UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 28, 2011

Vice President, Operations
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: APPROVAL OF LEAK-BEFORE-BREAK OF THE PRESSURIZER SURGE LINE (TAC NO. ME3420)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 232 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3 (Waterford 3). This amendment consists of changes to the Waterford 3 Final Safety Analysis Report (FSAR) in response to Entergy Operations, Inc.'s (the licensee's) application dated February 22, 2010, as supplemented by letters dated August 12, November 23, and December 21, 2010, and January 24, 2011.

The licensee was planning to replace the two Waterford 3 steam generators (SGs) during the forthcoming spring 2011 refueling outage. Based on design changes in the replacement SGs, piping systems will require rerouting in the steam generator cavity area. The rerouting of SG blowdown line cannot be effectively performed without removing the existing dynamic protection associated with the pressurizer surge line. The amendment revises the Waterford 3 FSAR to allow the removal of pipe break dynamic protection associated with the pressurizer surge line using leak-before-break methodologies. The licensee will include the revised information in the FSAR in the next periodic update in accordance with paragraph 50.71(e) of Title 10 of the *Code of Federal Regulations*. Currently, the licensee has postponed the replacement of the SGs to fall 2012 refueling outage.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures:
1. Amendment No. 232 to NPF-38
2. Safety Evaluation

cc w/encls: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 232
License No. NPF-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (EOI), dated February 22, 2010, as supplemented by letters dated August 12, November 23, and December 21, 2010, and January 24, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Waterford Steam Electric Station, Unit 3 (Waterford 3), Final Safety Analysis Report (FSAR) and, as indicated in the attachment to this license amendment, Paragraph 2.C.(2) of Facility Operating License No. NPF-38 is hereby amended to read as follows:

    2. Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 232, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance. In addition, the licensee shall include the revised information in the Waterford 3 FSAR in the next periodic update to the FSAR in accordance with 10 CFR 50.71(e), as described in the licensee's application dated February 22, 2010, as supplemented by letters dated August 12, November 23, and December 21, 2010, and January 24, 2011, and the NRC staff's safety evaluation for this amendment

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License No. NPF-38

Date of Issuance: February 28, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 232

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Facility Operating License

REMOVE	INSERT
-4-	-4-

or indirectly any control over (i) the facility, (ii) power or energy produced by the facility, or (iii) the licensees of the facility. Further, any rights acquired under this authorization may be exercised only in compliance with and subject to the requirements and restrictions of this operating license, the Atomic Energy Act of 1954, as amended, and the NRC's regulations. For purposes of this condition, the limitations of 10 CFR 50.81, as now in effect and as they may be subsequently amended, are fully applicable to the equity investors and any successors in interest to the equity investors, as long as the license for the facility remains in effect.

(b) Entergy Louisiana, LLC (or its designee) to notify the NRC in writing prior to any change in (i) the terms or conditions of any lease agreements executed as part of the above authorized financial transactions, (ii) any facility operating agreement involving a licensee that is in effect now or will be in effect in the future, or (iii) the existing property insurance coverages for the facility, that would materially alter the representations and conditions, set forth in the staff's Safety Evaluation enclosed to the NRC letter dated September 18, 1989. In addition, Entergy Louisiana, LLC or its designee is required to notify the NRC of any action by equity investors or successors in interest to Entergy Louisiana, LLC that may have an effect on the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. <u>Maximum Power Level</u>

EOI is authorized to operate the facility at reactor core power levels not in excess of 3716 megawatts thermal (100% power) in accordance with the conditions specified herein.

2. <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 232, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 232 TO

FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By letter dated February 22, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100550605), as supplemented by letters dated August 12, November 23, and December 21, 2010, and January 24, 2011 (ADAMS Accession Nos. ML102300176, ML103300039, ML110120526, and ML110270155, respectively), Entergy Operations, Inc. (Entergy, the licensee), submitted a license amendment request (LAR) that proposes to implement leak-before-break (LBB) for the pressurizer surge line at Waterford Steam Electric Station, Unit 3 (Waterford 3). The supplemental letters dated August 12, November 23, and December 21, 2010, and January 24, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 20, 2010 (75 FR 20632).

On November 10, 2010, the NRC held a public meeting with the licensee to discuss the capability of the reactor coolant system (RCS) leakage detection as part of the NRC staff's review. The summary for this meeting, dated November 17, 2010, is available in ADAMS Accession No. ML103190416.

The licensee had plans to replace the steam generators at Waterford 3 in the upcoming refueling outage in spring 2011. The steam generator replacement effort requires rerouting the steam generator blowdown piping which will be blocked by one of the pipe whip restraints on the pressurizer surge line. If the LBB is approved for the surge line, the pipe whip restraint can be removed to allow the rerouting of the blowdown pipe. The licensee proposes to implement LBB for the pressurizer surge line in accordance with General Design Criterion (GDC) 4, "Environmental and dynamic effects design bases," of Appendix A to Part 50 of Title 10 of the

Code of Federal Regulations (10 CFR 50) and is based on proprietary Westinghouse Electric Company, LLC report, WCAP-17187-P[1], "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Waterford 3 Steam Electric Station, Unit 3, Using Leak-Before-Break Methodology," February 2010.

The licensee has deferred its planned steam generator replacement until the fall 2012 outage, but asked the NRC to complete its ongoing review of this license amendment request on the original schedule requested.

1.1 Background

In the early 1990s, the NRC staff approved LBB on the RCS main coolant loop piping at Waterford 3 based on the ABB Combustion Engineering Nuclear Power topical report prepared for Combustion Engineering Owners Group (CEOG), CEN-367-A, "Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems," February 1991 (ADAMS Accession No. ML071070104). However, at the time, the pressurizer surge line was not included in the topical report CEN-367-A.

The pressurizer-to-surge line nozzle and hot-leg-to-surge line nozzle contain the nickel-based Alloy 82/182 dissimilar metal weld which is susceptible to primary water stress-corrosion cracking (PWSCC). By letter dated April 21, 2008, the NRC approved licensee's proposed weld overlay design in Request for Alternative W3-R&R-006[2] (ADAMS Accession No. ML080950273). Subsequently, in the refueling outage of spring 2008, the licensee installed weld overlays on the Alloy 82/182 welds at these two nozzle locations to mitigate PWSCC.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR Part 50, Appendix A, GDC 4 state, in part, that

> Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.... However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of a fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

The LBB concept is based, in part, on calculations and experimental data demonstrating that certain pipe material has sufficient fracture toughness (ductility) to prevent a small through-wall flaw from propagating rapidly and uncontrollably to catastrophic pipe rupture and to ensure that

---

[1] A non-proprietary version, WCAP-17187-NP, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Waterford 3 Steam Electric Station, Unit 3, Using Leak-Before-Break Methodology," February 2010, is publicly available under ADAMS Accession No. ML100550607.

[2] McCann, J. F., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Request for Alternative W3-R&R-006 Proposed Alternative to ASME Code Requirements for Weld Overlay Repairs," dated April 26, 2007 (ADAMS Accession No. ML071230223).

the probability of a pipe rupture is extremely low. The small leaking flaw is demonstrated to grow slowly and the limited leakage would be detected by the RCS leakage detection system early on such that licensees can shut down the plant to repair the degraded pipe long before the potential failure.

NRC Regulatory Issue Summary 2010-07, "Regulatory Requirements for Application of Weld Overlays and Other Mitigation Techniques in Piping Systems Approved for Leak-Before-Break," dated June 8, 2010 (ADAMS Accession No. ML101380231), provides guidance on submitting updated LBB analyses for those LBB piping systems that have overlaid Alloy 82/182 welds.

The implementation of LBB requires a license amendment under 10 CFR 50.90 because one or more of the criteria of 10 CFR 50.59(c)(2) applies to LBB. Once a proposed LBB LAR is approved, the licensee is required to amend its final safety analysis report (FSAR) to document that LBB becomes a part of licensing basis for the pressurizer surge line. In Attachment 4 to the Submittal dated February 22, 2010, the licensee has committed to revise the Waterford 3 FSAR to include additional details on the RCS leakage detection system capability.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Revision 1 (ADAMS Accession No. ML063600396), provides guidance on screening criteria, safety margins, and analytical methods for the piping systems to be qualified for LBB, including guidance for review of LBB application, guidance for determining an acceptable leakage crack, and the RCS leakage detection sensitivity based on the fracture mechanics analysis. The guidance describes that determination of leakage from a crack in a piping system under pressure involves uncertainties and, therefore, margins are needed. Sources of uncertainties include plugging of the leakage crack with particulate material over time, correlation of leakage rates with crack geometry, correlations of measured parameters (e.g., sump level changes or containment radiation levels) with leakage rate, and frequency and accuracy of leakage instrumentation monitoring. Section III.4 of SRP Section 3.6.3 describes that the NRC staff evaluates the proposed leakage detection systems to determine whether they are "sufficiently reliable, redundant, and sensitive so that a margin on the detection of unidentified leakage exists for through-wall flaws to support the deterministic fracture mechanics evaluation.' The guidance specifies that the predicted leakage rate from the postulated leakage crack should be a factor of 10 times greater than the minimum leakage the detection system is capable of sensing unless the licensee provides justification accounting for the effects of uncertainties in the leakage measurement.

NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," dated November 1984 (ADAMS Accession No. ML093170485), provides the technical basis for the LBB analyses.

Licensees need to submit, for NRC review and approval, a fracture mechanics evaluation of specific piping configurations to meet the requirements of GDC 4. A candidate pipe should satisfy the screen criteria of SRP Section 3.6.3, demonstrating that the pipe experiences no active degradation mechanisms. The pipe also needs to satisfy the safety margins in SRP Section 3.6.3 via fracture mechanics analyses. Finally, the LBB application is predicated on the ability of the RCS leakage detection systems to detect a certain leak rate, with certain margin, that corresponds to the flaw size of the candidate pipe. NRC Regulatory Guide (RG) 1.45, Revision 0, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973 (ADAMS Accession No. ML003740113), provides guidance on the RCS leakage detection system capability.

The guidance of SRP Section 3.6.3 also states that specifications for plant-specific leakage detection systems inside the containment should be equivalent to those in RG 1.45. GDC 30, "Quality of reactor coolant pressure boundary," of Appendix A to 10 CFR Part 50 requires, in part, that

> Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

In RG 1.45, the NRC staff described acceptable methods of implementing this requirement with regard to the selection of leakage detection systems for the reactor coolant boundary. The regulatory position of RG 1.45, Revision 0, is that at least three different detection methods should be employed. Two of these methods should be: (1) sump level and flow monitoring and (2) airborne particulate radioactivity monitoring. The third method may involve either monitoring of condensate flow rate from air coolers or monitoring of gaseous radioactivity. The RG recommends that the sensitivity and response time of each leakage detection system employed for detection of unidentified leakage should be adequate to detect a leakage rate, or its equivalent, of 1 gallon per minute (gpm) in less than 1 hour.

In Section 5.2.5, "Detection of Leakage Through the Reactor Coolant Pressure Boundary," of the Waterford Updated Final Safety Analysis Report (UFSAR), the licensee indicates the leakage detection systems comply with Revision 0 of RG 1.45, as follows:

> The reactor coolant pressure boundary (RCPB) Leakage Detection System is designed to detect and identify abnormal leakage within the limits given in the Technical Specifications. The Leakage Detection System is capable of reliably:
>
> a) Detecting unidentified sources of abnormal leakage as low as 1.0 gpm.
>
> b) Identifying particular sources of abnormal leakage as low as 1.0 gpm.
>
> The RCPB Leakage Detection System is consistent with the recommendations of NRC Regulatory Guide 1.45 (May 1973).

The requirements related to the content of the Technical Specifications (TSs) are contained in Section 50.36, "Technical specifications," of 10 CFR, which requires that the TSs include limiting conditions for operation (LCOs). The criteria defined by 10 CFR 50.36(c)(2)(ii) relevant to determining whether capabilities related to RCPB leakage detection should be included in the TS LCOs, are as follows:

> A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's LBB analysis in WCAP-17187-P using the acceptance criteria in SRP Section 3.6.3.

3.1 Scope of LBB Application

The licensee proposed to implement LBB on the pressurizer surge line which connects the pressurizer to the hot leg. The surge line is a 12-inch diameter nominal size, schedule 160 stainless steel pipe. The licensee stated that the minimum wall thickness is 1.1685 inches. The nominal thickness is 1.312 inches based on the standard piping handbook. The outside diameter is 12.75 inches.

3.2 Screening Criteria for Degradation Mechanisms

SRP Sections 3.6.3.I and 3.6.3.III specify that the candidate piping should not experience active degradation mechanisms such as fatigue, water hammer, corrosion, creep, or cleavage failure.

3.2.1 Fatigue

SRP Section 3.6.3.III.10 recommends that LBB not be applied to piping with a history of fatigue. In response to an NRC staff Request for Additional Information (RAI) dated April 21, 2010 (ADAMS Accession No. ML10110635), which requested Entergy to justify LBB for the pressurizer surge line in light of thermal stratification in the pressurizer surge line, by letter dated August 12, 2010, the licensee responded that the issue of thermal stratification in the surge line was initially documented in NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," dated December 20, 1988 (ADAMS Accession No. ML031220290). In a letter to the NRC dated March 3, 1989[3], the licensee indicated that Waterford 3 would participate in addressing several action items agreed on by the CEOG and the NRC. The licensee supplied plant data to support a bounding analysis of CEOG surge lines, which met the reporting requirements in NRC

---

[3] Burski, R. F., Louisiana Power & Light Co., letter to U.S. Nuclear Regulatory Commission, "Waterford 3 SES, Docket No. 50-382, License No. NPF-38, Response to NRC Bulletin No. 88-11, 'Pressurizer Surge Line Thermal Stratification,'" dated March 3, 1989 (ADAMS Legacy Accession No. 8903080265).

Bulletin 88-11. By letter dated May 5, 1992[4], the licensee provided NRC with the results of a visual inspection of the surge line, which concluded that neither the surge line nor its affiliated hardware suffered any discernable distress or structural distress as a result of thermal stratification.

The licensee stated that the only remaining NRC Bulletin 88-11 action item pertaining to Waterford 3 was to perform plant-specific activities to update the surge line design documentation, which was completed and reported to the Commission by letter dated December 23, 1993[5], demonstrating that the integrity of the Waterford 3 pressurizer surge line satisfied American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section III criteria for the life of the plant, taking into account the effects of thermal stratification. The licensee concluded that as a result of the ASME Code analyses and the absence of any indication of fatigue cracking in the Waterford 3 surge line, the existence of thermal fatigue does not preclude the application of LBB.

In addition, the licensee stated that based on known data there has been no cracking in surge lines in either Westinghouse or Combustion Engineering (CE) plants. There was one case in 1989 where the Trojan plant replaced a surge line nozzle for what was believed to have been a flaw. Subsequent examination of the nozzle did not identify any flaws. Therefore, there is no known history of surge line cracking. The licensee reviewed inservice inspections for the attached components located on the Waterford 3 surge line and has not found any reportable indications.

The NRC staff concludes that the licensee has addressed the thermal stratification in the pressurizer surge line and that thermal stratification would not likely result in fatigue in the surge line and, therefore, fatigue is not likely to be an active degradation mechanism.

3.2.2 Corrosion

Stress corrosion cracking (SCC) occurs when high tensile stresses, susceptible material, and a corrosive environment exist simultaneously. The surge line is made of stainless steel. Since some residual stresses and some degree of material susceptibility exist in stainless steel piping, the licensee minimizes the potential for stress corrosion by selecting a material resistant to SCC and preventing the occurrence of a corrosive environment. The licensee's material specifications consider compatibility with the system's operating environment (both internal and external) and other material in the system, applicable ASME Code rules, fracture toughness, welding, fabrication, and processing.

The elements of a water environment known to increase the susceptibility of austenitic stainless steel to stress corrosion are: oxygen, fluorides, chlorides, hydroxides, hydrogen peroxide, and reduced forms of sulfur. The licensee has cleaned internal and external pipe surfaces before

---

[4] Burski, R. F., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Waterford 3 SES, Docket No. 50-382, License No. NPF-38, NRC Bulletin No. 88-11, Pressurizer Surge Line Thermal Stratification," dated May 5, 1992 (ADAMS Legacy Accession No. 9205060264).

[5] Burski, R. F., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Waterford 3 SES, Docket No. 50-382, License No. NPF-38, NRC Bulletin No. 88-11, Pressurizer Surge Line Thermal Stratification," dated December 23, 1993 (ADAMS Legacy Accession No. 9312300153).

commercial operation and controlled water chemistry during plant operation to prevent the occurrence of a corrosive environment. During flushes and preoperational testing, water chemistry is controlled in accordance with written specifications. The licensee follows the acceptance criteria on chlorides, fluorides, conductivity, and pH level. During plant operation, the licensee monitors and maintains the reactor coolant water chemistry within specific limits. For example, the licensee controls charging flow chemistry and maintains hydrogen in the reactor coolant at specified concentrations to limit oxygen concentration in the RCS. Halogen concentrations are also controlled by maintaining concentrations of chlorides and fluorides within the specified limits.

Wall thinning by erosion and the erosion-corrosion effect will not occur because of the low velocity and the stainless steel's resistance to these degradation mechanisms.

PWSCC has occurred in Alloy 82/182 dissimilar metal butt welds in primary-water reactors (PWRs). In May 2008, the licensee installed weld overlays on the Alloy 82/182 dissimilar metal welds in the surge line hot leg and pressurizer nozzle joints to preemptively mitigate the potential of PWSCC. The weld overlays and Alloy 82/182 welds were examined by ultrasonic testing (UT) as part of the weld overlay installation in May 2008. By letter dated May 29, 2008[6], the licensee reported that the UT examinations did not reveal any PWSCC cracking or indications. By letter dated August 12, 2010, the licensee reiterated that no PWSCC flaws have been identified in the Waterford 3 surge line nozzles.

The NRC staff concludes that corrosion, specifically PWSCC, will not be an active degradation mechanism in the surge line because the licensee has mitigated Alloy 82/182 welds with a weld overlay.

3.2.3 Water Hammer

By letter dated August 12, 2010, in response to the NRC staff's RAI R-3 dated April 21, 2010, the licensee stated that the RCS is designed and operated to preclude any voiding condition in normally filled lines to minimize water hammer. A review of the Waterford 3 condition reports did not reveal any history of water hammer. Section 2.2, *Water Hammer*, of WCAP-17187-P states that the temperature and pressure of the reactor coolant are maintained within a narrow range to ensure dynamic stability in the system to prevent water hammer. By letter dated August 12, 2010, the licensee clarified that the pressurizer pressure controller is set to maintain a setpoint value of 2250 pounds per square inch absolute (psia), steady state operation. The procedurally allowed range for controlling RCS pressure is 2175 to 2265 psia. The Technical Specification range is 2125 to 2275 psia. As a result of controlling T_{Cold} and pressurizer pressure within the procedural requirements, a surge line temperature range of 641.5 degrees Fahrenheit (°F) ±1.5 °F is maintained under steady state conditions.

The NRC staff concludes that water hammer is not a likely active degradation mechanism in the Waterford 3 surge line because the line has not had a history of water hammer and the plant operating procedures maintain dynamic stability in the system to avoid potential water hammer.

---

[6] Murillo, R.J., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Summary of Weld Overlay Ultrasonic Examinations for Pressurizer and Hot Leg Nozzle Welds at Waterford 3 Steam Electric Station," dated May 29, 2008 (ADAMS Accession No. ML081540252).

3.2.4 Creep and Cleavage Failure

The NRC staff does not believe creep is an active degradation mechanism for the pressurizer surge line because the operating temperature of 650 °F is below the temperature that would cause significant creep damage to the piping. Also, cleavage failure would not be an active degradation mechanism because the operating temperatures are not in the range that would cause such failure. In addition, stainless steel used for the pressurizer surge piping does not lead to cleavage failure.

The NRC staff concludes that the pressurizer surge line satisfies the screening criteria of SRP Section 3.6.3.

3.2.5 Fracture Mechanics Analysis

The NRC staff notes that when compared with the postulated axial flaw, the postulated circumferential flaw is controlling and limiting in the LBB evaluations because the critical axial flaw is much longer than the critical circumferential flaw. The postulated leakage axial flaw would have higher margin than the postulated leakage circumferential flaw when compared to their respective critical flaw sizes. The conservative approach is to analyze the flaw with a lower margin. Therefore, the LBB fracture mechanics analysis focuses on the circumferential flaw only. The staff notes also that the LBB analysis assumes that the postulated circumferential flaw is 100 percent through wall; therefore, the crack size and crack growth discussed herein refers to the circumferential extent of the through-wall circumferential flaw.

3.2.5.1 Pipe Locations

SRP Section 3.6.3.III.11.C specifies how pipe loads should be applied in deriving critical and leakage flaw sizes. The licensee considered Nodes 10, 20, 70, 75, and 80 along the surge line because these locations have either the worst conditions (i.e., high loadings coupled with limiting material properties) or have weld overlays. Node 10 is located at the hot-leg-to-surge line nozzle. Node 20 is at the elbow that is closest to Node 10. Nodes 70 and 75 are located at the elbow downstream from the pressurizer-to-surge line nozzle, which is Node 80. Nodes 10 and 80 are Alloy 82/182 dissimilar metal welds that have been weld overlaid. Node 10 has worse conditions than Node 80 and its results are presented in the analysis. The licensee focused on Nodes 10, 20, 70, and 75 in the WCAP-17187-P report.

In RAI R-7 of the RAI dated April 21, 2010, the NRC staff asked the licensee to clarify the procedure for choosing the critical location in the LBB analyses. In its letter dated August 12, 2010, the licensee stated that the most critical LBB analyses location was chosen based on the highest faulted stress from all the weld locations. As shown in Table 3-5, "Summary of Loads and Stresses at the Critical Locations," of WCAP-17187-P, Node 70 has the highest faulted stress (Load Case F) and therefore is the most limiting location for the LBB analyses. In addition, the licensee's stability analyses results show that Node 70 has the most limiting locations. For Load Case A, Node 20 had a higher moment than Node 70. For leak rate calculation, Load Case A is limiting with smaller loads because for a given leak rate, smaller loads result in a larger flaw size and larger loads result in a smaller flaw size. A larger flaw size will result in a lower margin when compared to the critical crack size. Therefore, for a given leak rate, higher normal moments and stresses at Node 20 result in a lower leakage flaw size and in a higher margin when compared to the critical crack size. Node 70 location has a lower flaw size margin than Node 20. Thus, Node 20 is not the limiting location as compared to Node 70 for the leak rate calculation. The individual loading cases at each location are considered separately for the leakage and stability analyses.

3.2.5.2 Applied Loading and Load Combinations

3.2.5.2.1 Effect of Power Uprate and Steam Generator Replacement

Section 3.4, *Loading Conditions*, of WCAP-17187-P, states that the applied loads for the LBB analysis are obtained from the updated pipe stress reanalysis. In RAI N-3 of the RAI dated April 21, 2010, the NRC staff asked the licensee to discuss the impact of the steam generator replacement and power uprate on the original pipe stress analysis, pipe supports, routing, and their locations on the surge line. By letter dated August 12, 2010, the licensee responded that the stress and fatigue analyses for the surge line were updated (1) for the effects of extended power uprate (EPU) only, and (2) for the effects of EPU plus the replacement steam generators (RSGs).

The licensee reanalyzed pipe stresses under EPU conditions, including normal operation (NO), deadweight, and thermal loads. The design basis loads and thermal anchor motions (TAMs) were changed where necessary. The licensee stated that EPU did not have a significant effect on the seismic response of the system; therefore, the seismic design basis RCS loads and seismic anchor motions (SAMs) were retained. The licensee also developed new branch line pipe break (BLPB) input forcing functions for the surge line and analyzed their effects on the RCS (i.e., to determine new RCS loads and BLPB anchor movements (BAMs)).

The licensee used these RCS responses (hot-leg surge nozzle seismic and BLPB excitations, and surge line TAMs, SAMs, and BAMs) as inputs to the surge line pipe stress reanalysis, along with the appropriate thermal stratification transient inputs to update the design basis for EPU conditions. The licensee stated that surge line stresses were found to be acceptable for EPU conditions.

The licensee also reanalyzed the surge line for the updated seismic, BLPB, and NO inputs and the effects of thermal stratification. The surge line was found to be acceptable in accordance with the design criteria of the ASME Code, Section III.

The licensee stated that no changes in the surge line routing, pipe supports (i.e., dead weight hangers), or pipe whip restraints were required for either EPU, or EPU plus RSG conditions. The current analysis of the surge line for the LBB application is based on the latest set of design basis loads, which are based on EPU plus RSG conditions. There were some changes in the load and stress results with the inclusion of RSGs. However, all requirements of the ASME Code, Section III continue to be met and the resulting fatigue usage, 0.41, is below the ASME Code, Section III, allowable of 1.0.

The NRC staff concludes that the licensee has updated the stress analysis of the surge line to include the effects of EPU plus RSGs. The licensee used the updated loadings in the LBB analysis for the pressurizer surge line. Therefore, the staff concludes that the loading input is acceptable.

3.2.5.2.2 Load Combinations

The licensee designated Load Cases A, B, and C for normal operation conditions and D, E, F, and G for faulted conditions as shown in Section 3.4, *Loading Conditions*, of WCAP-17187-P. In RAI N-4 of the RAI dated April 21, 2010, the NRC staff asked the licensee to explain why load combinations A/E and B/D were not used in the LBB analysis. In its letter dated August 12, 2010, the licensee stated that Load Case E (faulted condition) has normal operating steady state stratification loads that should be combined with Load Case B which also has normal operating steady state stratification loads. Similarly, Load Case D (faulted condition) has normal operating thermal loads that should be combined with the normal operating Load Case A which has normal operating thermal loads. Therefore, the logical combinations are Load Cases A/D and B/E. The licensee stated that these combinations are appropriate for the LBB analyses and are considered acceptable.

The NRC staff concludes that the licensee has clarified the load combination adequately and the appropriate load combination of A/D and B/E are appropriate and were used in the LBB evaluation.

3.2.5.2.3 Thermal Stratification

In RAI N-5(a) of the RAI dated April 21, 2010, the NRC staff asked the licensee to clarify the temperature profiles of the surge line that were used in the thermal stress calculations. In its letter dated August 12, 2010, the licensee stated that a CEOG program gathered plant data and developed the methodology to determine the resulting temperatures in the surge line. CE NPSD-546-P, Volume 1[7], "Pressurizer Surge Line Flow Stratification Evaluation," CEOG Task 587, July 1989, evaluated the instrumentation data and provided the basis and details of the resulting thermal hydraulic evaluation.

The derivation of the top-to-bottom fluid temperatures in the horizontal portions of the surge line was based on measured plant data from various CE-fleet plants. The most extensive measurements were taken at Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Fort Calhoun Station, and San Onofre Nuclear Generating Station, Units 2 and 3. Surface-mounted thermocouples (in many locations, mounted circumferentially around the pipe at 600 intervals) were placed along the surge lines. Data was taken for both heatup and cooldown conditions. After data reduction was performed, temperatures versus time plots were developed.

According to the licensee, because only the outside pipe wall temperature measurable data was collected, the inside fluid conditions were not explicitly known, and therefore needed to be calculated. The licensee developed various fluid models to simulate fluid conditions inside of the surge line that would correspond to the measured pipe wall temperature profiles. These evaluations produced the top-to-bottom fluid temperature differences characterized in the design basis thermal stratification transients. The licensee used the thermal stratification

---

7 Combustion Engineering, Inc., CEN-387-NP, "Pressurizer Surge Line Flow Stratification Evaluation," July 1989 is publicly available in ADAMS Legacy Accession No. 8908040088.

transients in the stress and fatigue structural analyses of the surge line piping, nozzles, and supports.

The licensee determined a set of transients and associated cycles based on steady state thermal stratification, normal stratification, and high and low stratification (i.e., maximum stratification occurring at the high and low ends of the temperature range in the surge line) to represent the conditions observed during CE plant heatups and cooldowns. The licensee applied these top-to-bottom temperature differences along the entire length of the surge line horizontal run, along with a linear thermal expansion temperature input equal to the average of the top-to-bottom temperature difference. Since stratification in the vertical segment of the surge piping is negligible, loads in these portions of the surge line were only subjected to effects from linear thermal expansion.

The NRC staff asked the licensee to clarify whether the maximum stratification temperature differential was included in calculating the critical and leakage crack sizes. In its letter dated August 12, 2010, in response to RAI N-5(c) of the staff RAI dated April 21, 2010, the licensee stated that Load Case F has the maximum stratification temperature differential for the faulted condition and this case was used for the critical flaw size determination. Load Case C has the maximum stratification temperature differential for the normal operating condition and the stress for Load Case C is significantly higher than the normal Load Cases A and B. Load Case C will provide significantly lower leakage flaw size. Lower leakage flaw size will result in higher LBB flaw size margin. Therefore, Load Case C does not govern while Load Cases A or B with lower normal stresses (which will provide higher leakage flaw sizes and eventually lower flaw size margins) governs.

The NRC staff concludes that the licensee has considered the conservative stratification temperature profile in the analysis and, therefore, its input on temperature differential is acceptable.

3.2.5.2.4 Effect of Weld Overlay

In its RAI dated April 21, 2010, the NRC staff questioned whether the pipe loads used in the LBB analyses include the effects of the weld overlay. This effect may be minimal for normal operating loads, but may have a larger effect on the safe shutdown earthquake (SSE) load calculation. In addition, the staff believes that the weld overlay (WOL) significantly stiffens the Alloy 82/182 dissimilar metal weld joint. This stiffness may affect the stresses calculated at the weld location. In its letter dated August 12, 2010, the licensee stated that the effects of the weld overlays on the surge line (e.g., weld shrinkage and overlay mass) were included as part of the design requirements.

The licensee stated further that additional loads that would be impacted by the weld overlay on the surge line were not considered to significantly affect the existing pipe stress analyses including loads from dynamic excitation due to the SSE. The configuration of the vertically orientated axial pipe shrinkage on the "floating" [non-restrictive movement] design of the surge line from the pressurizer to the hot-leg surge nozzle was bounded by the pipe restraint gaps. The licensee concluded that these affects are acceptable given the length of the surge line between the pressurizer and hot-leg surge nozzle and the elbows in the surge line pipe.

The licensee modeled the pipe stresses from safe-end to pipe joint for both anchored ends of the surge line. The nozzle to safe-end locations were conservatively selected as anchor points to maximize piping loads. The additional stiffness of the nozzle to safe-end from the weld overlay presents no change to pipe stresses due to the selection of the nozzle safe-end to pipe as the anchor point.

In RAI R-5 of its RAI dated April 21, 2010, the NRC staff asked the licensee to describe how the weld overlay was modeled in the piping stress analysis. In its letter dated August 12, 2010, the licensee responded that the weld overlay on the nozzles were not included in the piping analyses. The weld overlay is evaluated as part of the nozzle stress analysis using end loads from the piping analyses. The licensee has identified that the two effects of nozzle weld overlays on the pipes are added nozzle weight due to the weld material deposits and axial shrinkage of the nozzles. However, the licensee has determined that these effects are negligible for the surge line. The added weight at what are essentially anchor points for the subsystem has minimal effect on the response of piping of this size. Furthermore, surge lines are designed for maximum flexibility (e.g., they are relatively long and have many bends) in order to accommodate the thermal contraction and expansion of the line that occurs during the course of normal operation. This added flexibility further mitigates any weld overlay induced effects on the surge line.

The NRC staff noted that axial shrinkage of the weld overlay can cause a tensile axial stress in the rest of the piping system when the weld overlay is in-situ with the piping system connected to the pressurizer and hot leg. By letter dated August 12, 2010, in response to RAI R-6 of the NRC staff's RAI dated April 21, 2010, the licensee responded that the effects of the surge line pipe nozzle axial shrinkage were evaluated and determined to not adversely impact the existing surge line pipe stress analyses. These as-built measurements confirm the minor contribution from the axial shrinkage that occurs over the 38 feet distance between the hot-leg surge line nozzle and the pressurizer surge line nozzle.

The NRC staff concludes that the licensee has appropriately addressed the impact of the weld overlay on the pipe stresses.

3.2.5.2.5 Effect of Displacement Controlled Loads

The NRC staff noted that the Materials Reliability Program, MRP-216[8], "Advanced FEA [Finite Element Analysis] Evaluation of Growth of Postulated Circumferential PWSCC Flaws in Pressurizer Nozzle Dissimilar Metal Welds," report, published by the Electric Power Research Institute (EPRI), suggested that some of the displacement controlled loads can be eliminated from the calculation of through-wall crack stability. MRP-216 suggests that for relatively long flaws only about 50 percent of the applied moment would be relieved. Because there was still question about how much of the displacement-controlled loads were used in critical crack calculations, MRP-216 assumed that the thermal expansion loads contributed to the throughwall crack stability.

---

[8] A pre-publication version, dated July 31, 2007, is publicly available in ADAMS Accession No. ML072200195.

In RAI R-8 of the RAI dated April 21, 2010, the NRC staff asked the licensee why displacement-controlled loads such as thermal expansion were not included in calculating the critical flaw size in Alloy 82/182 welds. In its letter dated August 12, 2010, the licensee stated that the critical flaw size for the Alloy 82/182 welds is calculated using plastic instability as a failure mode, also known as ductile limit load. The licensee stated that this approach has been demonstrated to be accurate in predicting failures of high toughness materials in hundreds of pipe tests conducted world-wide. The theory of plastic instability states that the component will fail in the presence of a crack when the remaining ligament becomes entirely plastic. The staff notes that NUREG-1061, Volume 3, provides additional information regarding the theory and experiments of plastic instability and the limit load analysis. The licensee contends that when this happens, all displacement-controlled stresses, such as thermal and residual stresses, relax. Therefore, these stresses do not affect the failure. This occurs in all of the structural materials used in the primary coolant system of a PWR, and is invariant of temperature in stainless steels and Nickel-based alloys such as Alloy 182 welds. This is the methodology used in Appendix C of the ASME Code, Section XI.

The licensee explained that MRP-216, Section 5.3, states, in part, that

> The results of this study support the conclusion that the surge nozzle piping thermal loads are completely relieved prior to nozzle rupture, since the supportable crack plane rotation is greater than the imposed rotation due to thermal expansion.

In MRP-216, the normal piping thermal loads were used to add conservatism to expeditiously resolve an issue for the industry. As stated in the quote above, there was no technical need to include the thermal loads.

The licensee stated further that this conclusion is consistent with the wording in the standard review plan, SRP Section 3.6.3, Revision 1. For base metal and welds made by gas tungsten arc welding, which have a high toughness similar to that of Alloy 82/182, no thermal stress is used in the calculation of the critical flaw size, as shown in equation 7 on SRP Section 3.6.3, page 3.6.3-11.

For through-wall cracks, the NRC staff believes that most of the displacement-controlled loads are relaxed and that the driving force does decrease. However, MRP-216 suggests that the displacement-controlled moments are not fully relaxed until the crack length is significantly long (i.e., 70-80 percent of pipe's circumference). The MRP-216 results suggest that there is uncertainty to whether the displacement-controlled loadings are fully relaxed and whether this assumption is conservative. In its RAI dated April 21, 2010, the NRC staff requested the licensee to provide additional technical basis to support the argument that the displacement-controlled loads are not needed to calculate the critical crack size.

In its letter dated April 21, 2010, the licensee stated that various industry studies have been conducted regarding the effects of displacement-controlled loads on crack opening displacement. Even though some uncertainty exists as to when the majority of the displacement controlled loads relax, it is generally understood that these loads are relaxed quickly, well before nozzle failure. Additional review of MRP-216 reveals that a crack length of 58 percent of the circumference can also become fully relaxed under the displacement controlled moments. For other smaller lines, the crack length can be as low as 36 percent when the displacement loads are relaxed. This is presented in Appendix B of MRP-216.

The licensee explained that based on SRP Section 3.6.3, Z-factors are not required for high toughness materials when determining critical flaw sizing. As discussed in MRP-140, Alloy 82/182 is classified as a high toughness material comparable to base metals and welds fabricated by gas tungsten arc welding (GTAW). Therefore, the secondary stresses need not be considered in the limit load analysis. According to the licensee, this approach is further supported by ASME Code Sections III and XI in which thermal expansion stresses are not considered for high toughness materials (i.e., GTAW process). However, for the Waterford 3 surge line analysis, a Z-factor was conservatively applied which addresses uncertainties from secondary thermal loads. Therefore, the licensee concluded that the approach applied for the critical flaw sizing determination provides adequate conservatism.

The NRC staff recognizes that the issue regarding the displacement-controlled load needs to be studied further on a generic basis. However, in its confirmatory analysis[9], the staff included the displacement-controlled loads due to thermal expansion in the critical crack size calculation. The staff has verified that the critical crack size is twice as long as the leakage crack size and that the margin of 2 in SRP Section 3.6.3 has been satisfied.

3.2.6 Material Properties

SRP Sections 3.6.3.III.11.A and 3.6.3.III.11.B specify that material specifications and material properties be identified. As shown on Page 4-1 of WCAP-17187-P, the licensee used material properties based on ambient, not operating, temperature (with temperature correction). In RAI R-9 of its RAI dated April 21, 2010, the NRC staff questioned why the material properties at the ambient temperatures were used in lieu of the properties at operating temperatures. The staff suggested that the Battelle Memorial Institute data (both at room temperature and operating temperature) that corresponds to the certified material test report (CMTR) for this material be used. By letter dated August 12, 2010, the licensee stated that yield and ultimate strength data in the CMTR information is available only at room temperature. Hence, the Battelle stress-strain curve at room temperature was considered for the Ramberg-Osgood (R-O) law fit. Temperature variation for yield strength, ultimate strength and modulus of elasticity shown in Table 4-1, "Tensile Material Properties for Cast Stainless Steel Base and Stainless Steel Weld Metal at the Critical Locations," and Table 4-2, "Tensile Material Properties for Alloy 82/182 and Alloy 52M at the Critical SWOL [Structural Weld Overlay] Location (Node 10)," of WCAP-17187-P were considered.

The licensee clarified that the NRC-Battelle report, "Pipe Fracture Encyclopedia" [PIFRAC], (Volume 1, 1997)[10], has stress-strain properties for cast austenitic stainless steel (CASS) base materials at temperatures of 68 °F (20 degrees Celsius [°C]), 300 °F (149 C), and 550.4 °F (288 °C). The licensee presented a graph of the true stress-strain data for PIFRAC samples

---

9 Csontos, A., interoffice memorandum to Timothy Lupold, U.S. Nuclear Regulatory Commission, "Review of Waterford Unit 3 Surge Line License Amendment Request for Leak-Before-Break (LBB) Evaluation and Office of Nuclear Regulatory Research (RES) Confirmatory Analyses to Evaluate Safety Margins," dated October 7, 2010 (not publicly available).

10 Pipe Fracture Encyclopedia, Battelle Columbus Laboratories, Columbus Ohio, 1996.

A37-1 through A37-6. The data includes two samples at each of the three temperatures. The strain range of interest for the LBB applications is from 0 percent to 5 percent. The licensee presented the Ramberg-Osgood (R-O) law equation that would estimate this material data.

The licensee stated that this form of the material curve idealization has limited parameters to represent the actual material stress-strain curves. In the low strain range of interest, the data show similar trends with temperature. The licensee stated further that the material properties at room temperature curve was applied to fit the R-O law and was used for all temperatures of interest keeping the parameters "α" and "n" in the R-O law constant. However, it represented the temperature variation through the reference stress (yield strength of the material) and the modulus of elasticity. This variation is taken from the ASME Code. The yield strength of the actual material in the surge line at room temperature is obtained from the actual CMTRs. Yield strength at higher temperatures is obtained by scaling CMTR values proportionately with the ratio of the ASME Code yield strength at ambient temperature to the evaluation temperature.

According to the licensee, the yield stress inferred from the PIFRAC data was compared to the yield stress for the actual surge line materials. It was observed that the room temperature yield strength of the CASS material in PIFRAC is significantly lower than the CMTR values of the actual material. Also, the yield stress from the limited amount of PIFRAC data variation with temperature is inconsistent and has no trend, as the data at two higher temperatures, 300 °F (149 °C) and 550.4 °F (288 C), are almost identical and do not show any temperature variations. The trend of yield stress of the PIFRAC samples with temperature was not smoothly decreasing as expected. For these reasons, the licensee stated that the PIFRAC data cannot be used directly. However, the shape of stress versus strain as represented by the R-O law fit curve can be scaled to the CMTR yield stress at room temperature. Then, the curves at higher temperatures are obtained by scaling the room temperature curve by the ratio of the ASME Code yield stress at the evaluation temperature over the yield strength at room temperatures.

The licensee explained that this alternate approach is consistent with higher temperatures given the similarity of the PIFRAC stress-strain properties in the low strain range. This approach has also been verified by using the R-O law fit to the actual PIFRAC stress-strain curve data at higher temperatures of 300 °F (149 °C), and 550.4 °F (288 °C) and was observed to represent this data appropriately thereby supporting the methodology used.

A similar approach to the cast austenitic stainless steel material was also used for the stainless steel (SS) weld materials. The ambient temperature stress-strain properties available for samples A8-1-1 and A8-1-2 at temperature of 71.6 °F (22 °C) were fitted to R-O law, and then extrapolated to higher temperatures using the ASME Code scale factors. The resulting R-O law was also checked for the PIFRAC data available for sample A8-1-4 at 550.4 °F (288 °C) and found to be appropriately represented.

Based on the NRC staff's confirmatory analysis, the staff concludes that the surge line satisfies the safety margins on crack size and leakage and that the material properties used in the licensee's LBB analysis are appropriate.

3.2.7 Leakage Crack Size Calculations

SRP Section 3.6.3.III.11.C provides guidance on the calculation of a postulated leakage crack size. SRP Section 3.6.3.III.11.C.ii specifies that the pipe location with the worst material property should be selected. SRP Section 3.6.3.III.11.C.iii specifies that the postulated leakage crack size be calculated to be sufficient large so that the estimated leak rate during normal operation is 10 times greater than the minimum RCS leakage detection system capability. SRP Section 3.6.3 further stated that the normal operating loads (i.e., deadweight, thermal expansion, and pressure) are to be combined based on the algebraic sum of individual values and applied to the leakage flaw size.

The leakage crack size calculation for Nodes 20, 70, and 75 should be different from Node 10 because Node 10 represents the overlaid Alloy 82/182 weld which has a different configuration (i.e., thicker pipe wall, different material properties, and additional degradation mechanism) from the rest of pipe locations (e.g., Nodes 20, 70, and 75). The discussion of these two methods is presented separately below.

3.2.7.1 Leakage Crack Size Calculations for Pipe Nodes 20, 70, and 75

The licensee calculated leakage crack sizes at Nodes 20, 70, and 75 are based on a given leak rate of 2.5 gpm. This provides a margin of 10 to the RCS leakage detection system capability of 0.25 gpm at Waterford 3. To calculate the leakage crack size at Nodes 20, 70, and 75, the licensee used the fluid dynamics theory considering two-phase flow phenomena, friction pressure drop, and crack surface roughness. The licensee assumed fatigue as the degradation mechanism for Nodes 20, 70, and 75. The licensee obtained leakage crack sizes for these three nodes ranging from 3.1 to 3.24 inches. The NRC staff identified the following two issues in the licensee's leakage crack calculation.

The crack surface roughness is a major factor in the leakage crack calculations. Section 5.3, *Calculation Method,* of WCAP-17187-P, states that the crack relative roughness was obtained from fatigue crack data on stainless steel samples. In RAI N-6 of its RAI dated April 21, 2010, the NRC staff asked the licensee to discuss the source of the stainless steel samples and how the roughness value was obtained. In its letter dated August 12, 2010, the licensee stated that Westinghouse developed crack relative roughness values which have been previously accepted by the NRC. Westinghouse proprietary report, WCAP-9558, Revision 2, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-wall Crack," May 1981, documented the crack relative roughness that was used for the leak rate calculations. Since 1981, Westinghouse has applied this crack relative roughness value for similar LBB applications.

Section 5.4, *Leak Rate Calculations,* of WCAP-17187-P states that the air-fatigue crack surge roughness used for its leak rate calculation is about 50 percent higher than the typical surface roughness used by the industry. In its letter dated August 12, 2010, the licensee clarified that the air-fatigue crack surface roughness used for the leak rate calculation is higher than the typical surface roughness of 200 micro-inches used by the industry. The NRC staff concludes that the licensee used a surface roughness factor that the staff had approved in LBB amendment requests from other licensees and, in the absence of new experimental data for the crack surface roughness in stainless steel, the staff does not object the crack surface roughness used in the licensee's calculation.

In its letter dated August 12, 2010, in response to RAI N-7 of the NRC staff's RAI dated April 21, 2010, the licensee explained that the crack opening area estimation was based on NUREG/CR-3464[11] using linear-elastic fracture mechanics (LEFM), including effects of shell corrections. The imposed loads on the circumferential flaw are the axial tensile force and bending moment due to normal operating conditions (e.g., deadweight, pressure, thermal expansion). The crack opening area for the tensile loading is obtained by the energy method (Castigliano's theorem). The licensee obtained the crack opening area for the bending load by further derivation as shown on pages 76 through 80 of NUREG/CR-3464. The two crack opening area components (tension and bending) are combined to determine the total crack opening area. The licensee stated that since 1981, Westinghouse has used this crack area calculation method for other LBB applications that have been reviewed by the NRC. The staff concludes that the licensee has used an acceptable methodology that provides a reasonable estimation of the crack opening area.

3.2.7.2 Leakage Crack Size for Overlaid Alloy 82/182 Dissimilar Metal Welds

To calculate the leakage crack size for the overlaid Alloy 82/182 dissimilar metal welds (Nodes 10 and 80), the licensee used a different approach from the method discussed above because the pipe configuration and degradation mechanism at the overlaid welds are different from the rest of pipe locations. Instead, the licensee derived the leakage crack size for the overlaid dissimilar metal weld (Node 10) by multiplying an effective penalty factor of 1.44 to the leakage crack size derived for the original Alloy 82/182 without the weld overlay. The penalty factor of 1.44 is derived based on a weighted average of a penalty factor of 1.69 for the unmitigated Alloy 82/182 weld and a penalty factor of 1.0 for the Alloy 52M weld. The licensee obtained the penalty factor of 1.69 for the Alloy 82/182 welds based on the ratio of the leakage crack size derived for PWSCC and the leakage crack size derived for fatigue as shown in the 2003 presentation on "Impact of PWSCC and Current Leak Detection on Leak-Before-Break" [12] (2003 presentation).

In its letter dated August 12, 2010, the licensee explained that the 2003 presentation evaluates flow paths and the morphology of PWSCC flaws against previously applied LBB analyses to provide bounding leakage flaw penalty factors. The particular crack morphology parameters of interest are crack surface roughness, number of turns, and actual flow path. Historically, leakage cracks were generally assumed to be characterized as fatigue cracks with the depth equal to the wall thickness. The 2003 presentation shows that circumferential cracks in Alloy 82/182 welds are parallel to the long direction of dendritic grains which is considered a primary factor in PWSCC leakage flaw determination. The 2003 presentation provides the parameters

[11] Paris, P. C., and Tada, H., "The Application of Fracture Proof Design Methods using Tearing Instability Theory to Nuclear Piping Postulating Circumferential Through Wall Cracks," NUREG/CR-3464, U.S. Nuclear Regulatory Commission, Washington D.C., September 1983.

[12] Rudland, D. L., Wolterman, R., Wilkowski G., and Tregoning, R., "Impact of PWSCC and Current Leak Detection on Leak-Before-Break," Proceedings of Conference on Vessel Head Penetration Inspection, Cracking, and Repairs, Sponsored by U.S. Nuclear Regulatory Commission, Marriott Washingtonian Center, Gaithersburg, MD, September 29 to October 2, 2003, NUREG/CP-0191, Volume 1 (ADAMS Accession No. ML052370273).

for the crack surface roughness, number of turns and actual flow path. It also provides the mean and standard deviation of the data obtained. The licensee stated that these statistics produce a fairly wide variability in results from the data analyzed including that for PWSCC which can contribute to overly conservative results.

The licensee stated also that only crack length parameters are closely banded. These numbers are the ratio of crack lengths to thicknesses and are stand-alone values. The 2003 presentation listed both K_G (large crack opening displacement (COD) relative to global roughness) and K_{G+L} (small COD relative to global roughness). The 2003 presentation notes that the local roughness and global roughness for the PWSCC cracks (growing parallel to the long direction of the dendritic grains) are higher than those for IGSCC cracks, while also noting that the number of turns is lower. This suggests that crack length factor (i.e., K_{G+L}) may have a dominating role in the weld metal response to the PWSCC crack growth. The length of the actual flow path is simply K_{G+L} times t (weld thickness).

The licensee stated that based on the limited data available, the analyses in the 2003 presentation are believed to provide overly conservative results. Specifically, the 69 percent increase in leakage flaw size is believed to be exaggerated due to lack of data. Therefore, the licensee believes that the use of a penalty factor of 1.69 for Alloy 82/182 materials in WCAP-17187-P is concluded to be conservative.

To demonstrate the conservatism in using the penalty factor of 1.69, the licensee discussed a similar study during a Pressure Vessel and Piping Conference in July 2006 (July 2006 study)[13]. The licensee believes that the July 2006 study provides a better representation of the affects of PWSCC in Alloy 82/182 materials because it provides benchmarking results for the leak rate program KRAKFLO. Secondly, the laboratory test results for intergranular stress-corrosion cracking (IGSCC) leak rate studies were analyzed and found to be in agreement with the measured and predicted leak rates with conservative predictions found for small leak rates. The further benchmarking performed for field data would tend to provide additional validation of the inputs. Finally, the July 2006 study provides a sensitivity study for five different piping systems containing Alloy 82/182 welds.

The licensee stated that KRAKFLO predicts on average a 37 percent increase in leakage crack length when considering the crack with SCC morphology over conventional fatigue morphology. The results of this approach provide a more highly validated penalty factor of 1.37 in fatigue leakage flaw size for LBB applications. It should be noted that IGSCC crack morphology which was used in the July 2006 study is conservative since the PWSCC crack morphology is less severe than the IGSCC. The licensee stated further that based on the variability of results in the 2003 presentation, the application of a penalty factor of 1.69 for Alloy 82/182 welds provides highly conservative results for the original weld joint.

According to the licensee, the structural material at the weld overlaid locations is Alloy 52M, which is resistant to PWSCC. Considering that the new weld joint consists of both Alloy 52M and Alloy 82/182 materials, it would be even more conservative to apply this same penalty

---

13 Nana, A. D., Yoon, K. K., "Comparison of Leak Rates from Alloy 82/182 Butt Weld Cracks for Leak-Before-Break Applications," American Society of Mechanical Engineers, Pressure Vessel and Piping Conference, PVP2006-ICPVT11-93767, Vancouver, Canada, July 23-27, 2006.

factor throughout both materials. Therefore, a penalty factor of 1.0 for PWSCC is utilized for the thickness of the Alloy 52/152 weld material (consistent with SRP Section 3.6.3.III.3).

The licensee stated that the mechanical material properties of Alloy 82/182 and Alloy 52M are essentially the same. From a stress standpoint there are no interface differences to consider at the transition between the Alloy 82/182 base layer and the Alloy 52M weld overlay layer. The weld with the overlay can be addressed as a thicker layer of material. Absent the additional PWSCC crack morphology affects, the leakage flaw size for the Alloy 52M weld overlay will be appropriately smaller than that for the base weld. The leakage flaw without the penalty factor is a conservative estimation of the leakage flaw size for the weld overlay itself, the only difference being the effect of thickness.

The licensee explained that first, the leakage flaw size using fatigue crack morphology parameters is determined for the original Alloy 82/182 weld based on a given leak rate of 2.5 gpm (using a margin of 10 on leak detection). Subsequently, its size is increased (i.e., multiplied) by the penalty factor to obtain the PWSCC leakage flaw size for the overlaid Alloy 82/182 weld. The leak rate for the overlaid weld would be greater than 2.5 gpm because overlay crack morphology is smoother than the original weld. However, the leak rate is limited by the leakage inventory of 2.5 gpm from the original weld. These elements are combined primarily because the assumed leakage flaw would have to penetrate the Alloy 82/182 material for a "potential" leakage path. Also the post weld overlay residual stresses demonstrate that the inside surface of the pipe will be in a compressive stress zone. Therefore, the weld overlay application will prevent any crack initiation and arrest any PWSCC cracks at the inside surface.

The NRC staff does not agree with the licensee's argument that limited data in the 2003 presentation suggests overly conservative results (i.e., the penalty factor of 1.69 is overly conservative) for leakage in PWSCC in Alloy 82/182 welds. From a statistics standpoint, limited data can be used to make a good estimate of the mean. However, limited data suggest that the lack of knowledge uncertainty is large, thus producing a large scatter in the predicted distributions in the 2003 presentation. Obtaining more data will decrease this uncertainty and reduce the scatter, but will not have a large impact on the mean.

The NRC staff noted that the Battelle Memorial Institute conducted the experiments that were used to benchmark the KRAKFLO predictions. The COD measurements made during the leakage experiments were taken near the crack tip and not at the crack centerline. Therefore, the actual experimental crack opening was much larger than measured in the experiments. If an analysis was used to predict these experiments and assuming the COD (actually measured off center) was measured at the centerline, the analyses would under-predict the effects of morphology. Therefore, the staff believes that the penalty factor of 1.37 may under-predict the effects of crack morphology parameters.

By letter dated August 12, 2010, the licensee stated that the comparison to the July 2006 study, which concluded a penalty factor of 1.37, was only to show that the application of a penalty factor of 1.69 for the PWSCC-susceptible layer and the average penalty factor of 1.44 provided conservative results. The licensee contends that if a 1.69 penalty factor were to be applied for both the Alloy 82/182 and Alloy 52M weld layers, the leakage flaw size would only be increased by 17 percent which is within the currently analyzed LBB margins contained in WCAP-17187-P.

Therefore, the licensee concludes that the use of a PWSCC penalty factor of 1.69 for increasing the size of the leakage flaw for the base layer is conservative.

Based on the above, the NRC staff concludes that the licensee's methodology of using penalty factors to predict crack sizes is adequate and acceptable because the licensee has used the PWSCC data from studies in the July 2006 study and 2003 presentation. In its confirmatory analysis, the staff has verified that the critical crack length of the surge line is twice of the leakage crack length based on a leak rate that is 10 times than the RCS leakage detection system capability. Therefore, the staff concludes that Waterford 3 has satisfied the margins in SRP Section 3.6.3.

3.2.8 Weld Residual Stresses

Page 5-3 of WCAP-17187-P states that the weld residual stresses (WRS) from the WOL will generate compressive stresses on the inside diameter surface, thereby, making the leakage flaw size conservative. The NRC staff noted that the licensee's leakage crack calculation did not include WRS which may be non-conservative. By letter dated August 12, 2010, in response to RAI R-12 of the NRC staff's RAI dated April 21, 2010, the licensee stated that its approach regarding residual stresses was based on a defense-in-depth perspective to emphasize that surface cracking will not initiate due to the compressive stresses on the inside diameter of the pipe. The licensee stated further that for the LBB analysis, hypothetical through-wall flaws are postulated in order to calculate the leakage flaw size and critical flaw size. The WRS are localized in nature and are self-relieving for the postulated through-wall flaws used in the LBB analyses. SRP Section 3.6.3 does not require WRS to be included in the analysis.

The NRC staff noted that when SRP Section 3.6.3 was originally published in 1980s, PWSCC was not a concern in PWRs. SRP Section 3.6.3 was not developed to address welds that have been overlaid that created situations where WRS is an important factor. The effects of WRS on the leakage were unknown and, therefore, WRS were not specified in SRP Section 3.6.3. The staff does not believe that the WRS are self-relieved by the presence of the through wall crack. The staff has performed analyses that demonstrate that the WRS changes the crack opening displacement, which will affect leak rate.

By letter dated August 12, 2010, the licensee responded that based on NUREG/CR-6300, "Refinement and Evaluation of Crack-Opening-Area Analyses for Circumferential Through-Wall Cracks in Pipes," April 1995, the impact of WRS on the leakage flaw size is insignificant. Using Table 8.3 of NUREG/CR-6300, the impact of WRS on the crack opening displacement (COD) is roughly 3.3 to 4.4 percent which is within engineering calculation accuracy. Therefore, WRS does not represent a significant load in the LBB analysis. These stresses have also not been considered at other non-Alloy 82/182 weld locations.

The licensee stated that other more recent studies have been conducted that considered the impact of WRS on COD such as documented in NUREG/CR-6765, "Development of Technical Basis for Leak-Before-Break Evaluation Procedures," May 2002 (ADAMS Accession No. ML021720594), and NUREG/CR-6837, "The Battelle Integrity of Nuclear Piping (BINP) Program Final Report," June 2005 (ADAMS Accession No. ML051660467). NUREG/CR-6837 indicates that the effect of WRS on COD is especially pronounced for thin-wall pipe operating at low stress level. Even though the affect on COD due to WRS may vary, the general conclusions reached is that for most practical applications, the effect of WRS on the COD, by itself, was not a major contributing factor for LBB analyses (i.e., less than a 15 to 20 percent effect on the margin or crack size). The licensee stated that the subsequent revision (Revision 1) to SRP Section 3.6.3 did not require the effect of WRS be included in LBB leakage flaw determination methods.

Although SRP Section 3.6.3 does not specify WRS to be included in the LBB analysis, the NRC staff believes that WRS should be considered in the LBB evaluation for the overlaid Alloy 82/182 welds. In its confirmatory analysis, the staff included WRS and has verified that the surge line satisfies the margins on crack size and leak rate per SRP Section 3.6.3.

3.2.9 Crack Stability Analysis

The licensee performed a crack stability analysis to demonstrate that the postulated leakage cracks in the surge pipe are stable per SRP Section 3.6.3.III.11.C. SRP Section 3.6.3 allows the use of J-integral method and limit load method for the crack stability analysis. To determine crack stability for Nodes 20, 70, and 75, the licensee used the J-integral method. To determine crack stability for Node 10, the licensee used the limit load method because of the Alloy 82/182 weld.

The acceptance criteria for the J-integral method stipulate that if $J_{applied} < J_{Ic}$, the crack is stable. If $J_{applied} > J_{Ic}$, the crack is stable only if $T_{applied} < T_{material}$ and $J_{applied} < J_{max}$ are satisfied. $J_{applied}$ is the J-integral value calculated based on applied loads. J_{Ic} is taken from the J-integral resistance curve of the pipe material. It represents the fracture toughness of the pipe material at which point a crack would initiate. $T_{applied}$ is the tearing modulus value based on the applied loading. $T_{material}$ is the tearing modulus based on the material property. J_{max} is the maximum value of J for which $T_{material}$ is greater than or equal to the $T_{applied}$. The J and T values calculated from the applied loads should be less than the J and T values of the pipe material. This is to demonstrate that the pipe material has sufficient fracture toughness to resist uncontrollable crack propagation (i.e., the crack is stable).

Table 6-3 of WCAP-17187-P provides the values of J_{Ic}, $J_{applied}$, J_{max}, $T_{applied}$, and $T_{material}$ for Nodes 20, 70, and 75. The licensee has demonstrated that cracks at these nodes are stable based on the above acceptance criteria.

The limit load method (also known as the net section collapse method) is applicable to very ductile (i.e., high fracture toughness) materials such as stainless steel and welds fabricated by gas tungsten arc welding. To demonstrate crack stability, the degraded pipe is analyzed to determine the remaining net section of the pipe (the section that is not cracked) that reaches a stress value at which a plastic hinge is formed as the pipe is bent. Node 10 represents the Alloy 82/182 and Alloy 52M weld which has the high fracture toughness as that of stainless steel. As specified in SRP Section 3.6.3, for the limited load method, the "Z" factor should be considered for those welds fabricated with submerged arc welding (SMAW). The purpose of the Z-factor application is to increase the loading to compensate for the slight lower toughness material properties of the SMAW weld.

In RAI R-14 of the RAI dated April 21, 2010, the NRC staff questioned, for the overlaid Alloy 82/182 weld, which diameter was used in the Z-factor calculation (e.g., the outside diameter before or after the overlay). By letter dated August 12, 2010, the licensee stated that for the overlaid Alloy 82 weld, the outside diameter after the overlay was applied was used in the Z-factor calculation. The licensee noted that the Z-factor for the Alloy 52M material is 1.0. However, conservatively the same Z-factor for Alloy 82/182 weld was also applied for the Alloy 52M material. Therefore, the calculated critical flaw size with the Z-factor is conservative. The staff concludes that the licensee has used conservative Z-factor in crack stability analysis and, therefore, it is acceptable. The licensee's limit load method showed that the leakage crack at Node 10 will be stable.

The NRC staff concludes that the licensee has demonstrated that postulated cracks at the worst locations (Nodes 10, 20, 70, and 75) of the surge line will be stable and will not propagate uncontrollably under applied load combinations.

3.2.10 Critical Crack Size Calculations

SRP Section 3.6.3.III.11.C specifies how the critical crack size should be calculated. The licensee used the limit load method to calculate the critical crack sizes for Nodes 10, 20, 70, and 75. The licensee calculated critical crack sizes ranging from 10.4 inches to 12.8 inches for Nodes 20, 70, and 75. The critical crack size for Node 10 is shown in WCAP-17187-P (proprietary). The NRC staff concludes that the licensee has demonstrated that the leakage crack size is half of the critical crack size at each of the nodal points and, therefore, the licensee has satisfied the margin of 2 for crack sizes.

3.2.11 Fatigue Crack Growth Calculation

SRP Section 3.6.3 does not specify that a fatigue crack growth calculation be performed as part of the LBB evaluation. However, as an added assurance, the licensee performed a fatigue crack growth calculation based on a postulated non-leaking flaw having a depth of 10 percent of the wall thickness at Node 70 subjected to various transients. The licensee calculated a final flaw size at Node 70 to be 38 percent through wall at the end of 40 years. Node 70 represents the pipe locations other than the Alloy 82/182 weld locations. The NRC staff concludes that a flaw having 38 percent through wall depth is acceptable because it is within the maximum allowable flaw size of 75 percent through wall stipulated in the ASME Code, Section XI, IWB-3600.

For the fatigue crack growth calculation of the Alloy 82/182 welds, the licensee assumed an initial flaw size of 75 percent through wall flaw initiated at the inside surface of the pipe wall at the original Alloy 82/182 welds. This assumption is based on the weld overlay design requirements in Request for Alternative W3-R&R-006[14]. The licensee calculated the shortest time for the postulated flaw to reach the interface between the Alloy 82/182 weld and the weld overlay is 16 years. To monitor the potential for cracking, the overlaid Alloy 82/182 welds will be inspected per the licensee's Request for Alternative W3-R&R-006. This issue has been reviewed in the NRC's safety evaluation dated April 21, 2008 (ADAMS Accession No. ML080950273).

[14] McCann, J. F., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Request for Alternative W3-R&R-006 Proposed Alternative to ASME Code Requirements for Weld Overlay Repairs," dated April 26, 2007 (ADAMS Accession No. ML071230223).

3.2.12 NRC Staff Confirmatory Analysis

In October 2010, the NRC staff performed a confirmatory analysis to verify the safety margins of the surge line. The staff used the SQUIRT (Seepage Quantification of Upsets In Reactor Tubes) computer code to determine the leakage flaw length based on a given leak rate and the NRCPIPE computer code to determine the critical crack size at worst pipe locations.

The NRC staff used material properties for Nodes 10 and 70 from PIFRAC database, and from the licensee's submittals dated February 22, and August 12, 2010. The staff selected Nodes 10 and 70 for analysis because Node 10 represents the overlaid dissimilar metal weld and Node 70 represent the worst location other than the dissimilar metal welds in the surge line.

The NRC staff calculated safety margins using the procedure outlined in SRP Section 3.6.3. The staff used SQUIRT to estimate the leakage flaw length at a given leak rate of 2.5 gpm, for the normal loading (Load Case A). The 2.5 gpm leak rate is based on the margin of 10 applied to the RCS leakage detection capability of 0.25 gpm. The critical flaw length was determined at the applied moment corresponding to the relevant faulted condition for Nodes 10 and 70. The ratio of critical flaw and leakage flaw lengths determines the safety margin. The staff used the following assumptions in its analysis that entail varying degrees of conservatism:

i) The tensile properties of the weld metal, rather than the base metal properties, were assumed for Nodes 10 and 70.

ii) Tensile properties and the J-R curve for Alloy 52 (WOL) material were assumed to be similar to that of Alloy 82/182 dissimilar metal weld (for Node 10).

iii) In calculating the leakage flaw length for Node 10, the PWSCC option was used to determine the crack morphology parameters throughout the entire wall thickness (including the WOL).

iv) In calculating the leakage flaw length for Node 70, the surface roughness (fatigue in air environment) was assumed to be 300 micro-inches.

v) In calculating the critical flaw length, displacement-controlled loading was applied.

vi) In calculating the leakage flaw length and critical flaw length, the GE/EPRI estimation scheme was used.

Based on its analysis, the NRC staff confirmed that the critical flaw length is twice as the leakage flaw length at Nodes 10 and 70 and that the leakage flaw size is calculated based on a leak rate that is 10 times that of the RCS leakage detection system capability. These two results show that the safety margin of 2 on the crack size and the safety margin of 10 on leakage as recommended by SRP Section 3.6.3 have been satisfied.

3.2.13 Fracture Mechanics Analysis Summary

The NRC staff concludes that the pressurizer surge line has satisfied SRP Section 3.6.3 in the following areas of review:

(1) The licensee has mitigated PWSCC in the Alloy 82/182 dissimilar metal welds in the surge line by installing weld overlays. This satisfied the degradation mechanism screening criteria of SRP Section 3.6.3.

(2) The critical crack size is twice the leakage crack size; therefore, the safety margin of 2 specified by SRP Section 3.6.3 is satisfied.

(3) The leak rate from the leakage crack size is 10 times of the RCS leakage detection system capability of 0.25 gpm; therefore, the safety margin of 10 on leakage specified by SRP Section 3.6.3 is satisfied.

(4) The licensee performed a plant-specific and component-specific LBB evaluation and used appropriate input parameters and methodology.

(5) The licensee has demonstrated the stability of the postulated leakage crack sizes at the worst pipe locations.

3.2.14 Summary

On the basis of its review, the NRC staff concludes that for the pressurizer surge line at Waterford, the licensee has demonstrated that (1) a margin of 10 exists between the leak rate from the leakage flaw size and the RCS leakage detection system capability; (2) a margin of 2 exists between the critical flaw size and the leakage flaw size; (3) input parameters for the fracture mechanics analysis are applied consistent with SRP Section 3.6.3; (4) the surge line satisfies the screening criteria for degradation mechanisms of SRP Section 3.6.3, and (5) the postulated leakage cracks at the worst locations have been demonstrated to be stable.

The NRC staff concludes that the pressurizer surge line at Waterford exhibits LBB behavior consistent with the guidance in SRP Section 3.6.3, Revision 1. Pursuant to GDC 4 of Appendix A to 10 CFR Part 50, the staff concludes that the licensee is permitted to implement LBB and to exclude from the current licensing basis consideration of the dynamic effects associated with the postulated rupture of the pressurizer surge line at Waterford.

As a result of this approval, the licensee needs to incorporate the approved LBB application for the pressurizer surge line into the Waterford 3 FSAR. Also, in accordance with commitment made in Attachment 4 to the submittal dated February 22, 2010, the licensee will revise the Waterford 3 FSAR to include additional details on RCS leakage detection system capability.

3.3 RCS Leakage Detection System Capability

3.3.1 Technical Evaluation

As determined by the fracture mechanics evaluation, the postulated leakage flaw in the pressurizer surge line, which included margin to the critical crack size, would produce a leakage rate of 2.5 gpm. Applying a factor of 10 margin for detection capability consistent with SRP Section 3.6.3 guidelines results in a necessary capability to detect an RCS unidentified leakage rate of < 0.25 gpm to satisfy the guidelines. The licensee identified the following two systems as having the necessary leakage detection capability: a plant monitoring computer (PMC) point that samples the containment sump level transmitter every second and the RCS water inventory balance measurements.

In its letter dated August 12, 2010, the licensee described that the containment sump level computer point that the licensee identified as one of the leakage detection systems credited for surge line LBB is the same computer point that is used for the containment sump level leak detection capability for TS 3/4.4.5, "RCS Leakage Detection Instrumentation." The NRC staff accepted this sump level computer point in Waterford 3's License Amendment No. 197[15]. This PMC containment sump computer point for TS 3/4.4.5 compliance is based on the ability to detect a 1 gpm containment inflow rate within 1 hour consistent with response time requirements in RG 1.45. The licensee determined that this installed containment sump instrumentation would be sensitive enough to detect less than a 0.25 gpm change in leakage rate within 20 minutes. The licensee based this determination on combining reference accuracies of individual instrument loop components and averaging the combination over one scan per second for 5 minutes. The level determined from the first 5 minutes of data would be compared to the next 5-minute scan average. Allowing 10 minutes to pump out the containment sump between the 5 minute scan periods, the minimum time to detect a 0.25 gpm change would be 20 minutes. This sensitivity satisfies the SRP Section 3.6.3 guidelines for detecting changes in leakage rates for the postulated 2.5 gpm leakage crack associated with surge line LBB. The NRC staff recognizes that the sump level change reflects water collected in the sump and substantial time may elapse before the rate of water collecting in the sump matches the rate of RCS leakage because some water would be held as vapor in the containment atmosphere and as liquid (water) in locations that may not immediately drain to the sump.

The RCS inventory balance provides an alternative method of detecting changes in RCS leakage of less than 0.25 gpm. TS Surveillance Requirement (SR) 4.4.5.2.1 requires that an RCS water inventory balance be performed every 72 hours while the plant is operating at steady-state to ensure that unidentified operational leakage is within its specified limits. The licensee stated in the LAR that the data from the RCS inventory balance would be used to establish unidentified leakage rate from the RCS (gpm).

In Westinghouse's WCAP-16465-NP, Revision 0, "Pressurized Water Reactor Owners Group Standard RCS Leakage Action Levels and Response Guidelines for Pressurized Water Reactors," September 2006 (ADAMS Accession No. ML070310082), the Pressurized Water Reactor Owners Group (PWROG) found that RCS inventory balance measurements can detect

---

[15] Alexion, T. W., U.S. Nuclear Regulatory Commission, letter to Joseph Venable, Entergy Operations, Inc., "Waterford Steam Electric Station, Unit 3 – Issuance of Amendment Re: Reactor Coolant System Leakage Detection (TAC No. MC3085)," dated July 30, 2004 (ADAMS Accession No. ML032150057).

small leaks (<0.1 gpm) when data is recorded for a sufficient period of time. In WCAP-16465-NP, the PWROG references the methodology for calculating the unidentified RCS leakage rate provided in Westinghouse's WCAP-16423-NP, Revision 0, "Pressurized Water Reactor Owners Group Standard Process and Methods for Calculating RCS Leak Rate for Pressurized Water Reactors," September 2006 (ADAMS Accession No. ML070310084). In WCAP-16423-NP, the PWROG specified an optimal duration of 2 hours for the RCS inventory balance measurements.

In its amendment submittal, the licensee reported it has been trending normal RCS leakage at levels below 0.1 gpm. Based on reviews associated with preparation of Revision 1 to RG 1.45, the NRC staff concludes that operating experience supports this level of unidentified leakage detection sensitivity for the RCS inventory balance method on a generic basis. Thus, within a reasonable period after onset of RCS leakage at 0.25 gpm and well in advance of reaching the TS 3.4.5 unidentified leakage rate limit of 1.0 gpm, the RCS inventory balance would support implementation of actions to indentify the leakage source and mitigate the consequences of the leak.

The licensee described that various action levels have been procedurally established and many of the action levels have been based on the unidentified leakage rate. These action levels were based on those established by the PWROG in WCAP-16465-NP. In its letter dated February 22, 2010, the licensee listed the following action levels based on RCS unidentified leakage:

- One seven (7) day rolling average of daily unidentified RCS leak rates > 0.1 gpm.
- Two consecutive daily unidentified RCS leak rates > 0.15 gpm.
- One daily unidentified RCS leak rate > 0.3 gpm.

The licensee described actions that would be initiated as part of the corrective action process to address the condition causing the action level to be exceeded and that these actions may include performing containment walkdowns of accessible areas or performing a plant shutdown to address and correct the RCS leakage condition.

The licensee provided a markup of the FSAR pages detailing the additional Waterford 3 RCS leakage detection and monitoring system capability in Attachment 3 to the amendment request dated February 22, 2010. The proposed FSAR changes included a description of how the PMC point for the containment sump level instrument achieves the necessary sensitivity to support the LBB analysis and a description of the RCS unidentified leakage action levels. In Attachment 4 to the amendment request dated February 22, 2010, Entergy made a commitment to modify the Waterford 3 FSAR to include additional details on RCS Leakage Detection System capability. This commitment is consistent with the requirements of 10 CFR 50.71(e), which requires that the effect of changes to the facility and changes to procedures described in the FSAR be described in the next FSAR update.

3.3.2 RCS Leakage Detection System Technical Specifications

The NRC staff evaluated the adequacy of the existing Waterford 3 RCS leakage detection system TSs. As noted above, Waterford 3 TS LCO 3/4.4.5.1, "Leakage Detection Instrumentation," requires that one containment sump monitor be operable during power operation, startup, hot standby, and hot shutdown. The related TS LCO 3/4.4.5.2, "Operational Leakage," specifies that RCS operational leakage be limited to no pressure boundary leakage and 1 gpm unidentified leakage. To verify the limits in TS LCO 3/4.4.5.2 are satisfied, TS SR 4.4.5.2.1 requires that, beginning 12 hours after establishment of steady-state operation, RCS leakage be demonstrated to be within the above leakage limits through performance of a RCS water inventory balance at least once per 72 hours.

In an attachment to a letter dated November 23, 2010, in response to an NRC staff RAI dated October 27, 2010 (ADAMS Accession No. ML103000070), the licensee described that the existing Waterford 3 TS LCO 3.4.5.1 included the instrumentation to detect RCS leakage specified by RG 1.45 guidelines and necessary for compliance with GDC 30. In part, GDC 30 specifies that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. For compliance with the LBB guidance included in SRP Section 3.6.3, Entergy credited both the containment sump level instrument, which is one of the leakage detection instruments included in TS LCO 3.4.5.1 and credited to satisfy GDC 30, and the Waterford 3 RCS leakage monitoring program, which provides supplemental detection to satisfy LBB guidelines. However, the licensee concluded that the RCS leakage monitoring program, as credited to satisfy LBB guidelines (GDC 4), did not result in the need for new instruments to be included for GDC 30 compliance under TS LCO 3.4.5.1. The NRC staff recognizes the basis for the existing TS was related to GDC 30, but the current criteria for in TS LCO 3.4.5.1 is prescribed by the requirements of 10 CFR 50.36(c)(2)(ii). Criterion 1 of 10 CFR 50.36(c)(2)(ii) applies to installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. Therefore, the criterion does not limit the instrumentation included in TS LCOs to that instrumentation credited to satisfy GDC 30.

In the attachment to its letter dated November 23, 2010, the licensee also addressed the applicability of Criterion 2 of 10 CFR 50.36(c)(2)(ii), which applies to operating restrictions that are an initial condition of a transient analysis that assumes the failure of the integrity of a fission product barrier. In the attachment, the licensee referenced material from a generic evaluation of material reliability prepared by EPRI and presented in EPRI MRP-109, "Materials Reliability Program: Alloy 82/182 Pipe Butt Weld Safety Assessment for the US PWR Plant Designs: Westinghouse and CE Design Plants (MRP-109NP)," (non-proprietary version available at ADAMS Accession No. ML042430093). The objective of this report was to address the safety significance of postulated flaws in bimetallic butt welds, which are among the most likely locations for through-wall leak development in Combustion Engineering PWR reactor coolant system piping. Based on representative industry data from EPRI MRP-109, the licensee found that the period between development of a leakage flaw of 2.5 gpm and its progression to a critical flaw was conservatively estimated to be in excess of 2 years. Similarly, a flaw leaking at 1 gpm leakage would provide 2 years of detection time prior to the flaw reaching the leakage flaw size of 2.5 gpm. Therefore, the licensee concluded the existing Waterford 3 unidentified RCS leakage TS LCO 3.4.5.2 limit of 1.0 gpm provided ample control room operator response time for detection of an initial RCS pressure boundary leak.

After reviewing the response, the NRC staff requested that the licensee explain how the EPRI's MRP-109 data cited in the response satisfied SRP Section 3.6.3 guidance regarding leakage detection margin to account for uncertainties in the determination of leakage from postulated cracks in piping. The licensee provided its response in an attachment to its letter dated December 21, 2010. Entergy reported that the leakage detection uncertainty margin of 10 provided substantial conservatism, in part, because the Waterford 3 RCS water chemistry is maintained free of debris to the extent practical through feed and bleed chemistry control and minute wear particulates would thus be limited and not conducive to a flaw plugging environment. Also, RCS monitoring programs have been maintained sensitive to RCS leakage and the Waterford 3 sump level instrumentation has been maintained with low uncertainty values. In addition, the licensee provided additional description of the margin provided by the modeling of the leakage flaws in EPRI MRP-109 because the model included flaw calculations, material conditions, and loads that conservatively bound the Waterford 3 plant surge line. Finally, Entergy stated that the EPRI MRP-109 data demonstrated that, if no action were taken prior to reaching the shutdown action requirement under TS LCO 3.4.5.2 for unidentified operational leakage of 1.0 gpm and that leakage corresponded to an actual leakage flaw size of 10 gpm, there would remain in excess of an operating cycle (>1.5 years) prior to the 10 gpm leakage flaw having grown to a critical flaw state. The NRC staff concluded that the EPRI MRP-109 report provides an adequate basis to consider the relative growth rate of through-wall leakage cracks. Therefore, the existing TS LCO would provide margin to the potential unstable rupture of the pressurizer surge line.

As noted in the *Federal Register* notice accompanying the issuance of 10 CFR 50.36 (60 FR 36953, July 19, 1995), the rule reflects that technical specifications were intended to be reserved for those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. The available margin between leakage cracks of a size that assures detectable leakage reaches the leakage detection instrumentation and larger cracks that could progress to unstable pipe ruptures provides reasonable assurance that the existing RCS leakage TS remain acceptable with respect to the requirements of 10 CFR 50.36. Therefore, the pressurizer surge line LBB leakage rate of 0.25 gpm would not pose an immediate threat to safety and both the instruments used to detect this leakage rate and the leakage value itself may be appropriately controlled as FSAR information.

3.3.3 Summary

On the basis of its review of the LBB evaluation for the pressurizer surge line at Waterford 3, the NRC staff concludes that the licensee has demonstrated that (1) the availability of diverse instrumentation to detect leakage a factor of 10 below the calculated leak rate from the leakage flaw size; and (2) sufficient margin exists between the existing RCS leakage TS LCO for unidentified leakage and leakage likely to be associated with a crack that could progress to an unstable rupture that the leakage detection capability associated with LBB would be appropriately controlled as FSAR information. Therefore, the NRC staff concludes that the proposed leakage detection capability for pressurizer surge line LBB is consistent with the guidance of SRP Section 3.6.3, Revision 1, and is acceptable.

3.4 Commitments

In its letter dated February 22, 2010, the licensee made the following regulatory commitment:

> Entergy will modify the Waterford 3 Final Safety Analysis Report to include additional details on RCS Leakage Detection System capability.

The licensee's regulatory commitment is a "One Time Action" item and scheduled to be completed when the next regularly scheduled Waterford 3 FSAR Update submittal is done.

The NRC staff concludes that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the regulatory commitment are best provided by the licensee's administrative processes, including its commitment management program. The regulatory commitment does not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on April 20, 2010 (75 FR 20632). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. Tsao
S. Jones
R. Grover

Date: February 28, 2011

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures:
1. Amendment No. 232 to NPF-38
2. Safety Evaluation

cc w/encls: Distribution via Listserv

DISTRIBUTION:
PUBLIC
LPLIV r/f
RidsAcrsAcnw_MailCTR Resource
RidsNrrDciCpnb Resource
RidsNrrDirsItsb Resource
RidsNrrDorlLpl4 Resource
RidsNrrDorlDpr Resource
RidsNrrDssSbpb Resource
RidsNrrPMWaterford Resource
RidsNrrLAJBurkhardt Resource
RidsOgcRp Resource
RidsRgn4MailCenter Resource
JTsao, NRR/DCI/CPNB
SJones, NRR/DSS/SBPB
RGrover, NRR/DIRS/ITSB

ADAMS Accession No. ML110410119 *SE memo dated

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	NRR/DCI/CPNB/BC	NRR/DSS/SBPB/BC
NAME	NKalyanam	JBurkhardt	TLupold*	GCasto*
DATE	2/16/11	2/14/11	2/8/11	1/13/11
OFFICE	OGC NLO	NRR/LPL4/BC	NRR/LPL4/PM	
NAME	LSubin	MMarkley	NKalyanam	
DATE	2/17/11	2/28/11	2/28/11	

OFFICIAL RECORD COPY