaw initiates, the time required to grow to through wall is in excess of 20 years in most cases analyzed. The time to grow from a through wall leak to a crack equal to the critical crack size can be in excess of 40 years [4]. Furthermore, the chances of a flaw initiating in a colder location are very low, as will be shown below.

PROBABILISTIC EVALUATIONS

All of the analyses discussed to this point have been deterministic in nature. These deterministic analyses have assumed the existence of an initiated flaw and have used conservative inputs to determine the rate of crack growth. Probabilistic analyses can be used to determine the likelihood of a flaw initiating and growing through-wall. These analyses can be performed using probabilistic fracture mechanics and also using statistical methods. These two approaches are discussed in the following sections.

Probabilistic Fracture Mechanics Approach

As part of the original effort to develop the MRP-139 requirements, a probabilistic assessment was performed by Westinghouse for domestic Westinghouse, CE and B&W

design PWR plants using probabilistic fracture mechanics (PFM) methods. Detailed results of this work are provided in MRP-116 [13]. Though this assessment was performed in 2004, it is the most recent probabilistic assessment of PWSCC susceptible welds of different sizes and operating conditions. Though there have been advancements in the understanding of variables that effect PWSCC, the assessment still provides valuable insights into the likelihood of piping weld failure due to PWSCC.

The probabilistic assessment builds on the deterministic work and addresses the probability that a flaw could grow through the wall and could eventually lead to rupture and a resultant increase in core damage frequency. The evaluations documented in the report were intended to cover all the Alloy 82/182 butt weld locations in operating PWRs in the USA. The probabilistic safety assessment brings together the deterministic results, as well as complementary work to provide input on the effects of repairs and crack growth modeling.

Probabilistic fracture mechanics evaluations were performed to address the identified degradation mechanisms of PWSCC and FCG on alloy 82/182 dissimilar metal butt welds. The evaluations performed considered the limiting butt welds in large diameter pipes and smaller diameter pipes based on the deterministic evaluations for the Westinghouse, CE, and B&W NSSS designs. The RV inlet nozzle and RCP welds were not specifically evaluated because they were not limiting locations in the deterministic evaluations. Evaluations for each of the limiting locations considered the small axial leak and small circumferential leak failure modes. The results of the PFM evaluation for the circumferential leak probabilities, which represent a direct safety concern, are summarized in Table 4.

As shown in Table 4, the circumferential leak probabilities at 40-years are small. It must be noted that all of these probabilities are for cases evaluated at hot leg or pressurizer operating temperatures. Though not explicitly evaluated, the probabilities for locations at cold leg temperatures would be less.

As part of the MRP-116 probabilistic fracture mechanics evaluations, a sensitivity study was performed to determine the effects of ISI accuracy and frequency. This sensitivity study was performed for a weld that was considered to be representative of the welds included in the study. The results of the study are shown in Table 5. Though the weld considered in this study was not a cold leg weld, the results of the study would be expected to envelope the results for cold leg weld locations.

Based on the results of the probabilistic fracture mechanics analyses, it was concluded in MRP-116 that:

- Changes in inspection frequency or improvements in capability or accuracy have only a small benefit for the locations with the highest leak probabilities.
- Risk results do not justify shortening the 10-year ASME Code Section XI inspection interval, as long as all Alloy 182/82 locations are inspected. In other words, the shorter intervals specified in MRP-139 and code Case N-770 are not needed for the cold leg locations to satisfy risk objectives.

Statistical Approach: The Probability of Cracking

A statistical analysis was performed in reference [10] to assess the susceptibility of the RCP nozzle welds to PWSCC. The analysis considered available industry experience data for the locations of Alloy 82/182 DM welds. More specifically, the data analyzed included Alloy 82/182 DM welds in large diameter pipes, including:

1. Reactor vessel inlet and outlet nozzles,
2. Steam generator inlet and outlet nozzles,
3. Reactor coolant pump suction and discharge nozzles, and
4. Pressurizer surge nozzle.

The collected service experience data was fit to a Weibull distribution which was then used to calculate the probability of cracking as a function of EFPY. This was done for three different temperatures with the intent of covering the range of temperatures on the cold nozzle DM weld locations (548°F to 556°F), as well as a representative hot nozzle DM weld location (615°F). Three different cases were evaluated based on the data to which the Weibull distribution was fit. Case 1 is based on all the available inspection results, for reactor vessel nozzles, steam generator nozzles, pump nozzles, and pressurizer surge nozzles. Case 2 includes all the nozzles except the pressurizer nozzles, and Case 3 includes only the reactor vessel and RCP nozzles. The results of these cases at the three temperatures are shown in Table 6.

The results in Table 6 show there is no discernable difference between the cases at the cold leg temperatures. Furthermore, the predicted probability of cracking for the pump nozzle DM welds, operating at cold leg temperatures, is extremely low, even at 60 EFPY. The results of the Weibull fitting for the three cases indicate that even though DM welds have had many flaws at hot temperature locations, none have been found at cold temperature butt weld locations, and this gives a very low probability of flaws existing in cold temperature locations. Results in Table 6 show the highest probability of an indication at cold leg temperatures was only 1.42%, at 60 EFPY (Case 1 at 556°F). In comparison, the probability (60 EFPY) at hot leg temperatures is 23.71% (Case 1 at 615°F). These results are shown graphically in Figure 5.

Conclusions

Analyses have been performed to calculate the probability of failure for Alloy 82/182 welds using both probabilistic fracture mechanics and statistical methods. Both approaches, statistical and PFM have shown that the likelihood of either cracking or through-wall leaks, in large diameter cold leg welds is very small. Furthermore, sensitivity studies performed using probabilistic fracture mechanics have shown that even for the

more limiting high temperature locations, an inspection frequency greater than that required by Section XI, such as that in MRP-139 or Code Case N-770 has only a small benefit in terms of risk.

Though past service experience may not be an absolute indicator of the likelihood of future cracking, the experience does give an indication of the relative likelihood of cracking in cold leg temperature locations versus hot leg temperature locations. While there is a significant amount of PWSCC service experience in hot leg locations, the number of indications is still small relative to the number of potential locations. Also, all indications have been detected before they were a safety concern. Therefore, if hot leg PWSCC is a leading indicator for cold leg PWSCC, and the higher frequency of inspections will be maintained for the hot leg locations, it is reasonable to conclude that a moderately less rigorous inspection schedule would be capable of detecting any cold leg indications before they became large enough to be a concern.

OVERALL CONCLUSIONS

While there has been a large amount of service experience with primary water stress corrosion cracking of Alloy 82/182 welds, this experience has been limited to those welds operating at hot leg temperatures or higher. There have been no incidents of cracking in welds operating at cold leg temperatures that can be attributed to PWSCC. Though the MRP-139 and Code Case N-770 requirements for more frequent inspection were taken as a proactive measure, the accumulation of more positive service experience indicates that perhaps this increased inspection frequency for cold legs in particular is not necessary to maintain an acceptable level of safety and quality. Furthermore, it has been realized that accessing these cold leg weld locations for inspection presents a hardship to utilities and may present an increased risk due to the complications associated with removal of the reactor vessel core barrel.

There have been numerous studies performed to evaluate the likelihood of through-wall cracking and flaw tolerance in cold leg Alloy 82/182 welds. The analyses performed as the original basis for MRP-139 showed that the large diameter cold leg welds had high flaw tolerance and a very low probability of failure. More recent analyses, which considered design specific residual stress distributions, have confirmed the original conclusions that the flaw tolerance is high. Furthermore, the more recent analyses have shown that even large circumferential flaws, with a high likelihood of being detected during inservice inspection, will not grow to the maximum depth allowed by ASME Section XI in 10 years. These analyses have been performed based on the assumption that a flaw has initiated, which as shown by more recent probabilistic analyses based on service data is unlikely at the present time.

It is therefore concluded that an interval of 10 years for re-examination of large diameter cold leg Alloy 82/182 locations will provide a more than adequate level of safety and quality. Furthermore, this interval will reduce hardship on utilities and minimize the risks associated with movement of the reactor vessel core barrel. These conclusions can serve as a basis for revisions to MRP-139 and Code Case N-770 to incorporate a 10 year re-examination interval for large diameter cold leg butt welds.

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TABLE 1: TYPICAL LARGE-DIAMETER ALLOY 82/182 COLD LEG BUTT WELD LOCATIONS

Application	Typical Temperature (°F)	Typical ID (inches)	Typical Number
Westinghouse Plants[1]			
• Steam Generator Outlet Nozzles[2]	550-560	--	--
• Reactor Vessel Inlet Nozzles[3]		27.5	3
Combustion Engineering Plants			
• Reactor Coolant Pump Inlet Nozzles[4]	549-560	30	4
• Reactor Coolant Pump Outlet Nozzles[4]		30	4
Babcock and Wilcox Plants			
• Reactor Coolant Pump Inlet Nozzles		28	4
• Reactor Coolant Pump Outlet Nozzles	557	28	4
• Reactor Vessel Core Flood Nozzles		14	2
• Core Flood Tank Nozzle		14	2

1. Data is for a Westinghouse 3-loop plant. Number of typical locations is dependent on number of loops.
2. One Westinghouse plant has Alloy 82/182 butt welds between the reactor coolant piping and steam generator nozzles.
3. There are no Alloy 82/182 RPV nozzle welds in Westinghouse 2-loop plants and some early Westinghouse 3-loop and 4-loop plants.
4. Some CE plants do not have Alloy 82/182 RCP suction and discharge nozzle welds.

TABLE 2: SUMMARY OF CRACKING IN REACTOR VESSEL OUTLET NOZZLES

Plant	Temperature (F)	EFPY[1]
VC Summer	621	15.6
Seabrook	621	16.3
OHI 3	617	14.0
Ringhals 3	613	12.8
Ringhals 4	613	12.3
Salem 1	608	19.7

1. Effective Full Power Years of Operation at the time the indication was found.

TABLE 3: SUMMARY OF CRACKING IN JAPANESE STEAM GENERATOR INLET NOZZLE-TO-SAFE-END WELDS

Plant	Date	Number of Indications, Max. L, Max D		
		A Loop	B Loop	C Loop
Mihama Unit 2 500 MWe	September 2007	13 indications L=17mm D=13mm	0 indications	N/A
Tsuruga Unit 2 1110 MWe	November 2007	1 indications L=N/A D=N/A	5 indications L=21mm D=12mm	23 indications L=14mm D=13mm
Takahama Unit 2 780 MWe	December 2007	3 indications L=7mm D=N/A	2 indications L=7mm D=6mm	4 indications L=11mm D=8mm
Genkai Unit 1 529 MWe	January 2008	3 indications L=5mm D=N/A	0 indications	N/A
Takahama Unit 3 870 MWe	February 2008	7 indications L=28mm D=9mm	16 indications L=38mm D=15mm	9 indications L=14mm D=9mm
Tomari Unit 2 579 MWe	April 2008	3 indications L=13mm D=7mm	10 indications L=10mm D=5mm	N/A
Takahama Unit 4 870 MWe	October 2008	7 indications L=14mm D=12mm	8 indications L=30mm D=13mm	21 indications L=33mm D=16mm
D = Depth, L = Length, N/A = Not Applicable				

TABLE 4: SUMMARY OF 40-YEAR LEAK PROBABILITIES

Nozzle	Design	Circumferential 40-Year Small Leak Probability With ISI
Decay Heat	B&W	5.00E-05
RV Outlet Nozzle	W	2.00E-04
Safety/Relief	CE/W	9.81E-06
SDC	CE	2.70E-08
SG Inlet	CE	3.38E-06
Spray	CE/W	1.25E-04
Surge HL	CE	3.38E-06
Surge PZR	CE/W	2.00E-04
	B&W	2.00E-04

TABLE 5: RESULTS OF INSERVICE INSPECTION SENSITIVITY STUDY

Description	40-Year Circumferential Small Leak Probability	Risk[2] (Core Damage Frequency)
No ISI[1]	6.06E-05	4.55E-09
10 Year ISI[1]	5.92E-05	4.44E-09
1 Year ISI[1]	3.67E-05	2.75E-09

1. Standard inspection quality for 50% detection of a flaw 25% through the wall
2. For conditional core damage probability (CCDP) = 3.0E-03

TABLE 6: SUMMARY OF PROBABILITY OF CRACKING RESULTS FOR HOT AND COLD LEG WELDS

At EFPY	Case 1	Case 2	Case 3
Temperature 548°F			
20	0.25%	0.00%	0.01%
40	0.57%	0.03%	0.05%
60	0.93%	0.12%	0.15%
Temperature 556°F			
20	0.38%	0.01%	0.02%
40	0.88%	0.10%	0.13%
60	1.42%	0.35%	0.35%
Temperature 615°F			
20	6.98%	20.92%	9.84%
40	15.32%	86.63%	44.34%
60	23.71%	99.92%	80.10%

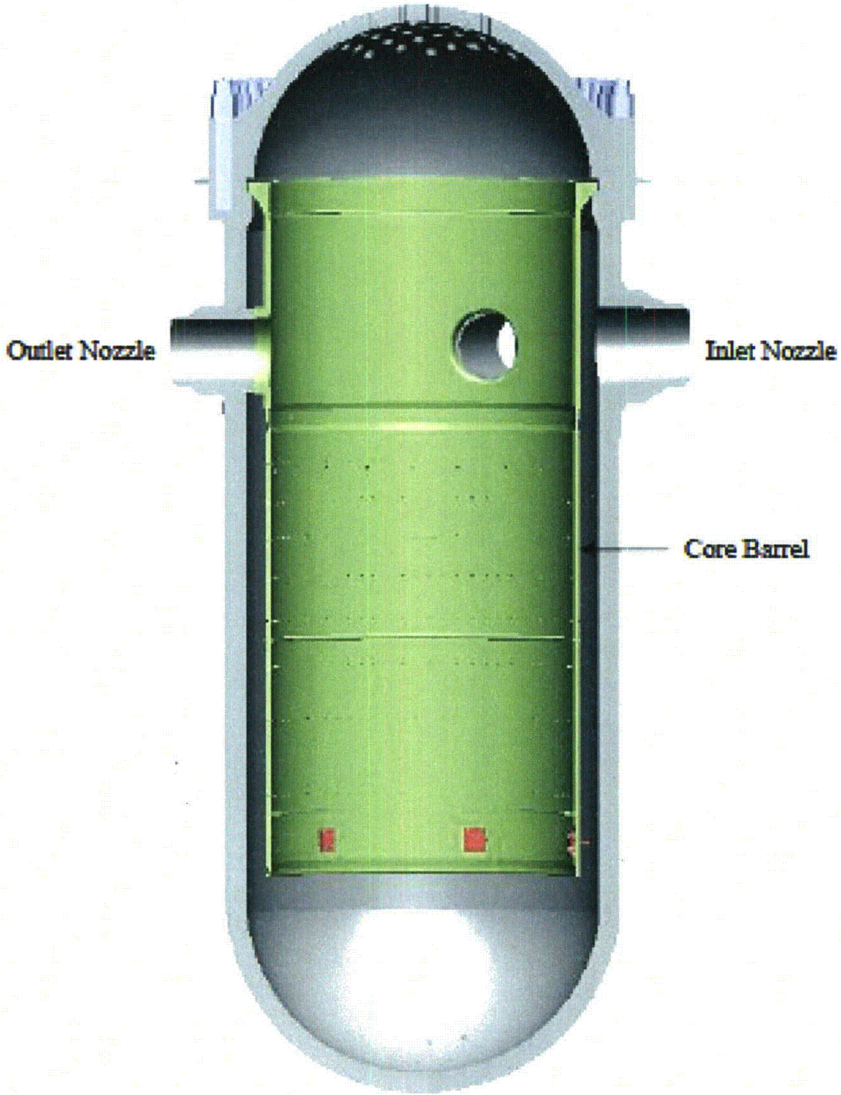

FIGURE 1: RELATIONSHIP OF THE CORE BARREL TO REACTOR VESSEL INLET AND OUTLET NOZZLES

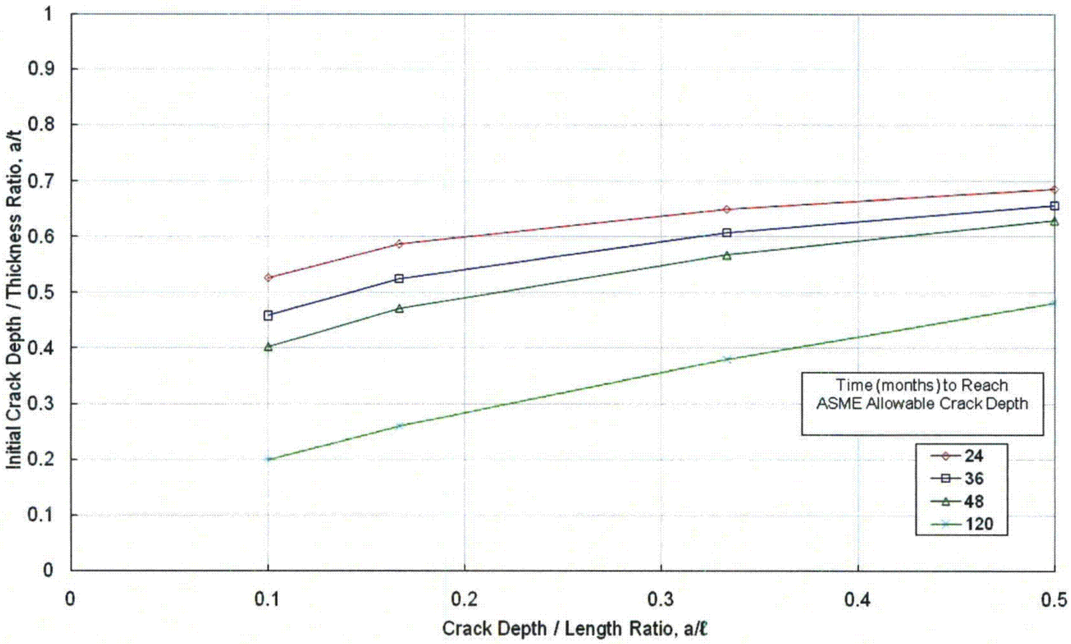

FIGURE 2: MAXIMUM ACCEPTABLE INITIAL CIRCUMFERENTIAL FLAWS, ACCOUNTING FOR PWSCC AND FCG, WITHOUT RESIDUAL STRESSES

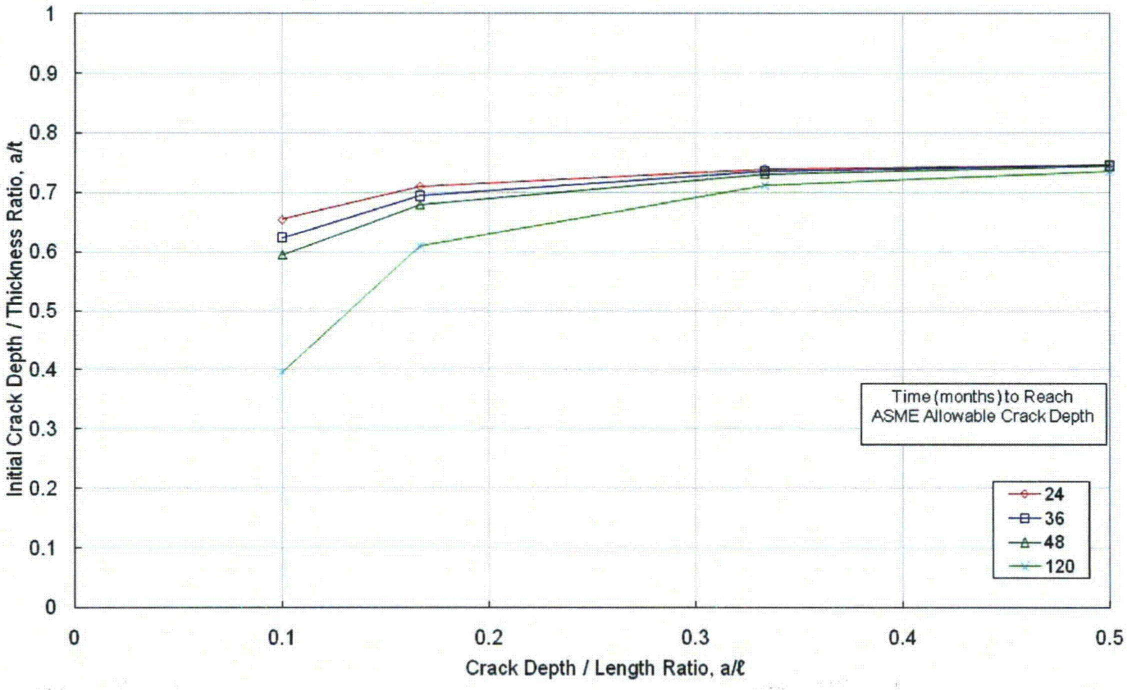

FIGURE 3: MAXIMUM ACCEPTABLE INITIAL CIRCUMFERENTIAL FLAWS, ACCOUNTING FOR PWSCC AND FCG, WITH FABRICATION RESIDUAL STRESSES AND AN INNER SURFACE WELD REPAIR

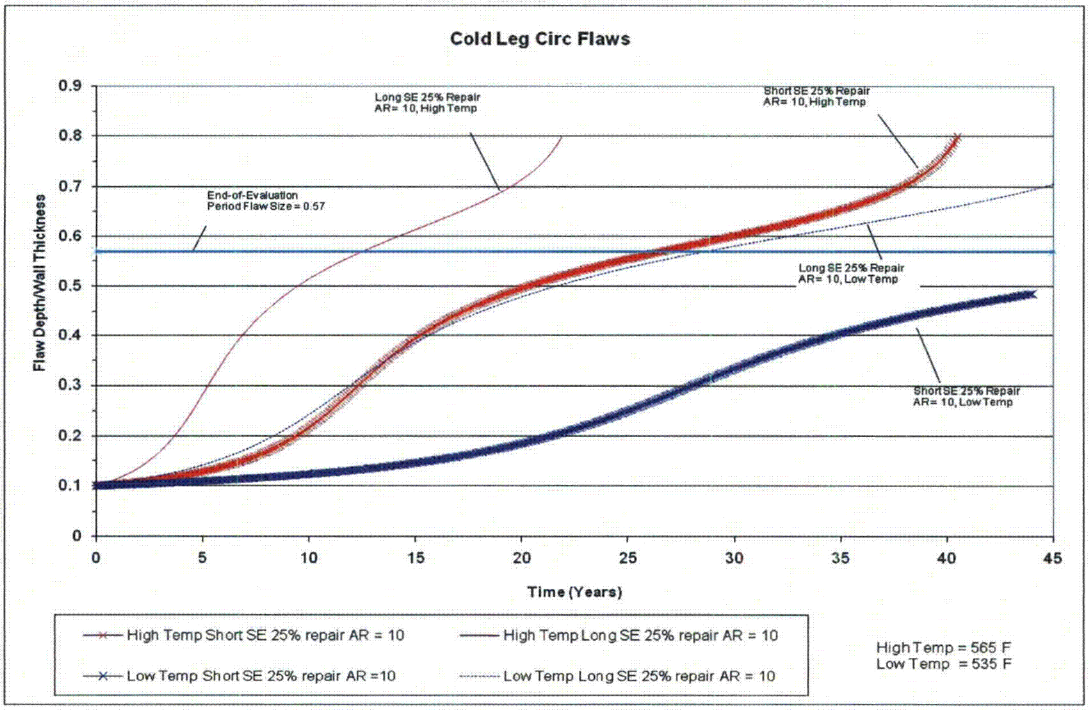

FIGURE 4: CIRCUMFERENTIAL FLAW PWSCC CRACK GROWTH AT THE RV INLET NOZZLE DM WELDS

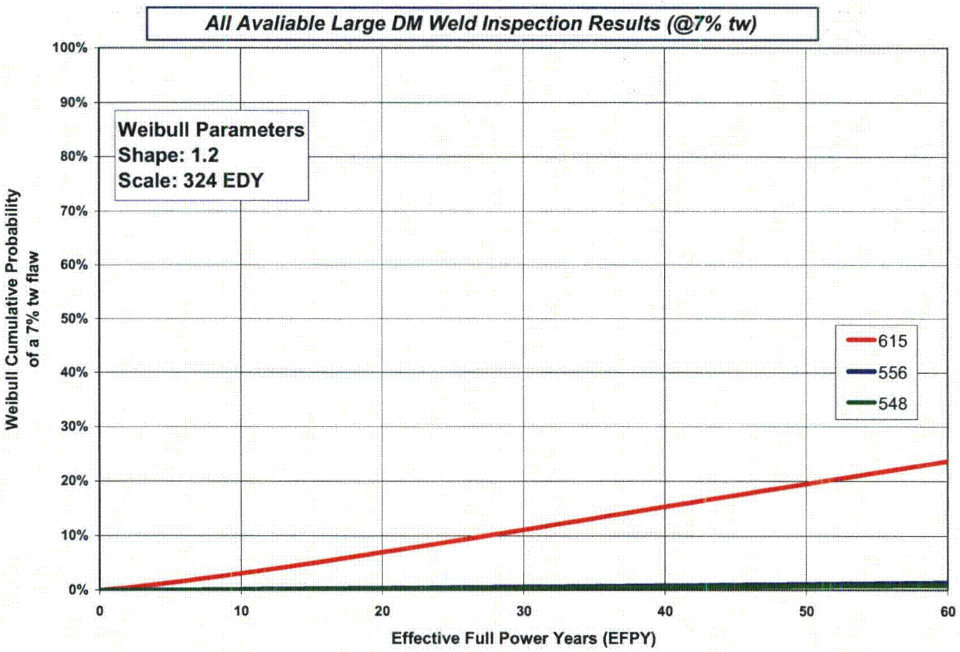

FIGURE 5: ALL AVAILABLE LARGE DM WELD INSPECTION RESULTS (7% THROUGH-WALL) – CASE 1