Laura A. Basta
H. B. Robinson Steam Electric Plant Unit 2
Site Vice President

Duke Energy
3581 West Entrance Road
Hartsville, SC 29550

O: 843 951 1701
Laura.Basta@duke-energy.com

Serial: RA-22-0290
August 30, 2023

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

H. B. Robinson Steam Electric Plant, Unit No. 2
Docket No. 50-261 / Renewed License Number DPR-23

SUBJECT: License Amendment Request to Exclude the Dynamic Effects of Specific Postulated Pipe Ruptures from the Design and Licensing Basis Based on Leak-Before-Break Methodology

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy) is submitting a request for an amendment to the Facility Operating License for H. B. Robinson Steam Electric Plant, Unit No. 2 (RNP). The proposed amendment uses the Leak-Before-Break (LBB) methodology to eliminate the dynamic effects of postulated pipe ruptures to auxiliary piping systems attached to the Reactor Coolant System (RCS) from the RNP design and licensing basis. There are no proposed changes to the Technical Specifications associated with this License Amendment Request (LAR).

This LAR is submitted in accordance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 4, "Environmental and dynamic effects design bases," following the guidance of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition", Section 3.6.3, "Leak-Before-Break Evaluation Procedures".

As noted in GDC 4, 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 the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. An evaluation of the proposed change is provided in Attachment 1.

The supporting technical bases for applying LBB methodology to the RCS auxiliary piping for RNP is provided by the following documents:

- WCAP-17776-P/NP, Revision 1, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for H. B. Robinson, Unit 2", March 2023,
- WCAP-17778-P/NP, Revision 1, "Technical Justification for Eliminating Residual Heat Removal (RHR) Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2", March 2023,
- WCAP-17779-P/NP, Revision 1, "Technical Justification for Eliminating Accumulator Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2", March 2023

WCAP-17776-P, Revision 1, WCAP-17778-P, Revision 1, and WCAP-17779-P, Revision 1 (Attachments 6, 7 and 8 respectively) include information proprietary to Westinghouse Electric Company, LLC (Westinghouse), and the Westinghouse documents CAW-23-008, CAW-23-009, and CAW-23-010 in Attachment 5 provide the affidavits signed by the owner of the information. Accordingly, it is respectfully requested that the proprietary information be withheld from public disclosure in accordance with 10 CFR 2.390. The redacted, non-proprietary versions are provided in Attachments 2, 3, and 4. Correspondence with respect to the proprietary aspects of the affidavits should be addressed to the Westinghouse representative identified in the respective affidavits.

Duke Energy has evaluated the proposed amendment and has determined it does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for this determination is included in Attachment 1.

Duke Energy requests approval of the proposed license amendment within one year of completion of the NRC's acceptance review to support the Subsequent License Renewal Application for RNP. Following NRC approval, Duke Energy will implement the amendment within 120 days.

In accordance with 10 CFR 50.91, Duke Energy is notifying the state of South Carolina of this license amendment request by transmitting a copy of this letter to the designated state officials.

Should you have any questions concerning this letter, or require additional information, please contact Ryan Treadway, Director - Nuclear Fleet Licensing at (980) 373-5873.

This submittal contains no new regulatory commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 30, 2023.

Sincerely,

Laura A. Basta
Site Vice President

Attachments:

1. Evaluation of the Proposed Change
2. WCAP-17776-NP, Revision 1, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2", March 2023 (Redacted)
3. WCAP-17778-NP, Revision 1, "Technical Justification for Eliminating Residual Heat Removal (RHR) Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2", March 2023 (Redacted)
4. WCAP-17779-NP, Revision 1, "Technical Justification for Eliminating Accumulator Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2", March 2023 (Redacted)
5. Westinghouse Affidavits
6. WCAP-17776-P, Revision 1, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2", March 2023 (Proprietary)
7. WCAP-17778-P, Revision 1, "Technical Justification for Eliminating Residual Heat Removal (RHR) Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2", March 2023 (Proprietary)
8. WCAP-17779-P, Revision 1, "Technical Justification for Eliminating Accumulator Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2", March 2023 (Proprietary)

cc: (Without Attachments)

L. Dudes, Regional Administrator USNRC Region II
J. Zeiler, NRC Senior Resident Inspector
L. Haeg, NRR Project Manager

A. Wilson, Attorney General (SC)
R. S. Mack, Assistant Bureau Chief, Bureau of Environmental Health Services (SC)
L. Garner, Manager, Radioactive and Infectious Waste Management Section (SC)

ATTACHMENT 1

EVALUATION OF THE PROPOSED CHANGE

EVALUATION OF THE PROPOSED CHANGE

TABLE OF CONTENTS

1.0 SUMMARY DESCRIPTION
2.0 DETAILED DESCRIPTION
  2.1 System Design and Operation
  2.2 Current Technical Specifications Requirements
  2.3 Reason for the Proposed Change
  2.4 Description of the Proposed Change
3.0 TECHNICAL EVALUATION
  3.1 Comparison of WCAP Topical Reports to NUREG-0800, Section 3.6.3.III, Review Procedures
  3.2 Margin Assessment
  3.3 Reactor Coolant System Leakage Detection Systems
  3.4 Conclusion
4.0 REGULATORY EVALUATION
  4.1 Applicable Regulatory Requirements/Criteria
  4.2 Precedent
  4.3 No Significant Hazards Consideration Determination
  4.4 Conclusions
5.0 ENVIRONMENTAL CONSIDERATION
6.0 REFERENCES

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy) requests an amendment to the Facility Operating License for H. B. Robinson Steam Electric Plant, Unit No. 2 (RNP). NRC approval is requested for application of the leak-before-break (LBB) methodology to auxiliary piping systems attached to the Reactor Coolant System (RCS) for RNP to eliminate the dynamic effects of postulated pipe ruptures.

This license amendment request (LAR) is submitted in accordance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 4, "Environmental and dynamic effects design bases," following the guidance of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 3.6.3, "Leak-Before-Break Evaluation Procedures." The LAR applies LBB methodology to demonstrate the risk of pipe rupture is extremely low for portions of auxiliary lines attached to the Reactor Coolant Loops (RCLs).

No changes to the Technical Specifications (TS) are required by this LAR.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

RNP is a three-loop Westinghouse pressurized water reactor. As described in the Updated Final Safety Analysis Report (UFSAR) Section 3.6.1, the current design basis includes application of LBB to the RCS primary loop piping. This LAR would expand the scope of the LBB methodology to include specific portions of piping systems attached to the RCS. The auxiliary lines attached to the RCLs included in the scope of this proposed change include:

- The Pressurizer Surge Line from the primary loop nozzle junction (i.e., weld that connects the nozzle to the surge line piping) to the pressurizer nozzle junction (i.e., weld that connects the pressure surge nozzle safe end to the pressurizer surge nozzle)
- The Residual Heat Removal (RHR) Lines, limited to the high energy Class 1 portion of the RHR lines (primary loop junction to the second isolation valve)
- The 10-inch Accumulator Lines (from the cold legs Loop A, Loop B and Loop C) and attached 8-inch line connected to 10-inch accumulator lines except for the piping upstream of Valves SI-875D, SI-875E, and SI-875F

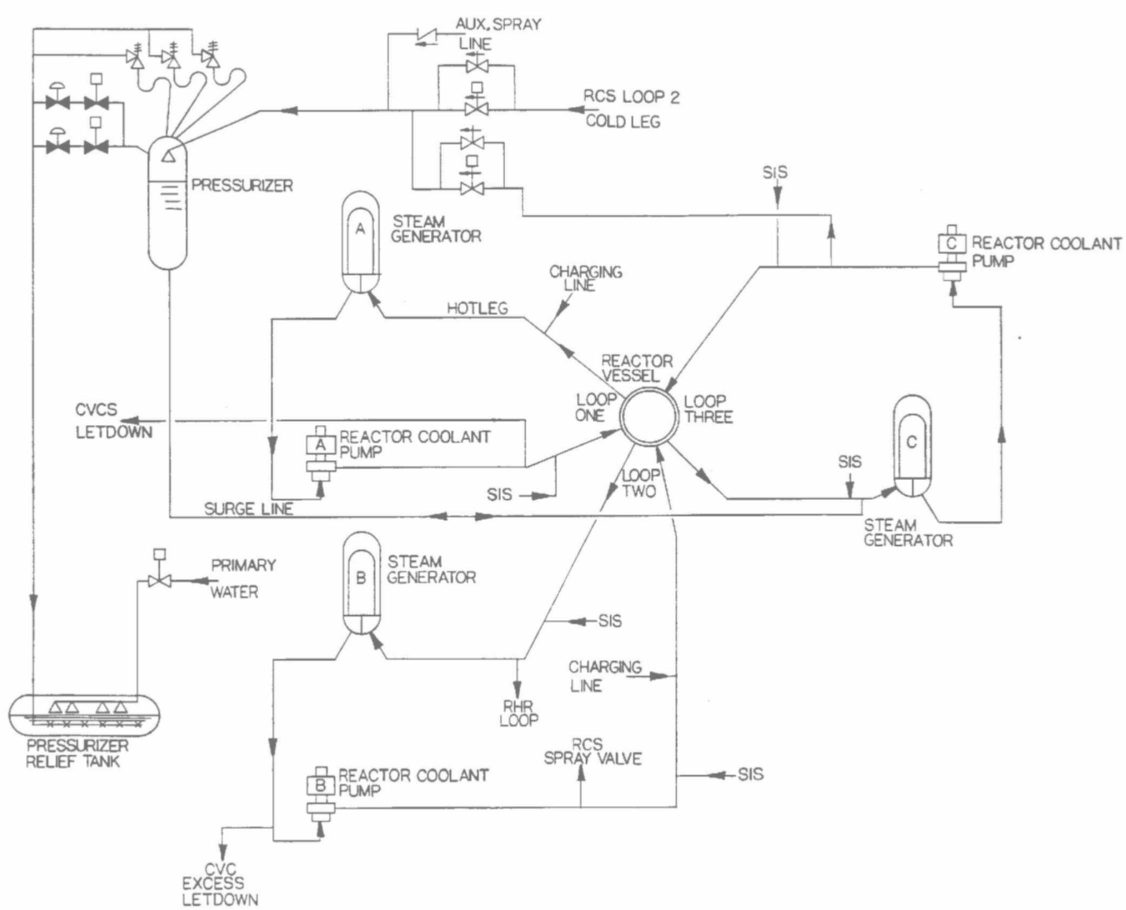
Pressurizer Surge Line - Pressurizer pressure is transmitted to the remainder of the RCS via the surge line that connects the bottom of the pressurizer with the RCS hot leg piping. The pressurizer surge line connects the bottom of the pressurizer to the hot leg of RCL 3.

Residual Heat Removal Lines - The RHR system is a low-pressure, low-temperature fluid system that is not used during power operation. The system is designed to operate at pressures less than 375 pounds per square inch gauge (psig) and at temperatures less than 350 degrees Fahrenheit (°F). The system is operated during plant cooldown after RCS pressure and temperature are within RHR system limitations. The primary purpose of the RHR system is to remove decay heat energy generated in the reactor core during plant cooldown and refueling operations. During plant shutdown and refueling, reactor coolant is drawn from the hot leg of RCS Loop 2 by the RHR pumps, discharged through the tube side of the RHR heat exchangers, and returned to the RCS via all three cold legs.

<u>Accumulator Lines</u> - An accumulator filled with borated water and pressurized with nitrogen is connected to each RCS cold leg. When RCS pressure drops below the nitrogen pressure setpoint, the accumulators discharge their borated water into the RCS. This action provides rapid refilling of the lower core plenum in the event of a large break in the RCS.

Materials that are susceptible to Primary Water Stress Corrosion Cracking (PWSCC), such as Alloy 600 and Alloy 82/182 weld metal, are not found at the RNP RHR line, Accumulator line, or Pressurizer Surge Line. In addition, the RNP RHR line, Accumulator line, and Pressurizer Surge Line do not contain Cast Austenitic Stainless Steel (CASS) material.

Figure 1
Reactor Coolant Loop Piping



2.2 Current Technical Specifications Requirements

RNP Technical Specifications Limiting Condition for Operation (LCO) 3.4.13, specifies that RCS operational leakage shall be limited to.

a) No pressure boundary leakage,
b) 1 gpm unidentified leakage,
c) 10 gpm identified leakage, and
d) 75 gallons per day primary to secondary leakage through any one Steam Generator

Applicability includes Modes 1-4. Actions for specified conditions are as described in LCO 3.4.13. Surveillance Requirements (SR) for RCS operational leakage are provided in SR 3.4.13.1 and 13.4.13.2

RNP Technical Specifications LCO 3.4.15, specifies that the following RCS leakage detection instrumentation shall be operable:

a) One containment sump level monitor,
b) One containment atmosphere radioactivity monitor (gaseous or particulate), and
c) One containment fan cooler condensate flow rate monitor

Applicability includes Modes 1-4. Actions for specified conditions are as described in LCO 3.14.15. SR for RCS leakage detection instrumentation are provided in SR 3.4.15.1 through 13.4.15.5

2.3 Reason for Proposed Change

Duke Energy is requesting the proposed amendment to apply LBB analyses to the RCS branch piping to facilitate potential future plant changes and Subsequent License Renewal Application (SLRA) for RNP. As stated in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 3.6.3, "Leak-Before-Break Evaluation Procedures," NRC staff approval of an LBB analysis permits an operating plant licensee to "remove protective hardware such as pipe whip restraints and jet impingement barriers, redesign pipe connected components, their supports and their internals, and other related changes."

During preparation of the SLRA for RNP it was determined that there were selected issues that would benefit significantly from a reduction in RCS loads through extending the LBB methodology to the branch lines in the scope of this LAR.

Reduced loads will be used to regain margin for the following NRC focus areas for subsequent license renewal:

1. Equivalent Margins Analyses of the Reactor Pressure Vessel (RPV) upper shell assembly and RPV nozzles,
2. Fracture Mechanics Evaluation of the RPV supports located beneath each of the three RPV inlet nozzles, and
3. Assessment of potential flaws in the core barrel for American Society of Mechanical Engineers Section XI or MRP-227 inspections

The RPV upper shell assembly, RPV nozzles, and RPV support assemblies are all susceptible to reduction of fracture toughness owing to additional neutron exposure associated with operation to 80-years. As such, structural integrity of these items with reduced fracture toughness will be evaluated as part of the SLRA as specified in NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants."

2.4 Description of Proposed Change

The proposed change would revise the RNP design and licensing basis to expand the scope of NRC staff's approval of LBB to auxiliary piping connected to the RCLs. The use of LBB for RNP is currently limited to the large, primary loop RCS piping, as discussed in UFSAR Section 3.6.1. The expanded scope LBB would eliminate the dynamic effects of postulated ruptures of specific portions of piping for the Pressurizer Surge Line, the RHR lines, and the Accumulator Lines. Upon implementation of this LAR, relevant sections of RNP UFSAR will be updated to reflect this expanded scope of LBB methodology.

3.0 TECHNICAL EVALUATION

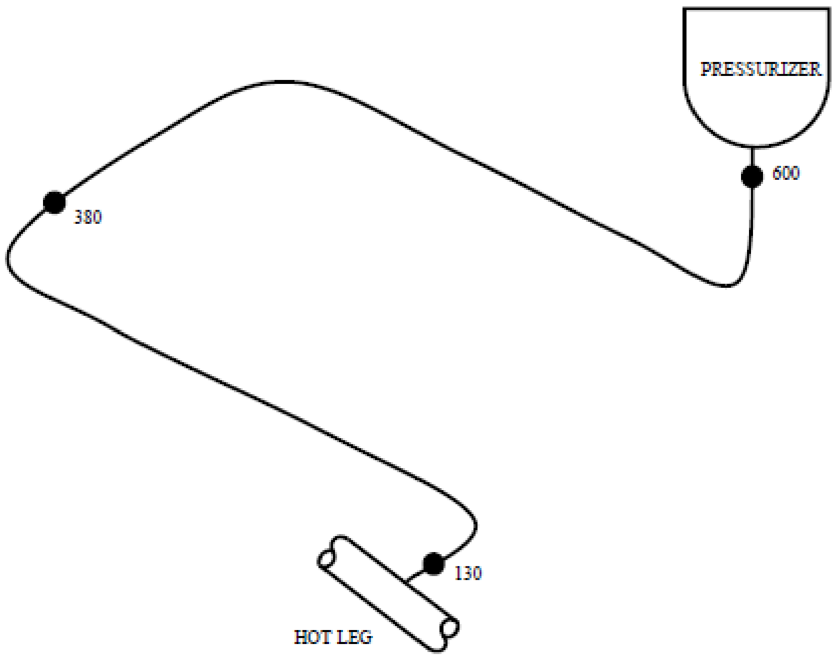
The LBB concept is based on calculations and experimental data demonstrating that certain pipe material has sufficient fracture toughness (ductility) to prevent a small through-wall flaw from propagating rapidly and uncontrollably to catastrophic pipe rupture and to ensure that the probability of a pipe 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 systems early on such that licensees can shut down the plant to repair the degraded pipe long before the potential catastrophic pipe rupture.

While the dynamic effects of pipe breaks have been eliminated for the RNP RCL piping, additional breaks remain applicable for the auxiliary piping systems connected to the RCLs. The auxiliary piping systems attached to the RCLs within the scope of this LAR include the following:

- The Pressurizer Surge Line from the primary loop nozzle junction (i.e., weld that connects the nozzle to the surge line piping) to the pressurizer nozzle junction (i.e., weld that connects the pressure surge nozzle safe end to the pressurizer surge nozzle)
- The Residual Heat Removal (RHR) Lines, limited to the high energy Class 1 portion of the RHR lines (primary loop junction to the second isolation valve)
- The 10-inch Accumulator Lines (from the cold legs Loop A, Loop B and Loop C) and attached 8-inch line connected to 10-inch accumulator lines except for the piping upstream of Valves SI-875D, SI-875E, and SI-875F

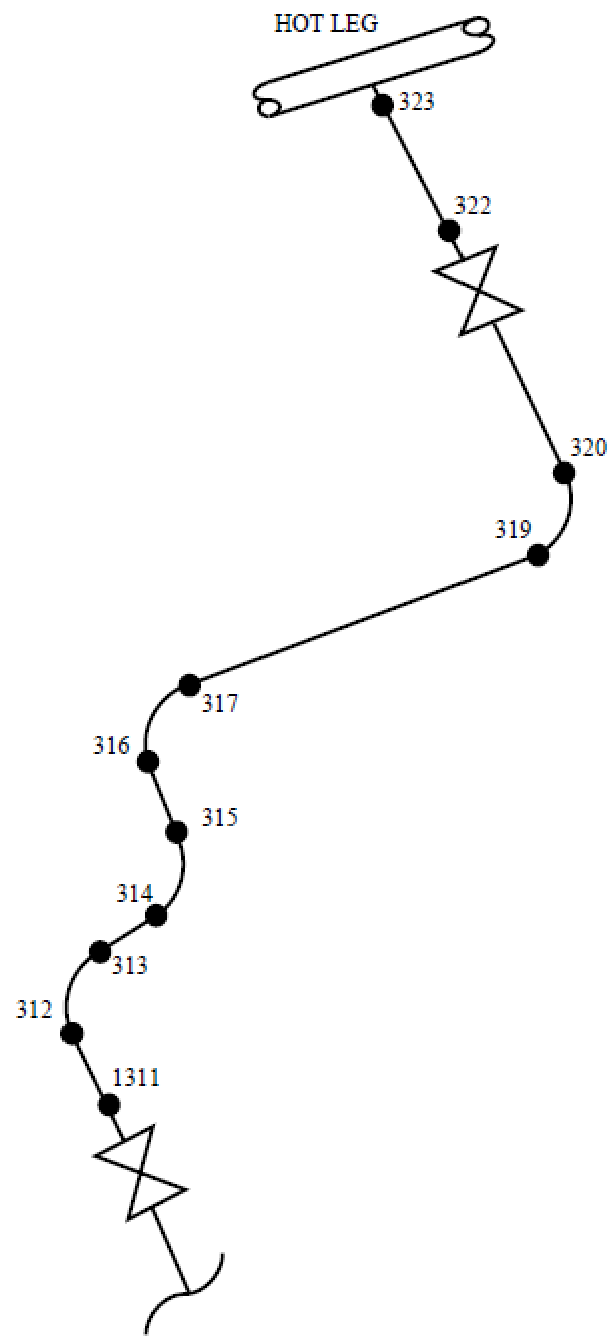
Attachment 6 to this LAR provides WCAP-17776-P, Revision 1, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for H. B. Robinson, Unit 2”. WCAP-17776-P, Revision 1 provides a description of a mechanistic pipe break evaluation method and the analytical results that can be used for establishing that a circumferential type of break will not occur within the pressurizer surge line. The evaluations consider that circumferentially oriented flaws cover longitudinal cases. Additionally, a fracture mechanics analysis that demonstrates the pressurizer surge line integrity for RNP consistent with the NRC’s position for exemption from consideration of postulated pipe rupture dynamic effects is presented. The pressurizer surge line is known to be subjected to thermal stratification and the effects of thermal stratification for the RNP pressurizer surge line have been used in the LBB evaluation presented in WCAP-17776-P, Revision 1. Attachment 2 to this LAR provides a non-proprietary version of WCAP-17776-P, Revision 1. Figure 2 below (from WCAP-17776-P Figure 3-1) shows the Pressurizer Surge line layout.

Figure 2
Pressurizer Surge Line Layout



Attachment 7 to this LAR provides WCAP-17778-P, Revision 1, "Technical Justification for Eliminating RHR Line Rupture as the Structural Design Basis for H. B. Robinson, Unit 2". WCAP-17778-P, Revision 1 provides a description of a mechanistic pipe break evaluation method and the analytical results that can be used for establishing that a circumferential type of break will not occur within the RHR lines. Consistent with the pressurizer surge line reported in WCAP-17776-P, Revision 1, the evaluations consider that circumferentially oriented flaws cover longitudinal cases. Additionally, a fracture mechanics demonstration of the RHR integrity for RNP consistent with the NRC position for exemption from consideration of postulated pipe rupture dynamic effects is presented. Attachment 3 to this LAR provides a non-proprietary version of WCAP-17778-P, Revision 1. Figure 3 below (from WCAP-17778-P Figure 3-1) shows the RHR line layout.

Figure 3
RHR Line Layout



Attachment 8 to this LAR provides WCAP-17779-P, Revision 1, "Technical Justification for Eliminating Accumulator Line Rupture as the Structural Design Basis for H. B. Robinson, Unit 2”. WCAP-17779-P, Revision 1 provides a description of a mechanistic pipe break evaluation method and the analytical results that can be used for establishing that a circumferential type of break will not occur within the accumulator lines that includes the 10-inch accumulator lines (from the cold legs Loop 1, Loop 2, and Loop 3) and attached 8-inch line connected to the 10-inch accumulator lines. NRC Standard Review Plan (SRP) Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Revision 1 requires that LBB should only be applied to high energy, ASME Code Class 1 or 2 piping. As noted in WCAP-17779-P, Revision 1, the scope includes the 10-inch Class 2 and Class 1 Safety Injection System piping from the accumulators to each of the cold legs. Although WCAP-17779-P, Revision 1 includes the Class 2 piping, the scope requested in this LAR does not include the Class 2 piping (piping upstream of Check Valves SI-875-D, SI-875-E, and SI-875-F) since this piping is not a high energy line (temperature less than 200 ^{O}F). Figures 4, 5, and 6 below (from WCAP-17779-P Figures 3-1, 3-2, and 3-3) show the Accumulator line layouts. Consistent with the pressurizer surge line reported in WCAP-17776-P, Revision 1, the evaluations consider that circumferentially oriented flaws cover longitudinal cases. Additionally, a fracture mechanics demonstration of accumulator line piping integrity for RNP consistent with the NRC position for exemption from consideration of postulated pipe rupture dynamic effects is presented. Attachment 4 to this LAR provides a non-proprietary version of WCAP-17779-P, Revision 1.

Figure 4
Loop A Accumulator Line Layout

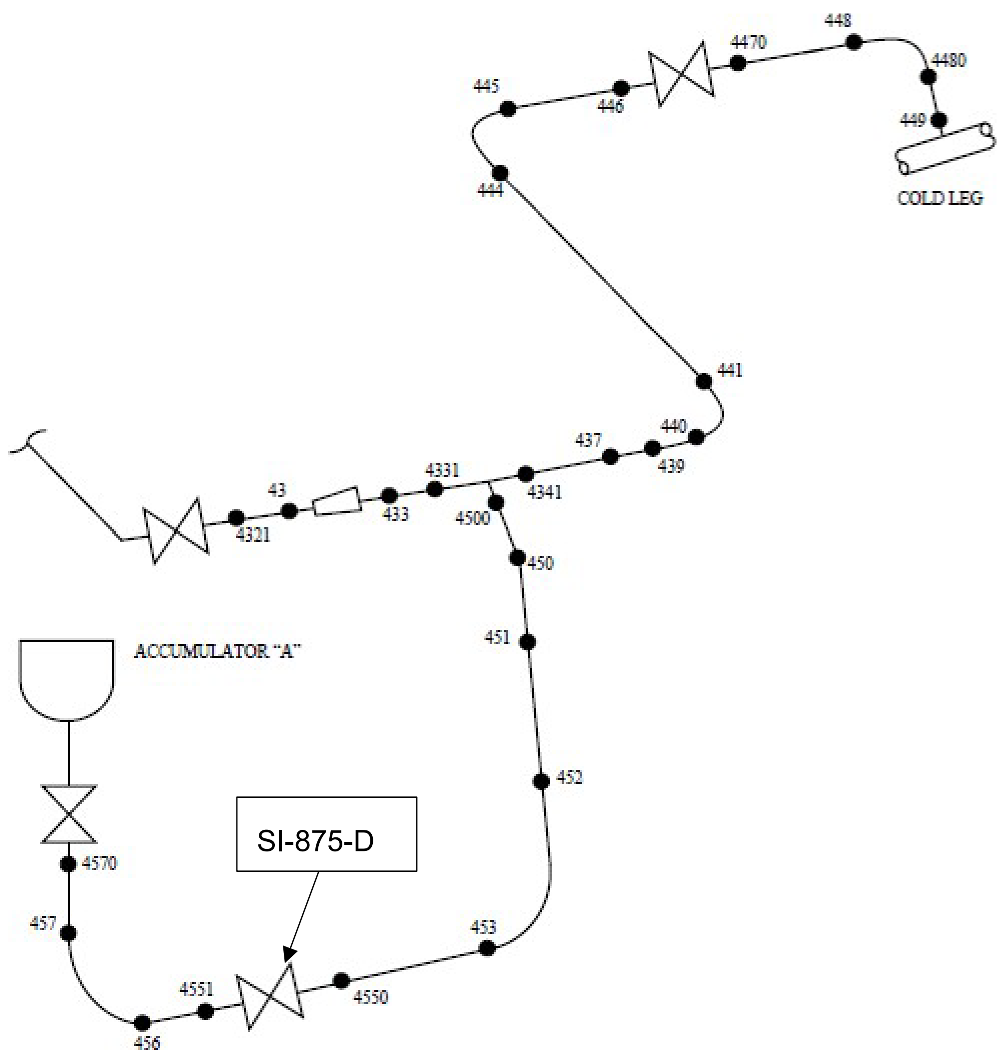


Figure 5
Loop B Accumulator Line Layout

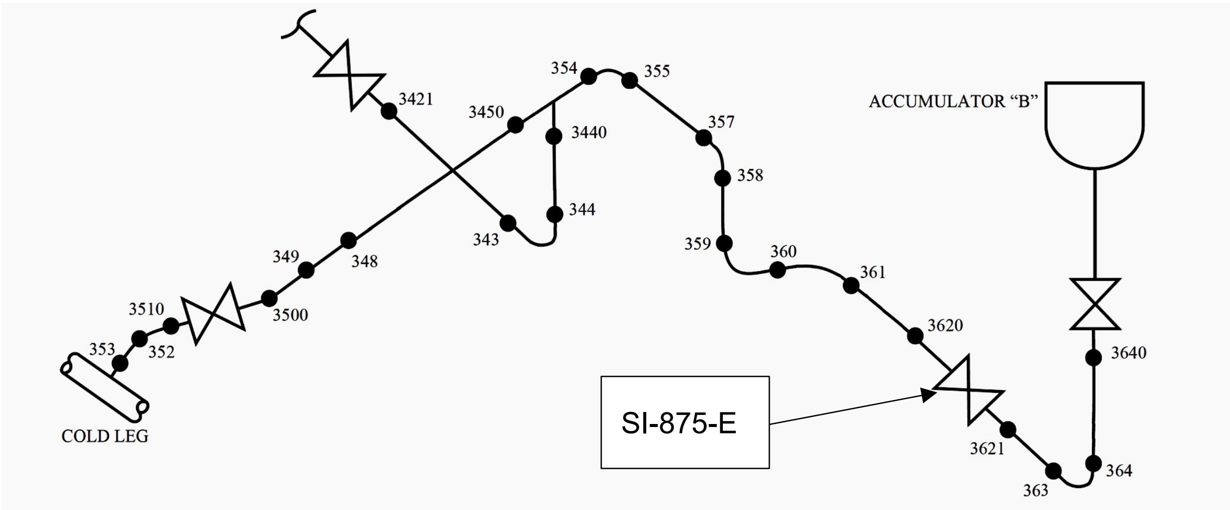
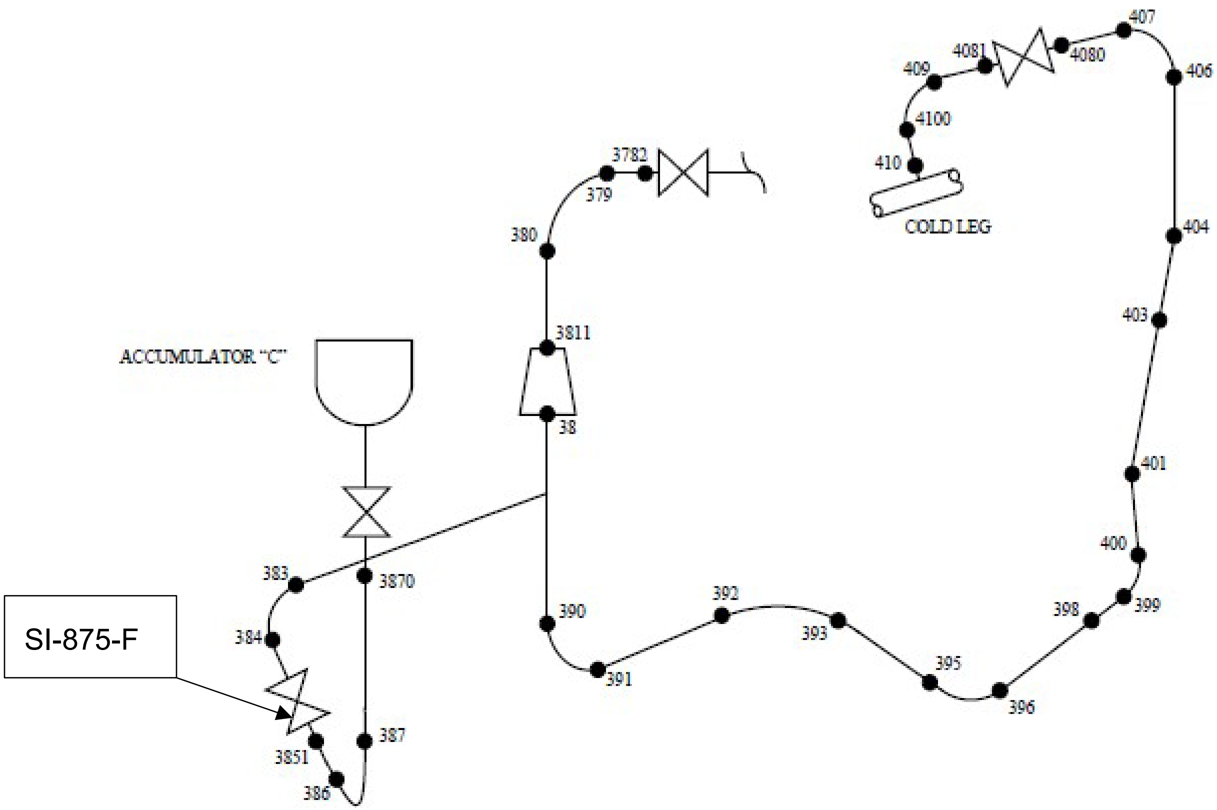


Figure 6

Loop C Accumulator Line Layout



3.1 Comparison of WCAP Topical Reports To NUREG-0800, Section 3.6.3.III, Review Procedures

The branch line Technical Documents (WCAP-17776, Revision 1; WCAP-17778, Revision 1; and WCAP-17779, Revision 1) have not been reviewed and approved by the NRC. Therefore, the following table presents SRP Section 3.6.3.III requirements and sections of the WCAP that address the specific requirements.

Table 1 Compliance with SRP 3.6.3	
SRP 3.6.3, III, Subparagraphs Requirement	**WCAP Sections That Address SRP Requirement**
1. "The reviewer should verify that the licensee's or applicant's LBB evaluation uses design basis loads and is based on the as-built piping configuration, as opposed to the design configuration…."	Sections 3.1 through 3.6 for WCAP-17776-P/NP, Revision 1, WCAP-17778-P/NP, Revision 1, and WCAP-17779-P/NP, Revision 1

Table 1 Compliance with SRP 3.6.3	
SRP 3.6.3, III, Subparagraphs Requirement	**WCAP Sections That Address SRP Requirement**
2. "The reviewer should evaluate the potential for degradation by erosion, erosion/corrosion, and erosion/cavitation due to unfavorable flow conditions and water chemistry. Industry experience for specific piping systems plays an important role in the evaluation of these degradation mechanisms. Additionally, an evaluation of wall thinning of elbows and other fittings is undertaken to ensure that American Society of Mechanical Engineers Code minimum wall requirements are met…"	Sections 2.1 through 2.4 WCAP-17776-P/NP, Revision 1, WCAP-17778-P/NP, Revision 1, and WCAP-17779-P/NP, Revision 1.
3. "The review should evaluate the material susceptibility to corrosion, the potential for high residual stresses, and environmental conditions that could lead to degradation by stress corrosion cracking. Primary water stress corrosion cracking (PWSCC) is considered to be an active degradation mechanism in Alloy 600/82/182 materials in pressurized water reactor plants…"	Section 2.1 for WCAP-17776-P/NP, Revision 1, WCAP-17778-P/NP, Revision 1, and WCAP-17779-P/NP, Revision 1.
4. "The reviewer should evaluate the adequacy of the leakage detection systems associated with the reactor coolant system…"	Section 6.4 for WCAP-17776-P/NP, Revision 1, WCAP-17778-P/NP, Revision 1, and WCAP-17779-P/NP, Revision 1 and section 2.2 of this Evaluation.
5. "The reviewer should verify that the potential for water hammer in the candidate piping systems is very low…"	Section 2.2 for WCAP-17776-P/NP, Revision 1, WCAP-17778-P/NP, Revision 1, and WCAP-17779-P/NP, Revision 1.
6. "The reviewer should verify that the candidate piping is not susceptible to creep and creep-fatigue. Operation below 700°F in ferritic steel piping and below 800°F in austenitic steel piping can alleviate concerns of creep."	Section 2.4 for WCAP-17776-P/NP, Revision 1, WCAP-17778-P/NP, Revision 1, and WCAP-17779-P/NP, Revision 1.
7. "The reviewer should evaluate the corrosion resistance of piping, which can be demonstrated by the frequency and degree of corrosion in the specific piping systems…"	Section 2.1 for WCAP-17776-P/NP, Revision 1, WCAP-17778-P/NP, Revision 1, and WCAP-17779-P/NP, Revision 1.

Table 1 Compliance with SRP 3.6.3	
SRP 3.6.3, III, Subparagraphs Requirement	**WCAP Sections That Address SRP Requirement**
8. "The reviewer should assess the potential for indirect sources of pipe ruptures to ensure that indirect failure mechanisms defined in the plant SAR are negligible causes of pipe rupture. Compliance with the snubber surveillance requirements of the technical specifications ensures that snubber failure rates are acceptably low."	All of the piping segments within the scope of this LAR are inside containment and protected relative to missiles and dynamic effects. Missiles and dynamic effects are discussed in UFSAR, Sections 3.5.1 and 3.6.1.
9. "The reviewer should determine that the piping material will not become susceptible to brittle cleavage-type failures over the full range of system operating temperatures (that is, the material is on the upper shelf of the Charpy Impact energy versus test temperature curve)."	There are no damage mechanisms that can lead to reduction of fracture toughness of the piping materials, radiation levels are low and there are no CASS product forms.
10. "The reviewer should determine that the candidate piping does not have a history of fatigue cracking or failure. An evaluation to ensure that the potential for pipe rupture due to thermal and mechanical induced fatigue is unlikely should be performed."	Sections 2.3 and 8.0 for WCAP-17776-P/NP, Revision 1, WCAP-17778-P/NP, Revision 1, and WCAP-17779-P/NP, Revision 1.
11. "The following steps constitute an acceptable deterministic LBB evaluation procedure:…"	Sections 4, 5, 6, and 7 for WCAP-17776-P/NP, Revision 1, WCAP-17778-P/NP, Revision 1, and WCAP-17779-P/NP, Revision 1.

3.2 Margin Assessment

The results of the leak rates of Section 6.4 and the corresponding stability evaluations of Section 7.2 for each of the three attached WCAPs are used in performing the assessment of margins. All the LBB recommended margins are satisfied. In summary, at all the critical locations relative to:

1. Flaw Size - Using faulted loads obtained by the absolute sum method, a margin of 2 or more exists between the critical flaw and the flaw having a leak rate of 10 gpm (the leakage flaw).
2. Leak Rate - A margin of 10 exists between the calculated leak rate from the leakage flaw and the plant leak detection capability of 1 gpm.
3. Loads - At the critical locations the leakage flaw was shown to be stable using the faulted loads obtained by the absolute sum method (i.e., a flaw twice the leakage flaw size is shown to be stable; hence the leakage flaw size is stable). A margin of 1 on loads using the absolute summation of faulted load combinations is satisfied.

3.3 Reactor Coolant System Leakage Detection Systems

As discussed in GL 80-04 and SRP 3.6.3, the licensee leakage detection systems should be sufficient to provide adequate margin to detect the leakage from a postulated circumferential throughwall flaw. The leak detection systems associated with the RCS are described in the UFSAR, Section 5.2.5, and consist of the following: (1) two radiation sensitive instruments, (2) humidity detector, (3) a condensate monitoring system which determines leakage losses from all water and steam systems within the containment, including that from the RCS, and (4) an increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer, or an increase in containment sump level are also used as leakage detection systems.

The following requirement is provided in Section 5.2.5 of the UFSAR: “To support the application of Leak Before Break methodology, at least one leakage detection system must be operable with a sensitivity capable of detecting a 1 gallon per minute leak within 4 hours.” This requirement was implemented when RNP incorporated WCAP-9558 and WCAP-9787 (References 1 and 2, respectively) into the Current Licensing Basis (CLB) for LBB of main coolant piping. This requirement is an exception to the guidance in Regulatory Guide 1.45, which requires detection of 1 gallon(s) per minute (gpm) leakage within 1 hour. The capability of the RNP leak detection systems to detect 1 gpm leakage is within 4 hours, consistent with the conditions of Generic Letter 84-04 (Reference 3).

3.4 Conclusion

The elimination of pressurizer surge line breaks, the RHR line breaks, and the accumulator line breaks from the structural design basis for RNP is justified as follows:

a. Stress corrosion cracking is precluded by use of fracture resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation.
b. Water hammer should not occur in the pressurizer surge piping, the RHR piping, and the accumulator piping because of system design, testing, and operational considerations.
c. The effects of low and high cycle fatigue on the integrity of the pressurizer surge line piping, the RHR line piping, and the accumulator line piping are negligible.
d. Ample margin exists between the leak rate of small stable flaws and the capability of the RNP, Unit 2 reactor coolant system pressure boundary leakage detection system.
e. Ample margin exists between the small stable flaw sizes of item (d) and larger stable flaws.
f. Ample margin exists in the material properties used to demonstrate end-of-service life stability of the critical flaws.
g. Fatigue crack growth results using the 40-year design transients and cycles (shown to be applicable for 60 years) show that there will be insignificant growth through the wall for the license renewal period (60-year plant life).

For the critical locations, flaws are identified that will be stable because of the ample margins described in d, e, and f above.

Based on loading, pipe geometry and pipe material properties considerations, enveloping critical (governing) locations were determined at which leak-before-break crack stability evaluations were made. Through-wall flaw sizes were postulated which would cause a leak at a rate of ten (10) times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, fatigue crack growth was shown not to be an issue for the pressurizer surge line piping, RHR line piping, and accumulator line piping.

Therefore, the LBB conditions and margins are satisfied for the RNP pressurizer surge line piping, RHR line piping, and accumulator line piping. It is demonstrated that the dynamic effects of the pipe rupture resulting from postulated breaks in the pressurizer surge piping, the RHR piping, and the accumulator piping need not be considered in the structural design basis of RNP for the license renewal period (60-year plant life).

4.0 REGULATORY EVALUATION

The proposed change would revise RNP licensing and design bases to expand the LBB scope to eliminate the dynamic effects of postulated ruptures of specific portions of piping for the pressurizer surge line, RHR lines, and accumulator lines. The following regulatory requirements have been reviewed and a No Significant Hazards Consideration Determination has been performed as discussed below.

4.1. Applicable Regulatory Requirements/Criteria

Current General Design Criteria (GDC), Criterion 4 – "Environmental and dynamic effects design bases" states 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, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. 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 fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

During the initial plant licensing of RNP, it was demonstrated that the design of the reactor coolant pressure boundary met the regulatory requirements in place at that time. The GDC included in Appendix A to 10 CFR Part 50 did not become effective until May 21, 1971. The construction permit for RNP was issued prior to May 21, 1971; consequently, RNP is not subject to current GDC requirements (SECY-92-223, dated September 18, 1992, ADAMS Accession Number ML18100B279). RNP's conformance with the GDC in existence at the time RNP was licensed (contained in Proposed Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," published in the Federal Register on July 11, 1967) is described in UFSAR Section 3.1.2. As defined in UFSAR Section 3.6.2, high energy piping systems are those whose service temperature exceeds 200°F and whose design pressure exceeds 275 psig. This LAR is based on evaluations to demonstrate that the piping in the scope of the request has an extremely low probability of rupture, consistent with the contemporary version of GDC 4.

The following information demonstrates compliance with GDC 14, 30, and 31 of 10 CFR 50, Appendix A. GDC 14, 30, and 31 of 10 CFR 50, Appendix A states:

> "Criterion 14 - Reactor coolant pressure boundary". The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
>
> "Criterion 30 - Quality of reactor coolant pressure boundary". Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

> “Criterion 31 - Fracture prevention of reactor coolant pressure boundary”. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

NRC GDC 14 and 30 are similar to Atomic Energy Commission (AEC) Criterion 9 (RNP UFSAR Section 3.1.2.9), “Reactor Coolant Pressure Boundary”, and AEC Criterion 16, “Monitoring Reactor Coolant Leakage”, (RNP UFSAR Section 3.1.2.16). NRC GDC 31 is similar to AEC Criterion 34, “Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention”, (RNP UFSAR Section 3.1.2.34).

The piping in the scope of this LAR is designated Class 1 (Class 1 portion is defined herein as piping that is within the American Society of Mechanical Engineers Section XI Subsection, IWB inspection boundary) reactor coolant pressure boundary piping whose materials, design and as-built configuration, analysis, fabrication, and testing preclude the possibility of gross rupture or significant leakage, as supported by the enclosed LBB evaluations based on as-built configuration, material properties, and design transients. This LAR also addresses the capability to detect and respond to piping system leakage prior to a potential flaw reaching a critical size. Therefore, the request is consistent with GDC 14, 30, and 31.

The reactor coolant pressure boundary is designed, fabricated, and constructed to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. Reactor coolant pressure boundary piping and components have provisions for inspection, testing and surveillance of critical areas by appropriate means to assess the structural and leak tight integrity of the boundary components during their service lifetime. The TS reactor coolant system leakage limits ensure the reactor coolant pressure boundary will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions.

NRC Standard Review Plan (SRP) Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Revision 1, provides guidance for review of the LBB application, including guidance for determining an acceptable leakage crack and the reactor coolant system leakage detection sensitivity based on the fracture mechanics analysis. The guidance states that determination of leakage from a crack in a piping system under pressure involves uncertainties and, therefore, margins are needed. Section III.4 of SRP 3.6.3 states 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. 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 Regulatory Guide (RG) 1.45, Revision 0, "Reactor Coolant Pressure Boundary Leakage Detection Systems." The RNP reactor coolant system pressure boundary leak detection system, consistent with the conditions of Generic Letter 84-04, meets the intent of Regulatory Guide 1.45 and meets a leak detection capability of 1 gpm. The conditions provided in Generic Letter 84-04 were implemented when RNP incorporated WCAP-9558 and WCAP-9787 (References 1 and 2, respectively) into the CLB for LBB of main coolant piping. Generic Letter 84-04 provides an exception to the guidance in Regulatory Guide 1.45, which requires detection of 1 gpm leakage within 1 hour. The capability of the RNP leak detection systems to detect 1 gpm leakage is within 4 hours, consistent with the conditions of Generic Letter 84-04.

Licensees are required 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 screening criteria of SRP, Section 3.6.3, by demonstrating that it experiences no active degradation. The candidate pipe should be demonstrated by the fracture mechanics analysis to satisfy the safety margins in SRP, Section 3.6.3. Finally, the licensee must demonstrate that the reactor coolant system leakage detection systems have the capability to detect a certain leak rate, with margins, when compared to the leak rate from the leakage flaw size of the candidate pipe.

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. When the proposed LBB LAR is approved by the NRC, the licensee is required to amend its final safety analysis report to document that the LBB methodology has become a part of the licensing basis for the candidate piping.

The requirements related to the content of the TS are contained in 10 CFR 50.36, which requires that the TS include LCOs. The criteria defined by 10 CFR 50.36(c)(2)(ii) relevant to determining whether capabilities related to reactor coolant pressure boundary (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.

The lowest flow rate calculated for the LBB leakage flaws is 10 gpm as stated in WCAP-17776, WCAP-17778 and WCAP-17779. The proposed change maintains the minimum required unidentified leakage detection capability of 1 gpm after applying the margin factor of 10, in accordance with SRP 3.6.3 criteria. The 1 gpm limit assures timely identification of reactor coolant pressure boundary degradation, and the measurement capability is sufficient to ensure reactor coolant system leakage can be detected well in advance of a through wall flaw propagating to a pipe rupture. The adequacy of the current TS is supported by the margins used in the LBB evaluations, i.e., a margin factor of 10 between leakage crack flow rate and leakage detection capability, and a factor of two between leakage crack size and critical crack size. These margins offset uncertainties associated with leakage detection and prediction.

In summary, the request is consistent with the GDC 4 provision that dynamic effects associated with postulated pipe ruptures may be removed from the design basis if NRC-approved analyses demonstrate an extremely low probability of pipe rupture occurring under design basis conditions. The proposed change maintains consistency with GDC 14 and GDC 30 criteria for maintaining the integrity of the reactor coolant pressure boundary and being able to detect reactor coolant system leakage. The existing TS for leakage detection and leakage limits are consistent with 10 CFR 50.36 and do not require revision to support this request.

4.2. Precedents

Several other licensees have requested and received approval to use the LBB methodology to eliminate the dynamic effects of pipe rupture for auxiliary piping systems attached to the reactor coolant system main piping including the following:

- Letter from US NRC to Entergy, "Waterford Steam Electric Station, Unit 3 – "Issuance of Amendment Re: Approval of Leak-Before-Break of the Pressurizer Surge Line", dated February 28, 2011 (ADAMS Accession Number ML110410119)
- Letter from US NRC to Northern States Power Company, "Prairie Island Nuclear Generating Plant, Units 1 and 2 - Issuance of Amendments Re: Request to Exclude the Dynamic Effects Associated with Certain Postulated Pipe Ruptures from the Licensing Basis based upon Application of Leak-Before-Break Methodology", dated October 27, 2011 (ADAMS Accession Number ML112200856)
- Letter from US NRC to Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Unit No. 1 – Issuance of Amendment Number 346 Re: Approval of Application of Proprietary Leak-Before-Break Methodology for Reactor Coolant System Small Diameter Piping", dated August 1, 2019 (ADAMS Accession Number ML19170A362)
- Letter from US NRC to PSE&G LLC "Salem Nuclear Generating Station, Unit Nos. 1 and 2 – Issuance of Amendment Numbers 336 and 317 Re: Leak-Before-Break for Accumulator, Residual Heat Removal, Safety Injection, and Pressurizer Surge Lines," dated February 23, 2021 (ADAMS Accession No. ML20338A038)
- Letter from US NRC to Dominion Energy Virginia "Surry Power Station, Units 1 and 2 - Issuance of Amendment Numbers 304 and 304 Re: Leak-Before-Break for Pressurizer Surge, Residual Heat Removal, Safety Injection Accumulator, Reactor Coolant System Bypass and Safety Injection Lines," dated August 20, 2021 (ADAMS Accession Number ML21175A185)

4.3 No Significant Hazards Consideration

Duke Energy Progress, LLC (Duke Energy) requests an amendment to the H. B. Robinson Steam Electric Plant, Unit No. 2 (RNP) Facility Operating License. The proposed amendment would change the RNP design and licensing basis as described in the RNP Updated Final Safety Analysis Report (UFSAR) to eliminate the dynamic effects of postulated pipe ruptures in specific portions of systems attached to the Reactor Coolant System (RCS) in accordance with 10 CFR 50, Appendix A, General Design Criterion 4, "Environmental and dynamic effects design bases." This License Amendment Request (LAR) uses Leak-Before-Break (LBB) methodology to demonstrate the risk of pipe rupture is extremely low for portions of the following systems piping connected to the RCS loop piping:

Duke Energy has evaluated the proposed changes using the criteria in 10 CFR 50.92 and determined that the proposed changes do not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change requests plant-specific approval of a previously approved Leak-Before-Break (LBB) evaluation methodology, in accordance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 4. The LBB evaluations demonstrate that the probability of a rupture of the piping in the scope of the request is extremely low under design basis conditions, such that the dynamic effects of postulated pipe ruptures may be removed from the design basis of RNP. The proposed change does not adversely affect accident initiators or precursors. Overall protection system performance will remain within the bounds of the previously performed accident analyses. The design of the protection systems will be unaffected. The Reactor Protection System (RPS) and Emergency Core Cooling System (ECCS) will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request will remain applicable.

There will be no change to normal plant operating parameters or accident mitigation performance. The proposed amendment will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the RNP Updated Final Safety Analysis Report (UFSAR).

Therefore, these proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change requests NRC approval of LBB methodology to demonstrate an extremely low probability of pipe rupture. It does not introduce any new accident scenarios, failure mechanisms, or single failures. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function. The proposed change has no adverse effects on any safety related systems or components and does not challenge the performance or integrity of any safety related system. Further, there are no changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of power operation or change any operating parameters.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   The proposed change does not adversely affect the ability of the fuel cladding, reactor coolant pressure boundary, or containment to perform their design basis functions as fission product barriers. The proposed change uses previously accepted analytical methods to demonstrate that the probability of a fluid system rupture is extremely low. It has no effect on the manner in which safety limits or limiting safety system settings are determined and it does not adversely affect any plant systems necessary to assure the accomplishment of protection functions.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the above, Duke Energy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusion

Based on the considerations discussed herein, Duke Energy concludes 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9) as follows:

(i) The proposed change involves no significant hazards consideration.

As described in Section 4.3 above, the proposed change involves no significant hazards consideration.

(ii) There are no significant changes in the types or significant increase in the amounts of any effluents that may be released off-site.

The proposed change would change the RNP design and licensing basis as described in the RNP UFSAR to eliminate the dynamic effects of postulated pipe ruptures in specific portions of systems attached to the RCS. The proposed change does not alter the design function or operation of any plant structure, system, or component. The reactor coolant pressure boundary will continue to meet its specific structural and leakage integrity performance criteria. The proposed change does not involve the installation of any new equipment or the modification of any equipment that may affect the types or amounts of effluents that may be released off-site. The proposed change will have no impact on normal plant releases and will not increase the predicted radiological consequences of accidents postulated in the UFSAR. Therefore, there are no significant changes in the types or significant increase in the amounts of any effluents that may be released off-site.

ii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change would change the RNP design and licensing basis as described in the RNP UFSAR to eliminate the dynamic effects of postulated pipe ruptures in specific portions of systems attached to the RCS. The proposed change does not implement plant physical changes or result in plant operation in a configuration outside the plant safety analyses or design basis. Furthermore, reactor coolant pressure boundary will continue to meet specific structural and leakage integrity performance criteria. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure associated with the proposed change.

Based on the above, Duke Energy concludes that, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. WCAP-9558, "Mechanistic Fracture Evaluation of Reactor Coolant Piping Containing a Postulated Circumferential Through-Wall Crack," dated May 1981.
2. WCAP-9787, "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation," dated May 1981.
3. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Report Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," dated February 1, 1984.

ATTACHMENT 2

WCAP-17776-NP, REVISION 1, "TECHNICAL JUSTIFICATION FOR ELIMINATING PRESSURIZER SURGE LINE RUPTURE AS THE STRUCTURAL DESIGN BASIS FOR H. B. ROBINSON UNIT 2”, MARCH 2023

(REDACTED)

WCAP-17776-NP
Revision 1

March 2023

Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2

WCAP-17776-NP
Revision 1

Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2

March 2023

Author: Nadia B. Petkova *
Operating Plants Piping and Supports

Reviewer: Momo Wiratmo*
Operating Plants Piping and Supports

Approved: Lynn A. Patterson, Manager*
Reactor Vessel and Containment
Vessel Design and Analysis

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

RECORD OF REVISIONS

Revision	Date	Revision Description
0	August 2013	Original Issue (WCAP-17776-NP). This is the non-proprietary class 3 version of WCAP-17776-P, Revision 0.
1	March 2023	This is the non-proprietary class 3 version of WCAP-17776-P, Revision 1.

TABLE OF CONTENTS

1.0 Introduction 1-1
1.1 Background 1-1
1.2 Scope and Objectives 1-1
1.3 References 1-2

2.0 Operation and Stability of the Reactor Coolant System 2-1
2.1 Stress Corrosion Cracking 2-1
2.2 Water Hammer 2-2
2.3 Low Cycle and High Cycle Fatigue 2-2
2.4 Summary Evaluation of Surge Line for Potential Degradation During Service 2-3
2.5 References 2-4

3.0 Pipe Geometry and Loading 3-1
3.1 Calculation of Loads and Stresses 3-1
3.2 Loads for Leak Rate Evaluation 3-1
3.3 Load Combination for Crack Stability Analyses 3-2
3.4 Loading Conditions 3-2
3.5 Summary of Loads 3-4
3.6 Pipe Geometry 3-4
3.7 References 3-4

4.0 Material Characterization 4-1
4.1 Pressurizer Surge Line Piping, Fittings, and Weld Materials 4-1
4.2 Tensile Properties 4-1
4.3 References 4-1

5.0 Critical Location and Evaluation Criteria 5-1
5.1 Critical Locations 5-1

6.0 Leak Rate Predictions 6-1
6.1 Introduction 6-1
6.2 General Considerations 6-1
6.3 Calculation Method 6-1
6.4 Leak Rate Calculations 6-2
6.5 References 6-2

7.0 Fracture Mechanics Evaluation 7-1
7.1 Global Failure Mechanism 7-1
7.2 Results of Crack Stability Evaluation 7-2
7.3 References 7-2

8.0 Assessment of Fatigue Crack Growth 8-1
8.1 Introduction 8-1
8.2 Results 8-2

8.3 References.. 8-3

9.0 Assessment of Margins 9-1

10.0 Conclusions.. 10-1

Appendix A: Limit Moment.. A-1

LIST OF TABLES

Table 3-1 Types of Loadings....3-5

Table 3-2 Normal and Faulted Loading Cases for Leak-Before-Break Evaluations 3-5

Table 3-3 Associated Load Cases for Analyses 3-6

Table 3-4 Summary of Leak-Before-Break Loads and Stresses at the Three Governing Stressed Weld Locations (Material A 376 TP316, Weld Process GTAW/SMAW Combination)...3-7

Table 4-1 Measure Tensile Properties for Pressurizer Surge Line Material A376 TP316....4-2

Table 4-2 Mechanical Properties for the Pressurizer Surge Line Material at Operating Temperatures 4-2

Table 6-1 Leakage Flaw Sizes 6-3

Table 7-1 Summary of Critical Flaw Sizes for the Pressurizer Surge Line 7-3

Table 8-1 Pressurizer Surge Line Fatigue Crack Growth Results....8-4

Table 9-1 Leakage Flaw Sizes, Critical Flaw Sizes, and Margins for the Pressurizer Surge Line ..9-2

LIST OF FIGURES

Figure 3-1 H. B. Robinson Unit 2 Pressurizer Surge Line Layout 3-9

Figure 6-1 Analytical Predictions of Critical Flow Rates of Steam-Water Mixtures 6-4

Figure 6-2 [ ]a,c,e Pressure Ratio as a Function of L/D 6-5

Figure 6-3 Idealized Pressure Drop Profile Through a Postulated Crack 6-6

Figure 7-1 [ ]a,c,e Stress Distribution 7-4

Figure 7-2 Critical Flaw Size Prediction – Node 130 Case D 7-5

Figure 7-3 Critical Flaw Size Prediction – Node 130 Case E 7-6

Figure 7-4 Critical Flaw Size Prediction – Node 130 Case F 7-7

Figure 7-5 Critical Flaw Size Prediction – Node 380 Case D 7-8

Figure 7-6 Critical Flaw Size Prediction – Node 380 Case E 7-9

Figure 7-7 Critical Flaw Size Prediction – Node 380 Case F 7-10

Figure 7-8 Critical Flaw Size Prediction – Node 600 Case D 7-11

Figure 7-9 Critical Flaw Size Prediction – Node 600 Case E 7-12

Figure 7-10 Critical Flaw Size Prediction – Node 600 Case F 7-13

Figure 8-1 Orientation of Stress Cuts for the Fatigue Crack Growth Analysis 8-5

Figure 8-2 Reference Crack Growth Curves for Stainless Steel in Air Environments 8-6

Figure A-1 Pipe with a Through-Wall Crack in Bending A-2

1.0 INTRODUCTION

1.1 BACKGROUND

The current structural design basis for the pressurizer surge line requires postulating non-mechanistic circumferential and longitudinal pipe breaks. This results in additional plant hardware (e.g. pipe whip restraints and jet shields) that would mitigate the dynamic consequences of the pipe breaks. It is therefore highly desirable to be realistic in the postulation of pipe breaks for the surge line. Presented in this report are the descriptions of a mechanistic pipe break evaluation method and the analytical results that can be used for establishing that a circumferential type break will not occur within the pressurizer surge line. The evaluations consider that circumferentially oriented flaws cover longitudinal cases. The pressurizer surge line is known to be subjected to thermal stratification and the effects of thermal stratification for the H. B. Robinson presurizer surge line have been evaluated and documented in WCAP-12962 (Reference 1-1). The results of the stratification evaluation as described in WCAP-12962 have been used in the Leak-Before-Break evaluation presented in this report.

1.2 SCOPE AND OBJECTIVES

The purpose of this investigation is to demonstrate Leak-Before-Break (LBB) for the H. B. Robinson Unit 2 pressurizer surge line. The scope of this work covers the entire pressurizer surge line from the primary loop nozzle junction to the pressurizer nozzle junction. A schematic drawing of the piping system is shown in Section 3.0. The recommendations and criteria proposed in SRP 3.6.3 (References 1-2 and 1-3) are used in this evaluation. The criteria and the resulting steps of the evaluation procedure can be briefly summarized as follows:

1. Calculate the applied loads. Identify the location at which the highest faulted stress occurs.

2. Identify the materials and the material properties.

3. Postulate a through-wall flaw at the governing location. The size of the flaw should be large enough so that the leakage is assured of detection with margin using the installed leak detection equipment when the pipe is subjected to normal operating loads. Demonstrate that there is a margin of 10 between the calculated leak rate and the leak detection capability.

4. Using maximum faulted loads in the stability analysis, demonstrate that there is a margin of 2 between the leakage size flaw and the critical size flaw.

5. Review the operating history to ascertain that operating experience has indicated no particular susceptibility to failure from the effects of corrosion, water hammer or low and high cycle fatigue.

6. For the materials types used in the plant, provide representative material properties.

7. Demonstrate margin on applied load.

8. Perform an assessment of fatigue crack growth. Show that a through-wall crack will not result.

This report provides a fracture mechanics analysis that demonstrates the pressurizer surge line integrity for H. B. Robinson Unit 2 consistent with the NRC's position for exemption from consideration of postulated pipe rupture dynamic effects (Reference 1-4).

It should be noted that the terms "flaw" and "crack" have the same meaning and are used interchangeably. "Governing location" and "critical location" are also used interchangeably throughout the report.

Note that there are several locations in this report where proprietary information has been identified and bracketed. For each of the bracketed locations, the reason for the proprietary classification is given using a standardized system. The proprietary brackets are labeled with three different letters, to provide this information, and the explanation for each letter is given below:

a. The information reveals the distinguishing aspects of a process or component, structure, tool, method, etc., and the prevention of its use by Westinghouse's competitors, without license from Westinghouse, gives Westinghouse a competitive economic advantage.

b. The information, if used by a competitor, would reduce the competitor's expenditure of resources or improve the competitor's advantage in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. The information reveals aspects of past, present, or future Westinghouse or customer-funded development plans and programs of potential commercial value to Westinghouse.

The proprietary information in the brackets which has been deleted in this version of this report are provided in the proprietary class 2 document (WCAP-17776-P, Revision 1).

1.3 REFERENCES

1-1 WCAP-12962, Revision 0, "Structural Evaluation of the H. B. Robinson Unit 2 and Shearon Harris Pressurizer Surge lines, Considering the Effects of Thermal Stratification," September 1991 including WCAP-12962 Supplement 1, Revision 0, October 1995 (Westinghouse Proprietary).

1-2 Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday August 28, 1987/Notices, pp. 32626-32633.

1-3 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

1-4 Nuclear Regulatory Commission, 10 CFR 50, Modification of General Design Criteria 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, Final Rule, Federal Register/Vol. 52, No. 207/Tuesday, October 27, 1987/Rules and Regulations, pp. 41288 41295.

2.0 OPERATION AND STABILITY OF THE REACTOR COOLANT SYSTEM

2.1 STRESS CORROSION CRACKING

The Westinghouse reactor coolant system pressurizer surge lines have an operating history that demonstrates the inherent operating stability characteristics of the design. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking (IGSCC)). This operating history totals over 1400 reactor-years, including 16 plants each having over 30 years of operation, 10 other plants each with over 25 years of operation, 11 plants each with over 20 years of operation and 12 plants each with over 15 years of operation.

In 1978, the United States Nuclear Regulatory Commission (USNRC) formed the second Pipe Crack Study Group. (The first Pipe Crack Study Group (PCSG) established in 1975 addressed cracking in boiling water reactors only.) One of the objectives of the second PCSG was to include a review of the potential for stress corrosion cracking in Pressurized Water Reactors (PWRs). The results of the study performed by the PCSG were presented in NUREG-0531 (Reference 2-1) entitled "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants." In that report the PCSG stated:

> "The PCSG has determined that the potential for stress-corrosion cracking in PWR primary system piping is extremely low because the ingredients that produce IGSCC are not all present. The use of hydrazine additives and a hydrogen overpressure limit the oxygen in the coolant to very low levels. Other impurities that might cause stress-corrosion cracking, such as halides or caustic, are also rigidly controlled. Only for brief periods during reactor shutdown when the coolant is exposed to the air and during the subsequent startup are conditions even marginally capable of producing stress-corrosion cracking in the primary systems of PWRs. Operating experience in PWRs supports this determination. To date, no stress corrosion cracking has been reported in the primary piping or safe ends of any PWR."

For stress corrosion cracking (SCC) to occur in piping, the following three conditions must exist simultaneously: high tensile stresses, susceptible material, and a corrosive environment. Since some residual stresses and some degree of material susceptibility exist in any stainless steel piping, the potential for stress corrosion is minimized by properly selecting a material immune to SCC as well as preventing the occurrence of a corrosive environment. The material specifications consider compatibility with the system's operating environment (both internal and external) as well as 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 (e.g., sulfides, sulfites, and thionates). Strict pipe cleaning standards prior to operation and careful control of water chemistry during plant operation are used to prevent the occurrence of a corrosive environment. Prior to being put into service, the piping is cleaned internally and externally. During flushes and preoperational testing, water chemistry is controlled in accordance with written specifications.

Requirements on chlorides, fluorides, conductivity, and pH are included in the acceptance criteria for the piping.

During plant operation, the reactor coolant water chemistry is monitored and maintained within very specific limits. Contaminant concentrations are kept below the thresholds known to be conducive to stress corrosion cracking with the major water chemistry control standards being included in the plant operating procedures as a condition for plant operation. For example, during normal power operation, oxygen concentration in the RCS is expected to be in the ppb range by controlling charging flow chemistry and maintaining hydrogen in the reactor coolant at specified concentrations. Halogen concentrations are also stringently controlled by maintaining concentrations of chlorides and fluorides within the specified limits. Thus, during plant operation, the likelihood of stress corrosion cracking is minimized.

During 1979, several instances of cracking in PWR feedwater piping led to the establishment of the third PCSG. The investigations of the PCSG reported in NUREG-0691 (Reference 2-2) further confirmed that no occurrences of IGSCC have been reported for PWR primary coolant systems.

Primary Water Stress Corrosion Cracking (PWSCC) occurred in V. C. Summer reactor vessel hot leg nozzle, Alloy 82/182 weld. It should be noted that this susceptible material is not found at the H. B. Robinson Unit 2 pressurizer surge line.

2.2 WATER HAMMER

Overall, there is a low potential for water hammer in the RCS and the connecting surge line since they are designed and operated to preclude the voiding condition in the normally filled surge line. The RCS and connecting surge line including piping and components, are designed for normal, upset, emergency, and faulted condition transients. The design requirements are conservative relative to both the number of transients and their severity. Pressurizer safety and relief valve actuation and the associated hydraulic transients following valve opening are considered in the system design. Only relatively slow transients are applicable to the surge line and there is no significant effect on the system dynamic loads. To ensure dynamic system stability, reactor coolant parameters are stringently controlled. Temperature during normal operation is maintained within a narrow range by the control rod positions. Pressure is also controlled within a narrow range for steady-state conditions by the pressurizer heaters and the pressurizer spray. The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters, namely system resistance and the reactor coolant pump characteristics, are controlled in the design process. Additionally, Westinghouse has instrumented typical reactor coolant systems to verify the flow and vibration characteristics of the system and the connecting auxiliary lines. Preoperational testing and operating experience have verified the Westinghouse approach. The operating transients of the RCS primary piping and the connected surge line are such that no significant water hammer can occur.

2.3 LOW CYCLE AND HIGH CYCLE FATIGUE

Fatigue considerations are accounted for in the surge line piping through the fatigue usage factor evaluation for the stratification analyses (Reference 2-3) to show compliance with the rules of Section III of the ASME Code. A further assessment of the low cycle fatigue loading is discussed in Section 8.0 as part of this study in the form of a fatigue crack growth assessment.

Pump vibrations during operation would result in high cycle fatigue loads in the piping system. During operation, an alarm signals the exceeding of the RC pump vibration limits. Field measurements have been made on the reactor coolant loop piping in a number of plants during hot functional testing. Stresses in the elbow below the RC pump have been found to be very small, between 2 and 3 ksi at the highest. Field measurements on a typical PWR plant indicate vibration amplitudes less than 1 ksi. When translated to the connecting surge line, these stresses would be even lower, well below the fatigue endurance limit for the surge line material and would result in an applied stress intensity factor below the threshold for fatigue crack growth. H. B. Robinson Unit 2 configurations are similar and the results are expected to be the similar.

2.4 SUMMARY EVALUATION OF SURGE LINE FOR POTENTIAL DEGRADATION DURING SERVICE

There has never been any service cracking or wall thinning identified in the pressurizer surge line of Westinghouse PWR design. The design, construction, inspection, and operation of the pressurizer surge line piping mitigate sources of such degradation.

There is no known mechanism for water hammer in the pressurizer/surge system. The pressurizer safety and relief piping system that is connected to the top of the pressurizer could have loading from water hammer events. However, these loads are effectively mitigated by the pressurizer and have a negligible effect on the surge line.

Wall thinning by erosion and erosion-corrosion effects should not occur in the surge line due to the low velocity, typically less than 1.0 ft/sec and the material, austenitic stainless steel, which is highly resistant to these degradation mechanisms. Per NUREG-0691 (Reference 2-1), a study on pipe cracking in PWR piping reported only two incidents of wall thinning in stainless steel pipe and these were not in the surge line. The cause of wall thinning is related to the high water velocity and is therefore clearly not a mechanism that would affect the surge line.

It is well known that the pressurizer surge line is subjected to thermal stratification and the effects of stratification are particularly significant during certain modes of heatup and cooldown operation. The effects of stratification have been evaluated for the H. B. Robinson Unit 2 surge line and the loads, accounting for the stratification effects, have been derived in WCAP-12962 (Reference 2-3). These loads are used in the Leak-Before-Break evaluation described in this report.

The H. B. Robinson Unit 2 surge line piping system is fabricated from forged products (see Section 4) which are not susceptible to toughness degradation due to thermal aging.

Finally, the maximum operating temperature of the pressurizer surge line piping, which is about 650°F, is well below the temperature that would cause any creep damage in stainless steel piping. Cleavage type failures are not a concern for the operating temperatures and the material used in the stainless steel piping of the pressurizer surge line.

2.5 REFERENCES

2-1 Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants, NUREG-0531, U.S. Nuclear Regulatory Commission, February 1979.

2-2 Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors, NUREG-0691, U.S. Nuclear Regulatory Commission, September 1980.

2-3 WCAP-12962, Revision 0, "Structural Evaluation of the H. B. Robinson Unit 2 and Shearon Harris Pressurizer Surge lines, Considering the Effects of Thermal Stratification," September 1991 including WCAP-12962 Supplement 1, Revision 0, October 1995 (Westinghouse Proprietary).

3.0 PIPE GEOMETRY AND LOADING

3.1 CALCULATION OF LOADS AND STRESSES

The stresses due to axial loads and bending moments are calculated by the following equation:

$$\sigma = \frac{F}{A} + \frac{M}{Z} \tag{3-1}$$

where,

σ = stress

F = axial load

M = moment

A = pipe cross-sectional area

Z = section modulus

The moments for the desired loading combinations are calculated by the following equation:

$$M = \sqrt{M_x^2 + M_y^2 + M_z^2} \tag{3-2}$$

where,

M_x = X component of moment, Torsion

M_y = Y component of bending moment

M_z = Z component of bending moment

The axial load and moments for leak rate predictions and crack stability analyses are computed by the methods to be explained in Sections 3.2 and 3.3.

3.2 LOADS FOR LEAK RATE EVALUATION

The normal operating loads for leak rate predictions are calculated by the following equations:

$$F = F_{DW} + F_{TH} + F_P \tag{3-3}$$

$$M_X = (M_X)_{DW} + (M_X)_{TH} \tag{3-4}$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{TH} \tag{3-5}$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{TH} \tag{3-6}$$

The subscripts of the above equations represent the following loading cases:

DW = deadweight

TH = normal thermal expansion or thermal stratification

P = load due to internal pressure

This method of combining loads is often referred to as the algebraic sum method (References 3-1 and 3-2).

The loads based on this method of combination are provided in Table 3-4 at the weld locations identified in Figure 3-1.

3.3 LOAD COMBINATION FOR CRACK STABILITY ANALYSES

In accordance with Standard Review Plan 3.6.3 (References 3-1 and 3-2), the absolute sum of loading components can be applied which results in higher magnitude of combined loads. If crack stability is demonstrated using these loads, the LBB margin on loads can be reduced from √2 to 1.0. The absolute summation of loads is shown in the following equations:

$$F = |F_{DW}| + |F_{TH}| + |F_{P}| + |F_{SSEINERTIA}| + |F_{SSEAM}| \quad (3\text{-}7)$$

$$M_X = |(M_X)_{DW}| + |(M_X)_{TH}| + |(M_X)_{SSEINERTIA}| + |(M_X)_{SSESAM}| \quad (3\text{-}8)$$

$$M_Y = |(M_Y)_{DW}| + |(M_Y)_{TH}| + |(M_Y)_{SSEINERTIA}| + |(M_Y)_{SSEAM}| \quad (3\text{-}9)$$

$$M_Z = |(M_Z)_{DW}| + |(M_Z)_{TH}| + |(M_Z)_{SSEINERTIA}| + |(M_Z)_{SSEAM}| \quad (3\text{-}10)$$

where subscript SSEINERTIA refers to safe shutdown earthquake inertia and SSEAM is safe shutdown earthquake anchor motion.

The loads so determined are used in the fracture mechanics evaluations (Section 7.0) to demonstrate the LBB margins at the locations established to be the governing locations. These loads at the three governing the weld locations (see Figure 3-1) are given in Table 3-4.

3.4 LOADING CONDITIONS

Because thermal stratification can cause large stresses during heatup and cooldown, a review of the stratification stresses was performed to identify the upper bound loadings. The identified types of loading are given in Table 3-1.

Seven loading cases were identified and are shown in Table 3-2. Cases A, B and C are the normal operating load cases and Cases D, E, F and G are the faulted load cases.

The cases postulated for Leak-Before-Break evaluation are summarized in Table 3-3. The cases of primary interest are the postulation of a detectable leak at normal 100% power [

]^{a,c,e}

Case Combination [

]^{a,c,e}

The case combination [

]^{a,c,e}

The realistic cases A/D or B/E or [

]^{a,c,e}

[

]^{a,c,e}

The logic for this system ΔT of [ ]^{a,c,e} is based on the following:

Actual practice, based on experience from other plants with this type of situation, indicates that the plant operators complete the cool down as quickly as possible once a leak in the primary system is detected. Technical Specifications may require cold shutdown within 36 hours, but actual practice is that the plant operators depressurize the system as soon as possible once a primary system leak is detected. Therefore, the hot leg is generally on the warmer side of the limits (~200°F) when the pressurizer bubble is quenched. Once the bubble is quenched, the pressurizer is cooled down fairly quickly reducing the ΔT in the system.

3.5 SUMMARY OF LOADS

The combined loads were evaluated at the various weld locations. Normal loads were determined using the algebraic sum method whereas faulted loads were combined using the absolute sum method. Table 3-4 shows loads and stresses at the three governing stressed weld locations for A376 TP316 material with GTAW/SMAW combination. Loads and stresses for Case C and Case G in Table 3-4 are shown for information only and they are not used in the LBB analysis.

3.6 PIPE GEOMETRY

The H. B. Robinson pressurizer surge line is 12-inch schedule 140; pipe outer diameter is 12.75 inch and a minimum pipe wall thickness, based on the maximum allowed counterbore at a butt weld (Reference 3-3), at the weld counterbore of 1.005 inches was used in the analysis.

3.7 REFERENCES

3-1 Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

3-2 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

3-3 American National Standards, Butt Welding Ends, ANSI B16.25-1979.

Table 3-1 Types of Loadings
Pressure (P)
Dead Weight (DW)
Normal Operating Thermal Expansion (TH)
Safe Shutdown Earthquake including Seismic Anchor Motion (SSE)
[ ]a,c,e
[ ]a,c,e
[ ]a,c,e

Table 3-2 Normal and Faulted Loading Cases for Leak-Before-Break Evaluations	
CASE A	This is the normal operating case at 653°F consisting of the algebraic sum of the loading components due to P, DW and TH.
CASE B	[ ]a,c,e
CASE C[1]	[ ]a,c,e
CASE D	This is the faulted operating case at 653°F consisting of the absolute sum (every component load is taken as positive) of P, DW, TH and SSE.
CASE E	[ ]a,c,e
CASE F	This is a forced cooldown case [ ]a,c,e with stratification [ ]a,c,e
CASE G[1]	[ ]a,c,e

[1] Case C and Case G are shown for information only.

Table 3-3 Associated Load Cases for Analyses	
A/D	This is the standard Leak-Before-Break evaluation.
A/F	This depicts a postulated forced cooldown event resulting from experiencing a detectable leak [ ] [a,c,e]
B/E	[ ] [a,c,e]
B/F	This depicts a postulated forced cooldown event resulting from experiencing a detectable leak [ ] [a,c,e]

Table 3-4 Summary of Leak-Before-Break Loads and Stresses at the Three Governing Stressed Weld Locations (Material A 376 TP316, Weld Process GTAW/SMAW Combination)

Node[a]	Case	Axial Force[b] (kips)	Axial Stress (ksi)	Moment (in-kips)	Moment Stress (ksi)	Total stress (ksi)
130	A	210.562	5.678	436.785	4.32	10.00
130	B	210.562	5.678	405.433	4.01	9.69
130	C[c]	44.846	1.209	3285.262	32.52	33.73
130	D	214.142	5.775	865.628	8.57	14.34
130	E	214.142	5.775	1046.293	10.36	16.13
130	F	45.076	1.216	2834.232	28.05	29.27
130	G[c]	48.426	1.306	3893.478	38.54	39.84
380	A	210.562	5.678	751.769	7.44	13.12
380	B	210.562	5.678	1042.106	10.31	15.99
380	C[c]	44.846	1.209	2411.412	23.87	25.08
380	D	212.011	5.717	805.097	7.97	13.69
380	E	212.011	5.717	1111.973	11.01	16.72
380	F	45.076	1.216	1920.339	19.01	20.22
380	G[c]	46.295	1.248	2522.435	24.97	26.21
600	A	203.837	5.497	1182.233	11.70	17.20
600	B	203.737	5.494	1083.418	10.72	16.22
600	C[c]	39.521	1.066	1424.814	14.10	15.17

Table 3-4 Summary of Leak-Before-Break Loads and Stresses at the Three Governing Stressed Weld Locations (Material A 376 TP316, Weld Process GTAW/SMAW Combination)

Node[a]	Case	Axial Force[b] (kips)	Axial Stress (ksi)	Moment (in-kips)	Moment Stress (ksi)	Total stress (ksi)
600	D	206.959	5.581	1513.316	14.98	20.56
600	E	206.859	5.578	1377.651	13.64	19.21
600	F	40.701	1.098	1158.077	11.46	12.56
600	G[c]	42.643	1.150	1802.016	17.84	18.99

Notes:

a. See Figure 3-1

b. Included Pressure

c. For information only

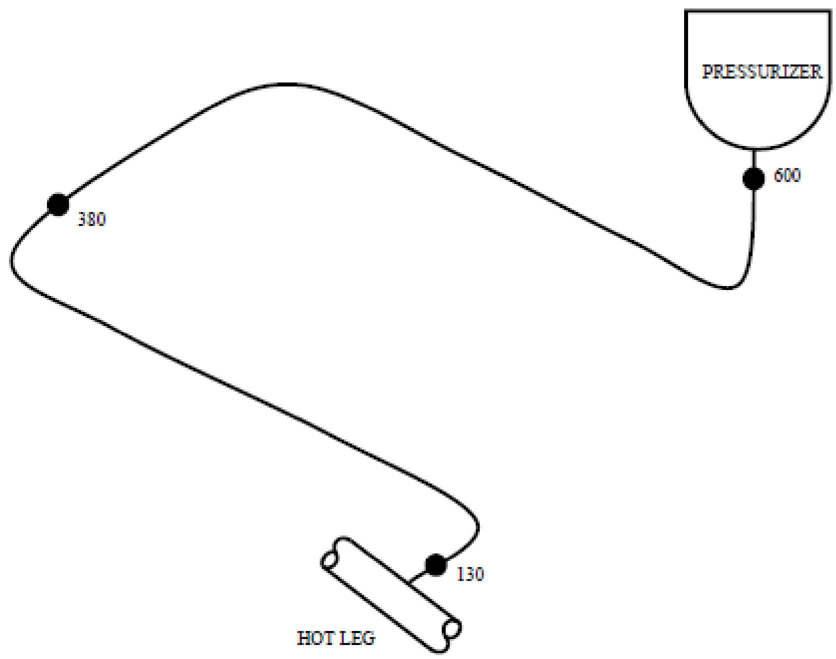


Figure 3-1 H. B. Robinson Unit 2 Pressurizer Surge Line Layout

4.0 MATERIAL CHARACTERIZATION

4.1 PRESSURIZER SURGE LINE PIPING, FITTINGS, AND WELD MATERIALS

The pipe material of the pressurizer surge line for H. B. Robinson Unit 2 is A376 TP316. This is a wrought product of the type used for the piping of several PWR Plants. The surge line is connected to the primary loop at one end and at the other end to the pressurizer nozzle. The surge line does not include any cast pipes or cast fittings. The welding processes used are Gas Tungsten Arc Weld (GTAW)/Shielded Metal Arc Weld (SMAW) combination. Figure 3-1 shows the schematic layout of the surge line and identifies the weld locations by node points.

In the following sections the tensile properties of the materials are presented for use in the Leak-Before-Break analyses.

4.2 TENSILE PROPERTIES

The Certified Materials Test Reports (CMTRs) for the H. B. Robinson Unit 2 pressurizer surge line were used to establish the tensile properties for the leak-before-break analyses. The tensile properties for the pipe material are provided in Table 4-1.

For H. B. Robinson Unit 2, specific data was used as a basis for determining tensile properties. The room temperature mechanical properties of the surge line material were obtained from the Certified Materials Test Reports (CMTRs) and are given in Table 4-1. The representative minimum and average tensile properties were established. The material properties at temperatures (653°F, 605°F, 455°F and 205°F) are required for the leak rate and stability analyses. The minimum and average tensile properties were calculated by using the ratio of ASME Boiler and Pressure Vessel Code Section II of the 2007 Edition with the 2008 Addenda (Reference 4-1) properties at the temperatures of interest stated above. Table 4-2 shows the tensile properties at various temperatures. The moduli of elasticity values were established at various temperatures from the ASME Code Section III (see Table 4-2). In the Leak-Before-Break evaluation, the representative minimum yield strength and minimum ultimate strength at temperature were used for the flaw stability evaluations and the representative average yield strength was used for the leak rate predictions.

The average and lower bound yield strengths and ultimate strengths for the pipe material are tabulated in Table 4-2. The ASME Code modulus of elasticity values are also given, and Poisson's ratio was taken as 0.3.

4.3 REFERENCES

4-1 ASME Boiler and Pressure Vessel Code, 2007 Edition with the 2008 Addenda, Section II, Part D – Properties (Customary) Materials.

Table 4-1 Measure Tensile Properties for Pressurizer Surge Line Material A376 TP316

Heat Number	Yield Strength (psi) Room Temp.	Ultimate Strength (psi) Room Temp.
8935(SER 2161)	39500	81600
8935(SER 2161)	40760	81600
8935(SER 2162)	37700	79400
8935(SER 2162)	40160	88200
8935(SER 2163)	35900	81000
8935(SER 2163)	38500	81000
8935(SER 2164)	41500	82600
8935(SER 2164)	39500	81800

Table 4-2 Mechanical Properties for the Pressurizer Surge Line Material at Operating Temperatures

Material	Temperature (°F)	Average Yield Strength (psi)	Lower Bound	
			Yield Stress (psi)	Ultimate Strength (psi)
A376 TP316	653	24144	22117	76012
A376 TP316	605	24637	22569	76012
A376 TP316	455	26950	24687	76060
A376 TP316	205	33671	30844	79289
Modulus of Elasticity:	E = 25.035 x 10^6 psi at 653°F ; E = 25.275 x 10^6 psi at 605°F; E = 26.125 x 10^6 psi at 455°F; E = 27.475 x 10^6 psi at 205°F			
Poisson's ratio: 0.3				

5.0 CRITICAL LOCATION AND EVALUATION CRITERIA

5.1 CRITICAL LOCATIONS

The leak-before-break (LBB) evaluation margins are to be demonstrated for the critical locations (governing locations). Such locations are established based on the loads (Section 3.0) and the material properties established in Section 4.0. These locations are defined below for the H. B. Robinson Unit 2 pressurizer surge line piping.

<u>Critical Locations</u>

Node 130 (hot leg nozzle to pipe weld location), Node 380 (intermediate elbow weld location) and Node 600 (pressurizer nozzle location, reducer to pipe weld location) are the governing weld locations identified for the LBB analysis. Node 130 is determined to be the critical location at the HL surge nozzle due to the limiting cross-sectional properties of the pipe, rather than the thicker reinforcement area at the nozzle to reactor coolant loop branch weld.

6.0 LEAK RATE PREDICTIONS

6.1 INTRODUCTION

The purpose of this section is to discuss the method which is used to predict the flow through postulated through-wall cracks and present the leak rate calculation results for through-wall circumferential cracks.

6.2 GENERAL CONSIDERATIONS

The flow of hot pressurized water through an opening to a lower back pressure causes flashing which can result in choking. For long channels where the ratio of the channel length, L, to hydraulic diameter, D_H, (L/D_H) is greater than [

]^{a,c,e}

6.3 CALCULATION METHOD

The basic method used in the leak rate calculations is the method developed by [

]^{a,c,e}

The flow rate through a crack was calculated in the following manner. Figure 6-1 from Reference 6-2 was used to estimate the critical pressure, P_c, for the pressurizer surge line enthalpy condition and an assumed flow. Once P_c was found for a given mass flow, the [

]^{a,c,e} was found from Figure 6-2 (taken from Reference 6-2). For all cases considered, since [

]^{a,c,e} Therefore, this method will yield the two-phase pressure drop due to momentum effects as illustrated in Figure 6-3, where P_o is the operating pressure. Now using the assumed flow rate, G, the frictional pressure drop can be calculated using

$$\Delta P_f = [\qquad\qquad]^{a,c,e} \tag{6-1}$$

where the friction factor f is determined using the [ ]^{a,c,e} The crack relative roughness, ε, was obtained from fatigue crack data on stainless steel samples. The relative roughness value used in these calculations was [ ]^{a,c,e}

The frictional pressure drop using equation 6-1 is then calculated for the assumed flow rate and added to the [ ]^{a,c,e} to obtain the total pressure drop from the primary system to the atmosphere.

That is, for the pressurizer surge line:

Absolute Pressure - 14.7 = [ ]^{a,c,e} (6-2)

for a given assumed flow rate G. If the right-hand side of equation 6-2 does not agree with the pressure difference between the pressurizer surge line and the atmosphere, then the procedure is repeated until equation 6-2 is satisfied to within an acceptable tolerance which in turn leads to flow rate value for a given crack size.

6.4 LEAK RATE CALCULATIONS

Leak rate calculations were made as a function of crack length at the governing locations previously identified in Section 5.1. The normal operating loads of Table 3-4 were applied, in these calculations. The crack opening areas were estimated using the method of Reference 6-3 and the leak rates were calculated using the two-phase flow formulation described above. The average material properties of Section 4.0 (see Table 4-2) were used for these calculations.

The flaw sizes to yield a leak rate of 10 gpm were calculated at the governing locations and are given in Table 6-1 for H. B. Robinson Unit 2. The flaw sizes so determined are called leakage flaw sizes.

The H. B. Robinson Unit 2 RCS pressure boundary leak detection system meets the intent of Regulatory Guide 1.45. Thus, to satisfy the margin of 10 on the leak rate, the flaw sizes (leakage flaw sizes) are determined which yield a leak rate of 10 gpm.

6.5 REFERENCES

6-1 [ ]^{a,c,e}

6-2 M. M, El-Wakil, "Nuclear Heat Transport, International Textbook Company," New York, N.Y, 1971.

6-3 Tada, H., "The Effects of Shell Corrections on Stress Intensity Factors and the Crack Opening Area of Circumferential and a Longitudinal Through-Crack in a Pipe," Section II-1, NUREG/CR-3464, September 1983.

Table 6-1 Leakage Flaw Sizes

Node Point	Load Case	Temperature (°F)	Leakage Flaw Size (in.) (for 10 gpm leakage)
130	A	653	4.91
130	B	[ ][a,c,e]	4.84
380	A	653	4.08
380	B	[ ][a,c,e]	3.48
600	A	653	3.26
600	B	[ ][a,c,e]	3.44

a,c,e

MASS VELOCITY (lb/(sec)(ft^2))

STAGNATION ENTHALPY (10^2 Btu/lb)

Figure 6-1 Analytical Predictions of Critical Flow Rates of Steam-Water Mixtures

a,c,e

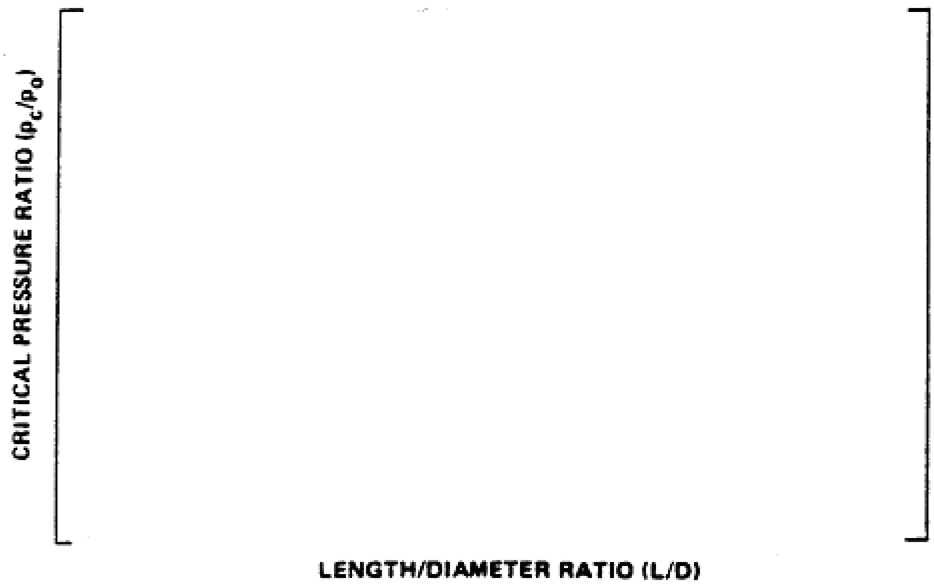


Figure 6-2 [ ]^{a,c,e} Pressure Ratio as a Function of L/D

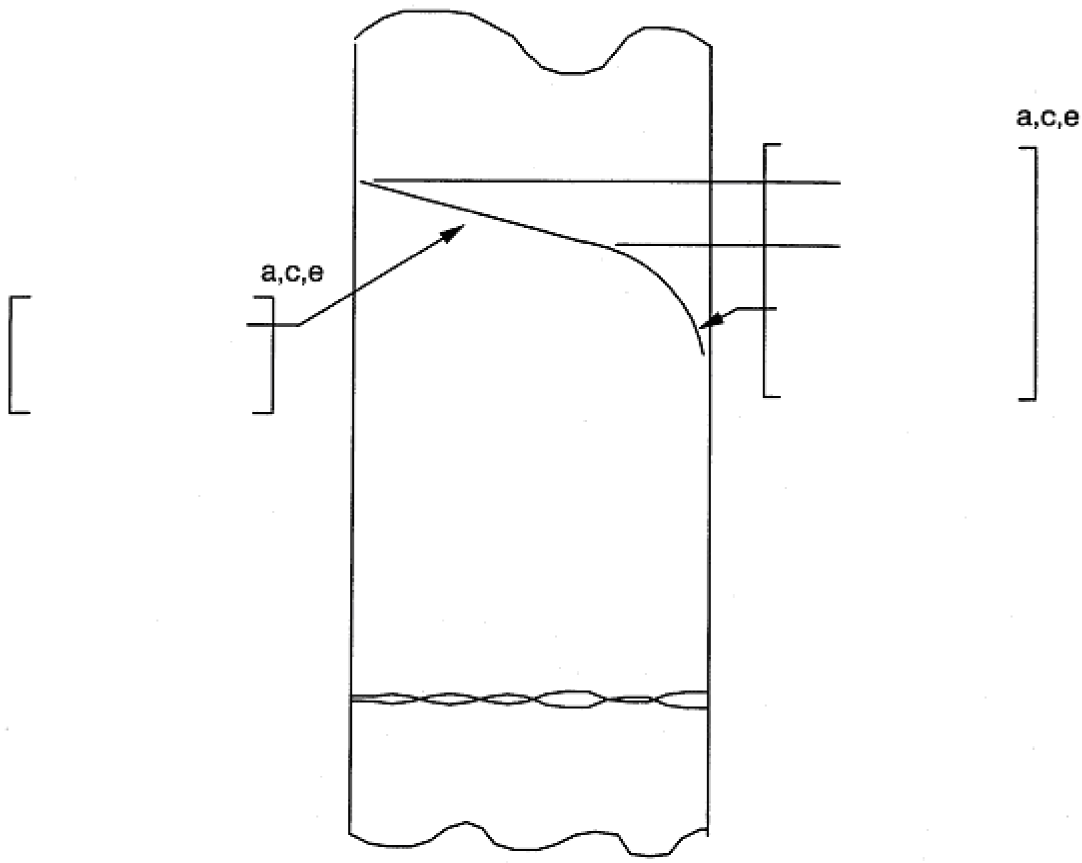


Figure 6-3 Idealized Pressure Drop Profile Through a Postulated Crack

7.0 FRACTURE MECHANICS EVALUATION

7.1 GLOBAL FAILURE MECHANISM

Determination of the conditions which lead to failure in stainless steel should be done with plastic fracture methodology because of the large amount of deformation accompanying fracture. One method for predicting the failure of ductile material is the plastic instability method, based on traditional plastic limit load concepts, but accounting for strain hardening and taking into account the presence of a flaw. The flawed pipe is predicted to fail when the remaining net section reaches a stress level at which a plastic hinge is formed. The stress level at which this occurs is termed as the flow stress. The flow stress is generally taken as the average of the yield and ultimate tensile strength of the material at the temperature of interest. This methodology has been shown to be applicable to ductile piping through a large number of experiments and will be used here to predict the critical flaw size in the primary coolant piping. The failure criterion has been obtained by requiring equilibrium of the section containing the flaw (Figure 7-1) when loads are applied. The detailed development is provided in Appendix A for a through-wall circumferential flaw in a pipe with internal pressure, axial force, and imposed bending moments. The limit moment for such a pipe is given by:

[ ]a,c,e

where:

[

]a,c,e

σ_f = $0.5 (\sigma_y + \sigma_u)$ = flow stress, psi

[

]a,c,e

The analytical model described above accurately accounts for the piping internal pressure as well as imposed axial force as they affect the limit moment. Good agreement was found between the analytical predictions and the experimental results (Reference 7-1). For application of the limit load methodology, the material, including consideration of the configuration, must have a sufficient ductility and ductile tearing resistance to sustain the limit load.

7.2 RESULTS OF CRACK STABILITY EVALUATION

A stability analysis based on limit load was performed for governing locations as described in Section 7.1. The weld process types, at these locations Node 130, Node 380 and Node 600 are used as GTAW and SMAW combination. The "Z" correction factor for SMAW (References 7-2 and 7-3) are as follows:

$$Z = 1.15\ [1.0 + 0.013\ (OD\text{-}4)] \text{ for SMAW}$$

where OD is the outer diameter of the pipe in inches.

The Z-factor for the GTAW weld is 1.0. The Z-factor for the SMAW was calculated for the governing locations, using the outer diameter of 12.75 inches. The applied faulted loads (Table 3-4) were increased by the Z factor and plots of limit load versus crack length were generated as shown in Figures 7-2 to 7-10. Lower bound material properties were used from Table 4-2. Table 7-1 summarizes the results of the stability analyses based on limit load. The leakage flaw sizes are also presented in the same table.

7.3 REFERENCES

7-1 Kanninen, M. F., et. al., "Mechanical Fracture Predictions for Sensitized Stainless Steel Piping with Circumferential Cracks," EPRI NP-192, September 1976.

7-2 Standard Review Plan; Public Comment Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

7-3 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

Table 7-1 Summary of Critical Flaw Sizes for the Pressurizer Surge Line

Node Point	Load Case	Temperature (°F)	Critical Flaw Size (in)
130	D	653	14.98
130	E	605	14.27
130	F	205	10.85
380	D	653	15.28
380	E	653	14.01
380	F	455	13.58
600	D	653	12.54
600	E	653	13.06
600	F	455	17.32

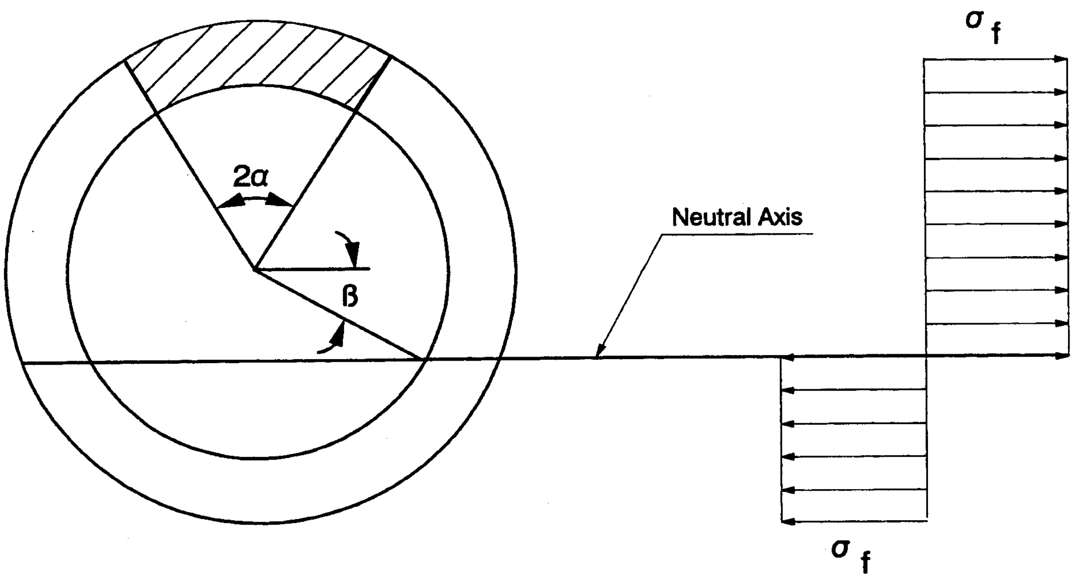


Figure 7-1 [ ]a,c,e Stress Distribution

a,c,e

OD = 12.75 in.	$\sigma_{y\text{-}min}$ = 22.117 ksi	F = 214.142 kips
t = 1.005 in.	$\sigma_{u\text{-}min}$ = 76.012 ksi	M = 865.628 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-2 Critical Flaw Size Prediction – Node 130 Case D

a,c,e

OD = 12.75 in.	$\sigma_{y\text{-}min}$ = 22.569 ksi	F = 214.142 kips
t = 1.005 in.	$\sigma_{u\text{-}min}$ = 76.012 ksi	M = 1046.293 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-3 Critical Flaw Size Prediction – Node 130 Case E

a,c,e

OD = 12.75 in.	$\sigma_{y\text{-}min}$ = 30.844 ksi	F = 45.076 kips
t = 1.005 in.	$\sigma_{u\text{-}min}$ = 79.289 ksi	M = 2834.232 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-4 Critical Flaw Size Prediction – Node 130 Case F

a,c,e

OD = 12.75 in.	$\sigma_{y\text{-}min}$ = 22.117 ksi	F = 212.011 kips
t = 1.005 in.	$\sigma_{u\text{-}min}$ = 76.012 ksi	M = 805.097 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-5 Critical Flaw Size Prediction – Node 380 Case D

a,c,e

OD = 12.75 in.	$\sigma_{y\text{-}min}$ = 22.117 ksi	F = 212.011 kips
t = 1.005 in.	$\sigma_{u\text{-}min}$ = 76.012 ksi	M = 1111.973 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-6 Critical Flaw Size Prediction – Node 380 Case E

a,c,e

OD = 12.75 in.	$\sigma_{y\text{-}min}$ = 24.687 ksi	F = 45.076 kips
t = 1.005 in.	$\sigma_{u\text{-}min}$ = 76.060 ksi	M = 1920.339 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-7 Critical Flaw Size Prediction – Node 380 Case F

a,c,e

OD = 12.75 in.	$\sigma_{y\text{-}min}$ = 22.117 ksi	F = 206.959 kips
t = 1.005 in.	$\sigma_{u\text{-}min}$ = 76.012 ksi	M = 1513.316 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-8 Critical Flaw Size Prediction – Node 600 Case D

a,c,e

OD = 12.75 in.	$\sigma_{y\text{-}min}$ = 22.117 ksi	F = 206.859 kips
t = 1.005 in.	$\sigma_{u\text{-}min}$ = 76.012 ksi	M = 1377.651 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-9 Critical Flaw Size Prediction – Node 600 Case E

a,c,e

OD = 12.75 in.	$\sigma_{y\text{-}min}$ = 24.687 ksi	F = 40.701 kips
t = 1.005 in.	$\sigma_{u\text{-}min}$ = 76.060 ksi	M = 1158.077 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-10 Critical Flaw Size Prediction – Node 600 Case F

8.0 ASSESSMENT OF FATIGUE CRACK GROWTH

8.1 INTRODUCTION

The fatigue crack growth (FCG) analysis is not a requirement for the LBB analysis (see References 8-1 and 8-2) since the LBB analysis is based on the postulation of through-wall flaw, whereas the FCG analysis is performed based on the surface flaw. However, a fatigue crack growth (FCG) assessment of the H. B. Robinson Unit 2 pressurizer surge line was performed. The fatigue crack growth (FCG) of the H. B. Robinson Unit 2 pressurizer surge line was determined by comparison with a fatigue crack growth analysis of a similar pressurizer surge line. The details of the fatigue crack growth analysis are presented below. By comparing the parameters critical to the fatigue crack growth analysis between H. B. Robinson Unit 2 and the similar pressurizer surge line analysis, it was concluded that the similar analysis would adequately cover the fatigue crack growth assessment of the H. B. Robinson Unit 2 pressurizer surge line. Due to similarities in Westinghouse PWR designs, it was possible to perform a representative fatigue crack growth assessment which would be applicable to H. B. Robinson Unit 2.

The methodology consists of first obtaining the local and structural transient stress analyses results and then superimposing the local and structural transient stresses. The design transients and cycles used in the FCG analyses were the similar ones used in Reference 8-3. An initial flaw size was postulated and the calculation of crack growth for the design plant life using the austenitic stainless steel crack growth law was performed. This fatigue crack growth analysis was performed at the hot leg nozzle location. At this location five through wall stress cuts were analyzed and their orientations are shown in Figure 8–1.

An extensive study was performed by the Materials Property Council Working Group on Reference Fatigue Crack Growth concerning the crack growth behavior of the austenitic stainless steels in an air environment, published in Reference 8-4. A reference fatigue crack growth curve for stainless steels in an air environment, is from Appendix C of the ASME Section XI Code, 2007 Edition (Reference 8-5). This curve is shown in Figure 8-2.

A compilation of data for austenitic stainless steels in a PWR water environment was made by Bamford (Reference 8-6), and it was found that the effect of the environment on the crack growth rate was small. For this reason it was conservatively estimated that the environmental factor should be set at [ ]^{a,c,e} in the crack growth rate equation from Reference 8-4. Based on these works (References 8-4 and 8-6) the stainless steel fatigue crack growth law used in the analyses is:

[

] a,c,e

8.2 RESULTS

Fatigue crack growth analyses were carried out along five stress cuts (Figure 8-1). The analyses were completed for postulated initial flaws oriented circumferentially. The flaws were assumed to be semi-elliptical with an aspect ratio of six to one. The initial flaw sizes were assumed to be 10% of the nominal wall thickness. The results of the fatigue crack growth analyses are presented in Table 8-1. For an initial flaw size of 0.14 inch, the result projects that the maximum final flaw size after 40 /60 years is about 14.8% of the nominal wall thickness. Therefore, flaw growth through the wall is not expected to occur during the 40/60 year design life of the plant and it is concluded that fatigue crack growth should not be a concern for the pressurizer surge line. Transients and cycles for the H. B. Robinson Unit 2 plant for a 40-year transient set will remain bounding for 60 years (Reference 8-7), the FCG results shown in Table 8-1 is also applicable for the 60 years.

8.3 REFERENCES

8-1 Standard Review Plan; Public Comment Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

8-2 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

8-3 WCAP-12962, Revision 0, "Structural Evaluation of the H. B. Robinson Unit 2 and Shearon Harris Pressurizer Surge lines, Considering the Effects of Thermal Stratification," September 1991 including WCAP-12962 Supplement 1, Revision 0, October 1995 (Westinghouse Proprietary).

8-4 James, L. A. and Jones, D. P., "Fatigue Crack Growth Correlations for Austenitic Stainless Steel in Air," in Predictive Capabilities in Environmentally Assisted Cracking, ASME publication PVP-99, December 1985.

8-5 ASME Boiler and Pressure Vessel Code Section XI, 2007 Edition with the 2008 Addenda, "Rules for Inservice Inspection of Nuclear Power Plant Components."

8-6 Bamford, W. H., "Fatigue Crack Growth of Stainless Steel Reactor Coolant Piping in a Pressurized Water Reactor Environment, ASME Trans. Journal of Pressure Vessel Technology" February 1979.

8-7 NUREG-1785, "Safety Evaluation Report Related to the License Renewal of H. B. Robinson Steam Electric Plant, Unit 2."

a,c,e

Table 8-1 Pressurizer Surge Line Fatigue Crack Growth Results				
Orientation (Figure 8-1)	**Crack Parameters**			
	Postulated Initial Flaw size (in)	**Initial Flaw (% of wall thickness)**	**Final Flaw Size (in) 40/60 Years***	**Final Flaw (% of wall thickness)**

Note: * Transients and cycles for the H. B. Robinson Unit 2 plant for a 40-year transient set will remain bounding for 60 years, the FCG results shown in Table 8-1 are also applicable for the 60 years.

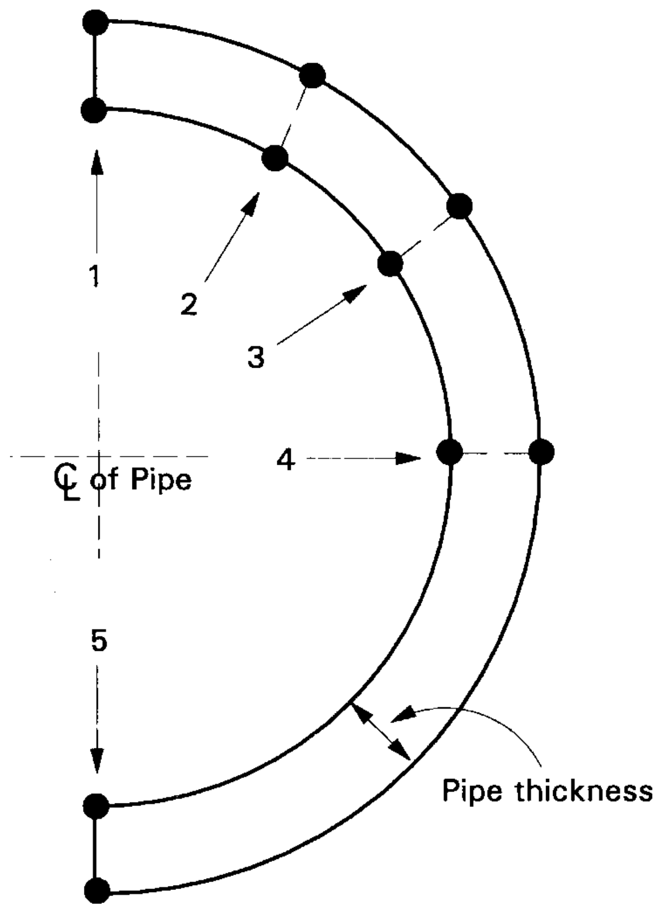


Figure 8-1 Orientation of Stress Cuts for the Fatigue Crack Growth Analysis

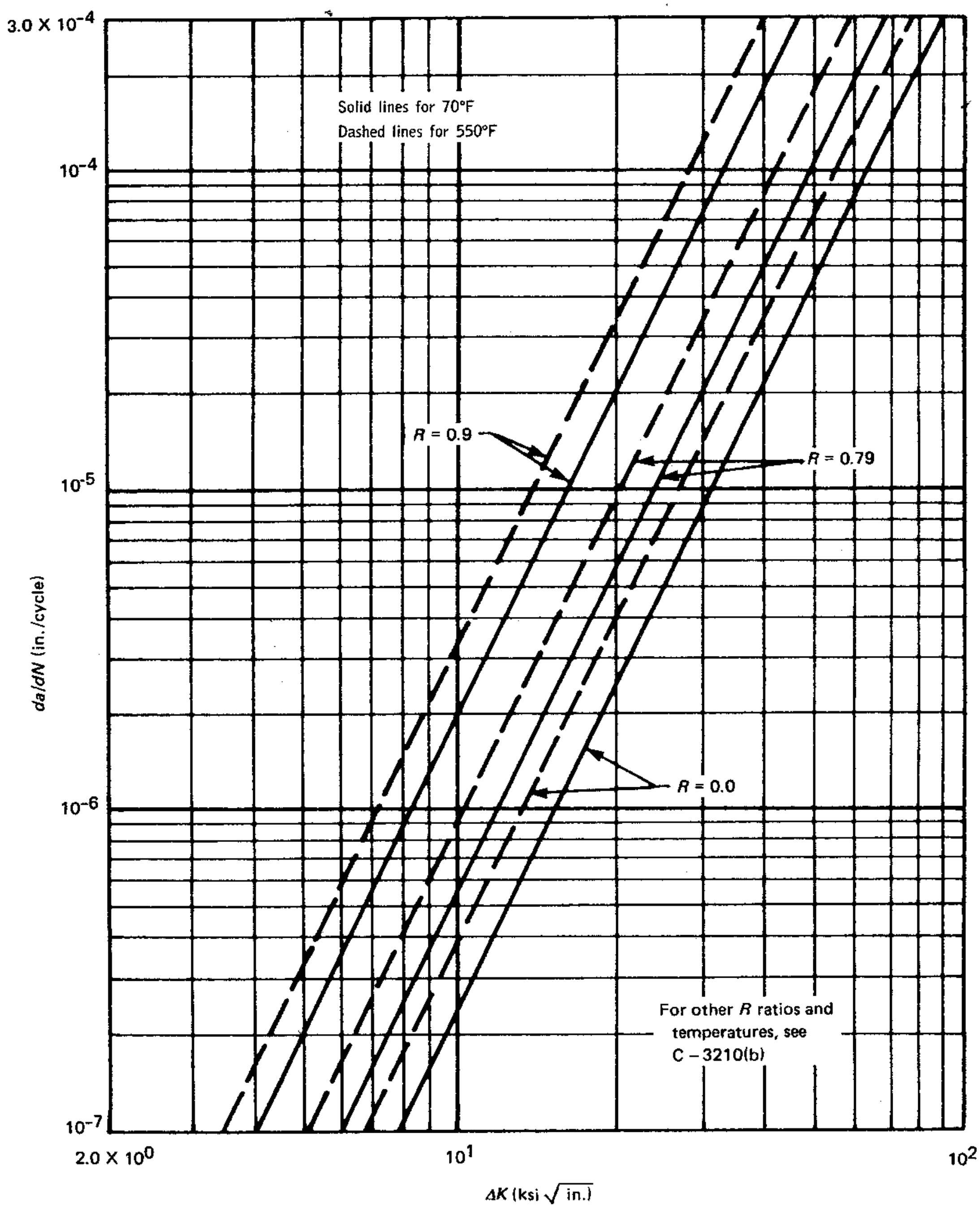


Figure 8-2 Reference Crack Growth Curves for Stainless Steel in Air Environments

9.0 ASSESSMENT OF MARGINS

The results of the leak rates of Section 6.4 and the corresponding stability evaluations of Section 7.2 are used in performing the assessment of margins. Margins are shown in Table 9-1. All the LBB recommended margins are satisfied.

In summary, at all the critical locations relative to:

1. Flaw Size - Using faulted loads obtained by the absolute sum method, a margin of 2 or more exists between the critical flaw and the flaw having a leak rate of 10 gpm (the leakage flaw).

2. Leak Rate - A margin of 10 exists between the calculated leak rate from the leakage flaw and the plant leak detection capability of 1 gpm.

3. Loads - At the critical locations the leakage flaw was shown to be stable using the faulted loads obtained by the absolute sum method (i.e., a flaw twice the leakage flaw size is shown to be stable; hence the leakage flaw size is stable). A margin of 1 on loads using the absolute summation of faulted load combinations is satisfied.

Table 9-1 Leakage Flaw Sizes, Critical Flaw Sizes, and Margins for the Pressurizer Surge Line

Node	Load Case	Critical Flaw Size (in)	Leakage Flaw Size (in)	Margin
130	A/D	14.98	4.91	3.0
130	A/F	10.84	4.91	2.2
130	B/E	14.27	4.84	2.9
130	B/F	10.84	4.84	2.2
380	A/D	15.28	4.08	3.7
380	A/F	13.58	4.08	3.3
380	B/E	14.01	3.48	4.0
380	B/F	13.58	3.48	3.9
600	A/D	12.54	3.26	3.8
600	A/F	17.32	3.26	5.3
600	B/E	13.06	3.44	3.8
600	B/F	17.32	3.44	5.0

10.0 CONCLUSIONS

This report justifies the elimination of pressurizer surge line breaks from the structural design basis for the H. B. Robinson Unit 2 license renewal period as follows:

a. Stress corrosion cracking is precluded by use of fracture resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation.

b. Water hammer should not occur in the pressurizer surge line piping because of system design, testing, and operational considerations.

c. The effects of low and high cycle fatigue on the integrity of the pressurizer surge line are negligible.

d. Ample margin exists between the leak rate of small stable flaws and the capability of the H. B. Robinson Unit 2 reactor coolant system pressure boundary Leakage Detection System.

e. Ample margin exists between the small stable flaw sizes of item (d) and larger stable flaws.

f. Ample margin exists in the material properties used to demonstrate end-of-service life (fully aged) stability of the critical flaws.

g. Fatigue crack growth results using the 40-year design transients and cycles (shown to be applicable for 60 years) show that there will be insignificant growth through the wall for the license renewal period (60-year plant life).

For the critical locations, flaws are identified that will be stable because of the ample margins described in d, e, and f above.

Based on loading, pipe geometry and pipe material properties considerations, enveloping critical (governing) locations were determined at which leak-before-break crack stability evaluations were made. Through-wall flaw sizes were postulated which would cause a leak at a rate of ten (10) times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, fatigue crack growth was shown not to be an issue for the pressurizer surge line piping. Therefore, the Leak-Before-Break conditions and margins are satisfied for the H. B. Robinson Unit 2 pressurizer surge line piping. It is demonstrated that the dynamic effects of the pipe rupture resulting from postulated breaks in the pressurizer surge line piping need not be considered in the structural design basis of H. B. Robinson Unit 2 for the license renewal period (60-year plant life).

APPENDIX A: LIMIT MOMENT

[

] a,c,e

a,c,e

Figure A-1 Pipe with a Through-Wall Crack in Bending

ATTACHMENT 3

WCAP-17778-NP, REVISION 1, "TECHNICAL JUSTIFICATION FOR ELIMINATING RESIDUAL HEAT REMOVAL (RHR) LINE RUPTURE AS THE STRUCTURAL DESIGN BASIS FOR H. B. ROBINSON UNIT 2”, MARCH 2023

(REDACTED)

Westinghouse Non-Proprietary Class 3

WCAP-17778-NP
Revision 1

March 2023

Technical Justification for Eliminating Residual Heat Removal (RHR) Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2

Westinghouse

WCAP-17778-NP
Revision 1

Technical Justification for Eliminating Residual Heat Removal (RHR) Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2

March 2023

Author: Nadia B. Petkova *
Operating Plants Piping and Supports

Reviewer: Momo Wiratmo*
Operating Plants Piping and Supports

Approved: Lynn A. Patterson, Manager*
Reactor Vessel and Containment
Vessel Design and Analysis

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

RECORD OF REVISIONS

Revision	Date	Revision Description
0	August 2013	Original Issue (WCAP-17778-NP). This is the non-proprietary class 3 version of WCAP-17778-P, Revision 0.
1	March 2023	This is the non-proprietary class 3 version of WCAP-17778-P, Revision 1.

TABLE OF CONTENTS

1.0 INTRODUCTION 1-1
1.1 PURPOSE 1-1
1.2 SCOPE AND OBJECTIVES 1-1
1.3 REFERENCES 1-2

2.0 OPERATION AND STABILITY OF THE REACTOR COOLANT SYSTEM 2-1
2.1 STRESS CORROSION CRACKING 2-1
2.2 WATER HAMMER 2-2
2.3 LOW CYCLE AND HIGH CYCLE FATIGUE 2-2
2.4 OTHER POSSIBLE DEGRADATION DURING SERVICE OF THE RHR LINES 2-3
2.5 REFERENCES 2-3

3.0 PIPE GEOMETRY AND LOADING 3-1
3.1 CALCULATION OF LOADS AND STRESSES 3-1
3.2 LOADS FOR LEAK RATE EVALUATION 3-1
3.3 LOAD COMBINATION FOR CRACK STABILITY ANALYSES 3-2
3.4 SUMMARY OF LOADS AND GEOMETRY FOR THE RHR LINES 3-2
3.5 GOVERNING LOCATIONS FOR THE RHR LINES 3-3
3.6 REFERENCES 3-3

4.0 MATERIAL CHARACTERIZATION 4-1
4.1 RHR LINE PIPE, FITTINGS AND WELD MATERIALS 4-1
4.2 TENSILE PROPERTIES 4-1
4.3 REFERENCES 4-1

5.0 CRITICAL LOCATION AND EVALUATION CRITERIA 5-1
5.1 CRITICAL LOCATIONS 5-1

6.0 LEAK RATE PREDICTIONS 6-1
6.1 INTRODUCTION 6-1
6.2 GENERAL CONSIDERATIONS 6-1
6.3 CALCULATION METHOD 6-1
6.4 LEAK RATE CALCULATIONS 6-2
6.5 REFERENCES 6-3

7.0 FRACTURE MECHANICS EVALUATION 7-1
7.1 GLOBAL FAILURE MECHANISM 7-1
7.2 RESULTS OF CRACK STABILITY EVALUATION 7-2
7.3 REFERENCES 7-3

8.0 ASSESSMENT OF FATIGUE CRACK GROWTH 8-1
8.1 INTRODUCTION 8-1
8.2 CRITICAL LOCATION FOR FATIGUE CRACK GROWTH ANALYSIS 8-1
8.3 DESIGN TRANSIENTS 8-1

8.4 STRESS ANALYSIS 8-1
8.5 OBE LOADS 8-2
8.6 TOTAL STRESS FOR FATIGUE CRACK GROWTH 8-2
8.7 FATIGUE CRACK GROWTH ANALYSIS 8-2
8.8 ANALYSIS PROCEDURE 8-2
8.9 RESULTS 8-4
8.10 REFERENCES 8-5

9.0 ASSESSMENT OF MARGINS 9-1

10.0 CONCLUSIONS 10-1

APPENDIX A: LIMIT MOMENT A-1

LIST OF TABLES

Table 3-1 Summary of H. B. Robinson Nuclear Power Plant Piping Geometry and Normal Operating Condition for the RHR Line....3-4

Table 3-2 Summary of H. B. Robinson Unit 2 Normal Loads and Stresses for Residual Heat Removal (RHR) Line 3-5

Table 3-3 Summary of H. B. Robinson Unit 2 Faulted Loads and Stresses for Residual Heat Removal (RHR) Line 3-6

Table 4-1 Measured Tensile Properties for RHR Line Material A376 TP316 4-2

Table 4-2 Mechanical Properties for RHR Line Material at Operating Temperatures....4-2

Table 6-1 Flaw Sizes Yielding a Leak Rate of 10 gpm for the RHR Line 6-4

Table 7-1 Stability Results for the RHR line Based on Limit Load....7-4

Table 8-1 Design Transients Considered for Fatigue Crack Growth Evaluation 8-6

Table 8-2 RHR Lines Fatigue Crack Growth Results 8-7

Table 9-1 Leakage Flaw Sizes, Critical Flaw Sizes and Margins for RHR Line 9-2

LIST OF FIGURES

Figure 3-1 H. B. Robinson Unit 2 RHR Line Layout....3-7

Figure 6-1 Analytical Predictions of Critical Flow Rates of Steam-Water Mixtures....6-5

Figure 6-2 [ ]a,c,e Pressure Ratio as a Function of L/D....6-6

Figure 6-3 Idealized Pressure Drop Profile Through a Postulated Crack....6-7

Figure 7-1 [ ]a,c,e Stress Distribution....7-5

Figure 7-2 Critical Flaw Size Prediction – Node 323....7-6

Figure 7-3 Critical Flaw Size Prediction – Node 320....7-7

Figure 8-1 Schematic of RHR Line at RCL Hot Leg Nozzle Weld Location....8-8

Figure 8-2 Reference Crack Growth Curves for Stainless Steel in Air Environments....8-9

Figure A-1 Pipe with a Through-Wall Crack in Bending....A-2

1.0 INTRODUCTION

1.1 PURPOSE

The current structural design basis for the Residual Heat Removal (RHR) lines requires postulating non-mechanistic circumferential and longitudinal pipe breaks. This results in additional plant hardware (e.g., pipe whip restraints and jet shields) which would mitigate the dynamic consequences of the pipe breaks. It is, therefore, highly desirable to be realistic in the postulation of pipe breaks for the RHR lines. Presented in this report are the descriptions of a mechanistic pipe break evaluation method and the analytical results that can be used for establishing that a circumferential type of break will not occur within the RHR lines.

1.2 SCOPE AND OBJECTIVES

The scope of this report is limited to the high energy Class 1 portion of the RHR lines (primary loop junction to the second isolation valve). A schematic drawing of the piping system is shown in Section 3. The recommendations and criteria proposed in SRP 3.6.3 (References 1-1 and 1-2) are used in this evaluation. The criteria and the resulting steps of the evaluation procedure can be briefly summarized as follows:

1. Calculate the applied loads. Identify the location(s) at which the highest faulted stress occurs.

2. Identify the materials and the material properties.

3. Postulate a surface flaw governing location. Determine fatigue crack growth. Show that a through-wall crack will not result.

4. Postulate a through-wall flaw at the governing location(s). The size of the flaw should be large enough so that the leakage is assured of detection with margin using the installed leak detection equipment when the pipe is subjected to normal operating loads. Demonstrate that there is a margin of 10 between the calculated leak rate and the leak detection capability.

5. Using maximum faulted loads in the stability analysis, demonstrate that there is a margin of 2 between the leakage size flaw and the critical size flaw.

6. Review the operating history to ascertain that operating experience has indicated no particular susceptibility to failure from the effects of corrosion, water hammer or low and high cycle fatigue.

7. For the material types used in the Plant, provide representative material properties.

This report provides a fracture mechanics demonstration of RHR integrity for H. B. Robinson Unit 2 consistent with the NRC position for exemption from consideration of postulated pipe rupture dynamic effects (Reference 1-3).

It should be noted that the terms “flaw” and “crack” have the same meaning and are used interchangeably. “Governing location” and “critical location” are also used interchangeably throughout the report.

Note that there are several locations in this report where proprietary information has been identified and bracketed. For each of the bracketed locations, the reason for the proprietary classification is given using a standardized system. The proprietary brackets are labeled with three different letters, to provide this information, and the explanation for each letter is given below:

a. The information reveals the distinguishing aspects of a process or component, structure, tool, method, etc., and the prevention of its use by Westinghouse's competitors, without license from Westinghouse, gives Westinghouse a competitive economic advantage.

b. The information, if used by a competitor, would reduce the competitor's expenditure of resources or improve the competitor's advantage in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. The information reveals aspects of past, present, or future Westinghouse or customer-funded development plans and programs of potential commercial value to Westinghouse.

The proprietary information in the brackets which has been deleted in this version of this report are provided in the proprietary class 2 document (WCAP-17778-P, Revision 1).

1.3 REFERENCES

1-1 Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday August 28, 1987/Notices, pp. 32626-32633.

1-2 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

1-3 Nuclear Regulatory Commission, 10 CFR 50, Modification of General Design Criteria 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, Final Rule, Federal Register/Vol. 52, No. 207/Tuesday, October 27, 1987/Rules and Regulations, pp. 41288 41295.

2.0 OPERATION AND STABILITY OF THE REACTOR COOLANT SYSTEM

2.1 STRESS CORROSION CRACKING

The Westinghouse reactor coolant system primary loop piping and connected Class 1 piping have an operating history that demonstrates the inherent operating stability characteristics of the design. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking (IGSCC)). This operating history totals over 1400 reactor-years, including 16 plants each having over 30 years of operation, 10 other plants each with over 25 years of operation, 11 plants each with over 20 years of operation and 12 plants each with over 15 years of operation.

In 1978, the United States Nuclear Regulatory Commission (USNRC) formed the second Pipe Crack Study Group. (The first Pipe Crack Study Group (PCSG) established in 1975 addressed cracking in boiling water reactors only.) One of the objectives of the second PCSG was to include a review of the potential for stress corrosion cracking in Pressurized Water Reactors (PWRs). The results of the study performed by the PCSG were presented in NUREG-0531 (Reference 2-1) entitled "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants." In that report the PCSG stated:

> "The PCSG has determined that the potential for stress-corrosion cracking in PWR primary system piping is extremely low because the ingredients that produce IGSCC are not all present. The use of hydrazine additives and a hydrogen overpressure limit the oxygen in the coolant to very low levels. Other impurities that might cause stress-corrosion cracking, such as halides or caustic, are also rigidly controlled. Only for brief periods during reactor shutdown when the coolant is exposed to the air and during the subsequent startup are conditions even marginally capable of producing stress-corrosion cracking in the primary systems of PWRs. Operating experience in PWRs supports this determination. To date, no stress corrosion cracking has been reported in the primary piping or safe ends of any PWR."

For stress corrosion cracking (SCC) to occur in piping, the following three conditions must exist simultaneously: high tensile stresses, susceptible material, and a corrosive environment. Since some residual stresses and some degree of material susceptibility exist in any stainless steel piping, the potential for stress corrosion is minimized by properly selecting a material immune to SCC as well as preventing the occurrence of a corrosive environment. The material specifications consider compatibility with the system's operating environment (both internal and external) as well as 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 (e.g., sulfides, sulfites, and thionates). Strict pipe cleaning standards prior to operation and careful control of water chemistry during plant operation are used to prevent the occurrence of a corrosive environment. Prior to being put into service, the piping is cleaned internally and externally. During flushes and preoperational testing, water chemistry is controlled in accordance with written specifications.

Requirements on chlorides, fluorides, conductivity, and pH are included in the acceptance criteria for the piping.

During plant operation, the reactor coolant water chemistry is monitored and maintained within very specific limits. Contaminant concentrations are kept below the thresholds known to be conducive to stress corrosion cracking with the major water chemistry control standards being included in the plant operating procedures as a condition for plant operation. For example, during normal power operation, oxygen concentration in the RCS is expected to be in the ppb range by controlling charging flow chemistry and maintaining hydrogen in the reactor coolant at specified concentrations. Halogen concentrations are also stringently controlled by maintaining concentrations of chlorides and fluorides within the specified limits. Thus, during plant operation, the likelihood of stress corrosion cracking is minimized.

During 1979, several instances of cracking in PWR feedwater piping led to the establishment of the third PCSG. The investigations of the PCSG reported in NUREG-0691 (Reference 2-2) further confirmed that no occurrences of IGSCC have been reported for PWR primary coolant systems.

Primary Water Stress Corrosion Cracking (PWSCC) occurred in V. C. Summer reactor vessel hot leg nozzle, Alloy 82/182 weld. It should be noted that this susceptible material is not found at the H. B. Robinson Unit 2 RHR line.

2.2 WATER HAMMER

Overall, there is a low potential for water hammer in the RCS and connecting RHR lines since they are designed and operated to preclude the voiding condition in normally filled lines. The RCS and connecting RHR lines including piping and components are designed for normal, upset, emergency, and faulted condition transients. The design requirements are conservative relative to both the number of transients and their severity. Relief valve actuation and the associated hydraulic transients following valve opening are considered in the system design. Other valve and pump actuations are relatively slow transients with no significant effect on the system dynamic loads. To ensure dynamic system stability, reactor coolant parameters are stringently controlled. Temperature during normal operation is maintained within a narrow range by the control rod positions; pressure is controlled also within a narrow range for steady-state conditions by the pressurizer heaters and pressurizer spray. The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters, namely system resistance and the reactor coolant pump characteristics are controlled in the design process. Additionally, Westinghouse has instrumented typical reactor coolant systems to verify the flow and vibration characteristics of the system and the connecting auxiliary lines. Preoperational testing and operating experience have verified the Westinghouse approach. The operating transients of the RCS primary piping and connected RHR lines are such that no significant water hammer can occur.

2.3 LOW CYCLE AND HIGH CYCLE FATIGUE

An assessment of the low cycle fatigue loadings is discussed in the form of a fatigue crack growth assessment, in Section 8.0.

Pump vibrations during operation would result in high cycle fatigue loads in the piping system. During operation, an alarm signals the exceedance of the RC pump shaft vibration limits. Field measurements have been made on the reactor coolant loop piping in a number of plants during hot functional testing. Stresses in the elbow below the RC pump have been found to be very small, between 2 and 3 ksi at the highest. Field measurements on typical PWR plants indicate vibration amplitudes less than 1 ksi. When translated to the connecting RHR lines, these stresses would be even lower, well below the fatigue endurance limit for the RHR line materials and would result in an applied stress intensity factor below the threshold for fatigue crack growth.

2.4 OTHER POSSIBLE DEGRADATION DURING SERVICE OF THE RHR LINES

Thermal stratification occurs when conditions permit hot and cold layers of water to exist simultaneously in a horizontal pipe. This can result in significant thermal loadings due to the high fluid temperature differentials. Changes in the stratification state result in thermal cycling, which can cause fatigue damage. This was an important issue in PWR feedwater line and pressurizer surge line piping, where temperature differentials of 300°F were not uncommon.

For the RHR piping in the H. B. Robinson Unit 2, thermal stratification due to NRC Bulletin 88-08 is not a concern (Reference 2-3).

The RHR Lines and the associated fittings for H. B. Robinson Nuclear Power Plant are forged product forms, which are not susceptible to toughness degradation due to thermal aging.

The maximum normal operating temperature of the RHR piping is about 605°F. This is well below the temperature that would cause any creep damage in stainless steel piping.

2.5 REFERENCES

2-1 Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants, NUREG-0531, U.S. Nuclear Regulatory Commission, February 1979.

2-2 Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors, NUREG-0691, U.S. Nuclear Regulatory Commission, September 1980.

2-3 NRC letter, "NRC Bulletin 88-08, Thermal Stresses in Piping Connected to Reactor Coolant Systems- H. B. Robinson Steam Electric Plant Unit no. 2 (TAC No. 69679)" Docket No. 50-261, October 1, 1991.

3.0 PIPE GEOMETRY AND LOADING

3.1 CALCULATION OF LOADS AND STRESSES

The stresses due to axial loads and bending moments are calculated by the following equation:

$$\sigma = \frac{F}{A} + \frac{M}{Z} \tag{3-1}$$

where,

σ = stress

F = axial load

M = moment

A = pipe cross-sectional area

Z = section modulus

The moments for the desired loading combinations are calculated by the following equation:

$$M = \sqrt{M_x^2 + M_y^2 + M_z^2} \tag{3-2}$$

where,

M_x = X component of moment, Torsion

M_y = Y component of bending moment

M_z = Z component of bending moment

The axial load and moments for leak rate predictions and crack stability analyses are computed by the methods to be explained in Sections 3.3 and 3.4.

3.2 LOADS FOR LEAK RATE EVALUATION

The normal operating loads for leak rate predictions are calculated by the following equations:

$$F = F_{DW} + F_{TH} + F_P \tag{3-3}$$

$$M_X = (M_X)_{DW} + (M_X)_{TH} \tag{3-4}$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{TH} \tag{3-5}$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{TH} \tag{3-6}$$

The subscripts of the above equations represent the following loading cases:

DW = deadweight

TH = normal thermal expansion

P = load due to internal pressure

This method of combining loads is often referred to as the algebraic sum method (References 3-1 and 3-2).

The as-built dimensions and normal operating conditions are given in Table 3-1. The loads based on this method of combination are provided in Table 3-2 at all the weld locations identified in Figure 3-1.

3.3 LOAD COMBINATION FOR CRACK STABILITY ANALYSES

In accordance with Standard Review Plan 3.6.3 (References 3-1 and 3-2), the absolute sum of loading components can be applied which results in higher magnitude of combined loads. If crack stability is demonstrated using these loads, the LBB margin on loads can be reduced from √2 to 1.0. The absolute summation of loads is shown in the following equations:

$$F = |F_{DW}| + |F_{TH}| + |F_P| + |F_{SSEINERTIA}| + |F_{SSEAM}| \quad (3\text{-}7)$$

$$M_X = |(M_X)_{DW}| + |(M_X)_{TH}| + |(M_X)_{SSEINERTIA}| + |(M_X)_{SSESAM}| \quad (3\text{-}8)$$

$$M_Y = |(M_Y)_{DW}| + |(M_Y)_{TH}| + |(M_Y)_{SSEINERTIA}| + |(M_Y)_{SSEAM}| \quad (3\text{-}9)$$

$$M_Z = |(M_Z)_{DW}| + |(M_Z)_{TH}| + |(M_Z)_{SSEINERTIA}| + |(M_Z)_{SSEAM}| \quad (3\text{-}10)$$

where subscript SSEINERTIA refers to safe shutdown earthquake inertia, SSEAM is safe shutdown earthquake anchor motion.

The loads so determined are used in the fracture mechanics evaluations (Section 7.0) to demonstrate the LBB margins at the locations established to be the governing locations. These loads at all the weld locations (see Figure 3-1) are given in Table 3-3.

3.4 SUMMARY OF LOADS AND GEOMETRY FOR THE RHR LINES

The load combinations were evaluated at the various weld locations. Normal loads were determined using the algebraic sum method whereas the faulted loads were combined using the absolute sum method. The normal operating loadings for the RHR lines are Pressure (P), Deadweight (DW) and Normal Operating Thermal Expansion (TH) loads. The faulted loadings consist of Normal Operating loads plus Safe Shutdown Earthquake (SSE) loads including the Seismic Anchor Motion.

Table 3-1 shows the piping geometry and normal operating condition for the RHR line at the weld locations. The minimum pipe wall thickness at the weld counterbore is used in the analysis, which is based on the maximum allowed counterbore at a butt weld (Reference 3-3). The normal and faulted loads are tabulated in Tables 3-2 and 3-3 respectively at the weld locations for RHR Line.

3.5 GOVERNING LOCATIONS FOR THE RHR LINES

All the welds at the RHR line are fabricated using the GTAW/SMAW combination or GTAW weld process procedures. The governing locations were established on the basis of the pipe schedules, material type, operating temperature, operating pressure, and the highest faulted stresses at the welds. Figure 3-1 shows the schematic layout of the RHR line and also identifies the governing weld locations.

The governing locations enveloping the RHR line are found to be: Node 323 and Node 320.

3.6 REFERENCES

3-1 Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

3-2 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

3-3 American National Standards, Butt Welding Ends, ANSI B16.25-1979.

Table 3-1 Summary of H. B. Robinson Nuclear Power Plant Piping Geometry and Normal Operating Condition for the RHR Line

Weld Location Node	Material Type	Outer Diameter (in)	Minimum Wall Thickness (in)	Normal Operating	
				Pressure (psig)	Temperature (°F)
323	A376 TP316	14.00	1.114	2235	605
322	A376 TP316	14.00	1.114	2235	605
320	A376 TP316	14.00	1.114	2235	350
319	A376 TP316	14.00	1.114	2235	350
317	A376 TP316	14.00	1.114	2235	350
316	A376 TP316	14.00	1.114	2235	350
315	A376 TP316	14.00	1.114	2235	350
314	A376 TP316	14.00	1.114	2235	350
313	A376 TP316	14.00	1.114	2235	350
312	A376 TP316	14.00	1.114	2235	350
1311	A376 TP316	14.00	1.114	2235	350

Table 3-2 Summary of H. B. Robinson Unit 2 Normal Loads and Stresses for Residual Heat Removal (RHR) Line

Weld Location[a] Node	Axial Force[b] (lbs)	Moment (in-lbs)	Axial Stress (psi)	Moment Stress (psi)	Total Stress (psi)
323	245667	244817	5446	1817	7263
322	245667	296448	5446	2200	7646
320	245643	299346	5446	2222	7667
319	243621	250051	5401	1856	7257
317	244046	64144	5410	476	5886
316	245471	44861	5442	333	5775
315	245464	36213	5442	269	5711
314	243585	43166	5400	320	5721
313	244024	109089	5410	810	6220
312	243958	130700	5408	970	6378
1311	243933	112207	5408	833	6241

Notes:

a. See Figure 3-1

b. Included Pressure

Table 3-3 Summary of H. B. Robinson Unit 2 Faulted Loads and Stresses for Residual Heat Removal (RHR) Line

Weld Location Node[a,b]	Axial Force[c] (lbs)	Moment (in-lbs)	Axial Stress (psi)	Moment Stress (psi)	Total Stress (psi)
323	252725	590504	5603	4382	9985
322	252725	559261	5603	4151	9753
320	252466	602162	5597	4469	10066
319	244997	448750	5431	3330	8762
317	249680	579326	5535	4299	9835
316	250893	581339	5562	4314	9877
315	250812	364615	5560	2706	8266
314	244609	419216	5423	3111	8534
313	249430	469278	5530	3483	9013
312	248953	348625	5519	2587	8106
1311	248680	380022	5513	2820	8333

Notes:

a. See Figure 3-1

b. See Table 3-1 for dimensions

c. Included Pressure

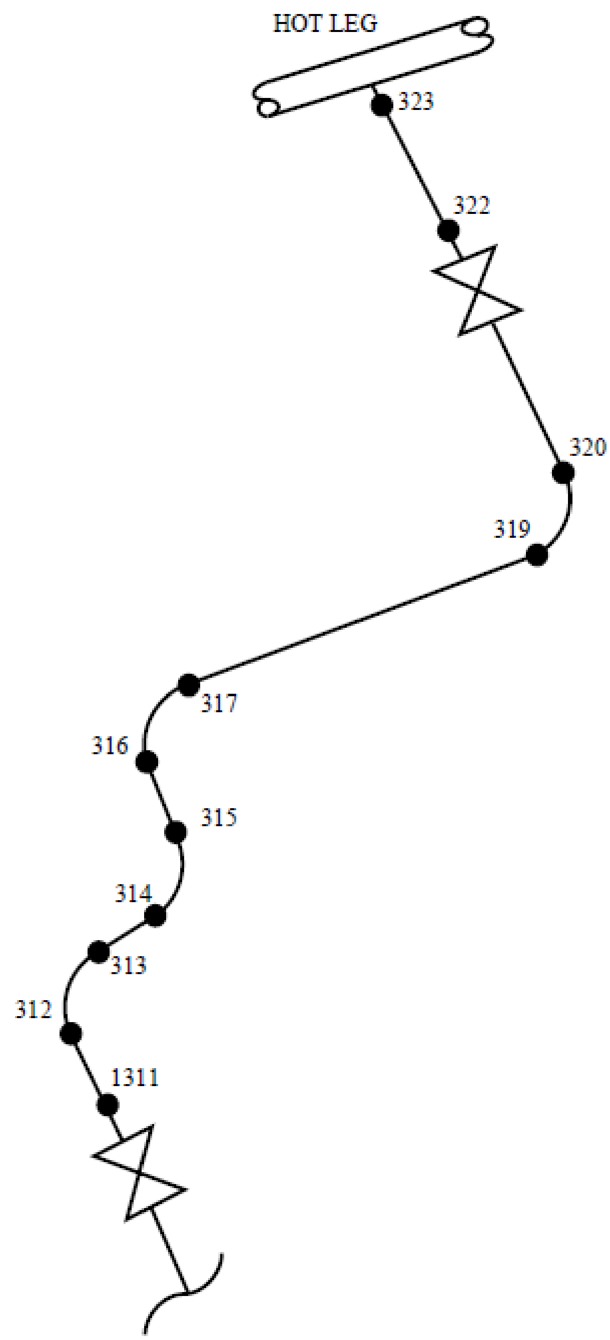


Figure 3-1 H. B. Robinson Unit 2 RHR Line Layout

4.0 MATERIAL CHARACTERIZATION

4.1 RHR LINE PIPE, FITTINGS AND WELD MATERIALS

The material type of the RHR line for the H. B. Robinson Unit 2 Nuclear Power Plant is A376 TP316. This is a wrought product of the type used for the piping in several PWR plants. The RHR line system does not include any cast pipes or cast fittings. The welding processes used are Gas Tungsten Arc Weld (GTAW) and Shielded Metal Arc Weld (SMAW) combination. Figure 3-1 show the schematic layout of the RHR line and also identifies the weld locations by node points.

In the following sections the tensile properties of the materials are presented for use in the Leak-Before-Break analyses.

4.2 TENSILE PROPERTIES

The Certified Materials Test Reports (CMTRs) for the H. B. Robinson Unit 2 RHR line were used to establish the tensile properties for the leak-before-break analyses. The tensile properties for the pipe material are provided in Table 4-1 for H. B. Robinson Unit 2.

For the A376 TP316 pipe material, the representative properties at operating temperatures of 605°F and 350°F are established from the tensile properties at room temperature given in Table 4-1 by utilizing Section II of the 2007 with the 2008 Addenda of ASME Boiler and Pressure Vessel Code (Reference 4-1). Code tensile properties at the operating temperatures were obtained by interpolating between 300°F, 400°F, 600°F and 700°F tensile Code properties. Ratios of the Code tensile properties at the operating temperatures to the corresponding properties were then applied to the room temperature tensile properties obtained from CMTRs (Table 4-1) to obtain the H. B. Robinson Unit 2 RHR line specific properties at operating temperatures of 605°F and 350°F.

The average and lower bound yield strengths and ultimate strengths for the pipe material are tabulated in Table 4-2. The ASME Code modulus of elasticity values are also given, and Poisson's ratio was taken as 0.3.

4.3 REFERENCES

4-1 ASME Boiler and Pressure Vessel Code, 2007 Edition with the 2008 Addenda, Section II, Part D – Properties (Customary) Materials.

Table 4-1 Measured Tensile Properties for RHR Line Material A376 TP316

Heat Number	Yield Strength (psi) At Room Temp.	Ultimate Strength (psi) At Room Temp.
J1434 (ser 3939)	40900	83800
J1434 (ser 3939)	46900	84800
J1434 (ser 3940)	44400	87400
J1434 (ser 3940)	41400	85500
J1434 (ser 3941)	43400	86400
J1434 (ser 3941)	43600	83400
J1434 (ser 3942)	48600	93200
J1434 (ser 3942)	43400	84900
DYEF(27009)	37200	78000
DYEF (27009)	37200	78000
DYEA (139788)	38600	83000

Table 4-2 Mechanical Properties for RHR Line Material at Operating Temperatures

Material	Temperature (°F)	Average Yield Strength (psi)	Lower Bound	
			Yield Stress (psi)	Ultimate Strength (psi)
A376 TP316	605	26610	23386	74672
A376 TP316	350	31604	27776	75296
Modulus of Elasticity:	$E = 25.275 \times 10^6$ psi at 605°F; $E = 26.700 \times 10^6$ psi at 350°F			
Poisson's ratio: 0.3				

5.0 CRITICAL LOCATION AND EVALUATION CRITERIA

5.1 CRITICAL LOCATIONS

The leak-before-break (LBB) evaluation margins are to be demonstrated for the critical locations (governing locations). Such locations are established based on the loads (Section 3.0) and the material properties established in Section 4.0. These locations are defined below for the H. B. Robinson RHR line piping. Tables 3-3 as well as Figure 3-1 are used for this evaluation.

<u>Critical Locations</u>

The highest stressed location for the RHR line from hot leg to first valve is at Node 323 (See Table 3-3 and Figure 3-1). The highest stressed location for the RHR line from first valve to second valve is at Node 320 (See Table 3-3 and Figure 3-1). Node 323 is determined to be the critical location at the HL RHR branch nozzle due to the limiting cross-sectional properties of the pipe, rather than the thicker reinforcement area at the nozzle to reactor coolant loop branch weld. Node 323 and Node 320 are the load critical locations for all the weld locations in the RHR line piping.

6.0 LEAK RATE PREDICTIONS

6.1 INTRODUCTION

The purpose of this section is to discuss the method which is used to predict the flow through postulated through-wall cracks and present the leak rate calculation results for through-wall circumferential cracks.

6.2 GENERAL CONSIDERATIONS

The flow of hot pressurized water through an opening to a lower back pressure causes flashing which can result in choking. For long channels where the ratio of the channel length, L, to hydraulic diameter, D_H, (L/D_H) is greater than [

]a,c,e

6.3 CALCULATION METHOD

The basic method used in the leak rate calculations is the method developed by [

]a,c,e

The flow rate through a crack was calculated in the following manner. Figure 6-1 from Reference 6-2 was used to estimate the critical pressure, P_c, for the RHR line enthalpy condition and an assumed flow. Once P_c was found for a given mass flow, the [ ]a,c,e was found from Figure 6-2 (taken from Reference 6-2). For all cases considered, since [

]a,c,e Therefore, this method will yield the two-phase pressure drop due to momentum effects as illustrated in Figure 6-3, where P_o is the operating pressure. Now using the assumed flow rate, G, the frictional pressure drop can be calculated using

$$\Delta P_f = [\qquad]^{a,c,e} \tag{6-1}$$

where the friction factor f is determined using the [ ]a,c,e The crack relative roughness, ε, was obtained from fatigue crack data on stainless steel samples. The relative roughness value used in these calculations was [ ]a,c,e

The frictional pressure drop using equation 6-1 is then calculated for the assumed flow rate and added to the [ ]a,c,e to obtain the total pressure drop from the primary system to the atmosphere.

That is, for the RHR line:

Absolute Pressure - 14.7 = [ ]^{a,c,e} (6-2)

for a given assumed flow rate G. If the right-hand side of equation 6-2 does not agree with the pressure difference between the RHR line and the atmosphere, then the procedure is repeated until equation 6-2 is satisfied to within an acceptable tolerance which in turn leads to flow rate value for a given crack size.

For the single phase cases with lower temperature (350°F at Node 320), leakage rate is calculated by the following equation (Reference 6-4) with crack opening area obtained by the method from Reference 6-3.

$$Q = A(2g\Delta P / k\rho)^{0.5} \quad \text{ft}^3\text{/sec;} \qquad (6\text{-}3)$$

Where, ΔP = pressure difference between stagnation and back pressure (lb/ft^2), g = acceleration of gravity (ft/sec^2), ρ = fluid density at atmospheric pressure (lb/ft^3), k = friction loss including passage loss, inlet and outlet of the through-wall crack, A = crack opening area (ft^2).

6.4 LEAK RATE CALCULATIONS

Leak rate calculations were made as a function of crack length at the governing locations previously identified in Section 5.1. The normal operating loads of Table 3-2 were applied in these calculations. The crack opening areas were estimated using the method of Reference 6-3 and the leak rates were calculated using the formulation described above. The average material properties of Section 4.0 (see Table 4-2) were used for these calculations.

The flaw sizes to yield a leak rate of 10 gpm were calculated at the governing locations and are given in Table 6-1 for H. B. Robinson Unit 2. The flaw sizes so determined are called leakage flaw sizes.

The H. B. Robinson Unit 2 RCS pressure boundary leak detection system meets the intent of Regulatory Guide 1.45. Thus, to satisfy the margin of 10 on the leak rate, the flaw sizes (leakage flaw sizes) are determined which yield a leak rate of 10 gpm.

6.5 REFERENCES

6-1 [

]a,c,e

6-2 M. M, El-Wakil, “Nuclear Heat Transport, International Textbook Company,” New York, N.Y, 1971.

6-3 Tada, H., “The Effects of Shell Corrections on Stress Intensity Factors and the Crack Opening Area of Circumferential and a Longitudinal Through-Crack in a Pipe,” Section II-1, NUREG/CR-3464, September 1983.

6-4 Crane, D. P., “Handbook of Hydraulic Resistance Coefficient.”

Table 6-1 Flaw Sizes Yielding a Leak Rate of 10 gpm for the RHR Line	
Location	**Leakage Flaw Size (in)**
Node 323	6.00
Node 320	5.25

a,c,e

MASS VELOCITY (lb/(sec)(ft^2))

STAGNATION ENTHALPY (10^2 Btu/lb)

Figure 6-1 Analytical Predictions of Critical Flow Rates of Steam-Water Mixtures

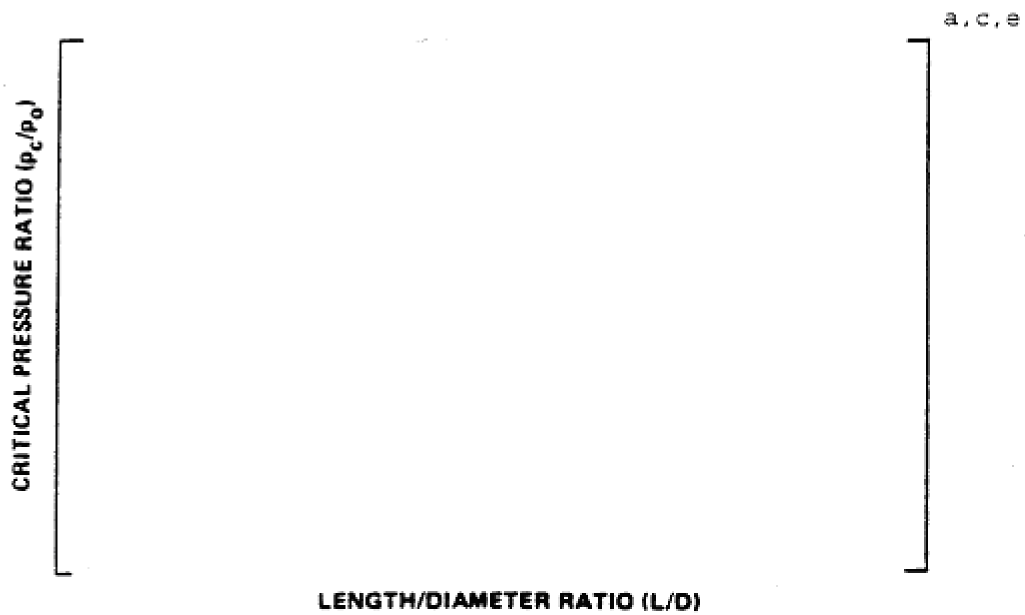


Figure 6-2 [ ]a,c,e Pressure Ratio as a Function of L/D

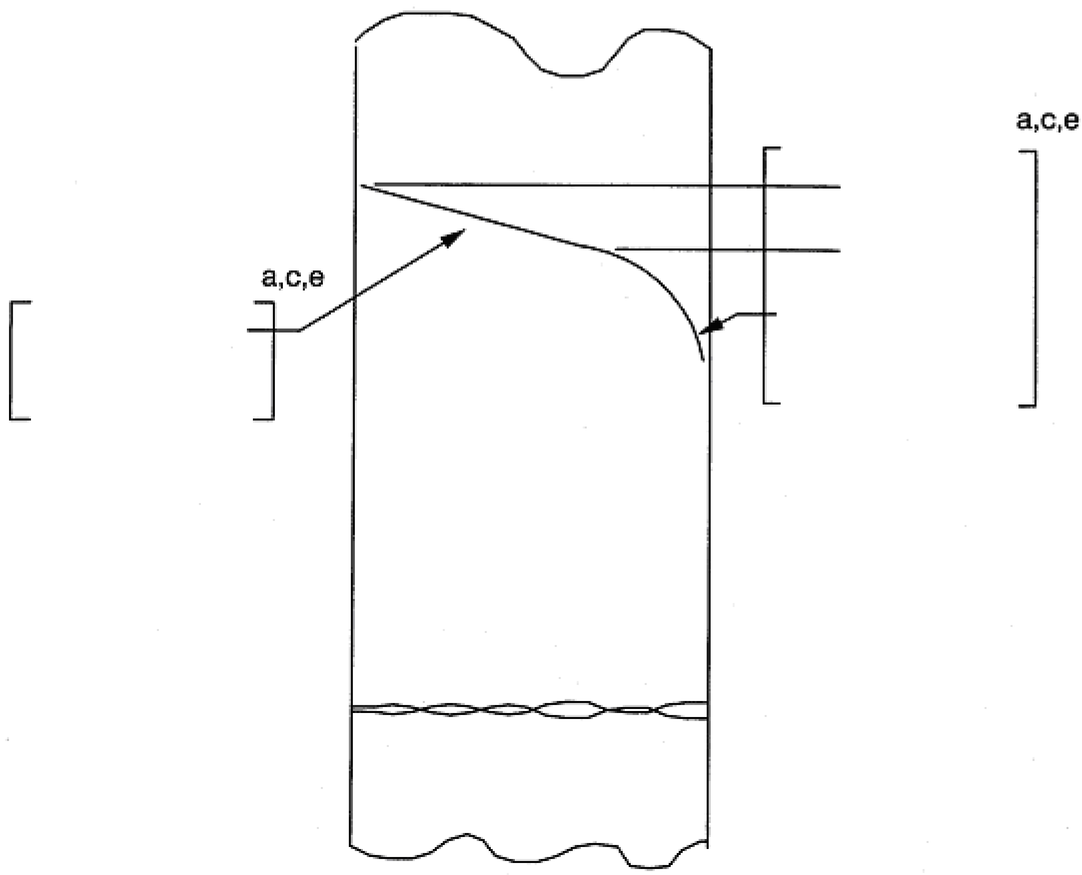


Figure 6-3 Idealized Pressure Drop Profile Through a Postulated Crack

7.0 FRACTURE MECHANICS EVALUATION

7.1 GLOBAL FAILURE MECHANISM

Determination of the conditions which lead to failure in stainless steel should be done with plastic fracture methodology because of the large amount of deformation accompanying fracture. One method for predicting the failure of ductile material is the plastic instability method, based on traditional plastic limit load concepts, but accounting for strain hardening and taking into account the presence of a flaw. The flawed pipe is predicted to fail when the remaining net section reaches a stress level at which a plastic hinge is formed. The stress level at which this occurs is termed as the flow stress. The flow stress is generally taken as the average of the yield and ultimate tensile strength of the material at the temperature of interest. This methodology has been shown to be applicable to ductile piping through a large number of experiments and will be used here to predict the critical flaw size in the primary coolant piping. The failure criterion has been obtained by requiring equilibrium of the section containing the flaw (Figure 7-1) when loads are applied. The detailed development is provided in Appendix A for a through-wall circumferential flaw in a pipe with internal pressure, axial force, and imposed bending moments. The limit moment for such a pipe is given by:

[ ]a,c,e

where:

[

]a,c,e

σ_f = $0.5\,(\sigma_y + \sigma_u)$ = flow stress, psi

[

]a,c,e

The analytical model described above accurately accounts for the piping internal pressure as well as imposed axial force as they affect the limit moment. Good agreement was found between the analytical predictions and the experimental results (Reference 7-1). For application of the limit load methodology, the material, including consideration of the configuration, must have a sufficient ductility and ductile tearing resistance to sustain the limit load.

7.2 RESULTS OF CRACK STABILITY EVALUATION

A stability analysis based on limit load was performed for these locations as described in Section 7.1. The weld process types, at the critical locations at Node 323 and Node 320 are used as GTAW and SMAW combination. The "Z" correction factor for SMAW (References 7-2 and 7-3) are as follows:

$$Z = 1.15\ [1.0 + 0.013\ (OD-4)] \text{ for SMAW}$$

where OD is the outer diameter of the pipe in inches.

The Z-factor for the GTAW weld is 1.0. The Z-factor for the SMAW was calculated for the critical locations, using the dimensions given in Table 3-1. The applied faulted loads (Table 3-3) were increased by the Z factor and plots of limit load versus crack length were generated as shown in Figures 7-2 and 7-3. Lower bound material properties were used from Table 4-2. Table 7-1 summarizes the results of the stability analyses based on limit load. The leakage flaw sizes are also presented in the same table.

7.3 REFERENCES

7-1 Kumar, V., German, M. D. and Shih, C. P., "An Engineering Approach for Elastic-Plastic Fracture Analysis," EPRI Report NP-1931, Project 1237-1, Electric Power Research Institute, July 1981.

7-2 Standard Review Plan; Public Comment Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

7-3 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

Table 7-1 Stability Results for the RHR line Based on Limit Load

Location	Critical Flaw Size (in)	Leakage Flaw Size (in)
Node 323	18.52	6.00
Node 320	19.13	5.25

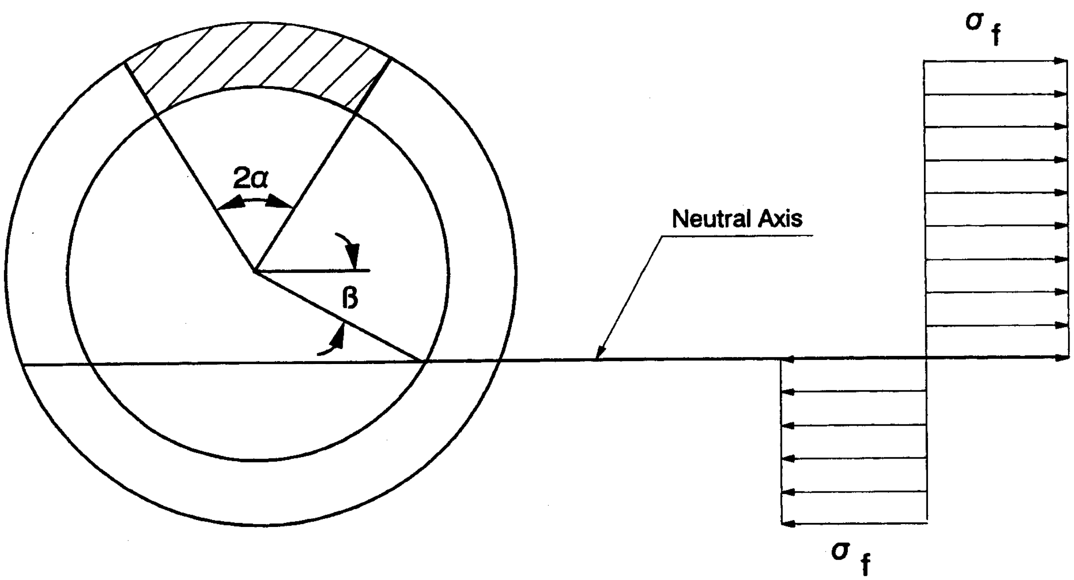


Figure 7-1 [ ]a,c,e Stress Distribution

a,c,e

OD = 14.00 in.	$\sigma_{y\text{-}min}$ = 23.386 ksi	F = 252.725 kips
t = 1.114 in.	$\sigma_{u\text{-}min}$ = 74.672 ksi	M = 590.504 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-2 Critical Flaw Size Prediction – Node 323

a,c,e

OD = 14.00 in.	$\sigma_{y\text{-}min}$ = 27.776 ksi	F = 252.466 kips
t = 1.114 in.	$\sigma_{u\text{-}min}$ = 75.296 ksi	M = 602.162 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-3 Critical Flaw Size Prediction – Node 320

8.0 ASSESSMENT OF FATIGUE CRACK GROWTH

8.1 INTRODUCTION

The fatigue crack growth (FCG) analysis is not a requirement for the LBB analysis (see References 8-1 and 8-2) since the LBB analysis is based on the postulation of a through-wall flaw, whereas the FCG analysis is performed based on the surface flaw. However, a fatigue crack growth (FCG) assessment of the H. B. Robinson Unit 2 RHR line was determined by comparison with a generic fatigue crack growth analysis of a similar piping system. The details of the generic fatigue crack growth analysis are presented below. By comparing the parameters critical to the fatigue crack growth analysis between H. B. Robinson and the generic analysis, it was concluded that the generic analysis would adequately cover the fatigue crack growth of the H. B. Robinson Unit 2 RHR lines.

Due to similarities in Westinghouse PWR designs, it was possible to perform a representative fatigue crack growth calculation which would be applicable to the H. B. Robinson Plant.

8.2 CRITICAL LOCATION FOR FATIGUE CRACK GROWTH ANALYSIS

The weld location at the RCL hot leg nozzle to RHR line (see Figure 8-1) was determined to be the most critical location for the fatigue crack growth evaluation. The nozzle configuration and weld location is shown in Figure 8-1. The geometry of the pipe was identical between the H. B. Robinson Unit 2 and the generic model (14″ Schedule 140). Both analyses used austenitic stainless steel at the critical location.

8.3 DESIGN TRANSIENTS

The transient conditions selected for this evaluation are based on conservative estimates of the magnitude and the frequency of the temperature fluctuations resulting from various operating conditions in the plant. These are representative of the conditions which are considered to occur during plant operation. The fatigue evaluation based on these transients provides confidence that the component is appropriate for its application over the design life of the plant. The normal operating and upset thermal transients were considered for this evaluation. Out of these 20 transients were used in the final fatigue crack growth analysis as listed in Table 8-1.

8.4 STRESS ANALYSIS

A thermal transient stress analysis was performed for a typical plant similar to the H. B. Robinson Unit 2 to obtain the through-wall stress profiles for use in the fatigue crack growth analysis. The generic RHR line design transients described in Section 8.3 were used.

A simplified analysis method was used to develop conservative maximum and minimum linear through wall stress distributions due to thermal transients. In this method, a 1-D computer program was used to perform the thermal analysis to determine the through wall temperature gradients as a function of time. The inside surface stress was calculated by using an equation, which is similar to the transient portion of ASME Section III NB 3600, Equation (11). The effect of discontinuity was included in the analysis by performing a separate 1-D thermal analysis for the pipe and nozzle. The maximum and minimum inside surface stresses were then obtained by searching the inside surface stress values calculated for each time step of the transient solution. The outside surface stresses corresponding to the maximum and minimum inside surface stresses were then calculated by a similar method. The maximum and minimum linear through wall stress distribution for each thermal transient was obtained by joining the corresponding inside and outside surface stresses by a straight line. These two stress profiles are called the maximum and minimum through wall stress distributions respectively, for convenience. The stresses due to the generic pressure and the generic moment loading were then superimposed on the through wall cyclical stresses to obtain the total maximum and minimum stress profile for each transient.

8.5 OBE LOADS

The stresses due to OBE loads were neglected in the fatigue crack growth analysis since these loads are not expected to contribute significantly to crack growth due to the small number of cycles.

8.6 TOTAL STRESS FOR FATIGUE CRACK GROWTH

The total through wall stress at a section was obtained by superimposing the generic pressure stress and the generic moment stresses on the thermal transient stresses. Thus, the total stress for fatigue crack growth at any point is given by the following equation:

Total Stress For Fatigue Crack Growth	=	Stress due to Internal Pressure	+	Stress due to Moment (DW + Thermal Expansion)	+	Thermal Transient Stress

8.7 FATIGUE CRACK GROWTH ANALYSIS

The fatigue crack growth analysis was performed to determine the effect of the design thermal transients tabulated in Table 8-1. The analysis was performed for the critical cross-section identified in Figure 8-1. A range of crack depths was postulated, and each was subjected to the transients in Table 8-1, which included pressure and moment loads.

8.8 ANALYSIS PROCEDURE

The fatigue crack growth analyses presented herein were conducted in the same manner as suggested by Section XI, Appendix A of the ASME Boiler and Pressure Vessel Code (Reference 8-3). The analysis procedure involves assuming an initial flaw exists at some point and predicting the growth of that flaw due to an imposed series of transient stresses. The growth of a crack per loading cycle is dependent on the range of applied stress intensity factor, ΔK_I, by the following:

$$\frac{da}{dN} = C_o \Delta K_I^n \qquad (8\text{-}1)$$

where "C_o" and the exponent "n" are material properties, and ΔK_I is defined later. For inert environments these material properties are constants, but for some water environments they are dependent on the level of mean stress present during the cycle. This can be accounted for by adjusting the value of "C_o" by a function of the ratio of minimum to maximum stress for any given transient, as will be discussed later. Fatigue crack growth properties of stainless steel in a pressurized water environment have been used in the analysis.

The input required for a fatigue crack growth analysis is basically the information necessary to calculate the parameter ΔK_I, which depends on crack and structure geometry and the range of applied stresses in the area where the crack exists. Once ΔK_I is calculated, the growth due to that particular cycle can be calculated by Equation (8-1). This increment of growth is then added to the original crack size, the ΔK_I adjusted, and the analysis proceeds to the next transient. The procedure is continued in this manner until all the transients have been analyzed.

The applied stresses at the flaw locations are resolved into membrane and bending stresses with respect to the wall thickness. Pressure, thermal, and discontinuity stresses are considered in the determination of the K_I factors.

The stress intensity factor at the point of maximum depth is calculated from the membrane and bending stresses using the following equation taken from the ASME Code (Reference 8-3):

$$K_I = \sqrt{\frac{\pi a}{Q}}[\sigma_m M_m + \sigma_b M_b]$$

where : σ_m, σ_b = Membrane and Bending Stress, respectively

a = Minor Semi-Axis (flaw depth)

Q = Flaw Shape Parameter Including A Plastic Zone Correction Factor for Plane Strain Condition

Q = $[\phi_1^2 - 0.212\ (\sigma / \sigma_{ys})^2]$

ϕ_1 = $\int_0^{\pi/2}\left[1-\left(\frac{b^2-a^2}{b^2}\right)\cos^2\Phi\right]^{1/2} d\Phi$

σ_{ys} = Yield Strength of the Material

σ = $\sigma_m + \sigma_b$

b = Major Semi-Axis (Flaw Length/2)

ϕ = Parametric Angle of the Ellipse

M_m = Correction Factor for Membrane Stress

M_b = Correction Factor for Bending Stress

The appropriate values of M_m and M_b as a function of crack geometry can be found in Reference 8-3. The range of stress intensity factor (ΔK_I) for fluctuation of applied stress is determined by first finding the maximum and minimum stress intensity factor ($K_{I\,max}$, $K_{I\,min}$) during a given transient and then calculating the range of stress intensity factor ($\Delta K_I = K_{I\,max} - K_{I\,min}$). At times $K_{I\,min}$ may go below zero, in these cases, $K_{I\,min}$ is set equal to zero before ΔK_I is determined.

Calculation of the fatigue crack growth for each cycle was then carried out using the reference fatigue crack growth rate law determined from consideration of the available data for stainless steel in a pressurized water environment. This law allows for the effect of mean stress or R ratio ($K_{I\,min}/K_{I\,max}$) on the growth rates.

The reference crack growth law used for the stainless steel RHR pipe system was taken from that developed by the Metal Properties Council - Pressure Vessel Research Committee Task Force In Crack Propagation Technology. The reference curve has the equation:

[(8-2)

]^{a,c,e}

This equation appears in Appendix C of ASME Section XI for air environments and its basis is provided in Reference 8-4 in Figure 8-2. For water environments, an environmental factor of [ ]^{a,c,e} was used, based on the crack growth tests in PWR environments reported in Reference 8-5.

8.9 RESULTS

Fatigue crack growth analyses were carried out at the critical cross-section. Analysis was completed for a range of postulated flaw sizes oriented circumferentially, and the results are presented in Table 8-2. The postulated flaws are assumed to have an aspect ratio of six to one. Even for the largest postulated flaw of 0.35 inch, which is about 35 percent of the wall thickness, the results project that the flaw growth through the wall will not occur during the 40/60 year design life of the plant. Transients and cycles for the H. B. Robinson Unit 2 plant for 40-year transient set will remain bounding for 60 years (Reference 8-6), the FCG results shown in Table 8-2 is also applicable for the 60 years.

Therefore, fatigue crack growth should not be a concern for the H. B. Robinson RHR Line.

8.10 REFERENCES

8-1 Standard Review Plan; Public Comment Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

8-2 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

8-3 ASME Boiler and Pressure Vessel Code Section XI, 2007 Edition with the 2008 Addenda, "Rules for Inservice Inspection of Nuclear Power Plant Components."

8-4 James, L. A., and Jones, D. P., "Fatigue Crack Growth Correlations for Austenitic Stainless Steel in Air," in Predictive Capabilities in Environmentally Assisted Cracking," ASME publication PVP-99, Dec. 1985.

8-5 Bamford, W. H., "Fatigue Crack Growth of Stainless Steel Piping in a Pressurized Water Reactor Environment," Trans ASME, Journal of Pressure Vessel Technology, Feb. 1979. Engineering Development Labs Report HEDL-TME-76-43, May 1976.

8-6 NUREG-1785, "Safety Evaluation Report Related to the License Renewal of H. B. Robinson Steam Electric Plant, Unit 2."

Table 8-1 Design Transients Considered for Fatigue Crack Growth Evaluation

Trans. No.	Description	No. of Occurrences
1	Unit Loading	13,200
2	Unit Unloading	13,200
3	Step Load Increase	2,000
4	Step Load Decrease	2,000
5	Large Step Load Decrease with Steam Dump	200
6	Feedwater Cycling	2000
7	Unit Loading Between 0 and 15% Power	500
8	Unit Unloading Between 0 and 15% Power	500
9	Loss of Load	80
10	Loss of Power	40
11	Partial Loss of Flow-Dead Loop	80
12	Partial Loss of Flow-Active Loop	80
13	Reactor Trip with no Inadvertent Cooldown	230
14	Reactor Trip with Cooldown; No Safety Injection	160
15	Reactor Trip with Cooldown Actuating Safety Injection	10
16	Inadvertent RCS Depressurization	20
17	Control Rod Drop	80
18	Inadvertent Safety Injection	60
19	Turbine Roll Test	20
20	Steady-State and Random Fluctuations	3.2×10^6

a,c,e

Table 8-2 RHR Lines Fatigue Crack Growth Results				
Initial	Crack Depth (in) After			
Crack Depth (in)	10 Years	20 Years	30 Years	40/60 Years*

Note: *Because transients and cycles for the H. B. Robinson Unit 2 plant for 40-year transient set will remain bounding for 60 years, the FCG results shown in Table 8-2 are also applicable for 60 years.

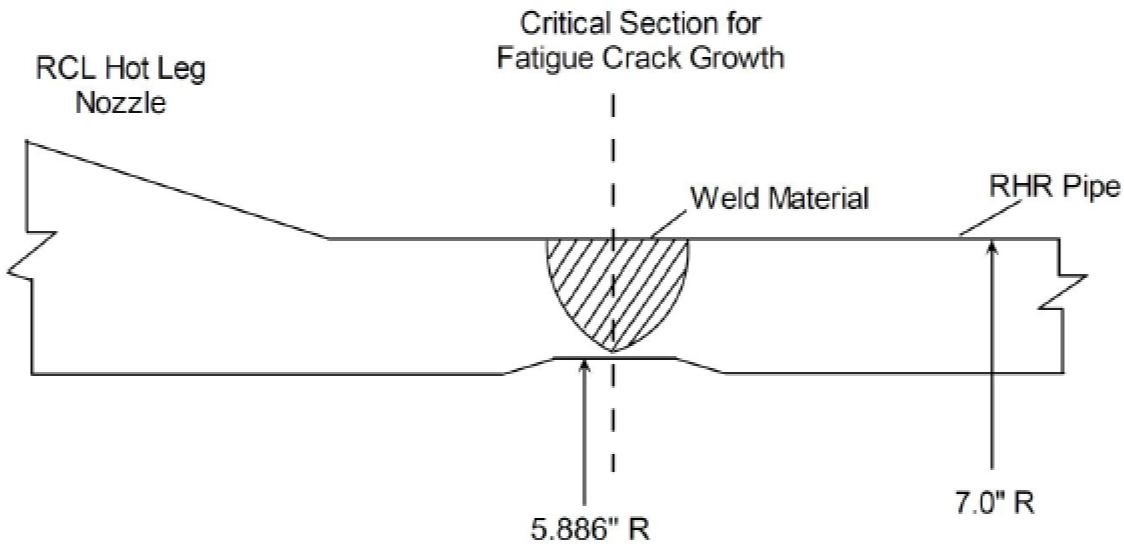


Figure 8-1 Schematic of RHR Line at RCL Hot Leg Nozzle Weld Location

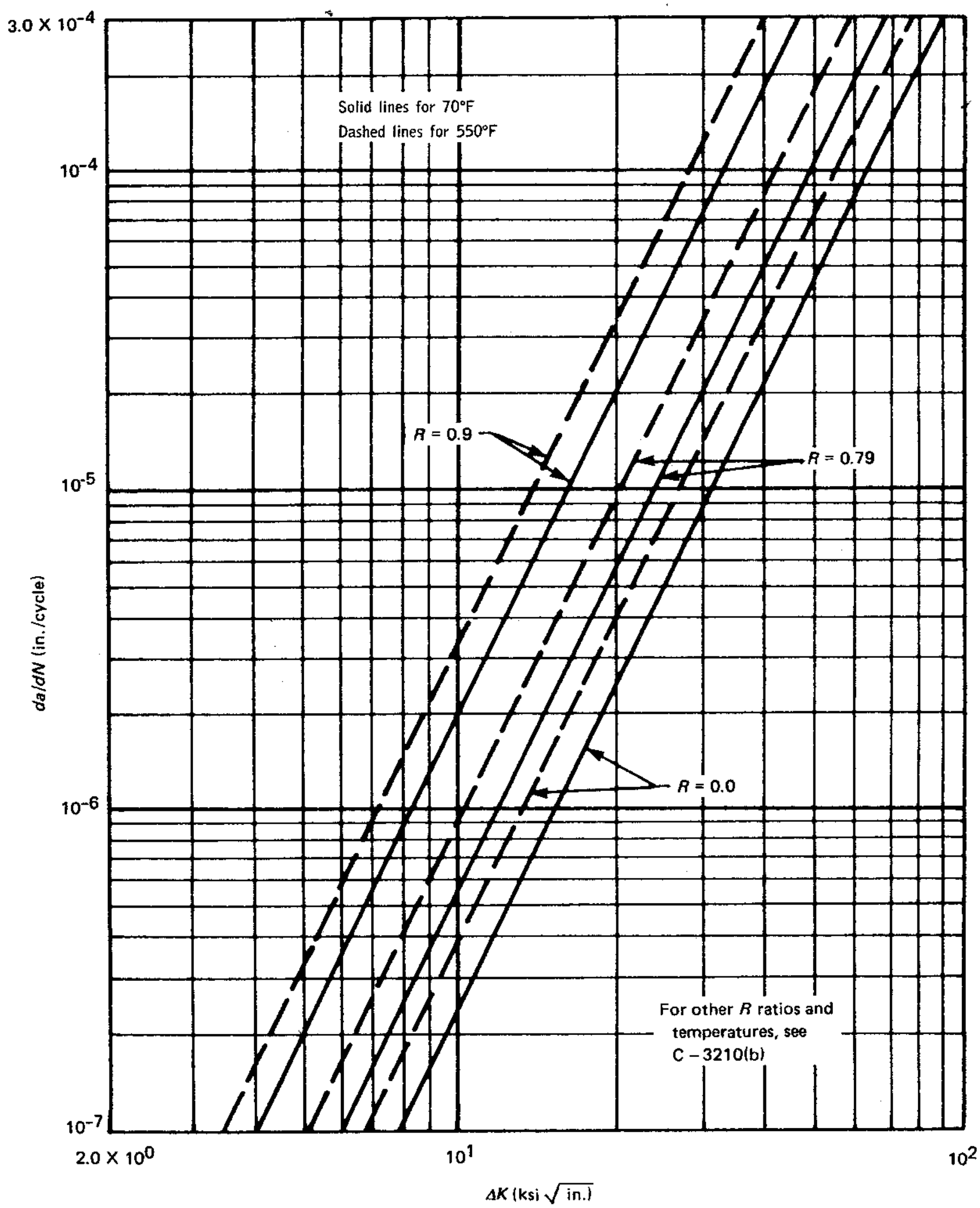


Figure 8-2 Reference Crack Growth Curves for Stainless Steel in Air Environments

9.0 ASSESSMENT OF MARGINS

The results of the leak rates of Section 6.4 and the corresponding stability evaluations of Section 7.2 are used in performing the assessment of margins. Margins are shown in Table 9-1. All the LBB recommended margins are satisfied.

In summary, at all the critical locations relative to:

1. Flaw Size - Using faulted loads obtained by the absolute sum method, a margin of 2 or more exists between the critical flaw and the flaw having a leak rate of 10 gpm (the leakage flaw).

2. Leak Rate - A margin of 10 exists between the calculated leak rate from the leakage flaw and the plant leak detection capability of 1 gpm.

3. Loads - At the critical locations the leakage flaw was shown to be stable using the faulted loads obtained by the absolute sum method (i.e., a flaw twice the leakage flaw size is shown to be stable; hence the leakage flaw size is stable). A margin of 1 on loads using the absolute summation of faulted load combinations is satisfied.

Table 9-1 Leakage Flaw Sizes, Critical Flaw Sizes and Margins for RHR Line

Location	Critical Flaw Size (in)	Leakage Flaw Size (in)	Margin
Node 323	18.52	6.00	3.1
Node 320	19.13	5.25	3.6

10.0 CONCLUSIONS

This report justifies the elimination of RHR line pipe break from the structural design basis for the H. B. Robinson Unit 2 during the 60 years plant life as follows:

a. Stress corrosion cracking is precluded by use of fracture resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation.

b. Water hammer should not occur in the primary loop piping and connected Class 1 piping because of system design, testing, and operational considerations.

c. The effects of low and high cycle fatigue on the integrity of the RHR line are negligible.

d. Ample margin exists between the leak rate of small stable flaws and the capability of the H. B. Robinson Unit 2 reactor coolant system pressure boundary Leakage Detection System.

e. Ample margin exists between the small stable flaw sizes of item (d) and larger stable flaws.

f. Ample margin exists in the material properties used to demonstrate end-of-service life (fully aged) stability of the critical flaws.

g. Fatigue crack growth results using the 40-year design transients and cycles (shown to be applicable for 60 years) show that there will be insignificant growth through the wall for the license renewal period (60-year plant life).

For the critical locations, flaws are identified that will be stable because of the ample margins described in d, e, and f above.

Based on loading, pipe geometry and material properties considerations, enveloping critical (governing) locations were determined at which leak-before-break crack stability evaluations were made. Through-wall flaw sizes were postulated which would cause a leak at a rate of ten (10) times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, fatigue crack growth was shown not to be an issue for the RHR line piping. Therefore, the Leak-Before-Break conditions and margins are satisfied for the H. B. Robinson Unit 2 RHR line piping. It is demonstrated that the dynamic effects of the pipe rupture resulting from postulated breaks in the RHR line piping need not be considered in the structural design basis of H. B. Robinson Unit 2 for the 60-years.

APPENDIX A: LIMIT MOMENT

[

] a,c,e

a,c,e

Figure A-1 Pipe with a Through-Wall Crack in Bending

ATTACHMENT 4

WCAP-17779-NP, REVISION 1, "TECHNICAL JUSTIFICATION FOR ELIMINATING ACCUMULATOR LINE RUPTURE AS THE STRUCTURAL DESIGN BASIS FOR H. B. ROBINSON UNIT 2”, MARCH 2023

(REDACTED)

WCAP-17779-NP
Revision 1

March 2023

Technical Justification for Eliminating Accumulator Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2

Westinghouse

WCAP-17779-NP
Revision 1

Technical Justification for Eliminating Accumulator Line Rupture as the Structural Design Basis for H. B. Robinson Unit 2

March 2023

Author: Nadia B. Petkova *
Operating Plants Piping and Supports

Reviewer: Momo Wiratmo*
Operating Plants Piping and Supports

Approved: Lynn A. Patterson, Manager*
Reactor Vessel and Containment
Vessel Design and Analysis

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

RECORD OF REVISIONS

Revision	Date	Revision Description
0	August 2013	Original Issue (WCAP-17779-NP). This is the non-proprietary class 3 version of WCAP-17779-P, Revision 0.
1	March 2023	This is the non-proprietary class 3 version of WCAP-17779-P, Revision 1.

TABLE OF CONTENTS

1.0 INTRODUCTION 1-1
1.1 PURPOSE 1-1
1.2 SCOPE AND OBJECTIVES 1-1
1.3 REFERENCES 1-2

2.0 OPERATION AND STABILITY OF THE REACTOR COOLANT SYSTEM 2-1
2.1 STRESS CORROSION CRACKING 2-1
2.2 WATER HAMMER 2-2
2.3 LOW CYCLE AND HIGH CYCLE FATIGUE 2-2
2.4 REFERENCES 2-3

3.0 PIPE GEOMETRY AND LOADING 3-1
3.1 CALCULATION OF LOADS AND STRESSES 3-1
3.2 LOADS FOR LEAK RATE EVALUATION 3-1
3.3 LOAD COMBINATION FOR CRACK STABILITY ANALYSES 3-2
3.4 REFERENCES 3-2

4.0 MATERIAL CHARACTERIZATION 4-1
4.1 ACCUMULATOR LINE PIPING AND WELD MATERIALS 4-1
4.2 TENSILE PROPERTIES 4-1
4.3 REFERENCE 4-1

5.0 CRITICAL LOCATION AND EVALUATION CRITERIA 5-1
5.1 CRITICAL LOCATIONS 5-1

6.0 LEAK RATE PREDICTIONS 6-1
6.1 INTRODUCTION 6-1
6.2 GENERAL CONSIDERATIONS 6-1
6.3 CALCULATION METHOD 6-1
6.4 LEAK RATE CALCULATIONS 6-2
6.5 REFERENCES 6-3

7.0 FRACTURE MECHANICS EVALUATION 7-1
7.1 GLOBAL FAILURE MECHANISM 7-1
7.2 RESULTS OF CRACK STABILITY EVALUATION 7-2
7.3 REFERENCES 7-2

8.0 ASSESSMENT OF FATIGUE CRACK GROWTH 8-1
8.1 INTRODUCTION 8-1
8.2 CRITICAL LOCATION FOR FATIGUE CRACK GROWTH ANALYSIS 8-1
8.3 DESIGN TRANSIENTS 8-1
8.4 STRESS ANALYSIS 8-1
8.5 OBE LOADS 8-2
8.6 TOTAL STRESS FOR FATIGUE CRACK GROWTH 8-2

8.7 FATIGUE CRACK GROWTH ANALYSIS 8-2
8.7.1 ANALYSIS PROCEDURE 8-2
8.8 RESULTS 8-3
8.9 REFERENCES 8-4

9.0 ASSESSMENT OF MARGINS 9-1

10.0 CONCLUSIONS 10-1

APPENDIX A: LIMIT MOMENT A-1

LIST OF TABLES

Table 3-1 Summary of H. B. Robinson Nuclear Power Plant Piping Geometry and Normal Operating Condition for Accumulator Lines Loop A, Loop B and Loop C...3-3

Table 3-2 Summary of Robinson Unit 2 Normal Loads and Stresses for Accumulator Line Loop A.....3-4

Table 3-3 Summary of Robinson Unit 2 Normal Loads and Stresses for Accumulator Line Loop B......3-5

Table 3-4 Summary of Robinson Unit 2 Normal Loads and Stresses for Accumulator Line Loop C 3-6

Table 3-5 Summary of Robinson Unit 2 Faulted Loads and Stresses for Accumulator Line Loop A 3-7

Table 3-6 Summary of Robinson Unit 2 Faulted Loads and Stresses for Accumulator Line Loop B 3-8

Table 3-7 Summary of Robinson Unit 2 Faulted Loads and Stresses for Accumulator Line Loop C 3-9

Table 4-1 Measured Tensile Properties for the Accumulator Line Material A376 TP316........................4-2

Table 4-2 Mechanical Properties for Accumulator Line Material at Operating Temperatures................4-2

Table 6-1 Flaw Sizes Yielding a Leak Rate of 10 gpm for the Accumulator Line 6-4

Table 7-1 Stability Results for the Accumulator Lines Based on Limit Load 7-3

Table 8-1 Design Transients Considered for Fatigue Crack Growth Evaluation.....................................8-5

Table 8-2 Accumulator Line Fatigue Crack Growth Results..8-6

Table 9-1 Leakage Flaw Sizes, Critical Flaw Sizes and Margins for Accumulator Line 9-2

LIST OF FIGURES

Figure 3-1 H. B. Robinson Unit 2 Loop A Accumulator Line Layout 3-10
Figure 3-2 H. B. Robinson Unit 2 Loop B Accumulator Line Layout…………………………...3-11
Figure 3-3 H. B. Robinson Unit 2 Loop C Accumulator Line Layout 3-12
Figure 6-1 Analytical Predictions of Critical Flow Rates of Steam-Water Mixtures 6-5
Figure 6-2 [ ]a,c,e Pressure Ratio as a Function of L/D 6-6
Figure 6-3 Idealized Pressure Drop Profile Through a Postulated Crack..6-7
Figure 7-1 [ ]a,c,e Stress Distribution 7-4
Figure 7-2 Critical Flaw Size Prediction – Node 409 Loop C..7-5
Figure 7-3 Critical Flaw Size Prediction – Node 360 Loop B...……………………………………..7-5
Figure 7-4 Critical Flaw Size Prediction – Node 364 Loop B..7-7
Figure 7-5 Critical Flaw Size Prediction – Node 3811 Loop C..7-8
Figure 8-1 Schematic of 10″ Accumulator Line at RCL Cold Leg Nozzle Weld Location.............8-7
Figure 8-2 Reference Crack Growth Curves for Stainless Steel in Air Enviroments......................8-8
Figure A-1 Pipe with a Through-Wall Crack in Bending A-2

1.0 INTRODUCTION

1.1 PURPOSE

The current structural design basis for the H. B. Robinson Unit 2, 10" accumulator lines (from the cold legs Loop A, Loop B and Loop C) and attached 8" line connected to 10" accumulator lines require postulating non-mechanistic circumferential and longitudinal pipe breaks. This results in additional plant hardware (e.g., pipe whip restraints and jet shields) which would mitigate the dynamic consequences of the pipe breaks. It is, therefore, highly desirable to be realistic in the postulation of pipe breaks for the accumulator lines. Presented in this report are the descriptions of a mechanistic pipe break evaluation method and the analytical results that can be used for establishing that a circumferential type of break will not occur within the accumulator lines. This report includes the 10" accumulator lines (from the cold legs Loop A, Loop B and Loop C) and attached 8" line connected to the 10" accumulator lines (see Figures in Section 3), for convenient purpose throughout the report it is called as accumulator line.

1.2 SCOPE AND OBJECTIVES

The scope of this report is limited to the accumulator line. Schematic drawings of the piping system are shown in Section 3. The recommendations and criteria proposed in SRP 3.6.3 (References 1-1 and 1-2) are used in this evaluation. The criteria and the resulting steps of the evaluation procedure can be briefly summarized as follows:

1. Calculate the applied loads. Identify the location(s) at which the highest faulted stress occurs.

2. Identify the materials and the material properties.

3. Postulate a surface flaw governing location. Determine fatigue crack growth. Show that a through-wall crack will not result.

4. Postulate a through-wall flaw at the governing location(s). The size of the flaw should be large enough so that the leakage is assured of detection with margin using the installed leak detection equipment when the pipe is subjected to normal operating loads. Demonstrate that there is a margin of 10 between the calculated leak rate and the leak detection capability.

5. Using maximum faulted loads in the stability analysis, demonstrate that there is a margin of 2 between the leakage size flaw and the critical size flaw.

6. Review the operating history to ascertain that operating experience has indicated no particular susceptibility to failure from the effects of corrosion, water hammer or low and high cycle fatigue.

7. For the materials types used in the Plant, provide representative material properties.

This report provides a fracture mechanics demonstration of accumulator line piping integrity for H. B. Robinson Unit 2 consistent with the NRC position for exemption from consideration of postulated pipe rupture dynamic effects (Reference 1-3).

It should be noted that the terms “flaw” and “crack” have the same meaning and are used interchangeably. “Governing location” and “critical location” are also used interchangeably throughout the report.

Note that there are several locations in this report where proprietary information has been identified and bracketed. For each of the bracketed locations, the reason for the proprietary classification is given using a standardized system. The proprietary brackets are labeled with three different letters, to provide this information, and the explanation for each letter is given below:

a. The information reveals the distinguishing aspects of a process or component, structure, tool, method, etc., and the prevention of its use by Westinghouse’s competitors, without license from Westinghouse, gives Westinghouse a competitive economic advantage.

b. The information, if used by a competitor, would reduce the competitor’s expenditure of resources or improve the competitor’s advantage in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. The information reveals aspects of past, present, or future Westinghouse or customer-funded development plans and programs of potential commercial value to Westinghouse.

The proprietary information in the brackets which has been deleted in this version of this report are provided in the proprietary class 2 document (WCAP-17779-P, Revision 1).

1.3 REFERENCES

1-1 Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday August 28, 1987/Notices, pp. 32626-32633.

1-2 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

1-3 Nuclear Regulatory Commission, 10 CFR 50, Modification of General Design Criteria 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, Final Rule, Federal Register/Vol. 52, No. 207/Tuesday, October 27, 1987/Rules and Regulations, pp. 41288-41295.

2.0 OPERATION AND STABILITY OF THE REACTOR COOLANT SYSTEM

2.1 STRESS CORROSION CRACKING

The Westinghouse reactor coolant system primary loops and attached class 1 piping have an operating history that demonstrates the inherent operating stability characteristics of the design. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking (IGSCC)). This operating history totals over 1400 reactor-years, including 16 plants each having over 30 years of operation, 10 other plants each with over 25 years of operation, 11 plants each with over 20 years of operation and 12 plants each with over 15 years of operation.

In 1978, the United States Nuclear Regulatory Commission (USNRC) formed the second Pipe Crack Study Group. (The first Pipe Crack Study Group (PCSG) established in 1975 addressed cracking in boiling water reactors only.) One of the objectives of the second PCSG was to include a review of the potential for stress corrosion cracking in Pressurized Water Reactors (PWRs). The results of the study performed by the PCSG were presented in NUREG-0531 (Reference 2-1) entitled "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants." In that report the PCSG stated:

> "The PCSG has determined that the potential for stress-corrosion cracking in PWR primary system piping is extremely low because the ingredients that produce IGSCC are not all present. The use of hydrazine additives and a hydrogen overpressure limit the oxygen in the coolant to very low levels. Other impurities that might cause stress-corrosion cracking, such as halides or caustic, are also rigidly controlled. Only for brief periods during reactor shutdown when the coolant is exposed to the air and during the subsequent startup are conditions even marginally capable of producing stress-corrosion cracking in the primary systems of PWRs. Operating experience in PWRs supports this determination. To date, no stress corrosion cracking has been reported in the primary piping or safe ends of any PWR."

For stress corrosion cracking (SCC) to occur in piping, the following three conditions must exist simultaneously: high tensile stresses, susceptible material, and a corrosive environment. Since some residual stresses and some degree of material susceptibility exist in any stainless steel piping, the potential for stress corrosion is minimized by properly selecting a material immune to SCC as well as preventing the occurrence of a corrosive environment. The material specifications consider compatibility with the system's operating environment (both internal and external) as well as 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 (e.g., sulfides, sulfites, and thionates). Strict pipe cleaning standards prior to operation and careful control of water chemistry during plant operation are used to prevent the occurrence of a corrosive environment. Prior to being put into service, the piping is cleaned internally and externally. During flushes and preoperational testing, water chemistry is controlled in accordance with written specifications.

Requirements on chlorides, fluorides, conductivity, and pH are included in the acceptance criteria for the piping.

During plant operation, the reactor coolant water chemistry is monitored and maintained within very specific limits. Contaminant concentrations are kept below the thresholds known to be conducive to stress corrosion cracking with the major water chemistry control standards being included in the plant operating procedures as a condition for plant operation. For example, during normal power operation, oxygen concentration in the RCS is expected to be in the ppb range by controlling charging flow chemistry and maintaining hydrogen in the reactor coolant at specified concentrations. Halogen concentrations are also stringently controlled by maintaining concentrations of chlorides and fluorides within the specified limits. Thus, during plant operation, the likelihood of stress corrosion cracking is minimized.

During 1979, several instances of cracking in PWR feedwater piping led to the establishment of the third PCSG. The investigations of the PCSG reported in NUREG-0691 (Reference 2-2) further confirmed that no occurrences of IGSCC have been reported for PWR primary coolant systems.

Primary Water Stress Corrosion Cracking (PWSCC) occurred in V. C. Summer reactor vessel hot leg nozzle, Alloy 82/182 weld. It should be noted that this susceptible material is not found at the H. B. Robinson Unit 2 accumulator line.

2.2 WATER HAMMER

Overall, there is a low potential for water hammer in the RCS and connecting accumulator lines since they are designed and operated to preclude the voiding condition in normally filled lines. The RCS and connecting accumulator lines including piping and components are designed for normal, upset, emergency, and faulted condition transients. The design requirements are conservative relative to both the number of transients and their severity. Relief valve actuation and the associated hydraulic transients following valve opening are considered in the system design. Other valve and pump actuations are relatively slow transients with no significant effect on the system dynamic loads. To ensure dynamic system stability, reactor coolant parameters are stringently controlled. Temperature during normal operation is maintained within a narrow range by the control rod positions; pressure is controlled also within a narrow range for steady-state conditions by the pressurizer heaters and pressurizer spray. The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters, namely system resistance and the reactor coolant pump characteristics are controlled in the design process. Additionally, Westinghouse has instrumented typical reactor coolant systems to verify the flow and vibration characteristics of the system and the connecting auxiliary lines. Preoperational testing and operating experience have verified the Westinghouse approach. The operating transients of the RCS primary piping and connected accumulator lines are such that no significant water hammer can occur.

2.3 LOW CYCLE AND HIGH CYCLE FATIGUE

An assessment of the low cycle fatigue loadings is discussed in the form of a fatigue crack growth assessment, in Section 8.0.

Pump vibrations during operation would result in high cycle fatigue loads in the piping system. During operation, an alarm signals the exceedance of the RC pump shaft vibration limits. Field measurements have been made on the reactor coolant loop piping in a number of plants during hot functional testing. Stresses in the elbow below the RC pump have been found to be very small, between 2 and 3 ksi at the highest. Field measurements on typical PWR plants indicate vibration amplitudes less than 1 ksi. When translated to the connecting accumulator lines, these stresses would be even lower, well below the fatigue endurance limit for the accumulator line materials and would result in an applied stress intensity factor below the threshold for fatigue crack growth.

2.4 REFERENCES

2-1 Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants, NUREG-0531, U.S. Nuclear Regulatory Commission, February 1979.

2-2 Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors, NUREG-0691, U.S. Nuclear Regulatory Commission, September 1980.

3.0 PIPE GEOMETRY AND LOADING

3.1 CALCULATION OF LOADS AND STRESSES

The stresses due to axial loads and bending moments are calculated by the following equation:

$$\sigma = \frac{F}{A} + \frac{M}{Z} \tag{3-1}$$

where,

σ = stress

F = axial load

M = moment

A = pipe cross-sectional area

Z = section modulus

The moments for the desired loading combinations are calculated by the following equation:

$$M = \sqrt{M_x^2 + M_y^2 + M_z^2} \tag{3-2}$$

where,

M_x = X component of moment, Torsion

M_y = Y component of bending moment

M_z = Z component of bending moment

The axial load and moments for leak rate predictions and crack stability analyses are computed by the methods to be explained in Sections 3.3 and 3.4.

3.2 LOADS FOR LEAK RATE EVALUATION

The normal operating loads for leak rate predictions are calculated by the following equations:

$$F = F_{DW} + F_{TH} + F_P \tag{3-3}$$

$$M_X = (M_X)_{DW} + (M_X)_{TH} \tag{3-4}$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{TH} \tag{3-5}$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{TH} \tag{3-6}$$

The subscripts of the above equations represent the following loading cases:

DW = deadweight

TH = normal thermal expansion

P = load due to internal pressure

This method of combining loads is often referred to as the algebraic sum method (References 3-1 and 3-2).

The as-built dimensions and normal operating conditions are given in Table 3-1. The minimum pipe wall thickness at the weld counterbore is used in the analysis, which is based on the maximum allowed counterbore at a butt weld (Reference 3-3). The loads based on this method of combination are provided in Tables 3-2 to 3-4 at all the weld locations identified in Figures 3-1 to 3-3.

3.3 LOAD COMBINATION FOR CRACK STABILITY ANALYSES

In accordance with Standard Review Plan 3.6.3 (References 3-1 and 3-2), the absolute sum of loading components can be applied which results in higher magnitude of combined loads. If crack stability is demonstrated using these loads, the LBB margin on loads can be reduced from √2 to 1.0. The absolute summation of loads is shown in the following equations:

$$F = |F_{DW}| + |F_{TH}| + |F_P| + |F_{SSEINERTIA}| + |F_{SSEAM}| \quad (3\text{-}7)$$

$$M_X = |(M_X)_{DW}| + |(M_X)_{TH}| + |(M_X)_{SSEINERTIA}| + |(M_X)_{SSESAM}| \quad (3\text{-}8)$$

$$M_Y = |(M_Y)_{DW}| + |(M_Y)_{TH}| + |(M_Y)_{SSEINERTIA}| + |(M_Y)_{SSEAM}| \quad (3\text{-}9)$$

$$M_Z = |(M_Z)_{DW}| + |(M_Z)_{TH}| + |(M_Z)_{SSEINERTIA}| + |(M_Z)_{SSEAM}| \quad (3\text{-}10)$$

where subscript SSEINERTIA refers to safe shutdown earthquake inertia, SSEAM is safe shutdown earthquake anchor motion.

The loads so determined are used in the fracture mechanics evaluations (Section 7.0) to demonstrate the LBB margins at the locations established to be the governing locations. These loads at all the weld locations (see Figures 3-1 to 3-3) are given in Tables 3-5 to 3-7.

3.4 REFERENCES

3-1 Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

3-2 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

3-3 American National Standards, Butt Welding Ends, ANSI B16.25-1979.

Table 3-1 Summary of H. B. Robinson Nuclear Power Plant Piping Geometry and Normal Operating Condition for Accumulator Lines Loop A, Loop B and Loop C

Loop	Weld Location Nodes (See Figures 3-1, 3-2 and 3-3)	Outer Diameter (in)	Minimum Wall Thickness (in)	Normal Operating	
				Pressure (psig)	Temperature (°F)
A	449 to 4470	10.750	0.896	2235	547
A	446 to 433	10.750	0.896	1500	280
A	43 to 4321	8.625	0.650	1500	280
A	4500 to 4550	10.750	0.896	1500	280
A	4551 to 4570	10.750	0.896	660	140
B	3421 to 3440	8.625	0.650	1500	280
B	354 to 3620	10.750	0.896	1500	280
B	3621 to 3640	10.750	0.896	660	140
B	3450 to 3500	10.750	0.896	1500	280
B	3510 to 353	10.750	0.896	2235	547
C	3782 to 3811	8.625	0.650	1500	280
C	38 to 4080	10.750	0.896	1500	280
C	4081 to 410	10.750	0.896	2235	547
C	383 to 384	10.750	0.896	1500	280
C	3851 to 3870	10.750	0.896	660	140

Table 3-2 Summary of Robinson Unit 2 Normal Loads and Stresses for Accumulator Line Loop A

Weld Location Node	Axial Force (lbs)	Moment (in-lbs)	Axial Stress (psi)	Moment Stress (psi)	Total Stress (psi)
449	138403	142003	4992	2249	7242
4480	138496	101836	4996	1613	6609
448	138895	89770	5010	1422	6432
4470	138895	83139	5010	1317	6327
446	92561	103376	3339	1638	4976
445	92561	179516	3339	2844	6182
444	92696	173839	3344	2754	6097
441	92696	355914	3344	5638	8982
440	92561	357607	3339	5665	9004
439	96537	327912	3482	5194	8677
437	95882	119449	3459	1892	5351
4341	95653	237310	3450	3759	7209
4331	94085	152532	3394	2416	5810
433	94078	149182	3393	2363	5757
43	62728	146149	3854	4838	8692
4321	62728	139785	3854	4627	8481
4500	94470	119442	3408	1892	5300
450	94634	130785	3413	2072	5485
451	97073	141437	3501	2240	5742
452	95800	136998	3456	2170	5626
453	96110	138587	3467	2195	5662
4550	96110	132529	3467	2099	5566
4551	43158	126048	1557	1997	3553
456	43158	125275	1557	1984	3541
457	42119	120578	1519	1910	3429
4570	42177	148761	1521	2356	3878

Table 3-3 Summary of Robinson Unit 2 Normal Loads and Stresses for Accumulator Line Loop B

Weld Location Node	Axial Force (lbs)	Moment (in-lbs)	Axial Stress (psi)	Moment Stress (psi)	Total Stress (psi)
3421	61438	47848	3775	1584	5359
343	50855	81499	3124	2698	5822
344	62368	91412	3832	3026	6858
3440	62410	93381	3834	3091	6926
354	90007	332774	3247	5271	8518
355	104571	448633	3772	7107	10879
357	80751	579267	2913	9176	12089
358	80853	598731	2916	9484	12401
359	79845	805253	2880	12756	15636
360	90007	900522	3247	14265	17512
361	80751	651617	2913	10322	13235
3620	80751	573146	2913	9079	11992
3621	27799	169356	1003	2683	3685
363	27799	320980	1003	5085	6087
364	58797	749985	2121	11880	14001
3640	58855	688251	2123	10902	13025
3450	91206	280065	3290	4436	7726
348	97910	123191	3532	1951	5483
349	91172	121410	3289	1923	5212
3500	91172	68726	3289	1089	4377
3510	137506	135966	4960	2154	7114
352	137506	161774	4960	2563	7523
353	137503	170760	4960	2705	7665

Table 3-4 Summary of Robinson Unit 2 Normal Loads and Stresses for Accumulator Line Loop C

Weld Location Node	Axial Force (lbs)	Moment (in-lbs)	Axial Stress (psi)	Moment Stress (psi)	Total Stress (psi)
3782	57397	337881	3526	11185	14712
379	69042	344978	4242	11420	15662
380	62720	275322	3853	9114	12968
3811	62250	437154	3825	14472	18296
38	93519	505161	3373	8002	11375
390	93386	201953	3368	3199	6568
391	95608	258517	3449	4095	7544
392	95608	267108	3449	4231	7680
393	97139	282209	3504	4470	7974
395	92013	149457	3319	2368	5686
396	88256	189171	3183	2997	6180
398	88832	150942	3204	2391	5595
399	88256	370601	3183	5871	9054
400	91541	365139	3302	5784	9086
401	89045	254825	3212	4037	7249
403	89045	376076	3212	5957	9169
404	89695	244909	3235	3880	7115
406	90679	331474	3271	5251	8522
407	101032	450150	3644	7131	10775
4080	101032	473827	3644	7506	11150
4081	147365	539677	5316	8549	13864
409	147366	558488	5316	8847	14162
4100	142698	495409	5147	7848	12995
410	142617	454623	5144	7202	12346
383	100765	469596	3635	7439	11073
384	89084	527800	3213	8361	11574
3851	36133	413573	1303	6551	7855
386	36131	398282	1303	6309	7612
387	43822	298125	1581	4723	6303
3870	43880	265969	1583	4213	5796

Table 3-5 Summary of Robinson Unit 2 Faulted Loads and Stresses for Accumulator Line Loop A

Weld Location Node	Axial Force (lbs)	Moment (in-lbs)	Axial Stress (psi)	Moment Stress (psi)	Total Stress (psi)
449	144964	410234	5229	6498	11727
4480	144871	386831	5226	6128	11353
448	144025	378465	5195	5995	11190
4470	143969	347372	5193	5503	10696
446	97268	313285	3509	4963	8471
445	97214	379897	3507	6018	9524
444	99502	397398	3589	6295	9884
441	97299	396402	3510	6279	9789
440	97268	398964	3509	6320	9828
439	97247	368204	3508	5833	9340
437	96901	253120	3495	4010	7505
4341	99013	472207	3571	7480	11052
4331	97719	546337	3525	8654	12179
433	97717	517809	3525	8203	11727
43	66394	491441	4079	16269	20348
4321	66394	421965	4079	13969	18048
4500	96324	457613	3474	7249	10723
450	96290	459802	3473	7284	10757
451	106295	428914	3834	6794	10628
452	105022	210080	3788	3328	7116
453	98958	225244	3569	3568	7138
4550	98960	277698	3570	4399	7969
4551	46077	293946	1662	4656	6318
456	46093	296695	1663	4700	6362
457	43540	272316	1570	4314	5884
4570	43599	258522	1573	4095	5668

Table 3-6 Summary of Robinson Unit 2 Faulted Loads and Stresses for Accumulator Line Loop B					
Weld Location Node	**Axial Force (lbs)**	**Moment (in-lbs)**	**Axial Stress (psi)**	**Moment Stress (psi)**	**Total Stress (psi)**
3421	67378	228446	4140	7562	11702
343	77663	294457	4771	9748	14519
344	65497	248602	4024	8230	12254
3440	65463	335167	4022	11095	15117
354	100313	511033	3618	8095	11714
355	109422	601889	3947	9534	13481
357	109282	722323	3942	11442	15384
358	111280	706172	4014	11186	15200
359	112303	965649	4051	15297	19347
360	100411	1077485	3622	17068	20690
361	109285	788847	3942	12496	16438
3620	109288	697501	3942	11049	14991
3621	56596	284563	2041	4508	6549
363	56670	435221	2044	6894	8938
364	61918	903679	2233	14315	16548
3640	61978	829002	2236	13132	15368
3450	99554	421100	3591	6671	10262
348	100896	350121	3639	5546	9186
349	100385	340525	3621	5394	9015
3500	100466	293526	3624	4650	8274
3510	147033	409511	5304	6487	11791
352	147069	468711	5305	7425	12730
353	145972	571906	5265	9059	14325

Table 3-7 Summary of Robinson Unit 2 Faulted Loads and Stresses for Accumulator Line Loop C

Weld Location Node	Axial Force (lbs)	Moment (in-lbs)	Axial Stress (psi)	Moment Stress (psi)	Total Stress (psi)
3782	71908	584336	4418	19344	23762
379	71899	595281	4417	19706	24124
380	74965	497991	4606	16486	21091
3811	74483	662826	4576	21942	26518
38	105746	757192	3814	11995	15809
390	96980	278708	3498	4415	7913
391	97597	329686	3520	5222	8743
392	97618	340846	3521	5399	8920
393	98446	364469	3551	5773	9324
395	98532	249644	3554	3955	7509
396	103096	305489	3719	4839	8558
398	101887	257901	3675	4085	7760
399	101887	535666	3675	8485	12160
400	98309	500908	3546	7935	11481
401	100892	409096	3639	6480	10120
403	101512	450033	3662	7129	10790
404	101169	480368	3649	7609	11259
406	100491	530129	3625	8398	12022
407	102086	608055	3682	9632	13314
4080	102090	626090	3682	9918	13600
4081	148633	710349	5361	11253	16614
409	148676	754046	5363	11945	17308
4100	145202	710407	5237	11253	16491
410	145283	665487	5240	10542	15782
383	101925	821895	3676	13019	16696
384	103802	712679	3744	11289	15034
3851	51072	569239	1842	9017	10859
386	51117	520793	1844	8250	10094
387	48608	438125	1753	6940	8694
3870	48551	389013	1751	6162	7914

Notes (for Table 3-2 to 3-7):

a. See Figures 3-1 to 3-3 for weld locations

b. Axial force included pressure

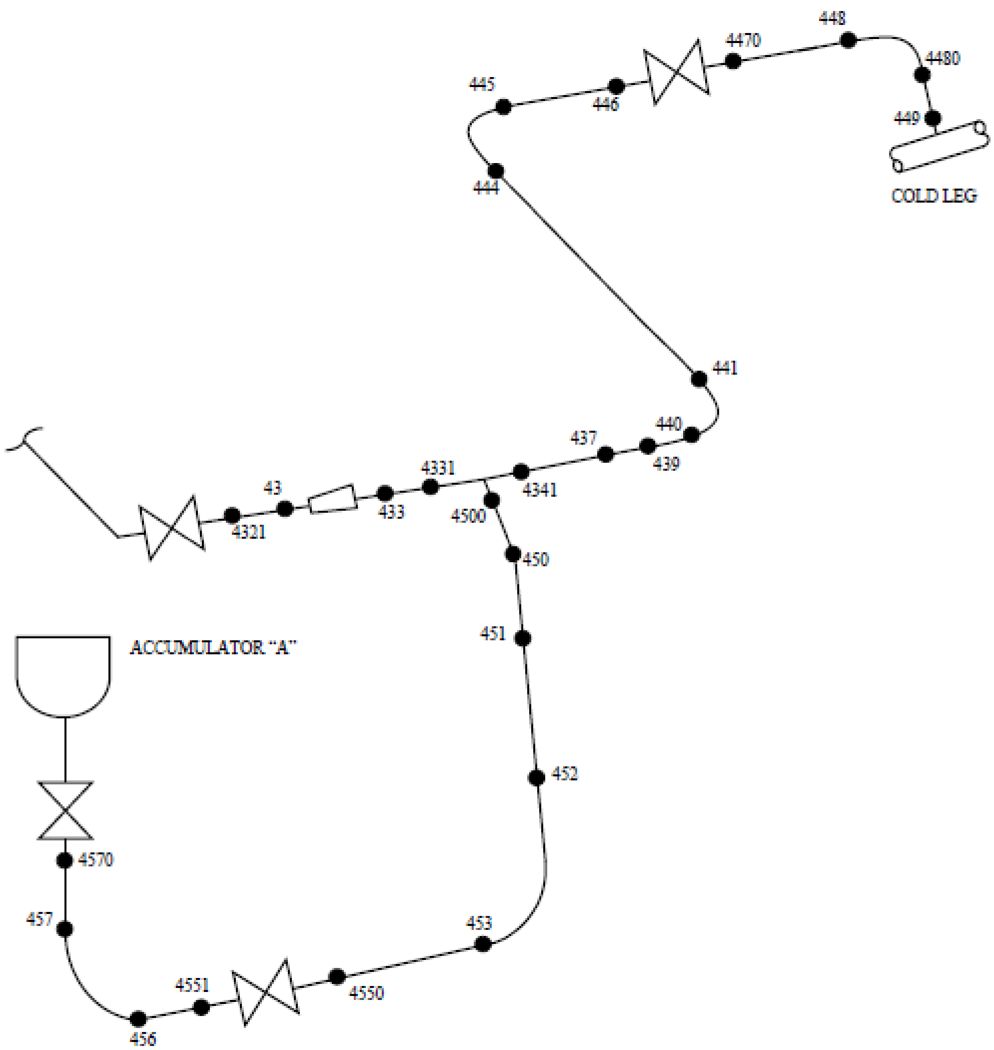


Figure 3-1 H. B. Robinson Unit 2 Loop A Accumulator Line Layout

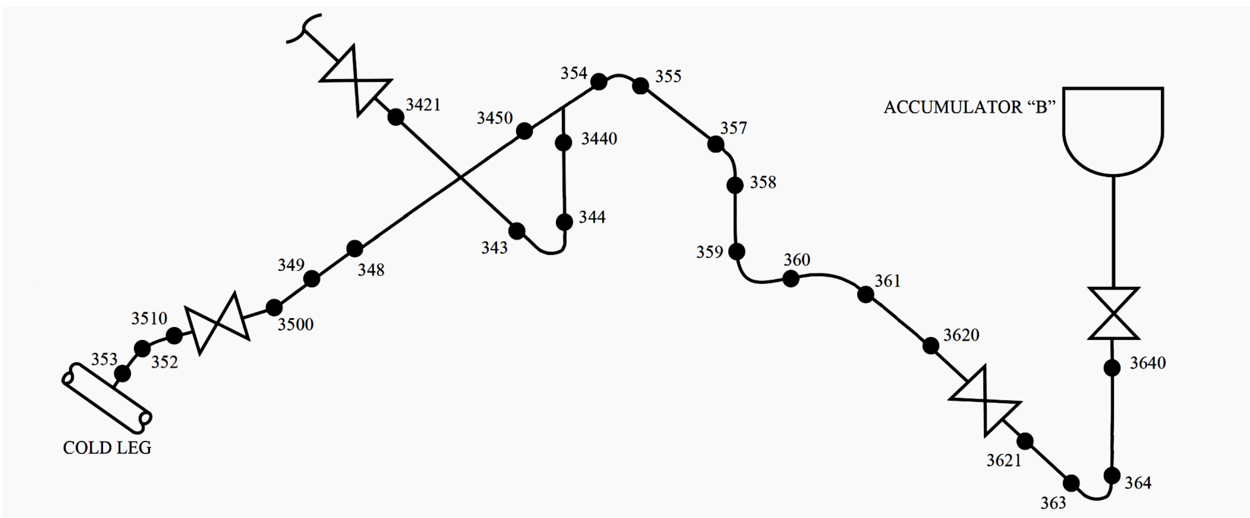


Figure 3-2 H. B. Robinson Unit 2 Loop B Accumulator Line Layout

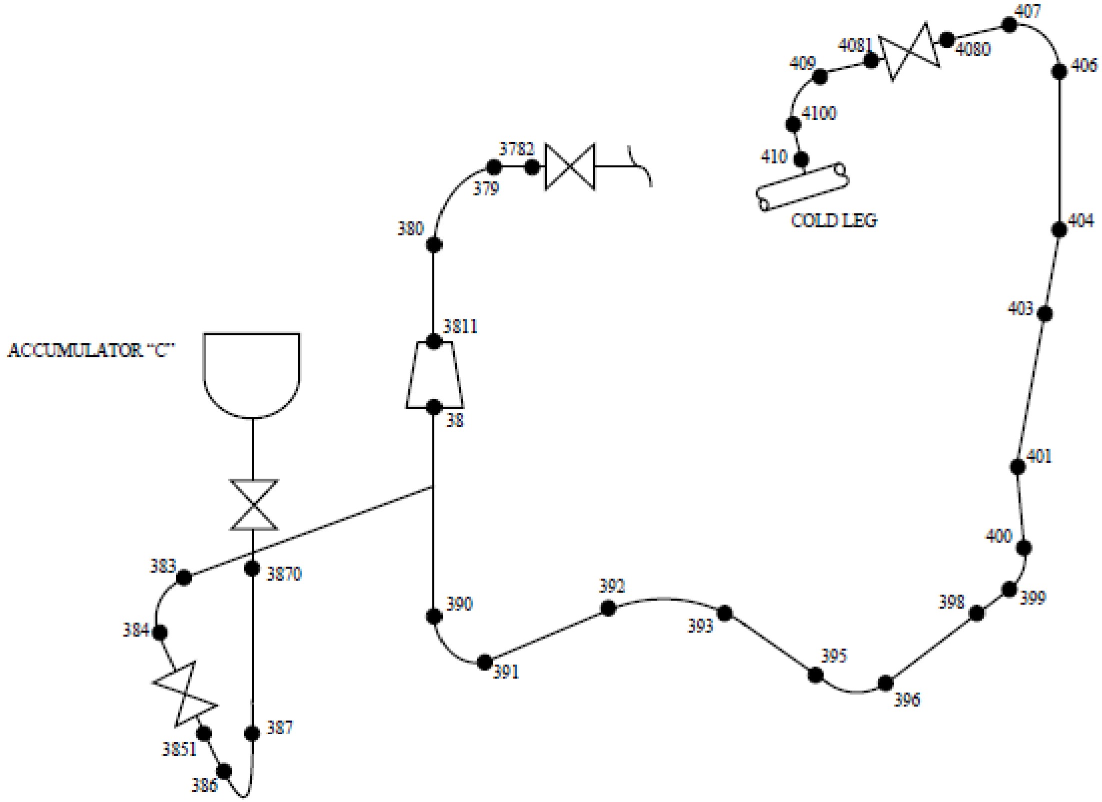


Figure 3-3 H. B. Robinson Unit 2 Loop C Accumulator Line Layout

4.0 MATERIAL CHARACTERIZATION

4.1 ACCUMULATOR LINE PIPING AND WELD MATERIALS

The material type of the accumulator line for H. B. Robinson Unit 2 is A376 TP316. This is a wrought product of the type used for the piping in several PWR plants. The accumulator line system does not include any cast pipes or cast fittings. The welding processes used are Gas Tungsten Arc Weld (GTAW) and Shielded Metal Arc Weld (SMAW) combination. Figures 3-1 to 3-3 show the schematic layout of the accumulator lines and also identify the weld location by node points.

In the following sections the tensile properties of the materials are presented for use in the Leak-Before-Break analyses.

4.2 TENSILE PROPERTIES

The Certified Materials Test Reports (CMTRs) for the H. B. Robinson Unit 2 accumulator lines were used to establish the tensile properties for the leak-before-break analyses. The tensile properties for the pipe material are provided in Table 4-1 for H. B. Robinson Unit 2.

For the A376 TP316 pipe material, the representative properties at operating temperatures of 140°F, 280°F and 547°F are established from the tensile properties at room temperature given in Table 4-1 by utilizing Section II of the 2007 Edition with the 2008 Addenda of the ASME Boiler and Pressure Vessel Code (Reference 4-1). Code tensile properties at the operating temperatures were obtained by interpolating between 70°F, 200°F, 300°F, 500°F and 600°F tensile Code properties. Ratios of the Code tensile properties at the operating temperatures to the corresponding properties were then applied to the room temperature tensile properties obtained from CMTRs (Table 4-1) to obtain the H. B. Robinson Unit 2 accumulator lines specific properties at operating temperatures of 140°F, 280°F and 547°F.

The average and lower bound yield strengths and ultimate strengths for the pipe material are tabulated in Table 4-2. The ASME Code modulus of elasticity values are also given, and Poisson's ratio was taken as 0.3.

4.3 REFERENCE

4-1 ASME Boiler and Pressure Vessel Code, 2007 Edition with the 2008 Addenda, Section II, Part D – Properties (Customary) Materials.

Table 4-1 Measured Tensile Properties for the Accumulator Line Material A376 TP316		
Heat Number	**Yield Strength (psi) Room Temp.**	**Ultimate Strength (psi) Room Temp.**
DXNS(139284)	36000	78800
DXNS(139284)	36000	78800
80275	35270	76950
48899	33710	81360
48977	31860	78940
48993	33850	79650
49096	32570	79790
80275	32860	78370
49065*	32570	76520
49083*	33990	76810

Note: *Applicable to the 8 inch line

Table 4-2 Mechanical Properties for Accumulator Line Material at Operating Temperatures				
			Lower Bound	
Material	**Temperature (°F)**	**Average Yield Strength (psi)**	**Yield Stress (psi)**	**Ultimate Strength (psi)**
A376 TP316	547	22090	20691	73667
A376 TP316	280	27099	25382	75226
A376 TP316	140	31512	29515	76950
A376 TP316	280*	26513	25947	74806
Modulus of Elasticity:	E = 25.618 x 10^6 psi at 547°F ; E = 27.100 x 10^6 psi at 280°F; E = 27.869 x 10^6 psi at 140°F			
Poisson's ratio: 0.3				

Note: *Applicable to the 8 inch line

5.0 CRITICAL LOCATION AND EVALUATION CRITERIA

5.1 CRITICAL LOCATIONS

The leak-before-break (LBB) evaluation margins are to be demonstrated for the critical locations (governing locations). Such locations are established based on the loads (Section 3.0) and the material properties established in Section 4.0. These locations are defined below for the H. B. Robinson accumulator lines. Tables 3-1 and 3-5 to 3-7 as well as Figures 3-1 to 3-3 are used for this evaluation.

Critical Locations

All the welds in the accumulator line are fabricated using the GTAW/SMAW combination. The pipe material type is A376 TP 316. The governing locations were established on the basis of the pipe geometry, material type, operating temperature, operating pressure, and the highest faulted stresses at the welds. Figures 3-1 to 3-3 show the schematic layout of the accumulator lines. The nozzle to reactor coolant loop branch welds is not shown in these figures, but based on the increased reinforcement thickness at these branch welds, the resulting faulted stress would not be limiting compared to the pipe weld thickness of the critical locations shown.

Critical Locations for the 10 inch Accumulator lines:

The highest faulted stress location is at Node 409 Loop C with temperature 547°F and pressure 2235 psig. The highest faulted stress location is at Node 360 Loop B with temperature 280°F and pressure 1500 psig. The highest faulted stress location is at Node 364 Loop B with temperature 140°F and pressure 660 psig. Therefore, Node 409 Loop C, Node 360 Loop B and Node 364 Loop B are the critical locations.

Critical Location for the 8 inch lines:

Highest faulted stress location is Node 3811 Loop C with temperature 280°F and pressure 1500 psig. Therefore, Node 3811 Loop C is the critical location.

6.0 LEAK RATE PREDICTIONS

6.1 INTRODUCTION

The purpose of this section is to discuss the method which is used to predict the flow through postulated through-wall cracks and present the leak rate calculation results for through-wall circumferential cracks.

6.2 GENERAL CONSIDERATIONS

The flow of hot pressurized water through an opening to a lower back pressure causes flashing which can result in choking. For long channels where the ratio of the channel length, L, to hydraulic diameter, D_H, (L/D_H) is greater than [

]^{a,c,e}

6.3 CALCULATION METHOD

The basic method used in the leak rate calculations is the method developed by [

]^{a,c,e}

The flow rate through a crack was calculated in the following manner. Figure 6-1 from Reference 6-2 was used to estimate the critical pressure, P_c, for the accumulator line enthalpy condition and an assumed flow. Once P_c was found for a given mass flow, the [

]^{a,c,e} Therefore, this method will yield the two-phase pressure drop due to momentum effects as illustrated in Figure 6-3, where P_o is the operating pressure. Now using the assumed flow rate, G, the frictional pressure drop can be calculated using

$$\Delta P_f = [\qquad\qquad]^{a,c,e} \tag{6-1}$$

where the friction factor f is determined using the [ ]^{a,c,e} The crack relative roughness, ε, was obtained from fatigue crack data on stainless steel samples. The relative roughness value used in these calculations was [ ]^{a,c,e}

The frictional pressure drop using equation 6-1 is then calculated for the assumed flow rate and added to the [ ]^{a,c,e} to obtain the total pressure drop from the primary system to the atmosphere.

That is, for the accumulator line:

Absolute Pressure - 14.7 = [ ]^{a,c,e} (6-2)

for a given assumed flow rate G. If the right-hand side of equation 6-2 does not agree with the pressure difference between the accumulator line and the atmosphere, then the procedure is repeated until equation 6-2 is satisfied to within an acceptable tolerance which in turn leads to flow rate value for a given crack size.

6.4 LEAK RATE CALCULATIONS

Leak rate calculations were made as a function of crack length at the governing locations previously identified in Section 5.1. The normal operating loads of Tables 3-2 through Table 3-4 were applied, in these calculations. The crack opening areas were estimated using the method of Reference 6-3 and the leak rates were calculated using the formulation described above. The average material properties of Section 4.0 (see Table 4-2) were used for these calculations.

For the single phase cases with lower temperature, leakage rate is calculated by the following equation (Reference 6-4) with crack opening area obtained by the method from Reference 6-3.

$$Q = A(2g\Delta P / k\rho)^{0.5} \quad \text{ft}^3\text{/sec;} \qquad (6\text{-}3)$$

Where, ΔP = pressure difference between stagnation and back pressure (lb/ft^2), g = acceleration of gravity (ft/sec^2), ρ = fluid density at atmospheric pressure (lb/ft^3), k = friction loss including passage loss, inlet and outlet of the through-wall crack, A = crack opening area (ft^2).

The flaw sizes to yield a leak rate of 10 gpm were calculated at the governing locations and are given in Table 6-1 for H. B. Robinson Unit 2. The flaw sizes so determined are called leakage flaw sizes.

The H. B. Robinson Unit 2 RCS pressure boundary leak detection system meets the intent of Regulatory Guide 1.45. Thus, to satisfy the margin of 10 on the leak rate, the flaw sizes (leakage flaw sizes) are determined which yield a leak rate of 10 gpm.

6.5 REFERENCES

6-1 [

]a,c,e

6-2 M. M, El-Wakil, "Nuclear Heat Transport, International Textbook Company," New York, N.Y, 1971.

6-3 Tada, H., "The Effects of Shell Corrections on Stress Intensity Factors and the Crack Opening Area of Circumferential and a Longitudinal Through-Crack in a Pipe," Section II-1, NUREG/CR-3464, September 1983.

6-4 Crane, D. P., "Handbook of Hydraulic Resistance Coefficient."

Table 6-1 Flaw Sizes Yielding a Leak Rate of 10 gpm for the Accumulator Line

Location	Leakage Flaw Size (in)
Node 409 Loop C	3.20
Node 360 Loop B	2.91
Node 364 Loop B	4.21
Node 3811 Loop C*	2.52

Note: *Applicable to the 8 inch line

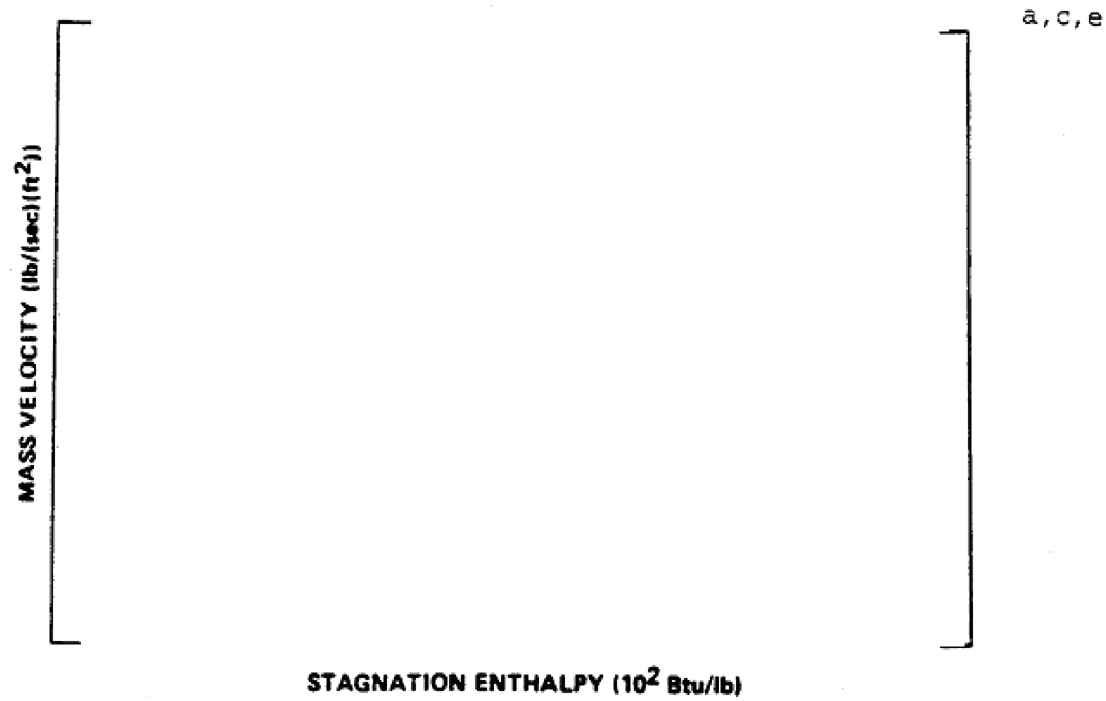


Figure 6-1 Analytical Predictions of Critical Flow Rates of Steam-Water Mixtures

a,c,e

CRITICAL PRESSURE RATIO (p_c/p_o)

LENGTH/DIAMETER RATIO (L/D)

Figure 6-2 [ ]^{a,c,e} Pressure Ratio as a Function of L/D

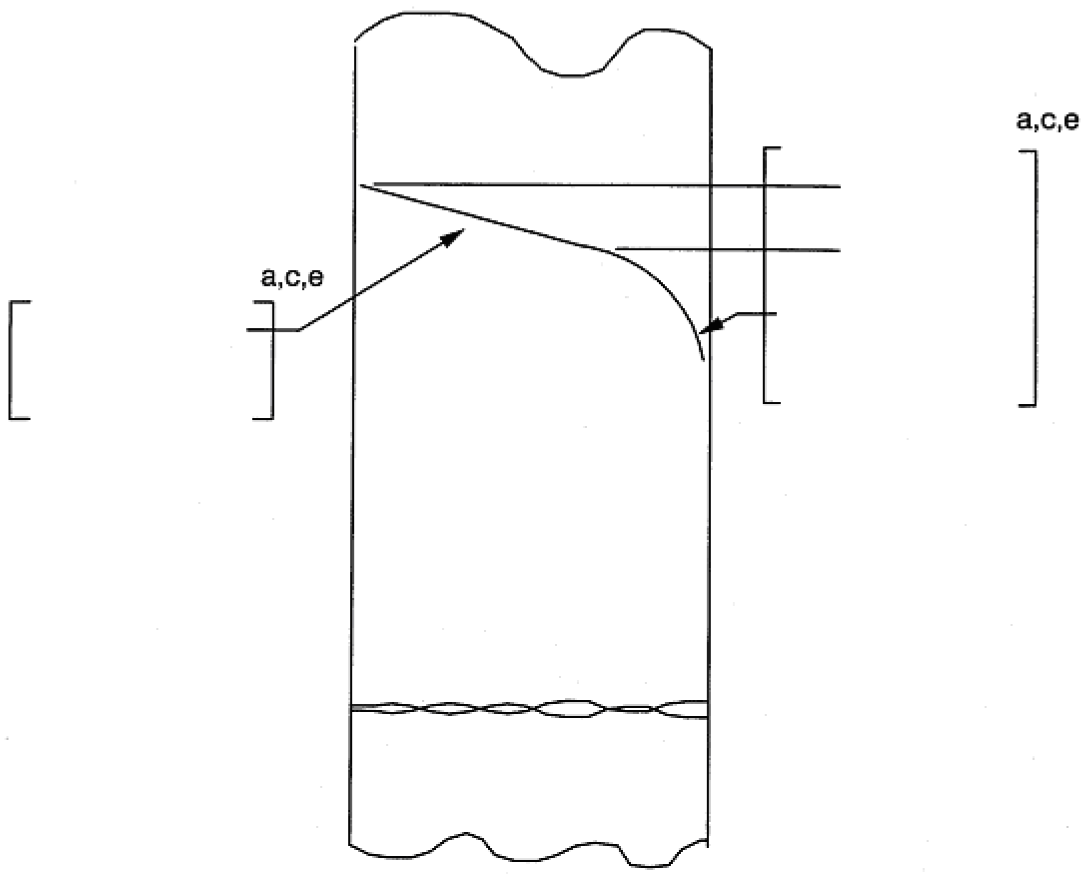


Figure 6-3 Idealized Pressure Drop Profile Through a Postulated Crack

7.0 FRACTURE MECHANICS EVALUATION

7.1 GLOBAL FAILURE MECHANISM

Determination of the conditions which lead to failure in stainless steel should be done with plastic fracture methodology because of the large amount of deformation accompanying fracture. One method for predicting the failure of ductile material is the plastic instability method, based on traditional plastic limit load concepts, but accounting for strain hardening and taking into account the presence of a flaw. The flawed pipe is predicted to fail when the remaining net section reaches a stress level at which a plastic hinge is formed. The stress level at which this occurs is termed as the flow stress. The flow stress is generally taken as the average of the yield and ultimate tensile strength of the material at the temperature of interest. This methodology has been shown to be applicable to ductile piping through a large number of experiments and will be used here to predict the critical flaw size in the primary coolant piping. The failure criterion has been obtained by requiring equilibrium of the section containing the flaw (Figure 7-1) when loads are applied. The detailed development is provided in Appendix A for a through-wall circumferential flaw in a pipe with internal pressure, axial force, and imposed bending moments. The limit moment for such a pipe is given by:

[ ]a,c,e

where:

[

]a,c,e

σ_f = $0.5 (\sigma_y + \sigma_u)$ = flow stress, psi

[

]a,c,e

The analytical model described above accurately accounts for the piping internal pressure as well as imposed axial force as they affect the limit moment. Good agreement was found between the analytical predictions and the experimental results (Reference 7-1). For application of the limit load methodology, the material, including consideration of the configuration, must have a sufficient ductility and ductile tearing resistance to sustain the limit load.

7.2 RESULTS OF CRACK STABILITY EVALUATION

A stability analysis based on limit load was performed for these locations as described in Section 7.1. The weld process types at the critical locations are used as GTAW and SMAW combination. The "Z" correction factor for SMAW (References 7-2 and 7-3) are as follows:

$$Z = 1.15\ [1.0 + 0.013\ (OD\text{-}4)] \text{ for SMAW}$$

where OD is the outer diameter of the pipe in inches.

The Z-factor for the GTAW weld is 1.0. The Z-factor for the SMAW was calculated for the critical locations, using the dimensions given in Table 3-1. The applied faulted loads (Table 3-5 through Table 3-7) were increased by the Z factor and plots of limit load versus crack length were generated as shown in Figure 7-2 through Figure 7-5. Lower bound material properties were used from Table 4-2. Table 7-1 summarizes the results of the stability analyses based on limit load. The leakage flaw sizes are also presented in the same table.

7.3 REFERENCES

7-1 Kanninen, M. F., et. al., "Mechanical Fracture Predictions for Sensitized Stainless Steel Piping with Circumferential Cracks," EPRI NP-192, September 1976.

7-2 Standard Review Plan; Public Comment Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

7-3 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

Table 7-1 Stability Results for the Accumulator Lines Based on Limit Load

Location	Critical Flaw Size (in)	Leakage Flaw Size (in)
Node 409 Loop C	11.53	3.20
Node 360 Loop B	11.17	2.91
Node 364 Loop B	13.19	4.21
Node 3811 Loop C*	7.62	2.52

Note: *Applicable to the 8 inch line

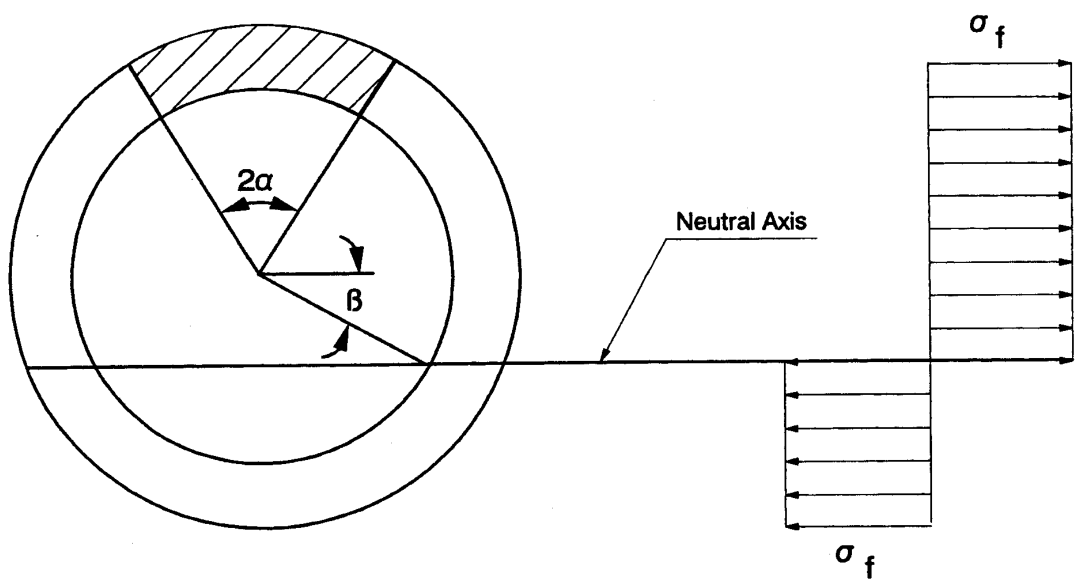


Figure 7-1 [ ][a,c,e] Stress Distribution

a,c,e

OD = 10.75 in.	$\sigma_{y\text{-}min}$ = 20.691 ksi	F = 148.676 kips
t = 0.896 in.	$\sigma_{u\text{-}min}$ = 73.667 ksi	M = 754.046 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-2 Critical Flaw Size Prediction – Node 409 Loop C

a,c,e

OD = 10.75 in.	$\sigma_{y\text{-}min}$ = 25.382 ksi	F = 100.411 kips
t = 0.896 in.	$\sigma_{u\text{-}min}$ = 75.226 ksi	M = 1077.485 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-3 Critical Flaw Size Prediction – Node 360 Loop B

a,c,e

OD = 10.75 in.	$\sigma_{y\text{-}min}$ = 29.515 ksi	F = 61.918 kips
t = 0.896 in.	$\sigma_{u\text{-}min}$ = 76.950 ksi	M = 903.679 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-4 Critical Flaw Size Prediction – Node 364 Loop B

a,c,e

OD = 8.625 in.	$\sigma_{y\text{-}min}$ = 25.947ksi	F = 74.483 kips
t = 0.650 in.	$\sigma_{u\text{-}min}$ = 74.806 ksi	M = 662.826 in-kips
A376 TP316 with SMAW Weld		

Note: OD = outer diameter, t = thickness

Figure 7-5 Critical Flaw Size Prediction – Node 3811 Loop C

8.0 ASSESSMENT OF FATIGUE CRACK GROWTH

8.1 INTRODUCTION

The fatigue crack growth (FCG) analysis is not a requirement for the LBB analysis. The LBB analysis is based on the postulation of through-wall flaw, whereas the FCG analysis is performed based on the surface flaw. However, a fatigue crack growth (FCG) assessment of the H. B. Robinson Unit 2 accumulator lines was determined by comparison with a generic fatigue crack growth analysis of a similar piping system. The details of the generic fatigue crack growth analysis are presented below. By comparing the parameters critical to the fatigue crack growth analysis between H. B. Robinson and the generic analysis, it was concluded that the generic analysis would adequately cover the fatigue crack growth of the H. B. Robinson Unit 2 accumulator lines.

Due to similarities in Westinghouse PWR designs, it was possible to perform a representative fatigue crack growth calculation which would be applicable to H. B. Robinson Unit 2.

8.2 CRITICAL LOCATION FOR FATIGUE CRACK GROWTH ANALYSIS

The weld location at the RCL cold leg nozzle to accumulator pipe was determined to be the most critical location for the fatigue crack growth evaluation. The nozzle configuration and weld location are shown in Figure 8-1. The geometry of the accumulator pipe was identical between the H. B. Robinson Unit 2 and the generic model (10″). Both analyses used austenitic stainless steel at the critical location.

8.3 DESIGN TRANSIENTS

The transient conditions selected for this evaluation are based on conservative estimates of the magnitude and the frequency of the temperature fluctuations documented in various operating plant reports. These are representative of the conditions which are considered to occur during plant operation. The normal operating and upset thermal transients, in accordance with the design specification and the applicable system design criteria document, were considered for this evaluation. Out of these, 15 transients were used in the fatigue crack growth analysis and are listed in Table 8-1. There are some differences between the generic transients used in the fatigue crack growth evaluation and the H. B. Robinson Unit 2 transients but these differences will have insignificant impact on the fatigue crack growth results.

8.4 STRESS ANALYSIS

A thermal transient stress analysis was performed for a typical plant similar to H. B. Robinson Unit 2 to obtain the through-wall stress profiles for use in the fatigue crack growth analysis. The generic accumulator line design transients described in Section 8.3 were used.

A simplified analysis method was used to develop conservative maximum and minimum linear through wall stress distributions due to minor thermal transients. In this method, a 1-D computer program was used to perform the thermal analysis to determine the through wall temperature gradients as a function of time. The inside surface stress was calculated by using an equation, which is similar to the transient portion of ASME Section III NB 3600, Equation (11). The effect of discontinuity was included in the analysis by performing a separate 1-D thermal analysis for the pipe and nozzle. The maximum and minimum inside surface stresses were then obtained by searching the inside surface stress values calculated for each time step of the transient solution. The outside surface stresses corresponding to the maximum and minimum inside surface stresses were then calculated by a similar method. The maximum and minimum linear through wall stress distribution for each thermal transient was obtained by joining the corresponding inside and outside surface stresses by a straight line. These two stress profiles are called the maximum and minimum through wall stress distributions respectively, for convenience.

The above methodology was used for minor thermal transients. For severe thermal transients, a 1-D axisymmetric finite element model of the accumulator piping was used to determine the nonlinear stress distributions. The effects of discontinuity at the critical location were included by increasing the magnitude of 1-D nonlinear through-wall stress by 20 percent at the inside one-third thickness of the pipe wall.

The stresses due to the generic pressure and the generic moment loading were superimposed on the through-wall cyclical stresses to obtain the total maximum and minimum stress profile for each transient.

8.5 OBE LOADS

The stresses due to OBE loads were neglected in the fatigue crack growth analysis since these loads are not expected to contribute significantly to crack growth due to the small number of cycles.

8.6 TOTAL STRESS FOR FATIGUE CRACK GROWTH

The total through-wall stress at a section was obtained by superimposing the generic pressure stress and the generic moment stresses on the thermal transient stresses. Thus, the total stress for fatigue crack growth at any point is given by the following equation:

Total Stress For Fatigue Crack Growth	=	Stress due to Internal Pressure	+	Stress due to Moment (DW + Thermal Expansion)	+	Thermal Transient Stress

8.7 FATIGUE CRACK GROWTH ANALYSIS

The fatigue crack growth analysis was performed to determine the effect of the design thermal transients tabulated in Table 8-1. The analysis was performed for the critical cross-section identified in Figure 8-1. A range of crack depths was postulated, and each was subjected to the transients in Table 8-1, which included pressure and moment loads.

8.7.1 Analysis Procedure

The fatigue crack growth analyses presented herein were conducted in the same manner as suggested by Section XI, Appendix A of the ASME Boiler and Pressure Vessel Code (Reference 8-1). The analysis procedure involves assuming an initial flaw exists at some point and predicting the growth of that flaw due to an imposed series of fluctuating stresses. The growth of a crack per loading cycle is dependent on the range of applied stress intensity factor ΔK_I, by the following:

$$\frac{da}{dN} = C_o \Delta K_I^n \qquad (8\text{-}1)$$

where "C_o" and the exponent "n" are material properties, and ΔK_I is defined as ($\Delta K_I = K_{max} - K_{min}$). For inert environments these material properties are constants, but for some water environments they are dependent on the level of mean stress present during the cycle. This can be accounted for by adjusting the value of "C_o" by a function of the ratio of minimum to maximum stress for any given transient. Fatigue crack growth properties of stainless steel in a pressurized water environment have been used in the analysis.

The input required for a fatigue crack growth analysis is basically the information necessary to calculate the parameter ΔK_I, which depends on crack size and structure geometry and the range of applied stresses in the area where the crack exists. Once ΔK_I is calculated, the growth due to that particular cycle can be calculated by Equation (8-1). This increment of growth is then added to the original crack size, the ΔK_I adjusted, and the analysis proceeds to the next transient. The procedure is continued in this manner until all the transients have been analyzed.

The reference crack growth law used for the stainless steel accumulator pipe system was taken from that developed by the Metal Properties Council - Pressure Vessel Research Committee Task Force In Crack Propagation Technology. The reference curve has the equation:

[(8-2)

]a,c,e

This equation appears in Appendix C of ASME Section XI for air environments and its basis is provided in Reference 8-2, and shown in Figure 8-2. For water environments, an environmental factor of [ ]a,c,e was used, based on the crack growth tests in PWR environments reported in Reference 8-3.

8.8 RESULTS

Fatigue crack growth analyses were carried out at the critical cross section. Analysis was completed for a range of postulated flaw sizes oriented circumferentially, and the results are presented in Table 8-2. The postulated flaws are assumed to have an aspect ratio of six to one. Even for the largest postulated flaw of 0.25 inch, which is about 28 percent of the wall thickness, the result projects that flaw growth through the wall will not occur during the 40/60 year design life of the plant. Transients and cycles for the H. B. Robinson Unit 2 plant for 40-year transient set will remain bounding for 60 years (Reference 8-4), the FCG results shown in Table 8-2 are also applicable for the 60 years. Therefore, fatigue crack growth should not be a concern for the H. B. Robinson Unit 2 accumulator line.

8.9 REFERENCES

8-1 ASME Boiler and Pressure Vessel Code Section XI, 2007 Edition with the 2008 Addenda, "Rules for Inservice Inspection of Nuclear Power Plant Components."

8-2 James, L. A., and Jones, D. P., "Fatigue Crack Growth Correlations for Austenitic Stainless Steel in Air," in Predictive Capabilities in Environmentally Assisted Cracking, ASME publication PVP-99, Dec. 1985.

8-3 Bamford, W. H., "Fatigue Crack Growth of Stainless Steel Piping in a Pressurized Water Reactor Environment," Trans ASME, Journal of Pressure Vessel Technology, Feb. 1979. Engineering Development Labs Report HEDL-TME-76-43, May 1976.

8-4 NUREG-1785, "Safety Evaluation Report Related to the License Renewal of H. B. Robinson Steam Electric Plant, Unit 2."

Table 8-1 Design Transients Considered for Fatigue Crack Growth Evaluation

Trans. No.	Description	No. of Occurrences
1	Unit Loading	13200
2	Unit Unloading	13200
3	Step Load Increase	2,000
4	Step Load Decrease	2,000
5	Feedwater Cycling	2,000
6	Reactor Trip with Cooldown No Safety Injection	160
7	Inadvertent RCS Depressurization	20
8	Control Rod Drop	80
9	Turbine Roll Test	20
10	Accumulator Actuation, Accident Operation	21
11	Accumulator Actuation, Inadvertent During Cooldown	4
12	High Head Safety Injection	110
13	Steady-State and Random Fluctuations	3.2×10^6
14	RHR Operations During Plant Cooldown	200
15	RHR Operations During Refueling	80

a,c,e

Table 8-2 Accumulator Line Fatigue Crack Growth Results				
Initial	Crack Depth (in) After			
Crack Depth (in)	10 Years	20 Years	30 Years	40/60 Years*

Note: * Transients and cycles for the H. B. Robinson Unit 2 plant for the 40-year transient set will remain bounding for 60 years, the FCG results shown in Table 8-2 is also applicable for the 60 years.

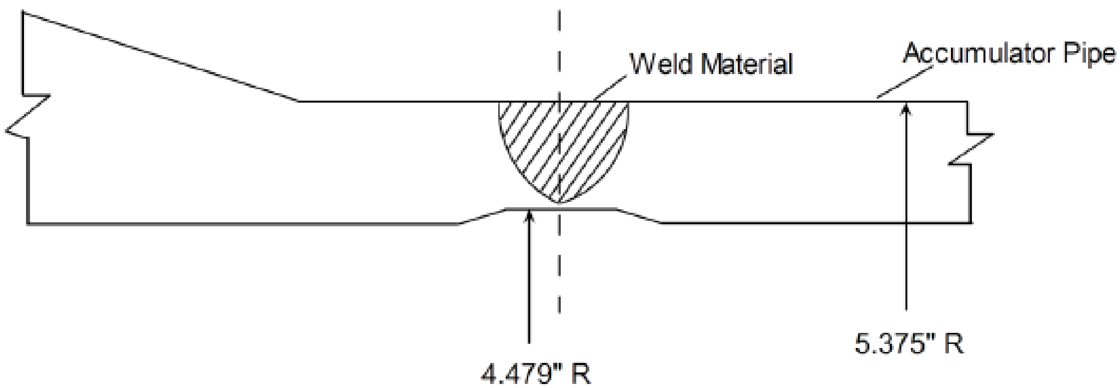


Figure 8-1 Schematic of 10″ Accumulator Line at RCL Cold Leg Nozzle Weld Location

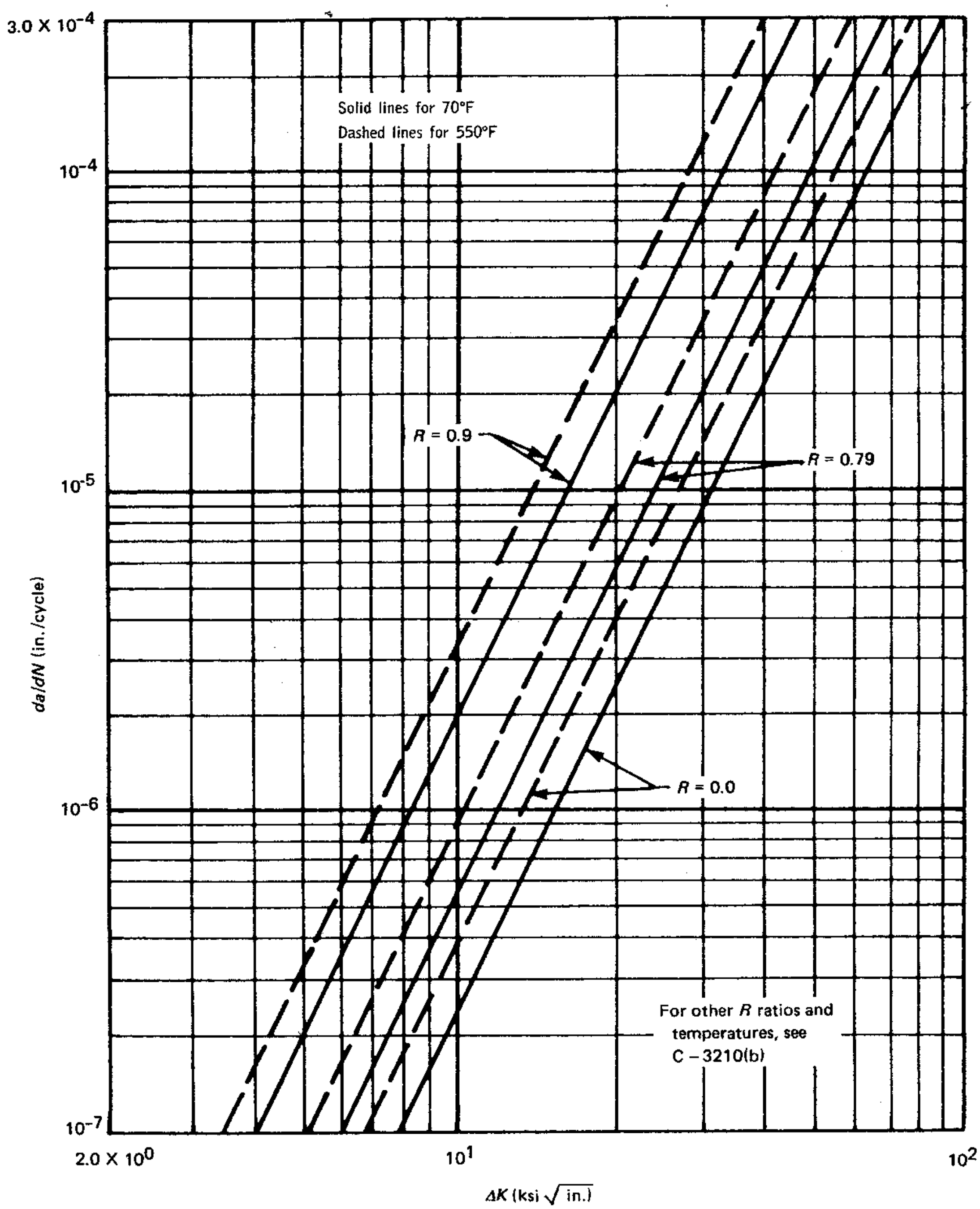


Figure 8-2 Reference Crack Growth Curves for Stainless Steel in Air Environments

9.0 ASSESSMENT OF MARGINS

The results of the leak rates of Section 6.4 and the corresponding stability evaluations of Section 7.2 are used in performing the assessment of margins. Margins are shown in Table 9-1. All the LBB recommended margins are satisfied.

In summary, at all the critical locations relative to:

1. Flaw Size - Using faulted loads obtained by the absolute sum method, a margin of 2 or more exists between the critical flaw and the flaw having a leak rate of 10 gpm (the leakage flaw).

2. Leak Rate - A margin of 10 exists between the calculated leak rate from the leakage flaw and the plant leak detection capability of 1 gpm.

3. Loads - At the critical locations the leakage flaw was shown to be stable using the faulted loads obtained by the absolute sum method (i.e., a flaw twice the leakage flaw size is shown to be stable; hence the leakage flaw size is stable). A margin of 1 on loads using the absolute summation of faulted load combinations is satisfied.

Table 9-1 Leakage Flaw Sizes, Critical Flaw Sizes and Margins for Accumulator Line

Location	Critical Flaw Size (in)	Leakage Flaw Size (in)	Margin
Node 409 Loop C	11.53	3.20	3.6
Node 360 Loop B	11.17	2.91	3.8
Node 364 Loop B	13.19	4.21	3.1
Node 3811 Loop C*	7.62	2.52	3.0

Note: *Applicable to the 8 inch line

10.0 CONCLUSIONS

This report justifies the elimination of accumulator line break from the structural design basis for H. B. Robinson Unit 2 during the 60 years plant life as follows:

a. Stress corrosion cracking is precluded by use of fracture resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation.

b. Water hammer should not occur in the accumulator line piping because of system design, testing, and operational considerations.

c. The effects of low and high cycle fatigue on the integrity of the accumulator line piping are negligible.

d. Ample margin exists between the leak rate of small stable flaws and the capability of the H. B. Robinson Unit 2 reactor coolant system pressure boundary Leakage Detection System.

e. Ample margin exists between the small stable flaw sizes of item (d) and larger stable flaws.

f. Ample margin exists in the material properties used to demonstrate end-of-service life (fully aged) stability of the critical flaws.

g. Fatigue crack growth results using the 40 year design transients and cycles (shown to be applicable for 60 years) show that there will be insignificant growth through the wall for the license renewal period (60 year plant life).

For the critical locations, flaws are identified that will be stable because of the ample margins described in d, e, and f above.

Based on loading, pipe geometry and material properties considerations, enveloping critical (governing) locations were determined at which leak-before-break crack stability evaluations were made. Through-wall flaw sizes were postulated which would cause a leak at a rate of ten (10) times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, fatigue crack growth was shown not to be an issue for the accumulator line piping. Therefore, the Leak-Before-Break conditions and margins are satisfied for H. B. Robinson Unit 2 accumulator line piping. It is demonstrated that the dynamic effects of the pipe rupture resulting from postulated breaks in the accumulator line piping need not be considered in the structural design basis of H. B. Robinson Unit 2 for the 60 years.

APPENDIX A: LIMIT MOMENT

[

] a,c,e

a,c,e

Figure A-1 Pipe with a Through-Wall Crack in Bending

ATTACHMENT 5

WESTINGHOUSE AFFIDAVITS

AFFIDAVIT CAW-23-008 for WCAP-17776-P, Revision 1

AFFIDAVIT CAW-23-009 for WCAP-17778-P, Revision 1

AFFIDAVIT CAW-23-010 for WCAP-17779-P, Revision 1

Commonwealth of Pennsylvania:
County of Butler:

(1) I, Camille Zozula, Manager/Interim Director, Management Systems & Regulatory Compliance, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of WCAP-17776-P Revision 1 be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

  (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

  (ii) The information sought to be withheld is being transmitted to the Commission in confidence and, to Westinghouse's knowledge, is not available in public sources.

  (iii) Westinghouse notes that a showing of substantial harm is no longer an applicable criterion for analyzing whether a document should be withheld from public disclosure. Nevertheless, public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

   (a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

   (b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

   (c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

   (d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

   (e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

   (f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower-case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower-case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 3/13/2023

Signed electronically by
Camille Zozula

Commonwealth of Pennsylvania:
County of Butler:

(1) I, Camille Zozula, Manager/Interim Director, Management Systems & Regulatory Compliance, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of WCAP-17778-P Revision 1 be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

(ii) The information sought to be withheld is being transmitted to the Commission in confidence and, to Westinghouse's knowledge, is not available in public sources.

(iii) Westinghouse notes that a showing of substantial harm is no longer an applicable criterion for analyzing whether a document should be withheld from public disclosure. Nevertheless, public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower-case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower-case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 3/13/2023

Signed electronically by
Camille Zozula

Commonwealth of Pennsylvania:
County of Butler:

(1) I, Camille Zozula, Manager/Interim Director, Management Systems & Regulatory Compliance, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of WCAP-17779-P Revision 1 be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

(ii) The information sought to be withheld is being transmitted to the Commission in confidence and, to Westinghouse's knowledge, is not available in public sources.

(iii) Westinghouse notes that a showing of substantial harm is no longer an applicable criterion for analyzing whether a document should be withheld from public disclosure. Nevertheless, public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower-case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower-case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 3/13/2023

Signed electronically by
Camille Zozula