PalisadesAppGNPEm Resource

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Cc:	ERICKSON, JEFFREY S; REYNA, MARY S
Subject:	LAR-PAL Equivalent Margin Analysis
Attachments:	LAR-PAL Equiv Margin Analysis 111214.pdf

Attached is a document submitted electronically to the NRC on November 12, 2014. Subject: "License Amendment Request for Approval of Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margins Analysis"

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PNP 2014-099

November 12, 2014

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

SUBJECT: License Amendment Request for Approval of Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margins Analysis

Palisades Nuclear Plant
Docket No. 50-255
License No. DPR-20

REFERENCES:

1. Palisades Nuclear Plant, *Application for Renewed Operating License*, dated March 22, 2005 (ADAMS Accession No. ML050940446).
2. Entergy Nuclear Operations, Inc. letter PNP 2013-028, *Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margins Analysis,* dated October 21, 2013 (ADAMS Accession No. ML13295A448).
3. NRC email to Entergy Nuclear Operations, Inc., *Request for Additional Information - Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margin Analysis - MF 2962,* dated May 13, 2014 (ADAMS Accession No. ML14133A684).
4. Entergy Nuclear Operations, Inc. letter PNP 2014-054, *Response to NRC Request for Additional Information - Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margin Analysis – MF 2962,* dated June 12, 2014 (ADAMS Accession No. ML14163A662).
5. Entergy Nuclear Operations, Inc. letter PNP 2014-066, *Supplemental Response to NRC Request for Additional Information - Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margin Analysis – MF 2962*, dated June 26, 2014 (ADAMS Accession No. ML14177A707).

Dear Sir or Madam:

In the Palisades Nuclear Plant (PNP) license renewal application (Reference 1), Nuclear Management Company (NMC), the former license holder for PNP, committed to submit an equivalent margins analysis (EMA) for Nuclear Regulatory Commission (NRC) approval at least three years before any reactor vessel beltline material Charpy upper-shelf energy (USE) decreases to less than 50 ft-lb, in accordance with 10 CFR 50 Appendix G, Section IV, "Fracture Toughness Requirements."

Entergy Nuclear Operations, Inc. (ENO) submitted the required EMA in Reference 2 under 10 CFR 50.4, "Written communications," as required by 10 CFR 50 Appendix G. In Reference 3, ENO received a request for additional information (RAI) concerning the EMA submittal. The ENO response to RAI questions 1, 3, 4, 5, and 6 was provided in Reference 4. The ENO response to RAI question 2 was provided in Reference 5.

During a conference call with ENO on October 21, 2014, the NRC requested that the EMA be submitted under 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," rather than under 10 CFR 50.4.

Pursuant to 10 CFR 50.90, ENO hereby submits an amendment application for the PNP operating license. The proposed amendment requests approval of an EMA completed in accordance with 10 CFR 50 Appendix G, Section IV.

This submittal contains no proprietary information.

ENO requests approval of the enclosed EMA by May 12, 2015.

In accordance with 10 CFR 50.91(b), a copy of this application, with attachments, is being provided to the designated State of Michigan official.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 12, 2014.

Sincerely,

ajv/jse

Attachments:
1. Description and Assessment of Requested Change
2. Responses to Request for Additional Information Questions
3. Westinghouse WCAP-17403-NP, Revision 1, "Palisades Nuclear Power Plant Extended Beltline Reactor Vessel Integrity Evaluation"
4. Westinghouse, WCAP-15353 – Supplement 2 – NP, Revision 0, "Palisades Reactor Pressure Vessel Fluence Evaluation"
5. Westinghouse WCAP-17651-NP, Revision 0, "Palisades Nuclear Power Plant Reactor Vessel Equivalent Margins Analysis"

cc: Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC
State of Michigan

Attachment 1

Description and Assessment of Requested Change

1.0 SUMMARY DESCRIPTION

Entergy Nuclear Operations, Inc. (ENO) requests Nuclear Regulatory Commission (NRC) approval of a proposed license amendment for Palisades Nuclear Plant (PNP) for an equivalent margins analysis (EMA) completed in accordance with 10 CFR 50 Appendix G, Section IV, “Fracture Toughness Requirements.”

2.0 DETAILED DESCRIPTION

In the PNP license renewal application (Reference 1), Nuclear Management Company (NMC), the former license holder for PNP, committed to submit an EMA for NRC approval at least three years before any reactor vessel beltline material Charpy upper-shelf energy (USE) decreases to less than 50 ft-lb, in accordance with 10 CFR 50 Appendix G, Section IV.

ENO submitted the required EMA in Reference 2 under 10 CFR 50.4, “Written communications,” as required by 10 CFR 50 Appendix G. In Reference 3, ENO received a request for additional information (RAI) concerning the EMA submittal. The ENO response to RAI questions 1, 3, 4, 5, and 6 was provided in in Reference 4. The ENO response to RAI question 2 was provided in Reference 5. The ENO responses to the six RAI questions are repeated in Attachment 2.

During a conference call with ENO on October 21, 2014, the NRC requested that the EMA be submitted under 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit,” rather than under 10 CFR 50.4.

Pursuant to 10 CFR 50.90, ENO hereby submits an amendment application for the PNP operating license. The proposed amendment requests approval of an EMA completed in accordance with 10 CFR 50 Appendix G, Section IV.

3.0 TECHNICAL EVALUATION

In the PNP license renewal application (Reference 1), NMC, the former license holder for PNP, committed to submit an EMA for NRC approval at least three years before any reactor vessel beltline material Charpy upper-shelf energy (USE) decreases to less than 50 ft-lb, in accordance with 10 CFR 50 Appendix G, Section IV.

The EMA is to demonstrate that material predicted to possess Charpy USE values less than 50 ft-lb will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

As documented in the PNP license amendment request for primary coolant system pressure-temperature limits (Reference 6), a plate material and a weld material in the PNP reactor vessel traditional beltline region are predicted to drop below the Appendix G 50 ft-lb screening criterion prior to the PNP end-of-license-extension (EOLE). The lower shell (LS) plate material, D-3804-1, is predicted to drop below the screening criterion in December 2016 and the intermediate shell (IS) to LS circumferential weld material, 9-112 (heat no. 27204), is predicted to drop below the criterion in November 2027.

As documented in WCAP-17403-NP (Attachment 3), the Charpy USE of an upper shell (US) plate material, D-3802-3, in the reactor vessel extended beltline region is predicted to remain above the 50 ft-lb Appendix G screening criterion at EOLE when considering an initial Charpy USE value based on a curve-fit of the available Charpy V-Notch data. However, this material is predicted to drop below the Appendix G screening criterion, to 47.5 ft-lb at EOLE, when considering an initial Charpy USE value based only on the available 95% shear Charpy V-Notch data for this material.

The remaining beltline and extended beltline materials in the reactor vessel are projected to maintain above the Charpy USE screening criterion of 50 ft-lb at EOLE.

The results of the fluence calculations for the extended beltline region materials of the PNP reactor vessel are provided in Table E-2 of WCAP-15353 – Supplement 2 – NP (Attachment 4). Supplement 2 was generated to address the neutron fluence experienced by materials located in the extended beltline regions above and below the reactor core that were not included in either Revision 0 of WCAP-15353 or in Supplement 1 of that report.

In accordance with 10 CFR 50 Appendix G, this letter transmits for NRC review and approval the EMA report WCAP-17651-NP (Attachment 5) for the two traditional beltline and one extended beltline reactor vessel materials discussed above. Extended beltline US plate material D-3802-3 was analyzed due to the possibility that it may fall below the 50 ft-lb limit if future operation includes higher flux levels, longer operating cycles, or changes to the reactor internals. The analysis of the three materials used the equivalent margins methodology specified in ASME Code Section XI, Division 1, Appendix K, “Assessment of Reactor Vessels with Low Upper Shelf Charpy Impact Energy Levels,” and concluded that all three of the reactor vessel materials are acceptable.

The work described herein was performed in accordance with industry and NRC accepted practices.

The extended beltline regions of the reactor pressure vessel with EOLE neutron fluence (E > 1.0 MeV) greater than 1.0 E+17 n/cm^2 have been included in the extended beltline evaluation in WCAP-17403-NP (Attachment 3). Figure 1-2 of the evaluation illustrates the boundary of the extended beltline region with neutron fluence in excess of 1.0 E+17 n/cm^2. It is noted that the neutron fluence for the inlet and outlet nozzles remain below 1.0 E+17 n/cm^2 at EOLE. The evaluation of these regions concluded that the materials are predicted to remain below the pressurized thermal shock screening criteria and the traditional beltline materials remain limiting. Also, all adjusted reference temperature values are predicted to remain below those contained in the analysis of record, so the pressure-temperature limit curves and low temperature overpressure protection (LTOP) setpoint limit curve continue to be governed by the traditional beltline materials.

The fluence evaluation in WCAP-15353 – Supplement 2 – NP (Attachment 4) used to assess the material properties is compliant with Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Previous PNP reactor pressure vessel neutron fluence evaluation submittals have been reviewed and approved by the NRC as being consistent with the requirements of Regulatory Guide 1.190. The methodology used for the WCAP-15353 fluence evaluation is detailed in

- WCAP-14040-NP-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," and

- WCAP-16083-NP-A, Revision 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry,"

which have been previously reviewed and approved by the NRC.

The EMA in WCAP-17651-NP is based upon ASME Code Section XI, Division 1, Appendix K, but has been supplemented with additional criteria specified in Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb," for material fracture toughness, transients, and fracture toughness resistance (J-R) model restrictions on sulfur content.

The USE values in the EMA have been matched to the proper orientation of the plate material. For axial flaws, the USE value for the lateral transverse "strong" orientation in the vessel wall has been used. Similarly, for circumferential flaws, the USE value for the transverse-lateral "weak" orientation has been used. In addition, the initial longitudinal USE values have been reduced to 65 percent per NUREG-0800 Branch Technical Position 5-3, "Fracture Toughness Requirements," to approximate the transverse "weak" direction.

Per Regulatory Guide 1.161, three additional cooldown transients at 100 °F/hr, 400 °F/hr, and 600 °F/hr have been considered in the EMA.

As discussed in the EMA, extended beltline plate material D-3802-3 and traditional beltline plate material D-3804-1 have sulfur content in excess of the J-R model 0.018 weight percent limitation. Additional analysis has been conducted using available information for the V-50 plate in NUREG/CR-5265, “Size Effects on J-R Curves for A 302-B Plate,” to demonstrate that the PNP reactor vessel high sulfur plate materials remains below the measured very conservative lower bound V-50 A-302 B plate J-R data.

Additional technical evaluation information is provided in Attachment 2.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

An assessment of the proposed changes concluded that there are no exceptions to any of the following regulations. Therefore, ENO would remain in compliance with the following regulations and guidance:

10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 1, "Quality Standards and Records," requires the structures, systems, and components important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

GDC 31, "Fracture prevention of the reactor coolant pressure boundary," requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle 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.

GDC 32, "Inspection of the reactor coolant pressure boundary," requires components that are part of the reactor coolant pressure boundary be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," requires that all lightwater reactors meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in 10 CFR 50, Appendix G and Appendix H.

10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," ensures that changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment are monitored. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel.

10 CFR 50, Appendix G, Section IV.A.1, "Reactor Vessel Charpy Upper-Shelf Energy Requirements," requires submission of an EMA for approval at least three years before any reactor vessel beltline material Charpy USE decreases to less than 50 ft-lb.

Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001, describes methods for determining reactor pressure vessel fluence.

Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 Ft-Lb," June 1995, describes methods for demonstrating that the margins of safety against ductile fracture are equivalent to those in Appendix G of the ASME Code.

4.2 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. (ENO) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No

   This amendment request is for approval of an equivalent margins analysis (EMA) in accordance in accordance with 10 CFR 50 Appendix G, Section IV, "Fracture Toughness Requirements." The EMA is to demonstrate that reactor vessel beltline material predicted to possess Charpy Upper Shelf Energy (USE) values less than 50 ft-lb will provide margins of safety against fracture equivalent to

those required by Appendix G of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

The EMA does not involve a significant increase in the probability or consequences of an accident, and does not result in physical alteration of a plant structure, system or component (SSC) or installation of new or different types of equipment. The EMA does not affect plant operation or any design function. The EMA verifies the capability of a SCC to perform a design function. Further, the EMA does not significantly affect the probability of accidents previously evaluated in the Updated Final Safety Analysis Report (UFSAR), or cause a change to any of the dose analyses associated with the UFSAR accidents because accident mitigation functions would remain unchanged.

Therefore, the proposed change does 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 amendment request is for approval of an EMA in accordance in accordance with 10 CFR 50 Appendix G, Section IV. The EMA is to demonstrate that reactor vessel beltline material predicted to possess Charpy USE values less than 50 ft-lb will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Boiler and Pressure Vessel Code. The EMA does not change the design function, operation, or integrity of the reactor vessel, and does not challenge the performance or integrity of any safety-related systems. No physical plant alterations are made as a result of the proposed change. The EMA will not create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing basis.

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

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The amendment request is for approval of an EMA in accordance in accordance with 10 CFR 50 Appendix G, Section IV. The EMA is to demonstrate that reactor vessel beltline material predicted to possess Charpy USE values less than 50 ft-lb will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Boiler and Pressure Vessel Code. As such, there is no significant reduction in the margin of safety as a result of the

EMA. No design bases or safety limits are exceeded or altered due to the EMA. The margin of safety associated with the acceptance criteria of accidents previously evaluated in the UFSAR is unchanged. The proposed change has no effect on the availability, operability, or performance of the safety-related systems and components.

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

4.3 Conclusions

In conclusion, based on the considerations discussed above, (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 does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, 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. Palisades Nuclear Plant, *Application for Renewed Operating License*, dated March 22, 2005 (ADAMS Accession No. ML050940446).
2. Entergy Nuclear Operations, Inc. letter PNP 2013-028, *Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margins Analysis,* dated October 21, 2013 (ADAMS Accession No. ML13295A448).
3. NRC email to Entergy Nuclear Operations, Inc., *Request for Additional Information - Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margin Analysis - MF 2962,* dated May 13, 2014 (ADAMS Accession No. ML14133A684).
4. Entergy Nuclear Operations, Inc. letter PNP 2014-054, *Response to NRC Request for Additional Information - Palisades Nuclear Plant 10 CFR 50*

Appendix G Equivalent Margin Analysis – MF 2962, dated June 12, 2014 (ADAMS Accession No. ML14163A662).

5. Entergy Nuclear Operations, Inc. letter PNP 2014-066, *Supplemental Response to NRC Request for Additional Information - Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margin Analysis – MF 2962*, dated June 26, 2014 (ADAMS Accession No. ML14177A707).
6. Palisades Nuclear Plant, *License Amendment Request for Primary Coolant System Pressure-Temperature Limits*, dated March 7, 2011 (ADAMS Accession No. ML110730082)

Attachment 2

Responses to Request for Additional Information Questions

Entergy Nuclear Operations, Inc., (ENO) previously submitted the 10 CFR 50 Appendix G equivalent margins analysis (EMA) in Reference A below under 10 CFR 50.4, as discussed in the Detailed Description section in Attachment 1 of this submittal. In Reference B, ENO received a request for additional information (RAI) concerning the EMA submittal that contained six questions. The ENO responses to RAI questions 1, 3, 4, 5, and 6 were provided in Reference C. The ENO response to RAI question 2 was provided in Reference D.

A. Entergy Nuclear Operations, Inc. letter PNP 2013-028, *Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margins Analysis*, dated October 21, 2013 (ADAMS Accession No. ML13295A448).

B. NRC email to Entergy Nuclear Operations, Inc., *Request for Additional Information - Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margin Analysis - MF 2962*, dated May 13, 2014 (ADAMS Accession No. ML14133A684).

C. Entergy Nuclear Operations, Inc. letter PNP 2014-054, *Response to NRC Request for Additional Information - Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margin Analysis – MF 2962*, dated June 12, 2014 (ADAMS Accession No. ML14163A662).

D. Entergy Nuclear Operations, Inc. letter PNP 2014-066, Supplemental Response to NRC Request for Additional Information - Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margin Analysis – MF 2962, dated June 26, 2014 (ADAMS Accession No. ML14177A707).

The ENO responses to the six RAI questions are repeated below.

NRC Request (May 13, 2014)

1. *The EMA is based on American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Section XI, Appendix K as supplemented by Regulatory Guide (RG) 1.161 "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb". ASME Code Section XI, Appendix K, Article K-4210 and RG 1.161 both include equations for calculating the stress intensity factor due to radial thermal gradients. In Section 5.1 of the EMA submittal, the licensee discusses through-wall thermal stress and states that typical through-wall stress and stress distribution during a heatup transient are shown in Figures 5-1 and 5-2. But Figures 5-1 and 5-2 of the EMA submittal do not show these stresses as discussed. Provide figures showing typical through-wall stress and stress distributions during a heatup transient to support the discussion in paragraph 5.1 of the EMA submittal.*

ENO Response to RAI-1

Figures detailing typical heatup thermal axial stress and typical through-wall axial stress for the PNP reactor vessel used in the equivalent margins analysis (EMA) submittal are provided below.

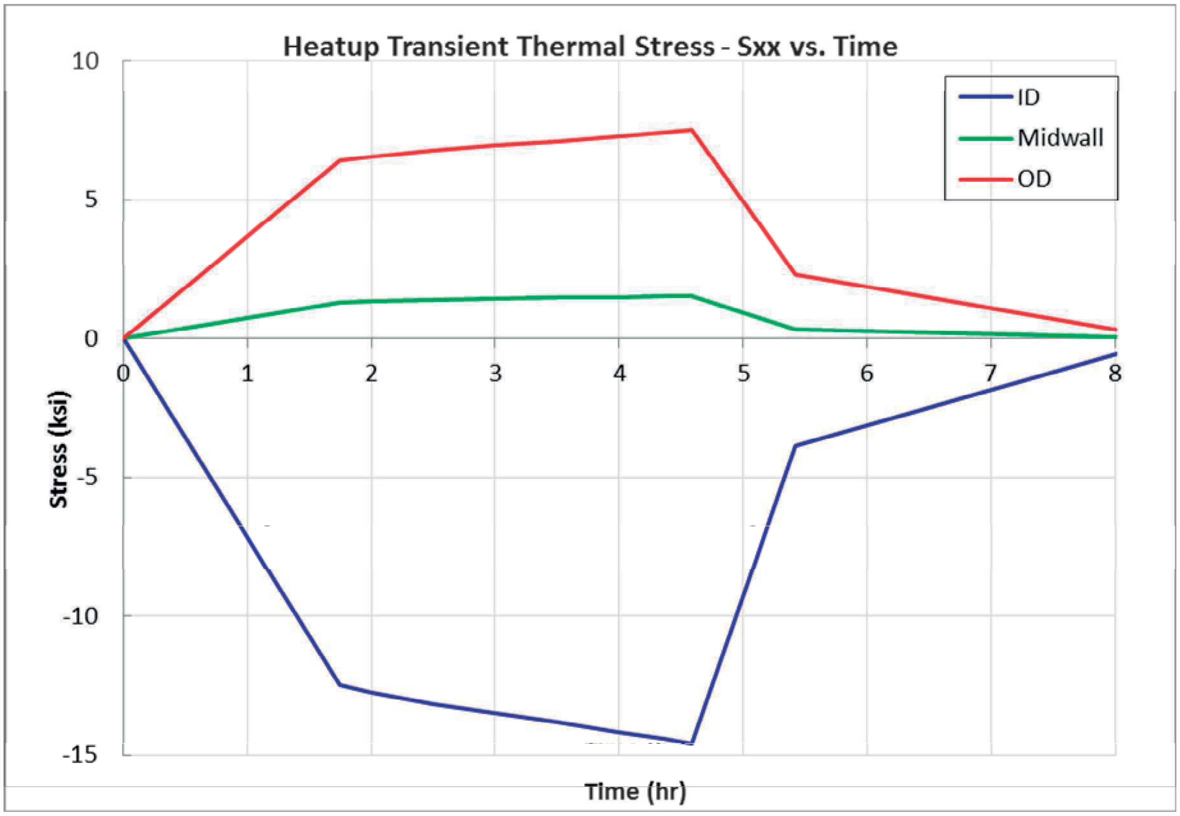


Figure 5-1(a) - PNP Typical Thermal Transient Axial Stress Profile – Stress versus Time

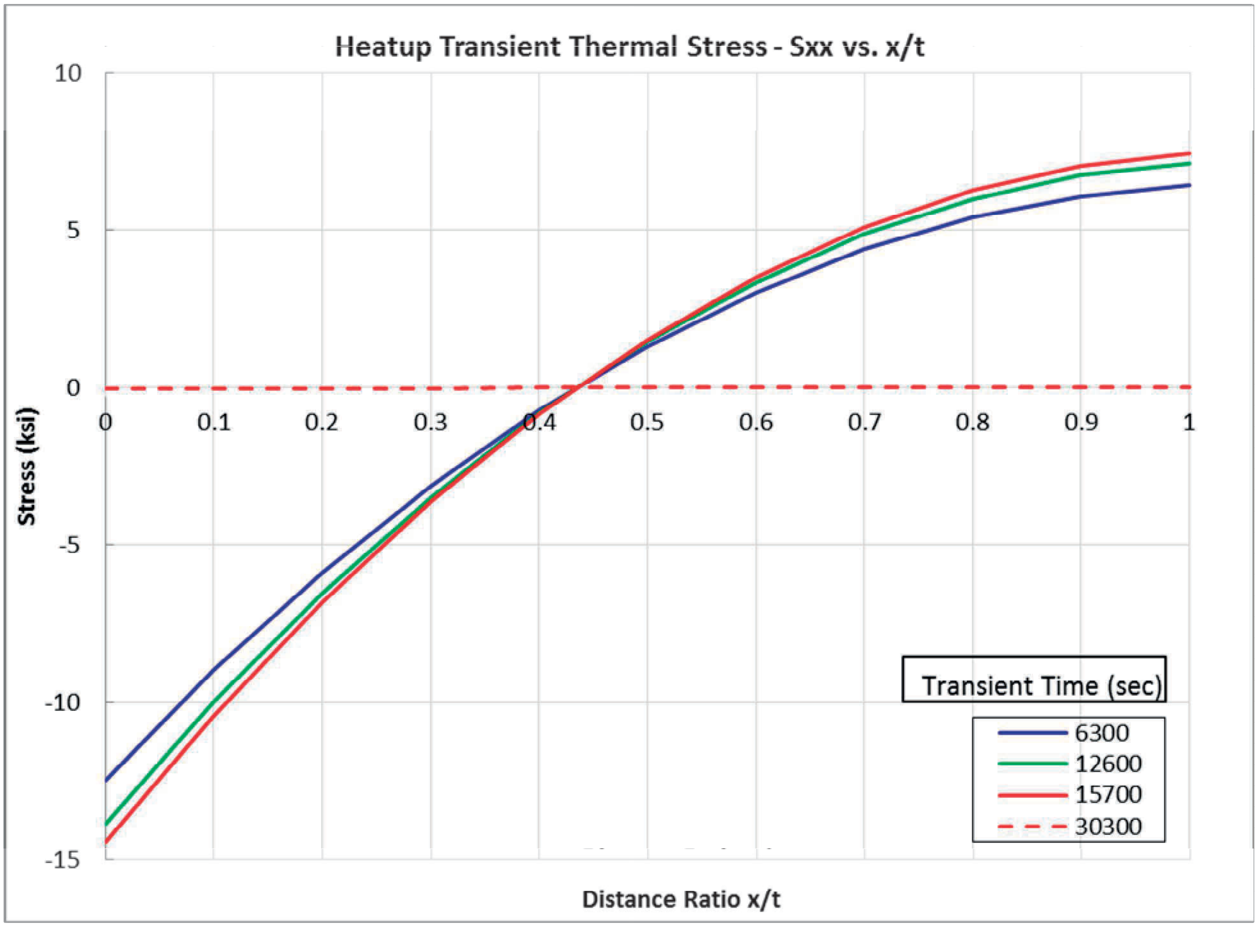


Figure 5-2(a) - PNP Typical Thermal Transient Axial Through-Wall Stress Distribution

NRC Request (May 13, 2014)

2. Section 5.1 states, "Only circumferential base metal flaws are considered in this analysis, because only the "weak" orientation USE is projected to drop below 50 ft-lbs as described below." Please demonstrate that assuming a circumferential flaw in the base metal with the weak Charpy V-Notch (CVN) value in the EMA is more limiting than assuming an axial flaw in the base metal with the strong CVN value. Please note that the significantly greater applied J integral associated with the axial flaw may challenge the fundamental assumption in the EMA submittal.

ENO Response to RAI-2

As documented in WCAP-17651-NP, Revision 0, the PNP reactor vessel plates have sulfur content greater than the 0.018 wt-% value provided in Regulatory Guide 1.161 (Reference 1). Therefore, lower bound high-sulfur fracture toughness data from the V-50 plate included in NUREG/CR-5265 (Reference 7)

was located, as documented in the WCAP, to provide justification for use of the J-R model included in Regulatory Guide 1.161.

However, since only transverse (T-L) direction, weak data, was available in this NUREG and since only the T-L upper-shelf energy (USE) dropped below 50 ft-lbs, only circumferential flaws were considered in the original WCAP submittal since there was no longitudinal (L-T) direction, strong data, for which to compare with the axial J-applied values for PNP. The ENO interpretation of Regulatory Guide 1.161 was that this was allowable and axial flaws did not need to be considered since the longitudinal final USE values of the PNP plate materials were over the 50 ft-lb limit of 10 CFR 50, Appendix G at end-of-license- extension (EOLE). However, per discussion during conference call between ENO and the NRC on June 6, 2014, axial flaws should still have been postulated, with longitudinal direction USE considered in the equivalent margins analysis.

Since the V-50 plate does not have L-T strong data reported in NUREG/CR-5265, the T-L data needs to be converted to L-T via an appropriate ratio to approximate the strong direction for direct comparison with axial flaws. The standard ratio is 65% per Regulatory Guide 1.161. Data was located in NUREG/CR-6426 (Reference 6), which had fracture toughness data for both orientations for five of the eight plate codes tested in this report. New Table 5-7 documents the material properties, initial USE values and available fracture toughness data for these materials. The average L-T/T-L fracture toughness conversion was 68% with consideration of all data, and 64% when the Z1/Z2 plate codes were excluded, as they appear to be an outlier compared to the other data points. Either calculated percent conversion supports the generic 65% conversion, which was then selected for use to ratio up the V-50 plate data to L-T orientation for comparison with axial flaw J-Applied values at PNP.

New Table 5-8 details the axial flaw safety factors for all transients, Levels A, B, C and D, with consideration of the Regulatory Guide 1.161 J-R model and limiting EOLE USE equal to 73 ft-lbs per WCAP-17651-NP. New Table 5-9 details the axial flaw safety factors for all transients, Levels A, B, C and D, with consideration of the V-50 plate data adjusted to the L-T orientation. New Figures 5-14 and 5-15 detail the applied J-Integral versus crack extension for axial flaws at 1/4t for Level A and B transients and applied J-Integral versus crack extension for axial flaws at 1/10t for Levels C and D transients, respectively. New Figure 5-16 details the axial flaw J-Integral versus crack extension at 1/4t for Level A and B transients for base metal with Regulatory Guide 1.161 model J-R curves and V-50 plate data included. New Figure 5-17 details the axial flaw J-Integral versus crack extension at 1/4t, pressure = 2.75 ksi, and a 100°F/hr cooldown transient for base metal with Regulatory Guide 1.161 model J-R curves and V-50 plate data included. Lastly, new Figure 5-18 details the axial flaw J-Integral versus crack extension at 1/10t for Levels C and

D transients, for base metal with Regulatory Guide 1.161 model J-R curves and V-50 plate data included.

It should be noted, as discussed in detail in WCAP-17651-NP, that the V-50 plate data has a lower weight percent Ni value (0.23 wt-%), due to being A 302 B steel, and not SA 302 B, Modified, that contribute to the V-50 plate having lower fracture toughness than the PNP-specific plate materials. The PNP plates are SA 302 B, Modified, which means that they have at least 0.4% Ni. Nickel was added to increase toughness. Conservatively, the lowest J-R curve test data reported in NUREG/CR-5265, which is from a 6T size specimen, is used for comparison to the J-Applied values. The 6T data is considerably lower than test data for the 1T J-R data, which is the standard size specimen typically used. Therefore, the V-50 plate 6T J-R data is a conservative lower bound, viewed as the worst possible case, and selected due to being the only available fracture toughness data with high-sulfur content.

The minimum safety factor with consideration of the Regulatory Guide 1.161 J-R model, L-T orientation USE values and the PNP-specific axial flaw J-Applied values is 1.7 while the minimum safety factor with relative to the V-50 plate data and the PNP-specific axial flaw J-applied values is 1.4 at 0.1-inch crack extension. All these cases have their structural factors above the minimum requirement of 1.15 per Regulatory Guide 1.161 and are deemed acceptable. The flaw extension figures demonstrate that the NRC Regulatory Guide 100°F/hr cooldown transient with the accumulation pressure levels governs the Level A and B transients, which is the limiting case. All cases, where the Regulatory Guide 1.161 J-R material correlation is considered with axial flaw J-Applied pressure loadings, are acceptable with the applied J-integral values at 0.1-inch crack extensions below the material J-resistance ($J_{0.1}$) as required by the ASME Code Appendix K. In some instances with consideration of the V-50 plate data adjusted to the L-T orientation, the J-Material curves, adjusted to transient temperature, are either slightly below or just over the J-applied values, specifically for the Regulatory Guide 1.161 100°F/hour cooldown transient. However, as discussed above, the V-50 data is a lower bound high-sulfur data set, that is not fully representative of the PNP actual plate materials, and this result can be considered acceptable with consideration of the associated Regulatory Guide 1.161 model, and the structural factor (SF) calculations shown in Tables 5-8 and 5-9. Finally, as discussed above, the Regulatory Guide 1.161 100°F/hour cooldown transient is more limiting than the PNP-specific transients, as shown in the comparison of J-Applied curves in Figures 5-16 and 5-17. Note that the Regulatory Guide 1.161 100°F/hour cooldown transient with pressure of 2.75 ksi is more conservative than PNP cooldown transient with pressure of 2.13 ksi.

Table 5-7: NUREG/CR-6426 L-T (Strong) vs. T-L (Weak) Charpy USE and Fracture Toughness Data

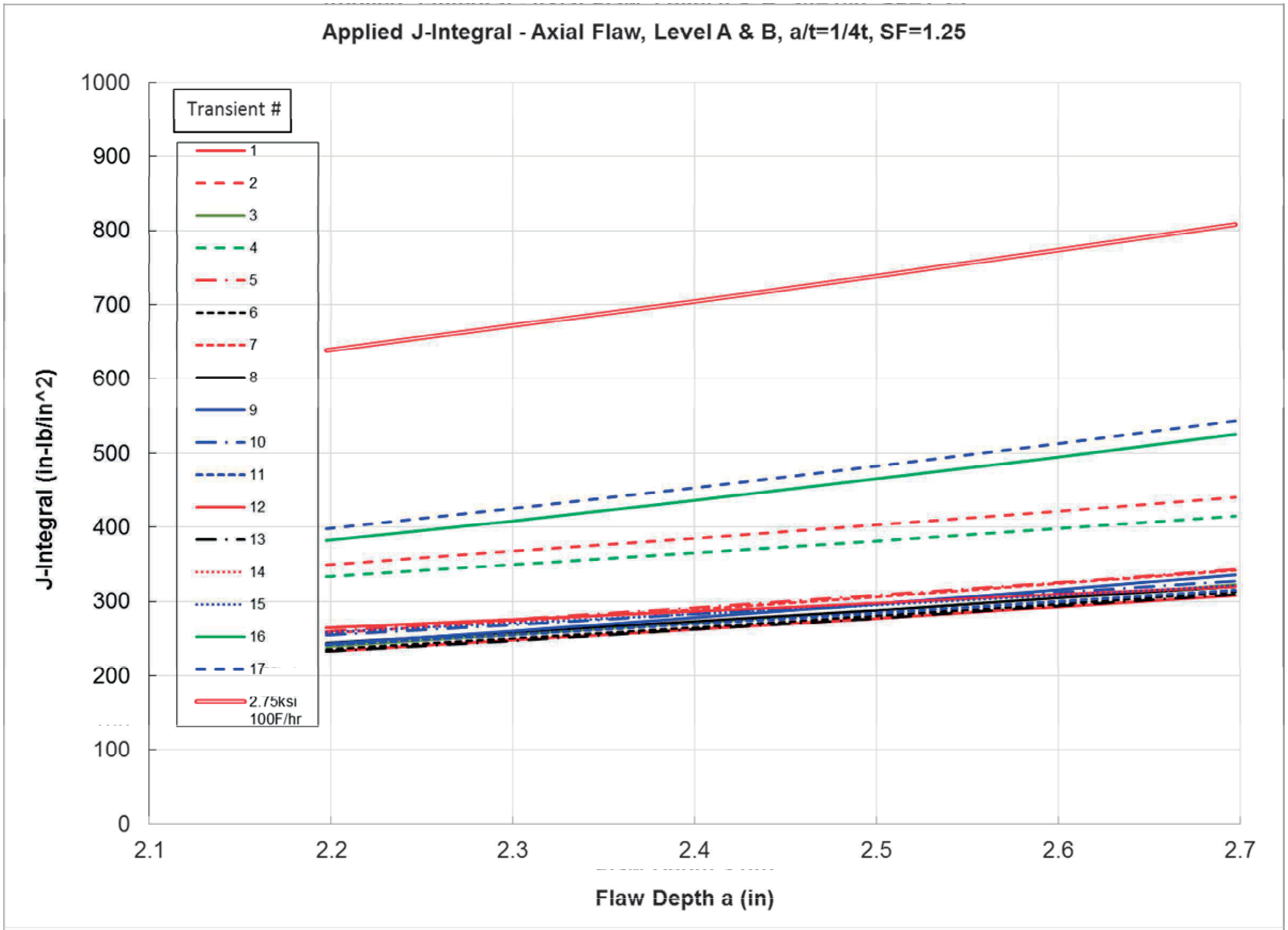
T= 180F	Plate Code	Chemistry			Initial USE (ft-lbs)		USE Ratio	$J_{0.1}$ (in/lb/in^2)		$J_{0.1}$ Ratio
		Cu	Ni	S	Longitudinal	Transverse		Longitudinal	Transverse	
Modified A302B	Z1, Z2	0.17	0.47	0.011	160	126	78.8%	3810	3300	86.6%
	Z5	0.16	0.60	0.016	153	95	62.1%	2640	1630	61.7%
	Z6A	0.18	0.49	0.013	129	113	87.6%	3570	2325	65.1%
	Z6B	0.21	0.51	0.023	117	64	54.7%	2360	1470	62.3%
	Z7	0.16	0.53	0.014	126	96	76.2%	4500	3000	66.7%
						Average (All)	71.9%		Average (All)	68.5%
						Average (Exclude Z1, Z2)	70.1%		Average (Exclude Z1, Z2)	64.0%

Table 5-8: Available Margins on Pressure Load for All Transients, Levels A, B, C and D, Axial Flaws

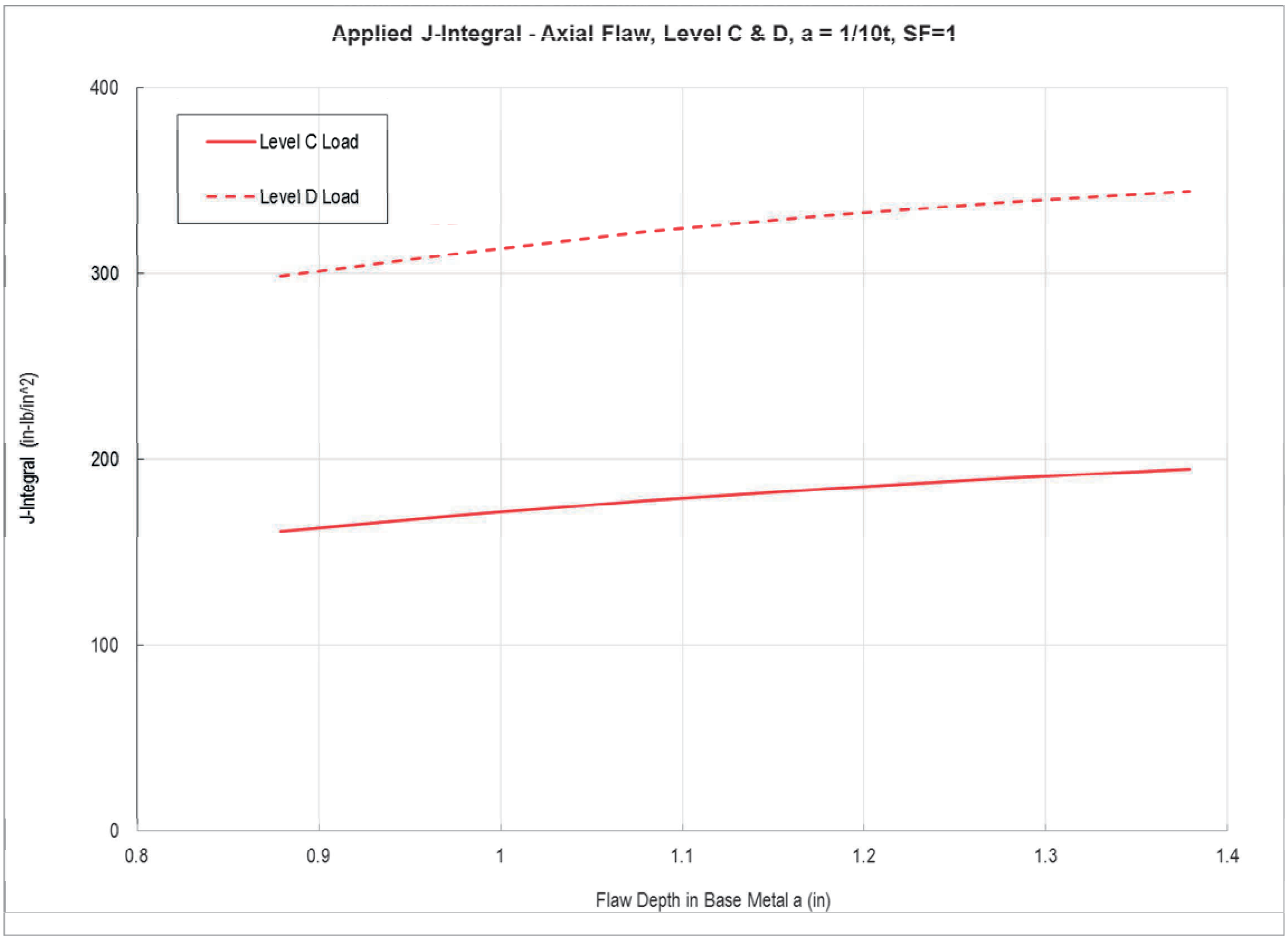
Level A and B	Base Metal – R. G. 1.161			Level C	Base Metal – R. G. 1.161			Level D	Base Metal – R. G. 1.161		
	Axial Flaw		$J_{0.1}$ material		Axial Flaw		$J_{0.1}$ material		Axial Flaw		$J_{0.1}$ material
Time (sec)	SF	J-applied x SF (in-lb/in^2)	(in-lb/in^2)	Time (sec)	SF	J-applied x SF (in-lb/in^2)	(in-lb/in^2)	Time (sec)	SF	J-applied x SF (in-lb/in^2)	(in-lb/in^2)
0	1.8	985	986	0	3.4	987	986	0	2.9	682	682
2800	1.7	1096	1096	1,197	5.0	1212	1213	798	4.1	831	830
3600	1.7	1138	1139	4,122	5.2	2218	2218	2,748	3.4	1490	1491
5400	1.7	1249	1249								
7200	1.8	1376	1376								
9000	1.9	1518	1518								
10800	18.5	1676	1676								
Minimum SF	**1.7**			Minimum SF	**3.4**			Minimum SF	**2.9**		

Table 5-9: Available Margins on Pressure Load for All Transients, Levels A, B, C and D, with Consideration of V-50 Plate Data and Axial Flaws

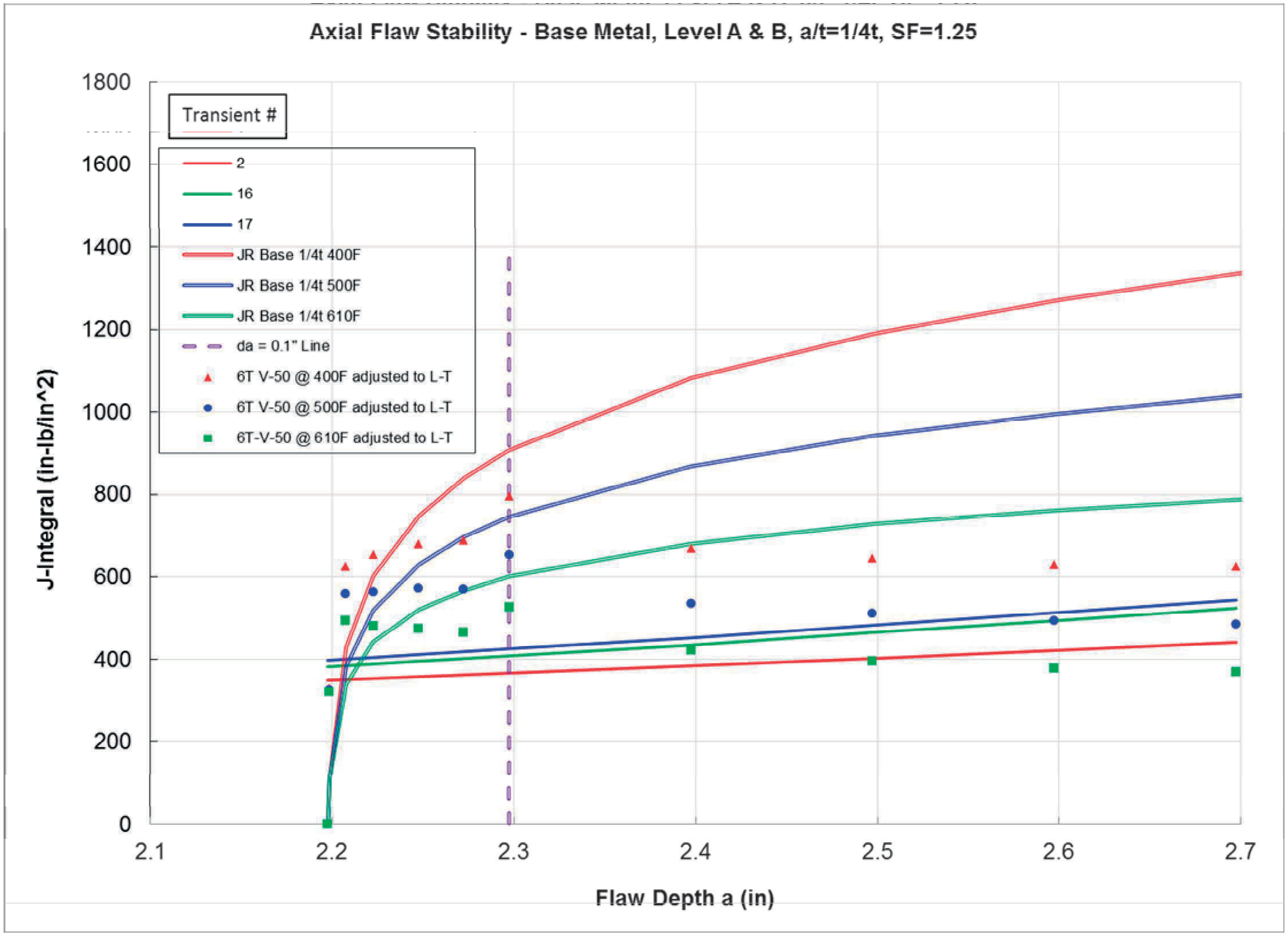
Level A and B	V-50 Plate			Level C	V-50 Plate			Level D	V-50 Plate		
	Axial Flaw		$J_{0.1}$		Axial Flaw		$J_{0.1}$		Axial Flaw		$J_{0.1}$
Time (sec)	SF	J-applied x SF (in-lb/in^2)	material (in-lb/in^2)	Time (sec)	SF	J-applied x SF (in-lb/in^2)	material (in-lb/in^2)	Time (sec)	SF	J-applied x SF (in-lb/in^2)	material (in-lb/in^2)
0	1.5	611	611	0	2.8	611	611	0	2.8	611	611
2800	1.4	679	679	1,197	4.0	751	751	798	3.9	743	743
3600	1.4	706	706	4,122	4.0	1,374	1,374	2,748	3.1	1335	1,335
5400	1.4	774	774								
7200	1.5	853	853								
9000	1.5	941	941								
10800	15.0	1039	1039								
Minimum SF	**1.4**			Minimum SF	**2.8**			Minimum SF	**2.8**		



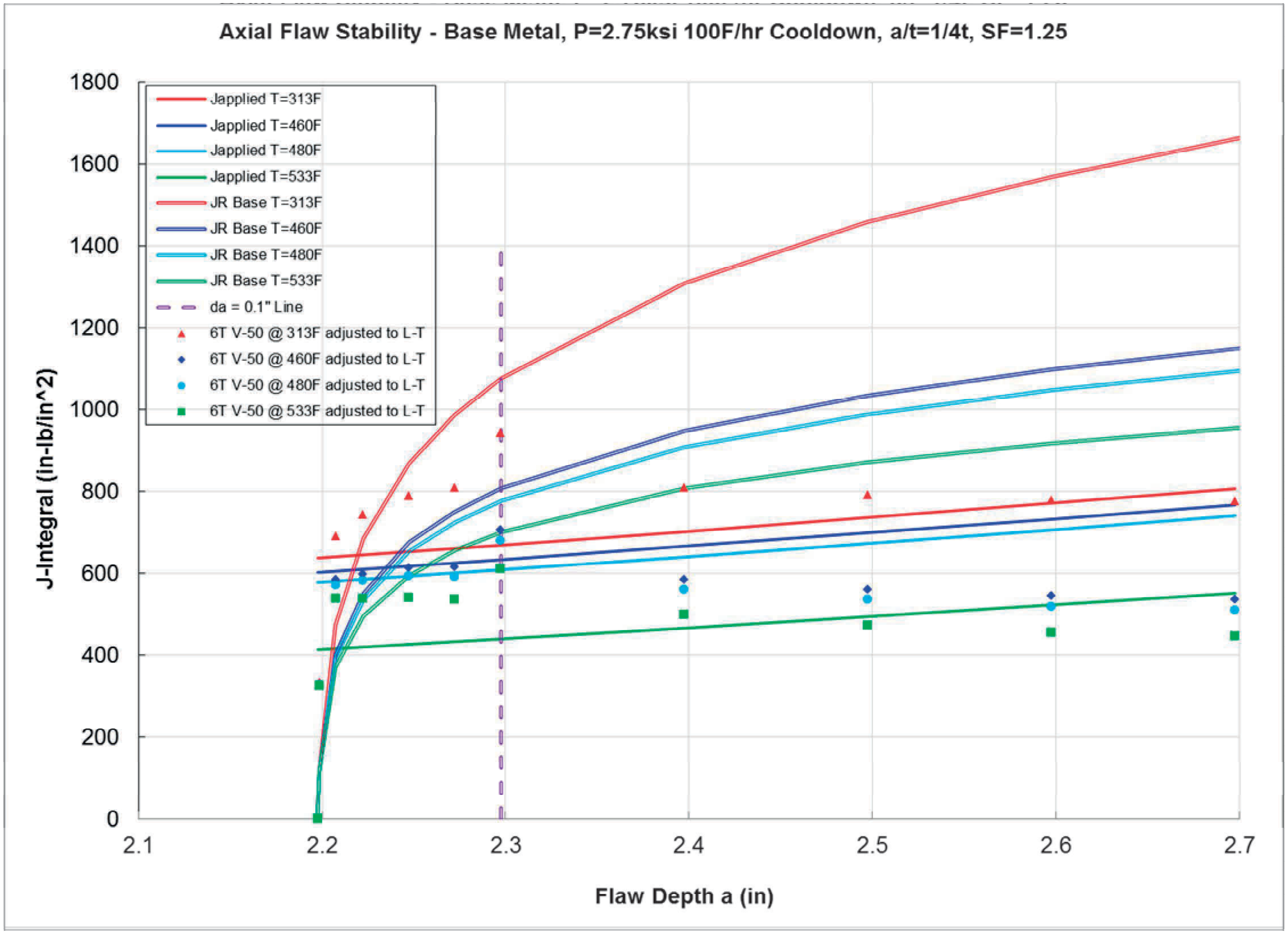
WCAP-17651-NP New Figure 5-14: Applied J-Integral versus Crack Extension for Axial Flaw – 1/4t, Level A and B



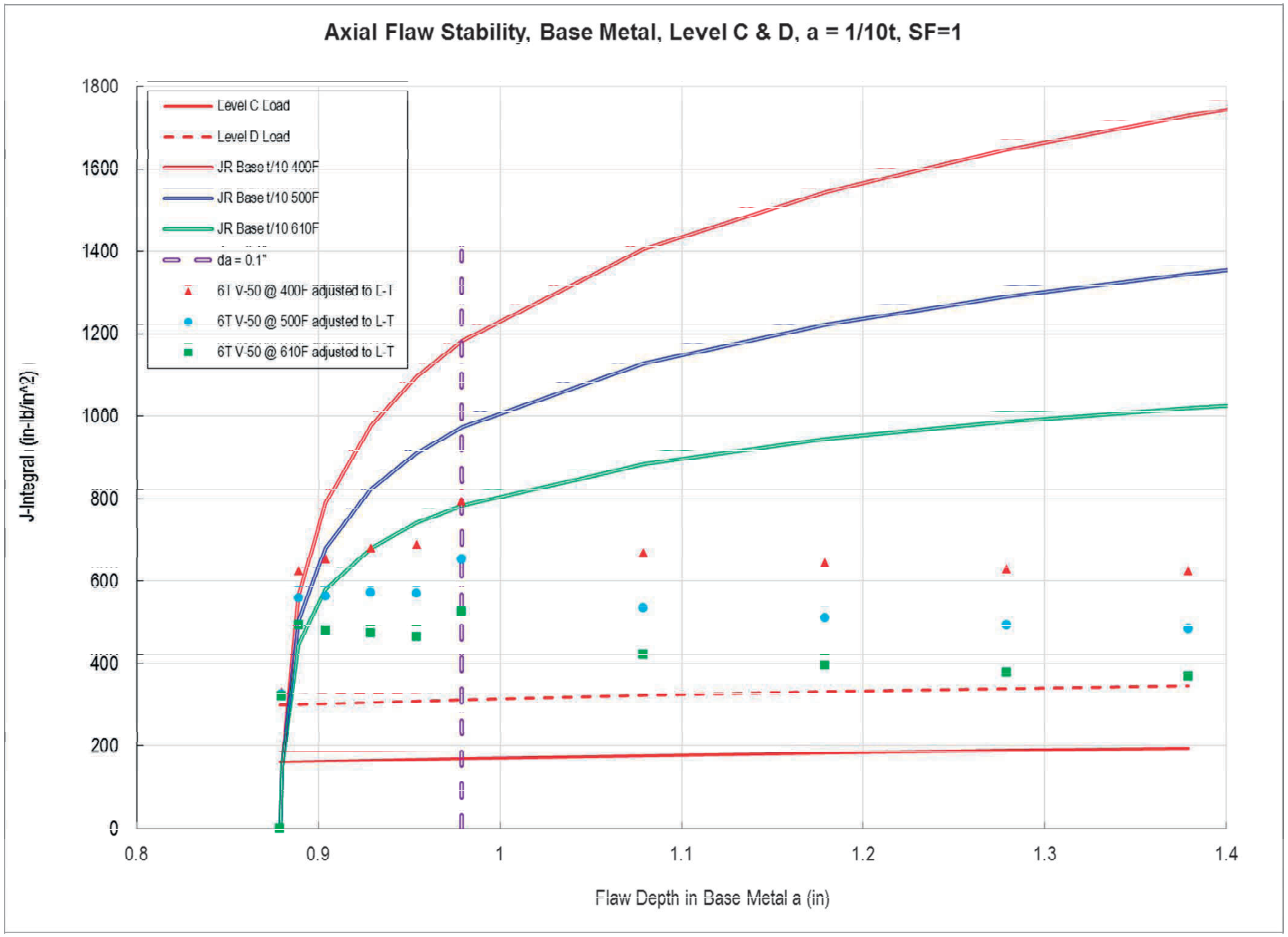
WCAP-17651-NP New Figure 5-15: Applied J-Integral versus Crack Extension for Axial Flaw – 1/10t, Levels C and D



WCAP-17651-NP New Figure 5-16: Axial Flaw J-Integral versus Crack Extension – t/4, Level A and B, Base Metal, with V-50 Plate Data Included



WCAP-17651-NP New Figure 5-17: Axial Flaw J-Integral versus Crack Extension – t/4, P=2.75 ksi, 100°F/hr Cooldown, Base Metal, with V-50 Plate Data Included



WCAP-17651-NP New Figure 5-18: Axial Flaw J-Integral versus Crack Extension – t/10, Levels C and D Loads, Base Metal, with V-50 Plate Data Included

NRC Request (May 13, 2014)

3. *The applied J-integral values for the circumferential flaws for all Level A and B service level conditions are shown in Figure 5-1, and the applied J-integral values for the circumferential flaws for Level C and D service level conditions are shown in Figure 5-2. Since Section 5.1 provides very limited information regarding the applied J-integral calculations, please confirm that the calculations underlying Figures 5-1 and 5-2 are based on the formulas in RG 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 FT-LB." If not, please describe, in addition to your response to RAI-1, your plant-specific calculations to support their acceptance in this application.*

ENO Response to RAI-3

Yes, the applied J-integral calculations underlying Figures 5-1 and 5-2 are based on formulas in RG 1.161.

NRC Request (May 13, 2014)

4. *Table 4-4 was presented but without being mentioned in Section 4 regarding how it was used in the EMA analysis. Therefore, please confirm that the calculated available margins presented in Table 5-3 for various time during cooldown are results, using the relevant J-R curves adjusted by the material margin factors of Table 4-4.*

ENO Response to RAI-4

As discussed in Section 2.2 of WCAP-17651-NP, "Palisades Nuclear Power Plant Reactor Vessel Equivalent Margins Analysis," Revision 0, RG 1.161 material margin factors (MF) in Table 4-4 were used for the J-R curves.

NRC Request (May 13, 2014)

5. *Section 5.2 provides justification for using the high-toughness/low-sulfur model from RG 1.161 in the proposed EMA for the high-sulfur plates, and Section 5.3 provides the corresponding EMA results. When the high-sulfur model (e.g., for the 6T specimen) of NUREG/CR-5265, "Size Effects on J-R Curves for A 302-B Plate," is used, please demonstrate that*

    - *The updated safety factors (see Table 5-3), after adjusting for temperature, will still be greater than 1.15.*

- *The updated applied J/J-R curves (see figures 5-8, 5-9, and 5-12), after adjusting for temperature, will still show that $dJ_{applied}/d_{\mathbf{a}} < dJ_{material}/d_{\mathbf{a}}$ at $J_{applied} = J_{material}$.*

If the above cannot be demonstrated, perform a sensitivity study, showing at what percentage of the proposed J-R curve (e.g., 90%), your EMA calculation results will meet the criteria on both crack extension and stability.

6. *Section 6 presents conclusions of this submittal. For Service Level C condition with 400°F/hr cooldown, it is concluded that, "The equivalent margins analyses for the plate materials are acceptable and bounded by the conservative test data reported in NUREG/CR-5265 in all cases for the Level C transient." This conclusion was repeated later for Service Level D condition with 600°F/hr cooldown, with "C" in the quote replaced by "D." Plot the relevant NUREG/CR 5265 6T data in Figure 5-12 and provide sufficient justification to support your conclusions.*

ENO Responses to RAI-5 and RAI-6

Updated Figures 5-9 and 5-12 with the V-50 plate data are provided below, along with added Tables 5-4 and 5-5 showing the Level C and D safety factors, respectively. Table 5-6 was added, which demonstrates the available margins on pressure loading with the V-50 plate data, adjusted for temperature, with consideration of all service loadings, Level A, B, C and D. The minimum safety factor (SF) with consideration of the V-50 plate data and the PNP-specific J-applied values is 1.5, which is above the minimum required SF of 1.15 per RG 1.161.

Figure 5-8 from WCAP-17651-NP, Revision 0, along with the updated Figures 5-9 and 5-12 below, all demonstrate that at $J_{applied} = J_{material}$, $dJ_{applied}/d_{\mathbf{a}} < dJ_{material}/d_{\mathbf{a}}$, is satisfied for all three cases (i.e., the slope of the $J_{applied}$ is smaller than the $J_{material}$ at the point of intersection).

Therefore, as demonstrated below and in WCAP-17651-NP, the equivalent margins of safety per ASME Code Section XI (References 4 and 5) are found to be acceptable for the PNP reactor vessel beltline and extended beltline regions with predicted Charpy upper-shelf energy levels falling below the 50 ft-lb 10 CFR 50, Appendix G requirements at end-of-license-extension.

Westinghouse discovered during the development of this RAI response that the Level C and D loading J-applied curves plotted in WCAP-17651-NP, Figure 5-12, were not the most limiting case. This error also propagated onto Figures 5-2 and 5-13 in the WCAP. This has been updated in the attached figures as part of this RAI response. Note that the conclusions to the report, including the safety factor determination, are unchanged; only

the figures were in error. This has been documented in the Westinghouse corrective action system, and will be corrected when the WCAP is revised to incorporate these RAI changes.

Lastly, note that the Level C and D margin tables (Tables 5-4 and 5-5 below) were originally omitted from WCAP-17651-NP because Service Level A and B, as discussed in Section 5.3 of WCAP-17651-NP, are the governing transients.

Table 5-4 Available Margins on Pressure Load for Level C, 400°F/hr Cooldown

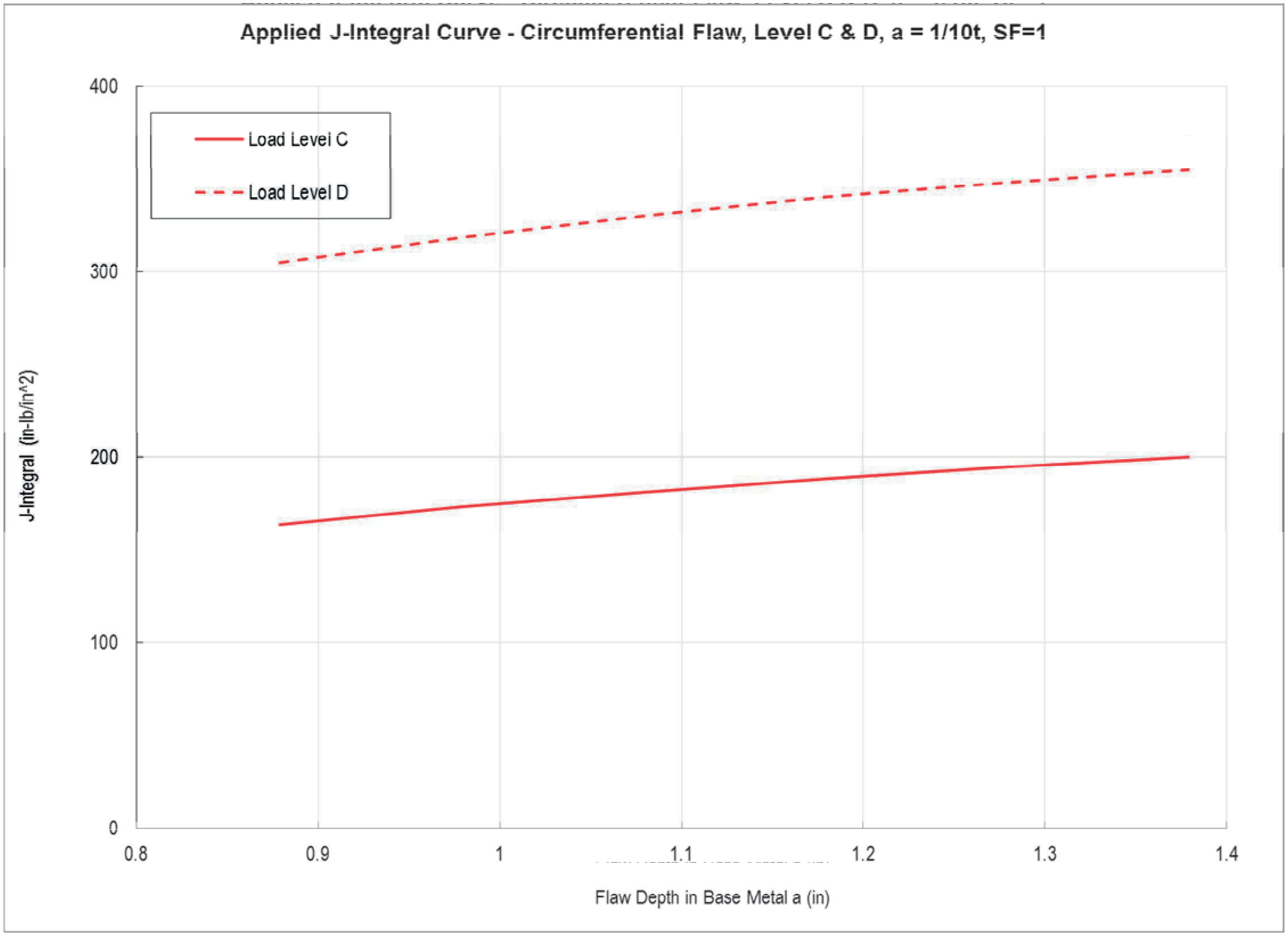
	Base Material			Weld Material		
	Circumferential Flaw			Circumferential Flaw		
Time (sec)	SF	J-applied x SF (in-lb/in^2)	$J_{0.1}$ material (in-lb/in^2)	SF	J-applied x SF (in-lb/in^2)	$J_{0.1}$ material (in-lb/in^2)
0	5.8	682	682	5.2	511	510
1,197	8.6	885	839	7.1	613	613
4,122	8.9	1,535	1,534	7.1	1,050	1,050
Minimum SF	**5.8**			**5.2**		

Table 5-5 Available Margins on Pressure Load for Level D, 600°F/hr Cooldown

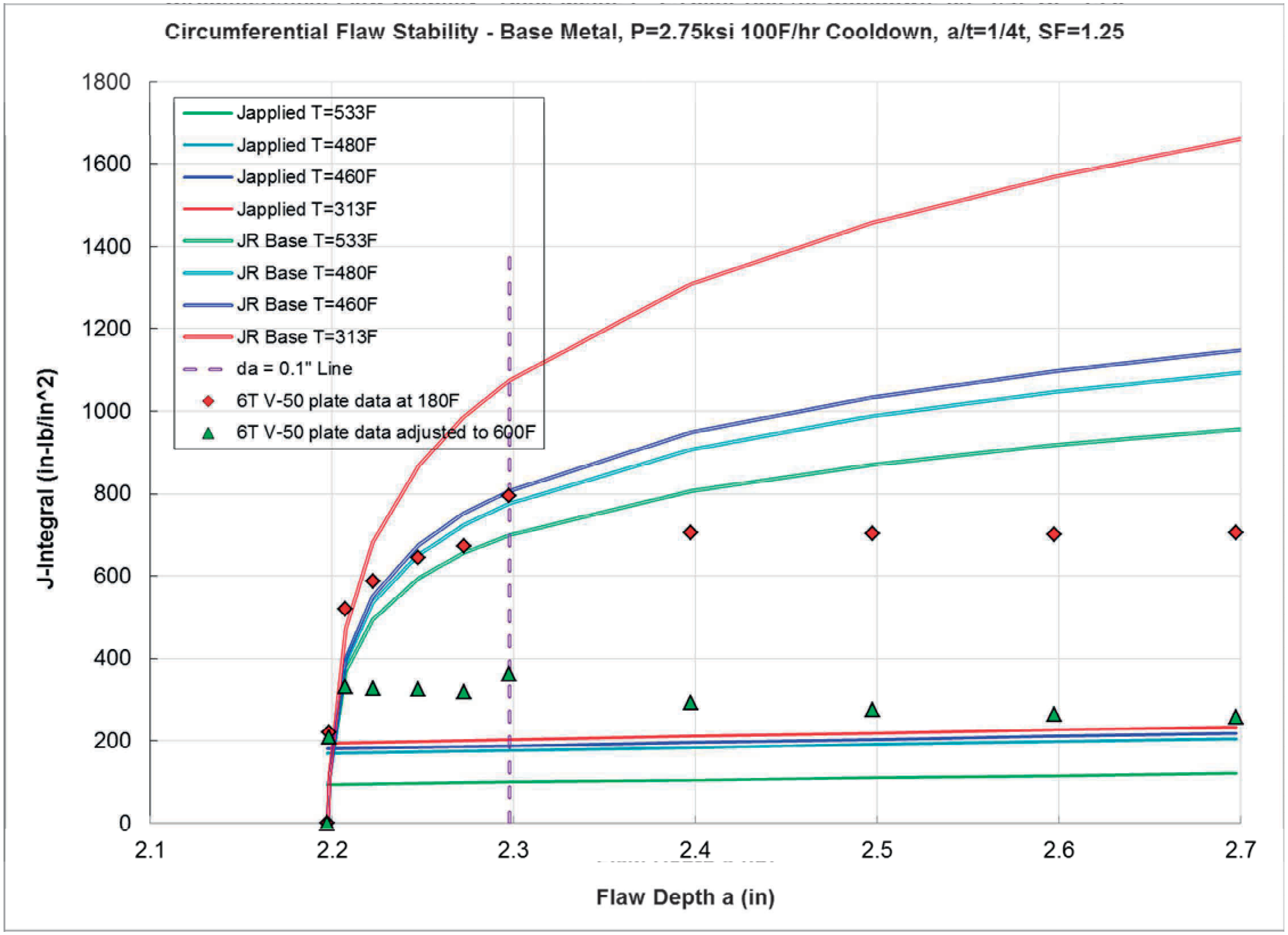
	Base Material			Weld Material		
	Circumferential Flaw			Circumferential Flaw		
Time (sec)	SF	J-applied x SF (in-lb/in^2)	$J_{0.1}$ material (in-lb/in^2)	SF	J-applied x SF (in-lb/in^2)	$J_{0.1}$ material (in-lb/in^2)
0	5.8	682	682	5.2	510	510
798	8.1	830	830	6.9	607	607
2,748	7.0	1,491	1,491	5.2	1,023	1,023
Minimum SF	**5.8**			**5.2**		

Table 5-6: Available Margins on Pressure Load for All Transients, Levels A, B, C and D, with Consideration of V-50 Plate Data

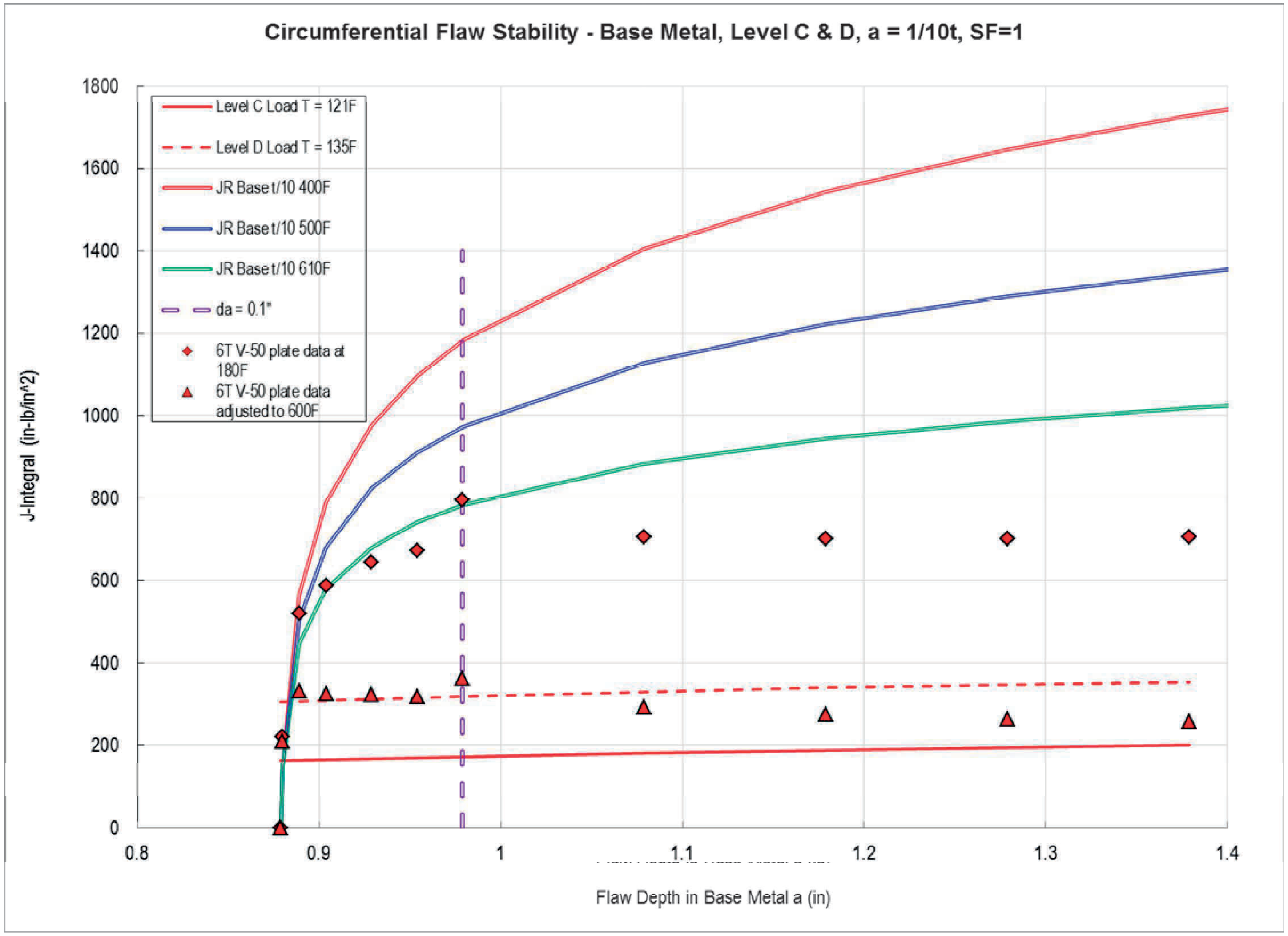
Level A and B	V-50 Plate			Level C	V-50 Plate			Level D	V-50 Plate		
	Circumferential Flaw		$J_{0.1}$ material (in-lb/in^2)		Circumferential Flaw		$J_{0.1}$ material (in-lb/in^2)		Circumferential Flaw		$J_{0.1}$ material (in-lb/in^2)
Time (sec)	SF	J-applied x SF (in-lb/in^2)		Time (sec)	SF	J-applied x SF (in-lb/in^2)		Time (sec)	SF	J-applied x SF (in-lb/in^2)	
0	**1.8**	397	397	0	3.4	397	397	0	3.4	397	397
2800	**1.6**	441	441	1,197	4.7	488	488	798	4.7	483	483
3600	**1.6**	459	459	4,122	1.9	893	893	2,748	1.5	868	868
5400	**1.7**	503	503								
7200	**1.8**	554	554								
9000	**1.9**	611	611								
10800	**18.0**	675	675								
Minimum SF	**1.6**			Minimum SF	**1.9**			Minimum SF	**1.5**		



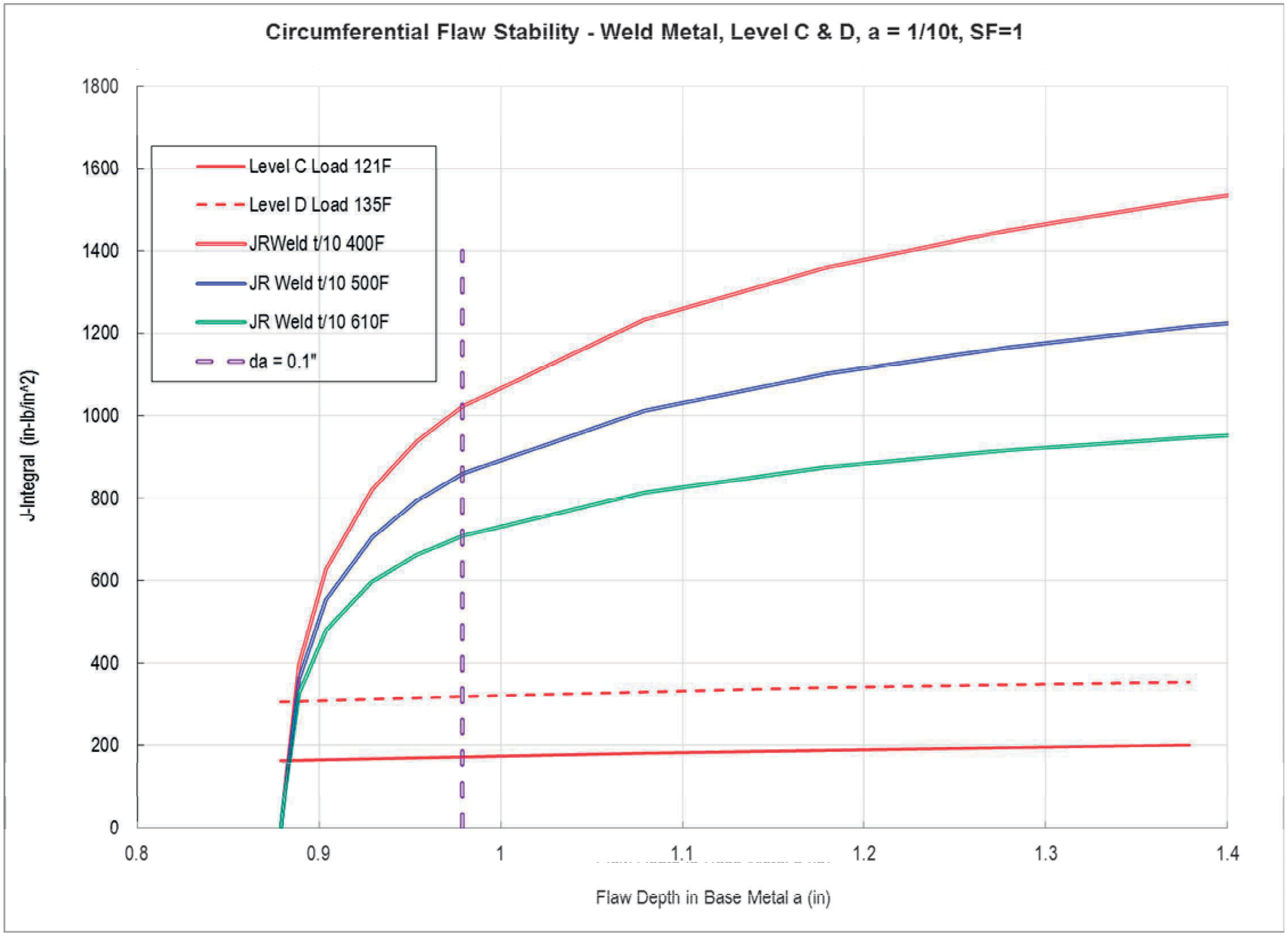
WCAP-17651-NP, Revision 0, Updated Figure 5-2 with Corrected, Limiting, Level C and D Transients



WCAP-17651-NP, Revision 0, Updated Figure 5-9 with V-50 Plate Data Included



WCAP-17651-NP, Revision 0, Updated Figure 5-12 with V-50 Plate Data Included and Corrected, Limiting, Level C and D Transients



WCAP-17651-NP, Revision 0, Updated Figure 5-13 with Corrected, Limiting, Level C and D Transients

RAI Response References

1. Regulatory Guide 1.161, “Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 Ft-Lb,” U. S. Nuclear Regulatory Commission, June 1995.
2. Westinghouse Report WCAP-17651-NP, Revision 0, “Palisades Nuclear Power Plant Reactor Vessel Equivalent Margins Analysis,” February 2013 (ADAMS Accession No. ML13295A451).
3. Code of Federal Regulations, 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.
4. ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, Appendix K, “Assessment of Reactor Vessels with Low Upper Shelf Charpy Impact Energy Levels,” 2007 Edition up to and including 2008 Addenda.
5. ASME B&PV Code, Section XI, Division 1, Appendix G, “Fracture Toughness Criteria for Protection Against Failure,” 1998 Edition up to and including 2000 Addenda.
6. NUREG/CR-6426, Volumes 1 and 2, “Ductile Fracture Toughness of Modified A 302 Grade B Plate Materials, Data Analysis,” U.S. Nuclear Regulatory Commission, January and February 1997.
7. NUREG/CR-5265, “Size Effects on J-R Curves for A 302-B Plate,” U.S. Nuclear Regulatory Commission, January 1989.

Attachment 3

Westinghouse WCAP-17403-NP
Revision 1

Palisades Nuclear Power Plant
Extended Beltline Reactor Vessel Integrity Evaluation

63 pages follow

Westinghouse Non-Proprietary Class 3

WCAP-17403-NP
Revision 1

January 2013

Palisades Nuclear Power Plant Extended Beltline Reactor Vessel Integrity Evaluation

Westinghouse

WCAP-17403-NP
Revision 1

Palisades Nuclear Power Plant Extended Beltline Reactor Vessel Integrity Evaluation

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Materials Center of Excellence – I

January 2013

Reviewer: Amy E. Freed*
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Approved: Frank C. Gift*, Manager
Materials Center of Excellence – I

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

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RECORD OF REVISION

Revision 0: Original Issue

Revision 1: This revision is being issued to correct the intermediate shell axial weld designations on Figures 1-1 and 1-2. In Revision 0, these welds are incorrectly labeled as 2-112 C – B – A, reading from left to right. The correct order is 2-112 B – C – A. No other changes were made to this report as a result of this correction; therefore, change bars are omitted.

TABLE OF CONTENTS

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EXECUTIVE SUMMARY

This report presents the Pressurized Thermal Shock (PTS), Upper-Shelf Energy (USE) and Adjusted Reference Temperature (ART) evaluations for the extended beltline region of the Palisades reactor vessel. PTS and USE evaluations must be shown to meet the applicable Nuclear Regulatory Commission (NRC) requirements through the end of the licensed operating period. Additionally, the calculated ART values must be shown to be less than those used in the current Palisades Analysis-of-Record (AOR) for Pressure-Temperature (P-T) limit curves. Palisades is currently licensed for 60 years of operation, which pertains to 42.1 effective full power years (EFPY). This is deemed end-of-license-extension (EOLE). Therefore, the PTS, USE and ART evaluations were performed at 42.1 EFPY in this report. The 42.1 EFPY extended beltline fluence values were determined by Westinghouse and are documented in WCAP-15353 – Supplement 2-NP. The conclusions to the PTS, USE and ART evaluations are as follows:

Extended Beltline Materials

Fluence calculations were performed for the Palisades reactor vessel upper (nozzle) shell plates, nozzle forgings, along with the associated upper shell and nozzle welds, and the lower shell to lower head weld to determine if any of these materials will exceed the 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) extended beltline material threshold at 42.1 EFPY. The Palisades reactor vessel materials that were identified as extended beltline materials in this report are the upper shell plates, the upper shell axial welds and the upper to intermediate shell circumferential weld. See Sections 3 and 4 for more details.

EOLE PTS Values

All of the extended beltline materials in the Palisades reactor vessel are projected to remain below the PTS screening criteria values of 270°F, for axially oriented welds and plates / forgings, and 300°F, for circumferentially oriented welds (per 10 CFR 50.61), through EOLE (42.1 EFPY). Additionally, the conclusions of this report, with regards to PTS, confirm that the traditional beltline materials remain limiting when compared to the extended beltline materials. This validates the conclusions of Structural Integrity Associates (SIA), Inc. Report No. 1000915.401. See Section 6 for more details.

EOLE USE Values

The limiting material in the Palisades extended beltline, Upper Shell (US) Plate D-3802-3, is predicted to drop below the USE screening criterion value of 50 ft-lb (per 10 CFR 50, Appendix G) through EOLE (42.1 EFPY) when considering an initial USE value based only on the available 95% shear Charpy V-Notch data for this material. However, this material is predicted to remain above the USE screening criterion value through EOLE when considering an initial USE value based on a CVGraph curve-fit of the available Charpy V-Notch data.

While US Plate D-3802-3 marginally meets the USE requirements at EOLE, based on a CVGraph curve-fit of the available Charpy V-Notch data, it is recommended that an Equivalent Margins Analysis (EMA) be performed for this material. It is recognized that two other materials located in the traditional beltline region of the reactor vessel are projected to drop below the 50 ft-lb screening criteria, which requires that an EMA be performed for these two materials. In addition, Palisades may elect to perform an uprate or submit for a second license extension in the future. Therefore, considering that an EMA will be performed for the two materials in the traditional beltline region of the Palisades reactor vessel that are

projected to drop below the 50 ft-lb screening criteria and future operation at Palisades may include higher flux levels, it is recommended that the EMA for the Palisades reactor vessel also include US Plate D-3802-3.

All of the remaining extended beltline materials in the Palisades reactor vessel are projected to remain above the USE screening criterion value of 50 ft-lb (per 10 CFR 50, Appendix G) through EOLE (42.1 EFPY). Therefore, the conclusions of Appendix D of WCAP-17341-NP, Revision 0 and this report confirm that three materials in the Palisades reactor vessel require a plant-specific EMA. See Section 7 for more details.

EOLE ART Values and P-T Limit Curve Applicability

The ART values for the extended beltline materials in the Palisades reactor vessel were calculated using the guidance provided in Regulatory Guide 1.99, Revision 2 through EOLE (42.1 EFPY). All of the ART values for the extended beltline materials are predicted to remain below those used in the AOR (WCAP-17341-NP, Revision 0) through EOLE (42.1 EFPY). Therefore, the P-T limit curves contained in the AOR for Palisades continue to be governed by the traditional beltline only. See Section 8 for more details.

1 INTRODUCTION

The definition of reactor vessel beltline, as given in the PTS Rule, 10 CFR 50.61 (Reference 1), is "*the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.*" Historically, only those materials directly adjacent to the active core, commonly referred to as the traditional beltline, have been evaluated with respect to Reactor Vessel Integrity (RVI).

The materials in the traditional beltline region of the Palisades reactor vessel, have been previously analyzed as part of several RVI evaluations including Pressurized Thermal Shock (PTS), Upper-Shelf Energy (USE) decrease, Adjusted Reference Temperature (ART) and Pressure-Temperature (P-T) limit curves. These RVI analyses for the traditional beltline have been completed and are documented in Structural Integrity Associates (SIA), Inc. Report No.'s 0901132.401, Revision 0 and 1000915.401, Revision 1 (References 2 and 3) for RT_{PTS} and Westinghouse Report WCAP-17341-NP, Revision 0 (Reference 4) for USE, ART and P-T limit curves.

As plants, such as Palisades, obtain License Renewal, additional materials are now being included in RVI analyses. The Generic Aging Lessons Learned (GALL) Report (NUREG-1801, Revision 2 (Reference 5)) states that any materials exceeding 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) must be monitored to evaluate the changes in fracture toughness. Any materials not previously evaluated for RVI that have predicted fluence levels greater than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) are now commonly referred to as the extended beltline.

Therefore, the purpose of this report is to evaluate the materials in the extended beltline region of the Palisades reactor vessel with respect to RVI. These materials are evaluated to determine their RT_{PTS}, USE and ART values at the end of life extension (EOLE), which corresponds to 42.1 Effective Full Power Years (EFPY). The applicability period of the current Palisades P-T limit curves is also analyzed to ensure that the curves applicability period is not impacted by the extended beltline materials.

Table 1-1 below gives a summary of the Palisades reactor vessel materials that have been previously analyzed and those that are considered for evaluation in this report. Figure 1-1 below shows the extent of the neutron fluence analysis of the Palisades reactor vessel per WCAP-15353 – Supplement 2-NP (Reference 6). Figure 1-2 below shows the extent of the 1 x 10^{17} n/cm^2 (E > 1.0 MeV) neutron fluence threshold for the Palisades reactor vessel. Note that these figures are for information only (FIO), and are not meant to delineate the exact, to scale, limits of the fluence analysis.

Section 2 of this report discusses the methodologies used to evaluate the various RVI analyses. The methodologies used to determine initial material property values are also given in this section. Section 3 identifies the materials in the Palisades reactor vessel extended beltline region that should be evaluated, and provides their associated neutron fluence values. Section 4 provides the extended beltline region material properties for Palisades. The material chemistry factors used in this analysis, along with applicable surveillance data are summarized in Section 5. The RT_{PTS}, USE, and ART values are determined in Sections 6 through 8, respectively. The P-T limit curve applicability period determination is also shown in Section 8.

Table 1-1 Palisades Reactor Vessel Materials and Analyses

Reactor Vessel Material[a]		Reference of RVI Analyses		
		RT_{PTS}[b]	USE[c]	ART and P-T Limit Curves[d]
Potential Extended Beltline Region[e]	Outlet Nozzle to Upper Shell Welds – Lowest Extent	This report	This report	This report
	Inlet Nozzle to Upper Shell Welds – Lowest Extent			
	Upper Shell Plates			
	Upper Shell Longitudinal Welds			
	Upper Shell to Intermediate Shell Circumferential Weld			
	Lower Shell to Lower Vessel Head Circumferential Weld			
Traditional Beltline Region	Intermediate Shell Plates	Ref. 3	Ref. 4	Ref. 4
	Intermediate Shell Longitudinal Welds	Refs. 2, 3		
	Intermediate to Lower Shell Circumferential Weld	Ref. 3		
	Lower Shell Plates	Ref. 3		
	Lower Shell Longitudinal Welds	Refs. 2, 3		

Notes for Table 1-1:

(a) The extent of the neutron fluence analysis (Reference 6) of the Palisades reactor vessel is shown on Figure 1-1 below.

(b) The RT_{PTS} methodology is defined in Section 2.2.1 of this report. In addition, supplemental information for this methodology is given in Section 2.2.4.

(c) The USE methodology is defined in Section 2.2.2 of this report.

(d) The ART methodology is defined in Section 2.2.3 of this report. The P-T limit curve applicability analysis methodology is defined in Section 2.2.5 of this report. Reference 4 documents the complete P-T limit curve development methodology.

(e) Extended beltline material properties were determined using the methodologies defined in Sections 2.1.1 and 2.1.2 of this report. The extent of the extended beltline is shown on Figure 1-2 below and defined in Section 3 of this report.

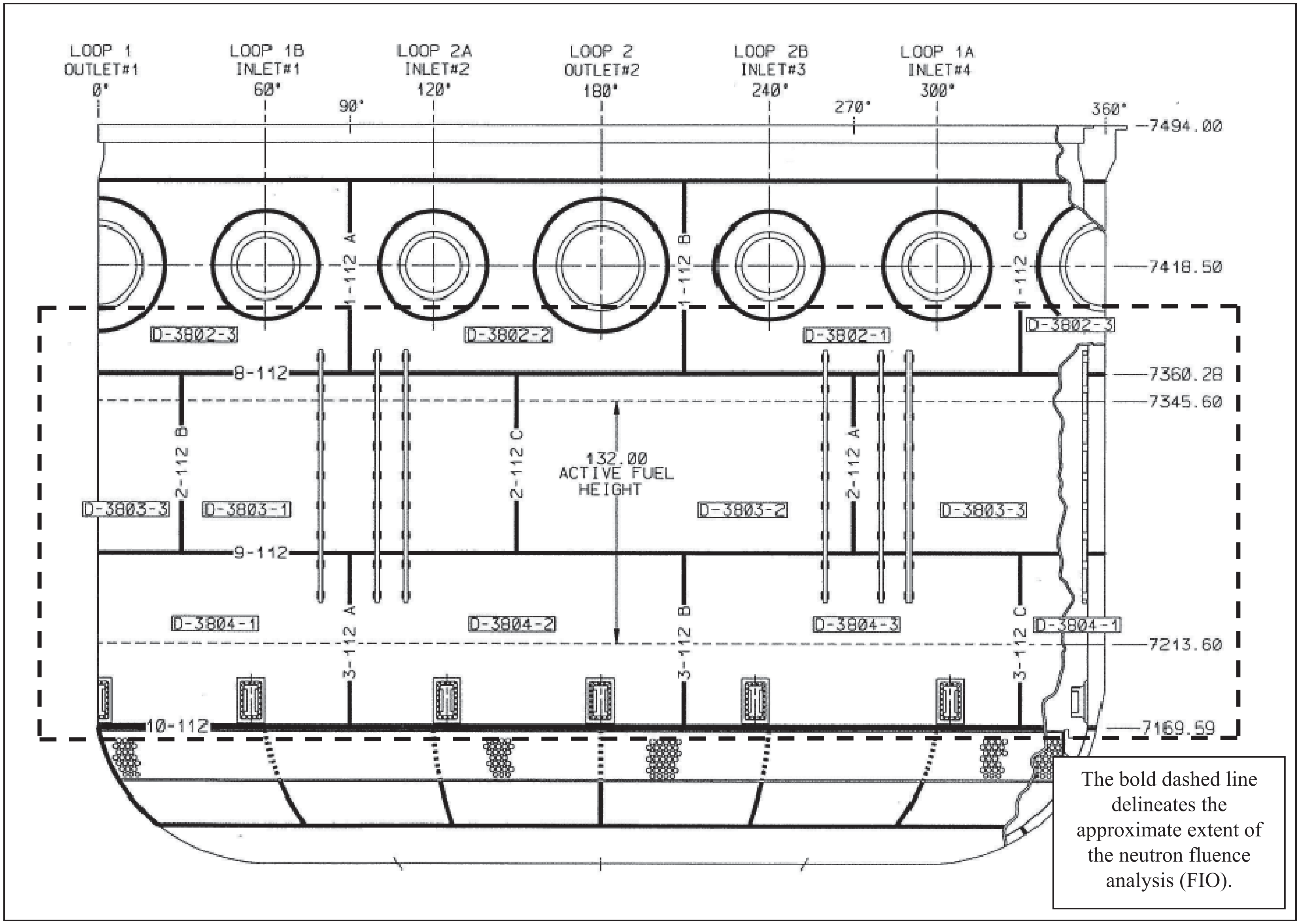


Figure 1-1 **Extent of the Neutron Fluence Analysis for the Palisades Reactor Vessel (FIO)**

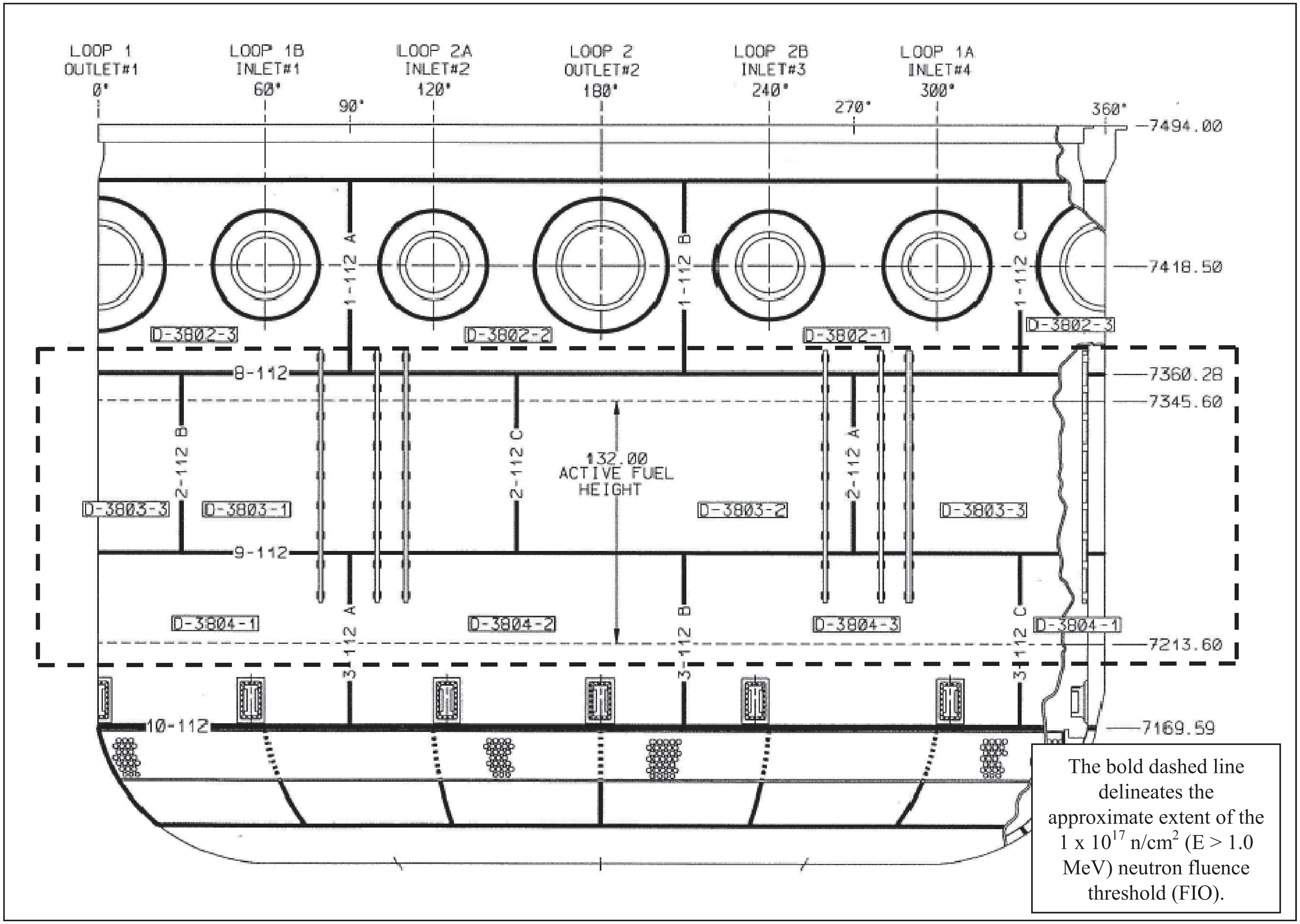


Figure 1-2 Extent of the 1 x 10^{17} n/cm^2 (E > 1.0 MeV) Neutron Fluence Threshold for the Palisades Reactor Vessel (FIO)

2 METHOD DISCUSSION

2.1 INITIAL MATERIAL PROPERTY DETERMINATION METHODOLOGY

2.1.1 Branch Technical Position 5-3 of NUREG-0800, Revision 2

Per NUREG-0800, Revision 2 Branch Technical Position MTEB 5-3 (Reference 7), the following equations and methodologies are to be used for determining initial (unirradiated) RT_{NDT} and USE values for ferritic materials.

Determination of RT_{NDT} for Vessel Materials

Temperature limitations are determined in relation to a characteristic temperature of the material, RT_{NDT}, that is established from the results of fracture toughness tests. Both drop-weight nil-ductility transition temperature (T_{NDT}) tests and Charpy V-Notch tests should be run to determine the RT_{NDT}. The T_{NDT} temperature, as determined by drop-weight tests (ASTM E-208-1969) is the RT_{NDT} if, at 60°F above the T_{NDT}, at least 50 ft-lbs of energy and 35 mils lateral expansion (LE) are obtained in Charpy V-Notch tests on specimens oriented in the weak direction (transverse to the direction of maximum working).

In most cases, the fracture toughness testing performed on vessel material for older plants did not include all tests necessary to determine the RT_{NDT} in this manner. Acceptable estimation methods for the most common cases, based on correlations of data from a large number of heats of vessel material, are provided below for guidance in determining RT_{NDT} when measured values are not available.

(1) If drop-weight tests were not performed, but full Charpy V-Notch curves were obtained, the T_{NDT} for SA-533 Grade B, Class 1 plate and weld material may be assumed to be the temperature at which 30 ft-lbs was obtained in Charpy V-Notch tests, or 0°F, whichever was higher.

(2) If drop-weight tests were not performed on SA-508, Class II forgings, the T_{NDT} may be estimated as the lowest of the following temperatures:

   (a) 60°F.
   (b) The temperatures of the Charpy V-Notch upper shelf.
   (c) The temperature at which 100 ft-lbs was obtained on Charpy V-Notch tests if the upper-shelf energy values were above 100 ft-lbs.

(3) If transversely-oriented Charpy V-Notch specimens were not tested, the temperature at which 50 ft-lbs and 35 mils LE would have been obtained on transverse specimens may be estimated by one of the following criteria:

(a) Test results from longitudinally-oriented specimens reduced to 65% of their value to provide conservative estimates of values expected from transversely oriented specimens.
(b) Temperatures at which 50 ft-lbs and 35 mils LE were obtained on longitudinally-oriented specimens increased 20°F to provide a conservative estimate of the temperature that would have been necessary to obtain the same values on transversely-oriented specimens.

(4) If limited Charpy V-Notch tests were performed at a single temperature to confirm that at least 30 ft-lbs was obtained, that temperature may be used as an estimate of the RT_{NDT} provided that at least 45 ft-lbs was obtained if the specimens were longitudinally oriented. If the minimum value obtained was less than 45 ft-lbs, the RT_{NDT} may be estimated as 20°F above the test temperature.

Estimation of Charpy V-Notch Upper-Shelf Energies

For the beltline region of reactor vessels, the upper-shelf toughness must account for the effects of neutron radiation. Reactor vessel beltline materials must have Charpy upper-shelf energy, in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lbs initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lbs.

If Charpy upper-shelf energy values were not obtained, conservative estimates should be made using results of tests on specimens from the first surveillance capsule removed.

If tests were only made on longitudinal specimens, the values should be reduced to 65% of the longitudinal values to estimate the transverse properties.

2.1.2 Charpy V-Notch Surveillance Data Curve Fitting

The hyperbolic tangent (TANH) curve approach is the most widely used model for determining the mean behavior of the Charpy energy-temperature relationship for reactor pressure vessel steels. The TANH fit provides parameters that have physical significance relative to the S-shape of the Charpy energy-temperature curve. For the symmetric TANH fit:

$$\textbf{Energy} = \boldsymbol{A + B}\ \textbf{tanh}\left[\frac{\boldsymbol{T - T_0}}{\boldsymbol{C}}\right]$$

Where,

A + B is the upper shelf

A – B is the lower shelf

T_0 is the mid-transition temperature at the energy level A

B/C is the slope of the mid-transition temperature T_0.

The computer code CVGraph (Version 5.3) is commonly used to determine the Charpy energy-temperature curve-fit parameters, and allows the engineering analyst flexibility in helping to provide the most meaningful fit to the Charpy energy data. Guidance for fitting Charpy energy data was developed in WCAP-14370 (Reference 8), and even though there is no standard method for using the TANH fit, some recommended guidance has been provided. This guidance starts first with fitting the lower shelf to a reasonably small number (i.e., A – B is set to either zero or some value slightly higher than zero (typically 2.2 ft-lb)); this assures that the curve does not extend below zero energy or give some excessively high level of lower shelf if there are little data at lower temperatures indicative of the lower shelf and the start of the transition region. Fixing the lower shelf reduces the fit to only three parameters.

The USE also can be fixed based upon some other definition of upper shelf using ASTM E185-82 (Reference 9), as an example. Following the ASTM E185-82 guidance, the USE can be determined for cases where there are ideally a minimum of three Charpy energy values that are on the upper shelf. Upper shelf generally means a fracture appearance of 100% shear, but for conservative estimates of USE, 95% shear data can be used. When evaluating USE, many different scenarios can arise. One scenario involves many data on the upper shelf, and one interpretation of ASTM E185-82 would allow the use of any three values (normally tested at one temperature) to define the USE. More typically, all data on the upper shelf are generally averaged (which can include data with 95% shear). One common problem with reactor pressure vessel surveillance data is not enough data with 100% shear, so that it is difficult to set an actual fixed USE (A + B).

When using the TANH model in CVGraph, the analyst can try different approaches which include a free fit upper shelf or some other predefined upper shelf levels. The recommendation for these limited data cases in WCAP-14370 is to fit the curve first fixing the lower shelf and leaving the upper shelf free to see what the curve fit predicts for the USE. If the USE value from this free upper shelf fit is within 3-5 ft-lb of some other estimate of USE, then the free fit parameters can be used for subsequent analyses. If the free upper shelf fit does not reasonably agree with the other estimate of USE, the analyst should choose a reasonable or conservative USE estimate and refit the data accordingly. The guidance in WCAP-14370 has been used in the USE analyses presented in this report.

2.2 REACTOR VESSEL INTEGRITY EVALUATIONS METHODOLOGY

2.2.1 Pressurized Thermal Shock (PTS)

The PTS Rule, 10 CFR 50.61 (Reference 1), requires that for each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected pressurized thermal shock reference temperature (RT_{PTS}) values accepted by the NRC for each reactor vessel material at the end-of-life fluence of the plant. This includes any reactor vessel material with an end-of-life fluence (E > 1.0 MeV) exceeding 1 x 10^{17} n/cm^2. This assessment must specify the basis for the projected value of RT_{PTS} for each vessel material, including the assumptions regarding core-loading patterns, and must specify the copper and nickel contents and the fluence value used in the calculation. This assessment must be updated whenever there is a significant change in projected values of RT_{PTS}, or upon request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are considered significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.

Per 10 CFR 50.61 (Reference 1), the following equations and variables are to be used for calculating RT_{PTS} values at the clad/base metal interface of the vessel. RT_{PTS} is also referred to as the EOL RT_{NDT} (reference nil-ductility transition temperature).

RT_{PTS} (°F) = Initial RT_{NDT} + M + ΔRT_{NDT}

Where,

Initial RT_{NDT} (°F) = $RT_{NDT(U)}$ = Initial Unirradiated RT_{NDT} value

M = Margin (°F) = $2 * \sqrt{\sigma_U^2 + \sigma_\Delta^2}$

Where,

σ_U = 0°F when $RT_{NDT(U)}$ is a measured value

σ_U = 17°F when $RT_{NDT(U)}$ is a generic value

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not credible or not used**

σ_Δ = 8.5°F when credible surveillance capsule data is used**

For welds:

σ_Δ = 28°F when surveillance capsule data is not credible or not used**

σ_Δ = 14°F when credible surveillance capsule data is used**

** σ_Δ not to exceed 0.5*ΔRT_{NDT} per 10 CFR 50.61 (Reference 1)

$$\Delta RT_{NDT}\ (^{o}F) = CF * FF$$

Where,

CF = chemistry factor (°F) calculated generically for copper (Cu) and nickel (Ni) content based on Tables 1 and 2 in Reference 1 for welds and plates, respectively (also referred to as Position 1.1). It can also be calculated using credible surveillance capsule data per Equation 5 of Reference 1 (also referred to as Position 2.1).

FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$, where the normalized neutron fluence at the clad/base metal interface on the inside surface of the vessel is $f = \Phi / (1.0 \times 10^{19})$. The units for Φ are n/cm^2, E > 1.0 MeV.

The RT_{PTS} screening criteria values are 270°F for plates, forgings and axial weld materials and 300°F for circumferential weld materials. All available surveillance data must be considered in the evaluation.

2.2.2 Upper-Shelf Energy (USE)

The predicted decrease in USE is determined as a function of fluence and copper content using either of the following:

- Figure 2 of Regulatory Guide 1.99, Revision 2 (Reference 10), Position 1.2, or
- Surveillance program test results and Figure 2 of Regulatory Guide 1.99, Revision 2, Position 2.2. Credibility Criterion 3 of Regulatory Guide 1.99, Revision 2 indicates that even if the surveillance data are not considered credible for determination of ΔRT_{NDT}, "they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82" (Reference 9).

Both methods require the use of the 1/4 thickness (1/4T) vessel fluence. Per Regulatory Guide 1.99, Revision 2, the following equation and variables are to be used for calculating 1/4T fluence values, which are then used to determine the predicted decrease in USE.

$$f = f_{surf} * e^{-0.24*(x)}$$

Where:

f = Vessel 1/4T fluence, $x10^{19}$ n/cm^2 (E > 1.0 MeV),

f_{surf} = Vessel inner wall surface fluence, $x10^{19}$ n/cm^2 (E > 1.0 MeV), and

x = The depth into the vessel wall from the inner surface, inches.

2.2.3 Adjusted Reference Temperature (ART)

Per Regulatory Guide 1.99, Revision 2 (Reference 10), the following equations and variables are to be used for calculating ART values at the clad/base metal interface and at the 1/4T and 3/4T locations.

ART (°F) = Initial RT_{NDT} + ΔRT_{NDT} + Margin

Where:

Initial RT_{NDT} (°F) = $RT_{NDT(U)}$ = Reference temperature of the unirradiated material

Margin (°F) = $2 * \sqrt{\sigma_I^2 + \sigma_\Delta^2}$

Where:

σ_I = 0°F when $RT_{NDT(U)}$ is a measured value

σ_I = 17°F when $RT_{NDT(U)}$ is a generic value

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not credible or not used**

σ_Δ = 8.5°F when credible surveillance capsule data is used**

For welds:

σ_Δ = 28°F when surveillance capsule data is not credible or not used**

σ_Δ = 14°F when credible surveillance capsule data is used**

** σ_Δ not to exceed 0.5*ΔRT_{NDT} per Regulatory Guide 1.99, Revision 2 (Reference 10)

ΔRT_{NDT} = CF * FF, (°F)

Where:

CF (°F) = chemistry factor based on the copper (Cu) and nickel (Ni) weight % of the material or based on the results of surveillance capsule test data. If the weight percent of copper and nickel is used to determine the CF, then the Position 1.1 CF is obtained from either Table 1 or 2 of Regulatory Guide 1.99, Revision 2. If surveillance capsule data is used to determine the CF, then the Position 2.1 CF is determined as follows:

$$CF = \frac{\sum_{i=1}^{n}[A_i * f_i^{(0.28-0.10\log f_i)}]}{\sum_{i=1}^{n}[f_i^{(0.56-0.20\log f_i)}]}$$

Where:

n = The number of surveillance data points

A_i = The measured value of ΔRT_{NDT}***

f_i = fluence for each surveillance data point

*** If the surveillance weld copper and nickel content differs from that of the vessel weld, then the measured values of ΔRT_{NDT} (A_i in the preceding equation for CF) shall be adjusted by multiplying them by the ratio of the chemistry factor for the vessel weld (CF_{VW}) to that for the surveillance weld (CF_{SW}) based on the copper and nickel content of the materials. In this case, ΔRT_{NDT} is determined as follows:

$$\Delta RT_{NDT} = (\text{measured } \Delta RT_{NDT}) * (CF_{VW} / CF_{SW})$$

FF = fluence factor = $f^{(0.28 - 0.10 \log (f))}$

Where:

f = Vessel inner wall surface fluence, 1/4T fluence, 3/4T fluence, or capsule fluence, $x10^{19}$ n/cm^2 (E > 1.0 MeV). The neutron fluence at any depth in the vessel wall is calculated as follows:

$f = f_{surf} * e^{-0.24 (x)}$

Where:

f	=	Vessel 1/4T or 3/4T fluence, $x10^{19}$ n/cm^2 (E > 1.0 MeV)
f_{surf}	=	Vessel inner wall surface fluence, $x10^{19}$ n/cm^2 (E > 1.0 MeV)
x	=	The depth into the vessel wall from the inner surface, inches

2.2.4 Supplemental Surveillance Data Guidance

Guidance was presented by the Nuclear Regulatory Commission (NRC) to industry at a meeting held by the NRC on February 12 and 13, 1998 (Reference 11), regarding adjustments of surveillance data for irradiation temperature and chemical composition differences when applying surveillance data from one plant to a different plant. This guidance also detailed how plants can determine the credibility status of their surveillance data, with or without consideration of sister plant data.

The guidance contained in Reference 11 was used by Structural Integrity Associates, Inc. in References 2 and 3 to determine the Position 2.1 chemistry factor values and credibility status of the Palisades surveillance materials. The information contained in those reports was directly utilized in WCAP-17341-NP, Revision 0 (Reference 4) and will be followed in the calculations detailed in this report.

2.2.5 Pressure-Temperature (P-T) Limit Curve Applicability

P-T limit curves, also referred to as Heatup and Cooldown limit curves, are based on the most limiting 1/4T and 3/4T adjusted reference temperature values for a given reactor vessel. Westinghouse has previously calculated P-T limit curves for Palisades under normal operating conditions through 42.1 EFPY in WCAP-17341-NP, Revision 0 (Reference 4), considering only the materials contained in the traditional beltline region. This current evaluation determines if the applicability period is affected by any materials in the extended beltline by comparing the ART values contained in the AOR with the ART values calculated for the extended beltline materials. If the ART values used in the previous analysis are *higher* than the ART values calculated for the extended beltline materials, the applicability period of the current curves will remain bounding. If the ART values used in the previous analysis are *lower* than the ART values calculated for the extended beltline materials, the applicability period of the current curves will be shortened. This new period of applicability is calculated based on a comparison of the ART values via linear interpolation.

3 CALCULATED FLUENCE

Fast neutron fluence (E > 1.0 MeV) values for the Palisades extended beltline were calculated in WCAP-15353 – Supplement 2-NP (Reference 6). The assessment was performed based on the guidance specified in Regulatory Guide 1.190 (Reference 12). The use of Regulatory Guide 1.190 has been approved by the NRC for the Palisades reactor vessel in an NRC Safety Evaluation Report (SER) from November of 2000 (Reference 13).

The industry accepted definition of "extended beltline" is given in Item IV.A2.R-84 of NUREG-1801, Revision 2 (Reference 5), and states that any materials exceeding 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) must be monitored to evaluate the changes in fracture toughness. Reactor vessel materials that are not traditionally thought of as being plant limiting because of low levels of neutron radiation must now be evaluated to determine the accumulated fluence at EOLE (42.1 EFPY). Therefore, fluence calculations were performed for the Palisades reactor vessel upper (nozzle) shell plates, nozzle forgings, along with the associated upper shell and nozzle welds, and the lower shell to lower head weld to determine if they will exceed 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at 42.1 EFPY. The materials that exceed this threshold are referred to as extended beltline materials in this report and are evaluated to assure that the PTS and USE acceptance criteria are met through EOLE. Additionally, ART values are calculated in this report to assure that no material's ART values exceed those used in the current Palisades P-T limit curve analysis of record.

For the PTS, USE and ART evaluations, the surface, 1/4T and 3/4T fluence values and fluence factors are needed. These values are summarized in Table 3-1 for the Palisades extended beltline materials. The neutron fluence information is summarized at 42.1 EFPY (EOLE). Note that the inlet and outlet nozzles and nozzle to shell welds, along with the lower shell to lower head circumferential weld did not reach the 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) fluence threshold; therefore, they are not included in Table 3-1 or any subsequent RVI evaluations contained in this report.

The reactor vessel thickness for Palisades is 8.79 inches (Reference 6). Hence, for the 1/4T and 3/4T fluence calculations, the depth into the vessel wall is as follows:

$x_{1/4T}$ = (0.25 * 8.79 inches) = 2.20 inches

$x_{3/4T}$ = (0.75 * 8.79 inches) = 6.59 inches

Table 3-1 Palisades Surface, 1/4T and 3/4T Fluence and Fluence Factor Values at 42.1 EFPY

Material Description	Surface fluence, f[(a)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	Surface FF	1/4T f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	3/4T FF
Extended Beltline Materials						
Upper Shell Plates (D-3802-1, D-3802-2, D-3802-3)	0.1529	0.5071	0.0902	0.3966	0.0314	0.2256
Upper Shell Axial Welds 1-112 A/B/C	0.09707	0.4109	0.0573	0.3148	0.0200	0.1718
Upper Shell to Intermediate Shell Circumferential Weld 8-112	0.1529	0.5071	0.0902	0.3966	0.0314	0.2256

Note for Table 3-1:

(a) All values taken from WCAP-15353 – Supplement 2-NP (Reference 6).

4 FRACTURE TOUGHNESS PROPERTIES

The fracture toughness properties of the ferritic materials in the reactor coolant pressure boundary are determined in accordance with the fracture toughness requirements in NUREG-0800, Revision 2, Branch Technical Position MTEB 5-3 (Reference 7) and the requirements of Subparagraph NB-2331 of Section III of the ASME B&PV Code (Reference 14), as specified by Paragraph II – D of 10 CFR Part 50, Appendix G (Reference 15). The extended beltline unirradiated material properties of the Palisades reactor vessel are presented in Table 4-1.

Table 4-1 Extended Beltline Material Properties for the Palisades Reactor Vessel[(a)]

Material Description	Heat Number	Chemical Composition		Fracture Toughness Properties	
		Cu Wt. %	Ni Wt. %	Initial RT_{NDT} (°F)	Initial USE (ft-lb)
Extended Beltline Materials					
Upper Shell (US) Plate D-3802-1	C-1279[(b)]	0.21	0.48	10[(e)]	75[(f)]
US Plate D-3802-2	C-1308	0.19	0.52	19[(e)]	73[(f)]
US Plate D-3802-3	C-1281	0.25	0.57	10[(e)]	62.2/59[(g)]
US Axial Welds 1-112 A/B/C	W5214[(c)]	0.213	1.007	-56[(d)]	118
US to Intermediate Shell (IS) Circumferential (Circ.) Weld 8-112	34B009[(c)]	0.192	0.98	-56[(d)]	111

Notes for Table 4-1:

(a) All values obtained from P-PENG-ER-006, Revision 0 (Reference 16), unless otherwise noted.

(b) US Plate D-3802-1 is the same heat of material (C-1279) as the traditional beltline region Intermediate Shell Plates D-3803-1 and D-3803-3 per WCAP-17341-NP, Revision 0 (Reference 4). Non-credible surveillance data, for use in the ΔRT_{NDT} determination, is available for this material. However, per Credibility Criterion 3 of Regulatory Guide 1.99, Revision 2, the surveillance data, for use in the percent decrease in USE determination, is deemed credible for this material.

(c) Weld Heat #'s W5214 and 34B009 are contained in the traditional beltline region of the Palisades reactor vessel. Their chemical composition and fracture toughness properties were taken from WCAP-17341-NP, Revision 0 (Reference 4). These welds were fabricated with Linde 1092 flux type. Note that weld Heat # W5214 has 'not fully credible' surveillance data associated with it per Reference 4, for use in the ΔRT_{NDT} determination. However, per Credibility Criterion 3 of Regulatory Guide 1.99, Revision 2, the surveillance data, for use in the percent decrease in USE determination, is deemed credible for both weld heats.

(d) Generic initial RT_{NDT} values for weld Heat #'s W5214 and 34B009 were taken from WCAP-17341-NP, Revision 0 (Reference 4).

(e) Initial RT_{NDT} values were determined using the Charpy V-Notch data contained in the Certified Material Test Reports (CMTRs) and in accordance with the requirements of Subparagraph NB-2331 of Section III of the ASME B&PV Code (Reference 14), as specified by Paragraph II – D of 10 CFR Part 50, Appendix G (Reference 15). Furthermore, portions of the fracture toughness requirements in NUREG-0800, Revision 2, Branch Technical Position MTEB 5-3 (Reference 7) were used in the determination of the initial RT_{NDT} values. Following the guidance provided in MTEB 5-3, Section 1.1(3)(b), the initial RT_{NDT} values associated with the longitudinal (strong) orientation were increased by 20°F to provide a conservative estimate for a transversely-oriented specimen. Lastly, per Reference 7, the determined Charpy V-Notch initial RT_{NDT} was compared to the T_{NDT} as reported in the P-PENG-ER-006 report (Reference 16) because the initial RT_{NDT} must be greater than or equal to the T_{NDT}. Appendix A of this report contains the Charpy V-Notch plots, which were refitted from the CMTRs using a hyperbolic tangent curve fitting program and were used to obtain the initial RT_{NDT} values.

(f) Initial USE values were determined using the guidance contained in ASTM E185-82 (Reference 9) and the data contained in P-PENG-ER-006, Revision 0 (Reference 16). Note that the Charpy V-Notch Tests were performed on longitudinally (strong direction) oriented specimens. Following the guidance provided in MTEB 5-3, Section 1.2, the upper-shelf energy values were reduced to 65% of the values associated with the strong direction (longitudinally-oriented) in order to approximate the properties in the weak direction (transversely-oriented).

(g) Initial USE value for US Plate D-3802-3 was determined using an unconstrained TANH fit to the available Charpy energy data in P-PENG-ER-006, Revision 0 (Reference 16) as described below. The initial upper-shelf energy value of 62.2 ft-lb, after the 65% reduction in Note (f), is reasonable when comparisons are made with the two other plate materials in the extended beltline, which had data available at 100% shear fracture appearance resulting in much higher initial USE levels. If the data were treated as described above in Note (f) using 95% shear data, the corresponding initial USE value would be 59 ft-lb.

Determination of an Initial Curve-Fit USE Value for US Plate D-3802-3

The definition for USE is given in ASTM E185-82 (Reference 9), Sections 4.17 and 4.18, and reads as follows:

> "*Charpy transition curve* – a graphic representation of Charpy data, including absorbed energy, lateral expansion, and fracture appearance, extending over a range including the lower shelf energy (<5 % shear), transition region, and the upper shelf energy (>95 % shear)."
>
> "*upper shelf energy level* – the average energy value for all Charpy specimens (normally three) whose test temperature is above the upper end of the transition region. For specimens tested in sets of three at each test temperature, the set having the highest average may be regarded as defining the upper shelf energy."

Using engineering judgment and ASTM E185-82, Sections 4.17 and 4.18, "Above the upper end of the transition region" is considered to be > 95% shear. Per P-PENG-ER-006, Revision 0 (Reference 16), the Charpy testing for US Plate D-3802-3 did not achieve 100% Shear values; a maximum of 95% Shear was reached. It is probable that the 95% shear data is indicative of the onset of upper shelf, but not the actual upper shelf, which would be > 95% shear. In addition, the actual upper-shelf energy is likely to have been greater than the average shown (91 ft-lbs, pre-MTEB 5-3 reduction) if Charpy impact tests were performed at a temperature above 160°F. Finally, the correlation coefficient of the unconstrained CVGraph curve fit of the available data for this material is very high (0.987), such that if this unirradiated Charpy data was actually surveillance capsule Charpy data, it would satisfy Credibility Criterion 2 of Regulatory Guide 1.99, Revision 2 (Reference 10).

Therefore, considering the above, the Charpy impact data was refitted using a hyperbolic tangent curve fitting program (CVGraph, Version 5.3) to predict the upper-shelf energy at 100% shear. To do this, the upper-shelf energy, which is typically fixed in the program (See Appendix A), was allowed to float such that a new upper-shelf energy was calculated by the curve fit. Figures 4-1a and 4-1b present the CVGraph figure, which documents the predicted initial USE value.

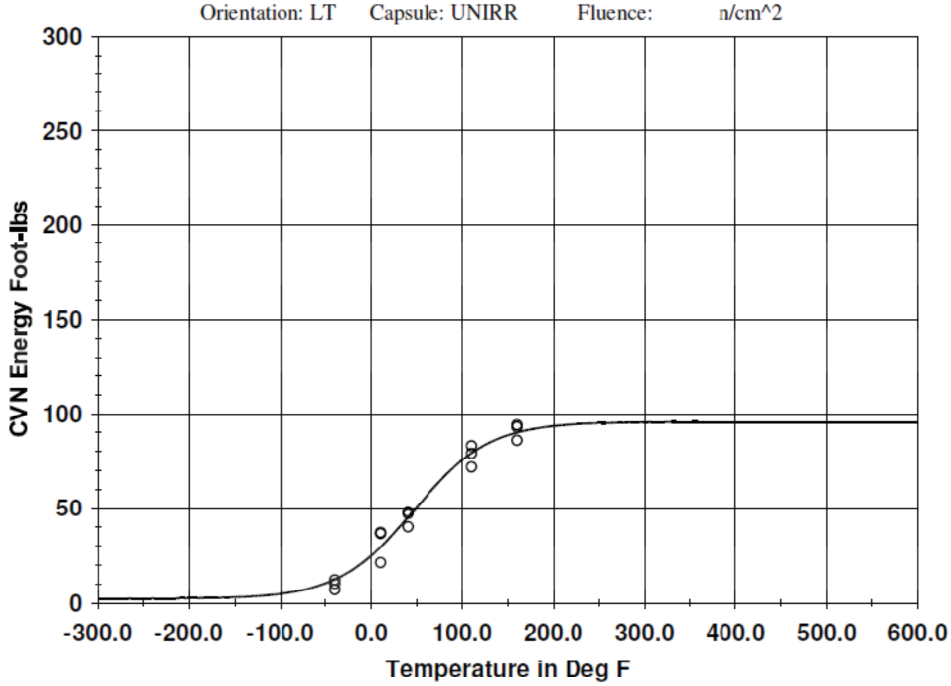
Initial Curve-Fit USE Value for US Plate D-3802-3 Conclusion

The initial curve-fit USE value based on longitudinal Charpy data was determined to be 95.7 ft-lbs by CVGraph, Version 5.3. This value is reasonable when compared with the as measured average initial USE value of 91 ft-lbs, which considered 95% shear Charpy data points determined from longitudinal specimens. The initial curve-fit USE value, after reduction to 65% per MTEB 5-3, is 62.2 ft-lbs.

Therefore, an initial curve-fit USE value of 62.2 ft-lb will also be assigned to US Plate D-3802-3. This initial USE value of 62.2 ft-lb will be used in Section 7 in order to calculate the projected EOLE USE for this material.

US PLATE D-3802-3 - IMPACT ENERGY - LT - UNIRRADIATED

CVGRAPH 5.3 Hyperbolic Tangent Curve Printed on 07/01/2011 07:25 AM
Page 1
Coefficients of Curve 1
A = 48.95 B = 46.75 C = 82.04 T0 = 46.6 D = 0.00E+00
Equation is A + B * [Tanh((T-To)/(C+DT))]
Upper Shelf Energy=95.7 Lower Shelf Energy=2.2(Fixed)
Temp@30 ft-lbs=11.4 Deg F Temp@50 ft-lbs=48.5 Deg F
Plant: Palisades Material: SA302BM Heat: C-1281-1
Orientation: LT Capsule: UNIRR Fluence: n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-40.00	7.00	12.30	-5.30
-40.00	10.00	12.30	-2.30
-40.00	12.00	12.30	-.30
10.00	21.00	29.38	-8.38
10.00	36.50	29.38	7.12
10.00	37.00	29.38	7.62
40.00	48.00	45.20	2.80
40.00	40.00	45.20	-5.20
40.00	47.00	45.20	1.80

Figure 4-1a CVGraph Unconstrained Fit of US Plate D-3802-3 Charpy Data for Use in Determination of the Initial USE Value

US PLATE D-3802-3 - IMPACT ENERGY - LT - UNIRRADIATED

Page 2
Plant: Palisades Material: SA302BM Heat: C-1281-1
Orientation: LT Capsule: UNIRR Fluence: n/cm^2

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
110.00	79.00	79.26	-.26
110.00	83.00	79.26	3.74
110.00	72.00	79.26	-7.26
160.00	86.00	90.15	-4.15
160.00	94.00	90.15	3.85
160.00	93.00	90.15	2.85

Correlation Coefficient = .987

Figure 4-1b CVGraph Unconstrained Fit of US Plate D-3802-3 Charpy Data for Use in Determination of the Initial USE Value

5 CHEMISTRY FACTORS AND UPPER-SHELF ENERGY SURVEILLANCE DATA

As described in Section 2 of this report, Position 1.1 chemistry factors for each reactor vessel extended beltline material are calculated using the best-estimate copper and nickel weight percent values of the material and Tables 1 and 2 of 10 CFR 50.61. The best-estimate copper and nickel weight percent values for the Palisades reactor vessel extended beltline materials were provided in Table 4-1 of this report. The Position 2.1 chemistry factors are calculated for the materials that have available surveillance program results. The calculation is performed using the method described in 10 CFR 50.61 and Regulatory Guide 1.99, Revision 2, as summarized in Sections 2.2.1 and 2.2.3 of this report, respectively.

Normally, no surveillance data exists for materials in the extended beltline region of reactor pressure vessels. However, for Palisades, the material heats for US Plate D-3802-1 (Heat # C-1279), US Axial Welds 1-112 A/B/C (Heat # W5214) and US to IS Circumferential Weld 8-112 (Heat # 34B009) are also contained in the traditional beltline region of the Palisades reactor vessel. The capsule Charpy data from all of the capsules withdrawn and tested to date were used in the calculation of the Position 2.1 chemistry factors. In addition to the chemistry factor data, measured USE percent decrease data is also available for the three material heats shared between the extended and traditional beltline.

The Position 1.1 chemistry factors are summarized along with the Position 2.1 chemistry factors in Table 5-1 for Palisades. The measured USE decrease data for Palisades is summarized in Table 5-2.

Table 5-1 Summary of Palisades Positions 1.1 and 2.1 Chemistry Factors

Material Description	Chemistry Factor (˚F)	
	Position 1.1[(a)]	**Position 2.1[(b)]**
Extended Beltline Materials		
US Plate D-3802-1[(c)]	139.4	147.71
US Plate D-3802-2	133.2	- - -
US Plate D-3802-3	171.8	- - -
US Axial Welds 1-112 A/B/C (Heat # W5214)	230.73[(b)]	227.74
US to IS Circ. Weld 8-112 (Heat # 34B009)	217.7[(b)]	- - -

Notes for Table 5-1:

(a) Position 1.1 Chemistry Factors for the extended beltline materials were calculated using the copper and nickel weight percents presented in Table 4-1 of this report and Tables 1 and 2 of 10 CFR 50.61 (Reference 1), unless otherwise noted.

(b) Position 1.1 Chemistry Factors for the extended beltline welds and the Position 2.1 Chemistry Factors are consistent with the values documented in Table 2-3 of WCAP-17341-NP, Revision 0 (Reference 4).

(c) The Cu and Ni wt. % values used to calculate the Position 1.1 Chemistry Factor were taken from as measured data summarized in P-PENG-ER-006, Revision 0 (Reference 16). These Cu and Ni wt. % values for the US Plate D-3802-1 are slightly different from the two plates that share this material heat # in the traditional beltline (Reference 4); however, since the upper shell plate is still the same heat of material, the Position 2.1 Chemistry Factor will still be applied to this material.

Table 5-2 Summary of Palisades Capsule Fluence Values and Measured USE Percent Decrease Data[a]

Material	Capsule	Capsule Fluence ($x10^{19}$ n/cm^2, E > 1.0 MeV)	Measured USE Decrease (%)
Weld Heat # W5214[b]	SA-60-1	1.50	46.9
	SA-240-1	2.38	48.9
Weld Heat # 34B009[b]	SA-60-1	1.50	51.5
	SA-240-1	2.38	49.6
Plate Heat # C-1279[b] (Longitudinal)	A-240	4.09	42.4
	W-290	0.938	27.7
	W-110	1.64	33.5
	W-100	2.09	34.1
Plate Heat # C-1279[b] (Transverse)	A-240	4.09	35.2
	W-290	0.938	17.6
	W-100	2.09	28.1

Notes for Table 5-2:

(a) Information taken from Table D-2 of WCAP-17341-NP, Revision 0 (Reference 4).
(b) Per Credibility Criterion 3 of Regulatory Guide 1.99, Revision 2, the surveillance data, for use in the percent decrease in USE determination, has been deemed credible for the weld and plate materials.

6 PRESSURIZED THERMAL SHOCK CALCULATIONS

A limiting condition on reactor vessel integrity known as PTS may occur during a severe system transient such as a loss-of-coolant accident or steam line break. Such transients may challenge the integrity of the reactor vessel under the following conditions: severe overcooling of the inside surface of the vessel wall followed by high repressurization; significant degradation of vessel material toughness caused by radiation embrittlement; and the presence of a critical-size defect anywhere within the vessel wall.

In 1985, the U.S. NRC issued a formal ruling (10 CFR 50.61) on PTS (Reference 1) that established screening criteria on reactor vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting reactor vessel component at the end-of-license, termed RT_{PTS}. RT_{PTS} screening values were set by the U.S. NRC for axial welds, forgings or plates, and circumferential weld seams for plant operation to the end of plant license. All domestic PWR vessels have been required to evaluate vessel embrittlement in accordance with the criteria through the end-of-license. The U.S. NRC revised 10 CFR 50.61 in 1991 and 1995 to change the procedure for calculating radiation embrittlement. These revisions make the procedure for calculating the reference temperature for pressurized thermal shock (RT_{PTS}) values consistent with the methods given in Regulatory Guide 1.99, Revision 2 (Reference 10).

These accepted methods were used with the surface fluence of Section 3 to calculate the following RT_{PTS} values for the Palisades reactor vessel extended beltline materials at 42.1 EFPY (EOLE). The EOLE RT_{PTS} calculations are summarized in Table 6-1.

PTS Conclusion

For Palisades, the limiting extended beltline RT_{PTS} values at 42.1 EFPY are 131.1°F and 119.9°F (see Table 6-1); these values correspond to US Plate D-3802-3 using Position 1.1 (Axially oriented welds and plates) and US to IS Circumferential Weld 8-112 (Heat # 34B009) using Position 1.1 (Circumferentially oriented weld). Therefore, all of the extended beltline materials in the Palisades reactor vessel are below the RT_{PTS} screening criteria values of 270°F, for axially oriented welds and plates / forgings, and 300°F, for circumferentially oriented welds through EOLE (42.1 EFPY).

The PTS conclusion confirms that the traditional beltline materials remain limiting when compared to the extended beltline materials. This validates the conclusions of Structural Integrity Associates (SIA), Inc. Report No. 1000915.401 (Reference 3).

Table 6-1 RT_{PTS} Calculations for the Palisades Reactor Vessel Extended Beltline Materials at 42.1 EFPY[a]

Material Description	CF[b] (°F)	Fluence[c] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF[c]	$RT_{NDT(U)}$[d] (°F)	ΔRT_{NDT} (°F)	σ_U[d] (°F)	σ_Δ[f] (°F)	Margin (°F)	RT_{PTS} (°F)
Extended Beltline Materials									
US Plate D-3802-1	139.4	0.1529	0.5071	10	70.7	0	17	34.0	114.7
Using Non-Credible Surveillance Data	147.71	0.1529	0.5071	10	74.9	0	17	34.0	118.9
US Plate D-3802-2	133.2	0.1529	0.5071	19	67.5	0	17	34.0	120.5
US Plate D-3802-3	171.8	0.1529	0.5071	10	87.1	0	17	34.0	131.1
US Axial Welds 1-112 A/B/C (Heat # W5214)	230.73	0.09707	0.4109	-56[e]	94.8	17[e]	28	65.5	104.3
Using Not Fully Credible Surveillance Data	227.74	0.09707	0.4109	-56[e]	93.6	17[e]	28	65.5	103.1
US to IS Circ. Weld 8-112 (Heat # 34B009)	217.7	0.1529	0.5071	-56[e]	110.4	17[e]	28	65.5	119.9

Notes for Table 6-1:

(a) The 10 CFR 50.61 methodology was utilized in the calculation of the RT_{PTS} values. See Section 2.2.1 of this report for details.
(b) Taken from Table 5-1 of this report.
(c) Taken from Table 3-1 of this report.
(d) Initial RT_{NDT} values are taken from Table 4-1 of this report and are measured values, unless otherwise noted. For measured initial RT_{NDT} values, σ_U = 0°F.
(e) Initial RT_{NDT} values are generic; therefore, σ_U = 17°F.
(f) Per WCAP-17341-NP, Revision 0 (Reference 4), surveillance data of the plate material and weld Heat # W5214 were considered to be non-credible and not fully credible, respectively. Per the guidance of 10 CFR 50.61, the base metal σ_Δ = 17°F for plate materials without surveillance data as well as for the plate material with non-credible surveillance data. The weld metal σ_Δ = 28°F for welds without surveillance data as well as for the weld metal with not fully credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT}.

7 UPPER-SHELF ENERGY CALCULATIONS

The requirements for USE are contained in 10 CFR 50, Appendix G (Reference 15). 10 CFR 50, Appendix G requires utilities to submit an analysis at least 3 years prior to the time that the USE of any reactor vessel material is predicted to drop below 50 ft-lb.

Regulatory Guide 1.99, Revision 2 defines two methods that can be used to predict the decrease in USE due to irradiation. The method to be used depends on the availability of credible surveillance capsule data. For vessel materials that are not in the surveillance program or are not credible, the Charpy USE (Position 1.2) is assumed to decrease as a function of fluence and copper content, as indicated in Regulatory Guide 1.99, Revision 2 (Reference 10).

When two or more credible surveillance data sets become available from the reactor vessel, they may be used to determine the Charpy USE of the surveillance materials. The surveillance data are then used in conjunction with Figure 2 of the Regulatory Guide to predict the decrease in USE (Position 2.2) of the reactor vessel materials due to irradiation.

The 42.1 EFPY (EOLE) Position 1.2 USE values of the vessel materials can be predicted using the corresponding 1/4T fluence projection, the copper content, and Figure 2 in Regulatory Guide 1.99, Revision 2. In applying either Position 1.2 or 2.2, the TANH fitted Charpy energy data followed the general principles described in Section 2.1.2 of this report.

The predicted Position 2.2 USE values are determined for the extended beltline materials whose heat numbers are contained in the surveillance program by using the reduced plant surveillance data along with the corresponding 1/4T fluence projection. The reduced plant surveillance data for the Palisades weld materials are contained in Appendix D of WCAP-17341-NP, Revision 0 (Reference 4). The reduced plant surveillance data for the Palisades plate materials are also contained in Appendix D of WCAP-17341-NP, Revision 0. This data is summarized in Table 5-2 of this report. The weld and plate reduced surveillance data were plotted on Regulatory Guide 1.99, Revision 2, Figure 2 (see Figures 7-1 and 7-2 of this report) using the surveillance capsule fluence values from WCAP-17341-NP, Revision 0. This data was fitted by drawing a line parallel to the existing lines as the upper bound of all the surveillance data. These reduced lines were used instead of the existing lines to determine the Position 2.2 EOLE USE values.

The projected USE values were calculated to determine if the Palisades reactor vessel extended beltline materials remain above the 50 ft-lb limit at EOLE. These calculations are summarized in Table 7-1.

USE Conclusion

All of the extended beltline materials are projected to have USE values greater than 50 ft-lb at 42.1 EFPY. However, it is noted that the initial USE for US Plate D-3802-3 has been determined using two approaches. The first approach used to project the USE at 100% shear is similar to the determination of the 30 ft-lb transition temperature (T_{30}) using a best fit TANH curve through all of the available data. By allowing a free, or unconstrained, fit to the upper shelf (since no data at 100% shear were measured), and only fixing the lower shelf level, a calculated initial upper-shelf energy can be obtained. This method of initial USE determination along with the orientation and projected drop in USE adjustments, per NUREG-0800, Revision 2 and Regulatory Guide 1.99, Revision 2, projects that the USE value for US Plate D-3802-3 is 50.1 ft-lb at EOLE fluence. It is noted that the correlation coefficient from CVGraph for the fit is equal to 0.987.

The second approach conservatively determines the USE for this plate from the highest temperature tested Charpy data, which corresponds to only 95% shear. Averaging these data and then making the orientation and projected drop in USE adjustments, per NUREG-0800, Revision 2 and Regulatory Guide 1.99, Revision 2, the USE at EOLE fluence will be below 50 ft-lbs at 42.1 EFPY.

It is noted that generic EMA evaluations for both the Westinghouse and Combustion Engineering fleets, WCAP-13587, Revision 1 (Reference 17) and CE NPSD-993 (Reference 18), respectively, predict that a lower-bound USE value below 50 ft-lb is justified. These evaluations show that there is considerable margin between the actual 10 CFR 50, Appendix G 50 ft-lb limit and the calculated lower-bound values. Therefore, using engineering judgment, the increase gained in the projected EOLE USE value by curve-fitting the Charpy data is minimal when compared to the margin that an EMA would justify. A plant-specific EMA should be completed on this material, as concluded above, in order to document the exact lower-bound USE value for this material at EOLE.

Reference 4 indicates that the USE at EOLE fluence will fall below the 50 ft-lb limitation for two materials in the traditional beltline region of the Palisades reactor vessel. Therefore, an Equivalent Margins Analysis (EMA) is to be performed for the Intermediate to Lower Shell Circumferential Weld 9-112 (Heat #27240) and the Lower Shell Plate D-3804-1. This EMA should also include the US Plate D-3802-3 material to provide a bounding case for all suspect reactor vessel materials.

Table 7-1 Palisades Predicted Positions 1.2 and 2.2 USE Values at 42.1 EFPY

Material Description		Wt. % Cu(a)	1/4T EOLE Fluence(b) (x10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE(a) (ft-lb)	Projected USE Decrease(c) (%)	Projected EOLE USE (ft-lb)
Extended Beltline Materials						
US Plate D-3802-1		0.21	0.0902	75	17	62.3
Using Surveillance Data(f)		0.21	0.0902	75	17(d)	62.3
US Plate D-3802-2		0.19	0.0902	73	16	61.3
US Plate D-3802-3	Using CVGraph Refitted Initial USE	0.25	0.0902	62.2	19.5	50.1(g)
	Using 95% Shear Initial USE	0.25	0.0902	59	19.5	47.5(g)
US Axial Welds 1-112 A/B/C (Heat # W5214)		0.213	0.0573	118	18.5	96.2
Using Surveillance Data		0.213	0.0573	118	30(d,e)	82.6
US to IS Circ. Weld 8-112 (Heat # 34B009)		0.192	0.0902	111	19	89.9
Using Surveillance Data		0.192	0.0902	111	35(d,e)	72.2

Notes for Table 7-1:

(a) From Table 4-1 of this report.
(b) From Table 3-1 of this report.
(c) Unless otherwise noted, percentage USE decrease values are based on Position 1.2 of Regulatory Guide 1.99, Revision 2 and calculated by plotting the 1/4T fluence values on Figure 2 of the Guide. The percent USE decrease values that corresponded to each material's specific Cu wt. % value were determined using interpolation between the existing Weld or Base Metal lines on Figure 2.
(d) Percentage USE decrease is based on Position 2.2 of Regulatory Guide 1.99, Revision 2, using data from Table 5-2 of this report. Credibility Criterion 3 in the Discussion section of Regulatory Guide 1.99, Revision 2, indicates that even if the surveillance data are not considered credible for determination of ΔRT_{NDT}, "they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E 185-82." Regulatory Guide 1.99, Revision 2, Position 2.2 indicates that an upper-bound line drawn parallel to the existing lines (in Figure 2 of the Guide) through the surveillance data points should be used in preference to the existing graph lines for determining the decrease in USE.
(e) Since the limiting surveillance data fell above the limiting line on Figure 2 of the Guide, the upper-bound line was drawn parallel to the "upper limit" line, and not the "% copper" lines. This was considered to be a conservative approach for the fluence levels being used in this evaluation.
(f) Plate surveillance data for these capsules used material from IS Plate D-3803-1; however, US Plate D-3802-1 is made from the same heat of material. Therefore, this USE data was used for US Plate D-3802-1 (Heat # C-1279) in this USE evaluation.
(g) The projected EOLE USE was determined for this material using an unconstrained TANH curve fit since there were no available data at 100% shear fracture appearance. Consistent with Credibility Criterion 2 of Regulatory Guide 1.99, Revision 2, if this unirradiated Charpy data was actually surveillance capsule Charpy data, it would be considered credible based on a very high correlation coefficient from CVGraph (0.987). However, a more conservative value could be calculated which would fall below 50 ft-lb. Therefore, an EMA is needed in the future for this material.

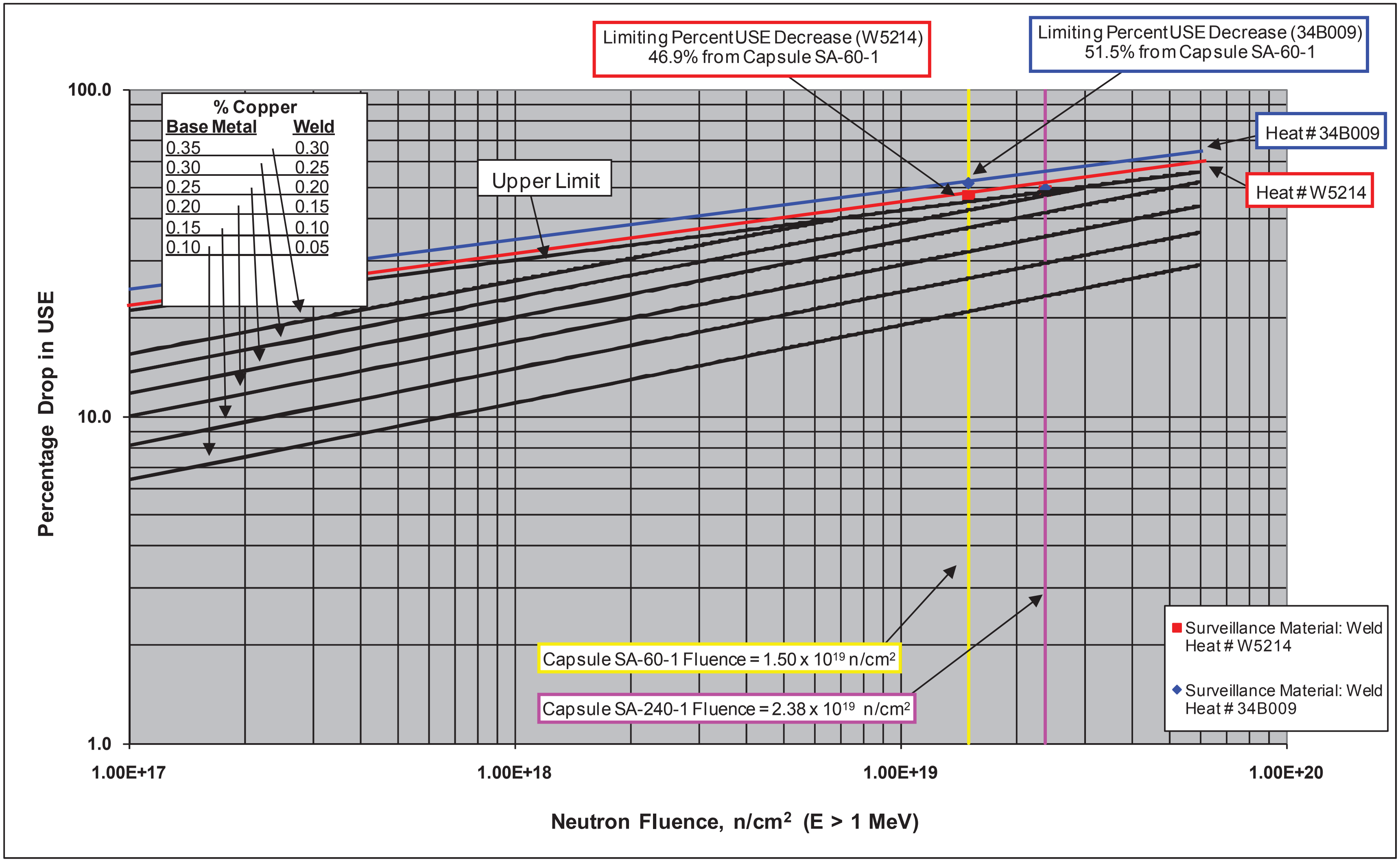


Figure 7-1 Regulatory Guide 1.99, Revision 2 Predicted Decrease in USE for Welds as a Function of Copper and Fluence for Palisades

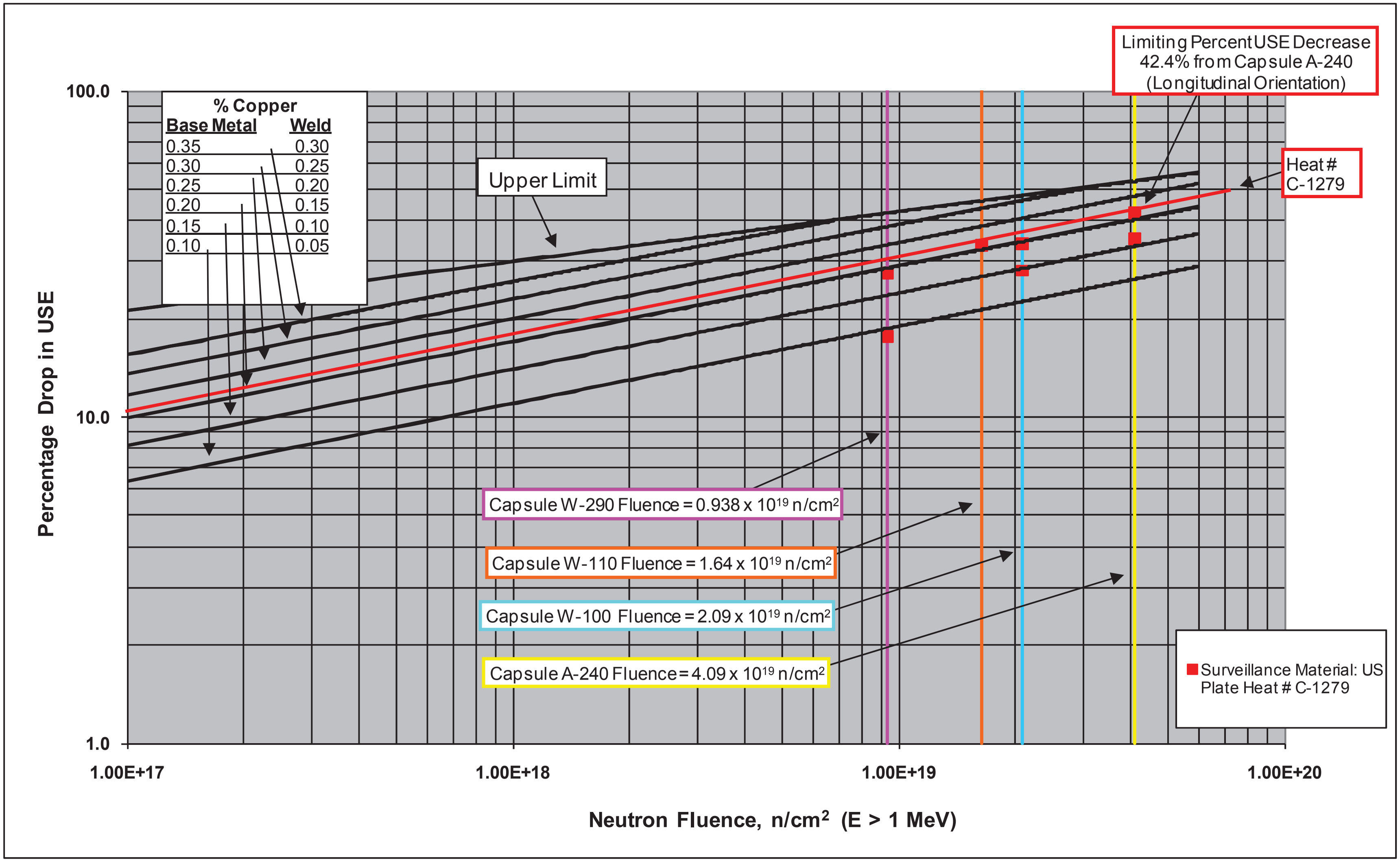


Figure 7-2 Regulatory Guide 1.99, Revision 2 Predicted Decrease in USE for Plates as a Function of Copper and Fluence for Palisades

8 PRESSURE-TEMPERATURE LIMIT CURVE APPLICABILITY

Heatup and cooldown limit curves are calculated using the most limiting values of RT_{NDT} (reference nil ductility transition temperature) corresponding to the limiting reactor vessel material. The most limiting reactor vessel material RT_{NDT} values are determined by using the unirradiated reactor vessel material fracture toughness properties and estimating the irradiation-induced shift (ΔRT_{NDT}).

RT_{NDT} increases as the material is exposed to fast-neutron irradiation; therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. Using the adjusted reference temperature (ART) values, pressure-temperature (P-T) limit curves are determined in accordance with the requirements of 10 CFR Part 50, Appendix G (Reference 15), as augmented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code (Reference 19).

The P-T limit curves for normal heatup and cooldown of the primary reactor coolant system for Palisades were previously developed in WCAP-17341-NP, Revision 0 (Reference 4) for 42.1 EFPY. The existing 42.1 EFPY P-T limit curves are based on the limiting beltline material ART values, which are influenced by both the fluence and the initial material properties of that material. The Palisades P-T limit curves were developed by calculating ART values utilizing the clad/base metal interface fluence that corresponded to each reactor vessel beltline material. The limiting ART values correspond to the Intermediate and Lower Shell Axial Welds 2-112 and 3-112 (Heat # W5214 using the Position 2.1 chemistry factor value based on not fully credible surveillance data with full margin term). Table 8-1 contains a summary of the limiting beltline material ART values at 42.1 EFPY.

To confirm the applicability of the P-T limit curves developed in WCAP-17341-NP, Revision 0 (Reference 4), the limiting extended beltline material ART values must be shown to be less than the limiting beltline material ART values summarized in Table 8-1. The accepted methods of Regulatory Guide 1.99, Revision 2 (Reference 10) were used along with the surface fluence of Section 3 to calculate the following ART values for the Palisades reactor vessel extended beltline materials at 42.1 EFPY (EOLE). The EOLE ART calculations are summarized below in Tables 8-2 and 8-3 at the 1/4T and 3/4T locations, respectively, for Palisades.

Per Regulatory Guide 1.99, Revision 2, σ_{Δ} need not exceed one half of ΔRT_{NDT} (See Section 2.2.3); however, for conservatism and consistency with References 2, 3 and 4, a full margin term will be used for all the Palisades extended beltline materials for the 3/4T ART calculations shown in Table 8-3. A full margin term was already necessary for the 1/4T ART calculations shown in Table 8-2. Note that the use of this conservative margin term, for the 3/4T location, caused the limiting material for the Palisades extended beltline to change from US Plate D-3802-3 for the 1/4T location to US Plate D-3802-2 for the 3/4T location.

Table 8-4 contains a summary of the limiting extended beltline materials ART values at 42.1 EFPY. Table 8-5 details the available margin between the limiting extended beltline and traditional beltline materials.

Existing P-T Limit Curve Applicability Conclusion

It is concluded that the Palisades reactor vessel extended beltline material ART values are bounded by the traditional beltline material ART values used in the development of the existing 42.1 EFPY P-T limit curves; therefore, the existing P-T limit curves remain valid as documented in WCAP-17341-NP (Reference 4).

Table 8-1 Summary of the Limiting Beltline ART Values Used in the Generation of the Palisades Heatup/Cooldown Curves at 42.1 EFPY

Limiting ART – Beltline Materials[(a)]	
Axial Welds 2-112 and 3-112 (Heat # W5214) Using the Position 2.1 Chemistry Factor Value Based on Not Fully Credible Surveillance Data with Full Margin Term (Limiting Axial Flaw Material)	
1/4T	**3/4T**
252.7°F	185.8°F

Note for Table 8-1:

(a) Values obtained from Table 4-4 of WCAP-17341-NP, Revision 0 (Reference 4).

Table 8-2 1/4T ART Calculations for the Palisades Reactor Vessel Extended Beltline Materials at 42.1 EFPY[(a)]

Material Description	CF[(b)] (°F)	1/4T Fluence[(c)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF[(c)]	$RT_{NDT(U)}$[(d)] (°F)	ΔRT_{NDT} (°F)	σ_I[(d)] (°F)	σ_Δ[(f)] (°F)	Margin (°F)	ART (°F)
Extended Beltline Materials									
US Plate D-3802-1	139.4	0.0902	0.3966	10	55.3	0	17	34.0	99.3
Using Non-Credible Surveillance Data	147.71	0.0902	0.3966	10	58.6	0	17	34.0	102.6
US Plate D-3802-2	133.2	0.0902	0.3966	19	52.8	0	17	34.0	105.8
US Plate D-3802-3	171.8	0.0902	0.3966	10	68.1	0	17	34.0	**112.1**
US Axial Welds 1-112 A/B/C (Heat # W5214)	230.73	0.0573	0.3148	-56[(e)]	72.6	17[(e)]	28	65.5	82.1
Using Not Fully Credible Surveillance Data	227.74	0.0573	0.3148	-56[(e)]	71.7	17[(e)]	28	65.5	81.2
US to IS Circ. Weld 8-112 (Heat # 34B009)	217.7	0.0902	0.3966	-56[(e)]	86.3	17[(e)]	28	65.5	95.9

Notes for Table 8-2:

(a) The Regulatory Guide 1.99, Revision 2 methodology was utilized in the calculation of the ART values. See Section 2.2.3 of this report for details.
(b) Taken from Table 5-1 of this report.
(c) Taken from Table 3-1 of this report.
(d) Initial RT_{NDT} values are taken from Table 4-1 of this report and are measured values, unless otherwise noted. For measured initial RT_{NDT} values, $\sigma_I = 0$°F.
(e) Initial RT_{NDT} values are generic; therefore, $\sigma_I = 17$°F.
(f) Per WCAP-17341-NP, Revision 0 (Reference 4), surveillance data of the plate material and weld Heat # W5214 were considered to be non-credible and not fully credible, respectively. Per the guidance of Reg. Guide 1.99, Revision 2, the base metal $\sigma_\Delta = 17$°F for both Positions 1.1 and 2.1 with non-credible surveillance data, and the weld metal $\sigma_\Delta = 28$°F for both Positions 1.1 and 2.1 with not fully credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

Table 8-3 3/4T ART Calculations for the Palisades Reactor Vessel Extended Beltline Materials at 42.1 EFPY[a]

Material Description	CF[b] (°F)	3/4T Fluence[c] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF[c]	$RT_{NDT(U)}$[d] (°F)	ΔRT_{NDT} (°F)	σ_I[d] (°F)	σ_Δ[f] (°F)	Margin (°F)	ART (°F)
Extended Beltline Materials									
US Plate D-3802-1	139.4	0.0314	0.2256	10	31.5	0	17	34.0	75.5
Using <u>Non-Credible</u> Surveillance Data	147.71	0.0314	0.2256	10	33.3	0	17	34.0	77.3
US Plate D-3802-2	133.2	0.0314	0.2256	19	30.1	0	17	34.0	**83.1**
US Plate D-3802-3	171.8	0.0314	0.2256	10	38.8	0	17	34.0	82.8
US Axial Welds 1-112 A/B/C (Heat # W5214)	230.73	0.0200	0.1718	-56[e]	39.6	17[e]	28	65.5	49.1
Using <u>Not Fully Credible</u> Surveillance Data	227.74	0.0200	0.1718	-56[e]	39.1	17[e]	28	65.5	48.6
US to IS Circ. Weld 8-112 (Heat # 34B009)	217.7	0.0314	0.2256	-56[e]	49.1	17[e]	28	65.5	58.6

Notes for Table 8-3:

(a) The Regulatory Guide 1.99, Revision 2 methodology was utilized in the calculation of the ART values. See Section 2.2.3 of this report for details.
(b) Taken from Table 5-1 of this report.
(c) Taken from Table 3-1 of this report.
(d) Initial RT_{NDT} values are taken from Table 4-1 of this report and are measured values, unless otherwise noted. For measured initial RT_{NDT} values, $\sigma_I = 0$°F.
(e) Initial RT_{NDT} values are generic; therefore, $\sigma_I = 17$°F.
(f) Per WCAP-17341-NP, Revision 0 (Reference 4), surveillance data of the plate material and weld Heat # W5214 were considered to be <u>non-credible</u> and <u>not fully credible</u>, respectively. Per the guidance of Reg. Guide 1.99, Revision 2, the base metal $\sigma_\Delta = 17$°F for both Positions 1.1 and 2.1 with <u>non-credible</u> surveillance data, and the weld metal $\sigma_\Delta = 28$°F for both Positions 1.1 and 2.1 with <u>not fully credible</u> surveillance data. Per Reg. Guide 1.99, Revision 2, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$; however, for conservatism and consistency with References 2, 3 and 4, a full margin term will be used for all Palisades extended beltline materials.

Table 8-4 Summary of the Limiting ART Values for the Palisades Reactor Vessel Extended Beltline at 42.1 EFPY

Limiting 1/4T ART	Limiting 3/4T ART
US Plate D-3802-3 using Position 1.1 Chemistry Factor Value with Full Margin Term	**US Plate D-3802-2 using Position 1.1 Chemistry Factor Value with Full Margin Term**
112.1°F	83.1°F

Table 8-5 Summary of the Available Margin Between the Palisades Reactor Vessel Beltline and Extended Beltline ART Values at 42.1 EFPY

Reactor Vessel Location	Limiting Material	Limiting ART Value (°F)	
		1/4T Location	3/4T Location
Traditional Beltline[a]	Axial Welds 2-112 and 3-112 (Heat # W5214) Using the Position 2.1 Chemistry Factor Value Based on Not Fully Credible Surveillance Data with Full Margin Term	252.7	185.8
Extended Beltline[b]	US Plate D-3802-3 using Position 1.1 Chemistry Factor Value with Full Margin Term	112.1	- - -
	US Plate D-3802-2 using Position 1.1 Chemistry Factor Value with Full Margin Term	- - -	83.1
Available Margin (°F)		140.6	102.7

Notes for Table 8-5:

(a) Traditional beltline limiting ART values used in development of the Palisades P-T limit curves, as documented in WCAP-17341-NP, Revision 0 (Reference 4), were taken from Table 8-1 of this report.

(b) Extended beltline limiting ART values were taken from Table 8-4 of this report.

9 REFERENCES

1. Code of Federal Regulations, 10 CFR Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

2. Structural Integrity Associates, Inc. Report No. 0901132.401, Revision 0, "Evaluation of Surveillance Data for Weld Heat No. W5214 for Application to Palisades PTS Analysis," Timothy J. Griesbach, April 2010.

3. Structural Integrity Associates, Inc. Report No. 1000915.401, Revision 1, "Revised Pressurized Thermal Shock Evaluation for the Palisades Reactor Pressure Vessel," Timothy J. Griesbach and Vikram Marthandam, November 12, 2010.

4. WCAP-17341-NP, Revision 0, "Palisades Nuclear Power Plant Heatup and Cooldown Limit Curves for Normal Operation and Upper-Shelf Energy Evaluation," E. J. Long and S. T. Byrne, February 2011.

5. NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," U.S. Nuclear Regulatory Commission, December 2010.

6. WCAP-15353 – Supplement 2-NP, Revision 0, "Palisades Reactor Pressure Vessel Fluence Evaluation," S. L. Anderson, July 2011.

7. "Fracture Toughness Requirements," Branch Technical Position 5-3, Revision 2, Contained in Chapter 5 of Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, NUREG-0800, March 2007.

8. WCAP-14370, Revision 0, "Use of the Hyperbolic Tangent Function for Fitting Transition Temperature Toughness Data," Thomas R. Mager et al., May 1995.

9. ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," American Society for Testing and Materials, 1982.

10. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, May 1988.

11. K. Wichman, M. Mitchel and A. Hiser, USNRC, Generic Letter 92-01 and RPV Integrity Workshop Handouts, *NRC/Industry Workshop on RPV Integrity Issues,* February 12, 1998.

12. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

13. NRC SER, Carl S. Hood (NRC) to Nathan L. Haskell (Palisades), "Palisades Plant – Reactor Vessel Neutron Fluence Evaluation and revised Schedule for Reaching Pressurized Thermal Shock Screening Criteria," November 14, 2000.

14. ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB, Section NB-2300, "Fracture Toughness Requirements for Material."

15. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

16. Combustion Engineering Report P-PENG-ER-006, Revision 0, "The Reactor Vessel Group Records Evaluation Program Phase II Final Report for the Palisades Reactor Pressure Vessel Plates, Forgings, Welds and Cladding," Combustion Engineering, Inc., October 1995.

17. WCAP-13587, Revision 1, "Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors," S. Tandon et al., September 1993.

18. Combustion Engineering Report CE NPSD-993, Revision 0, "Evaluation of Low Upper Shelf Energy for Reactor Vessel Beltline Weld and Base Metal Materials for Combustion Engineering Nuclear Steam Supply Systems Reactor Pressure Vessels," CEOG Task 821, C-E Owners Group, May 1995.

19. Appendix G to the 1998 through the 2000 Addenda Edition of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, "Fracture Toughness Criteria for Protection Against Failure."

APPENDIX A CVGRAPH PLOTS USED IN THE DETERMINATION OF THE INITIAL RT_{NDT} FOR THE PALISADES UPPER SHELL PLATES

The temperature representing a minimum of 50 ft-lbs and 35 mils lateral expansion (L.E.) is needed in order to determine initial RT_{NDT} values for the Palisades extended beltline plate material. The following CVGraph plots for Charpy V-Notch Impact Energy and L.E. were used, along with the methodology described in Table 4-1 footnote (e), to determine the initial RT_{NDT} values for the three plates.

Table A-1 contains the initial upper-shelf energy values used as input for the generation of the Charpy V-Notch plots in CVGraph, Version 5.3. The values shown in Table A-1 were fixed in the program. The lower-shelf energy values were fixed at 2.2 ft-lb for all cases. The lower-shelf Lateral Expansion values were fixed at 1.0 mils in all cases. Finally, note that all Charpy V-Notch data, obtained from P-PENG-ER-006, Revision 0 (Reference A-1), was tested in the strong-direction (longitudinal-orientation (LT)).

Table A-1 Summary of the USE Values Fixed in CVGraph for the Palisades Upper Shell Plates

Reactor Vessel Material	Unirradiated Initial USE (ft-lb)
US Plate D-3802-1	115[a]
US Plate D-3802-2	113[b]
US Plate D-3802-3	91[c]

Notes for Table A-1:

(a) Unirradiated initial USE value was determined based on one data point at 100% shear per Reference A-1.
(b) Unirradiated initial USE value was determined based on three data points at 100% shear per Reference A-1.
(c) Unirradiated initial USE value was determined based on three data points at 95% shear per Reference A-1.

Appendix A Reference

A-1 Combustion Engineering Report P-PENG-ER-006, Revision 0, "The Reactor Vessel Group Records Evaluation Program Phase II Final Report for the Palisades Reactor Pressure Vessel Plates, Forgings, Welds and Cladding," Combustion Engineering, Inc., October 1995.

A.1 US PLATE D-3802-1 (HEAT # C-1279)

US PLATE D-3802-1 - IMPACT ENERGY - LT - UNIRRADIATED

CVGRAPH 5.3 Hyperbolic Tangent Curve Printed on 05/11/2011 09:42 AM

Page 1

Coefficients of Curve 1

A = 58.6 B = 56.4 C = 57.79 T0 = 53.89 D = 0.00E+00

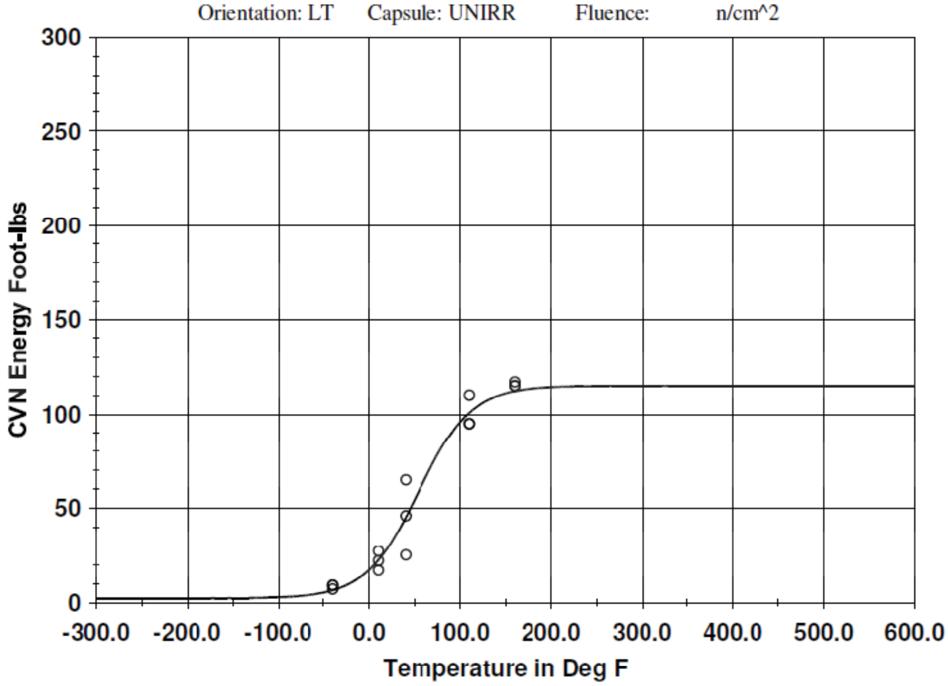
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=115.0(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=21.6 Deg F Temp@50 ft-lbs=45.1 Deg F

Plant: Palisades Material: SA302BM Heat: C-1279-2

Orientation: LT Capsule: UNIRR Fluence: n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-40.00	7.00	6.41	.59
-40.00	9.00	6.41	2.59
-40.00	9.00	6.41	2.59
10.00	22.00	22.46	-.46
10.00	27.00	22.46	4.54
10.00	17.00	22.46	-5.46
40.00	25.00	45.30	-20.30
40.00	65.00	45.30	19.70
40.00	46.00	45.30	.70

US PLATE D-3802-1 - IMPACT ENERGY - LT - UNIRRADIATED

Page 2
Plant: Palisades Material: SA302BM Heat: C-1279-2
Orientation: LT Capsule: UNIRR Fluence: n/cm^2

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
110.00	95.00	100.85	-5.85
110.00	95.00	100.85	-5.85
110.00	110.00	100.85	9.15
160.00	115.00	112.20	2.80
160.00	117.00	112.20	4.80
160.00	115.00	112.20	2.80

Correlation Coefficient = .981

US PLATE D-3802-1 - L. E. - LT - UNIRRADIATED

CVGRAPH 5.3 Hyperbolic Tangent Curve Printed on 05/11/2011 09:43 AM

Page 1

Coefficients of Curve 1

A = 44.62 B = 43.62 C = 66.71 T0 = 48.05 D = 0.00E+00

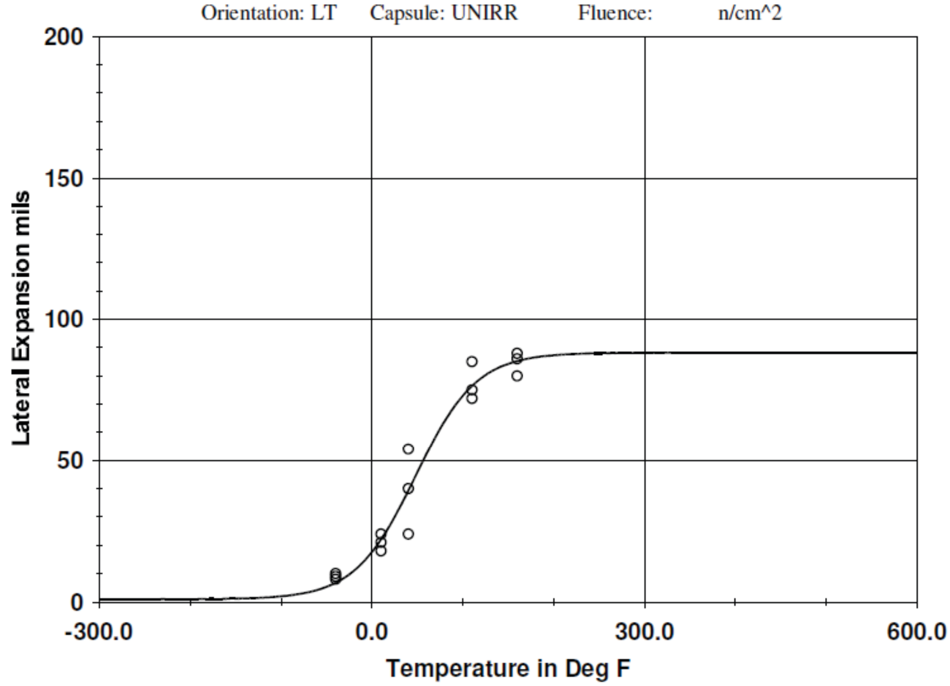
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf L.E.=88.2 Lower Shelf L.E.=1.0(Fixed)

Temp.@L.E. 35 mils=33.1 Deg F

Plant: Palisades Material: SA302BM Heat: C-1279-2

Orientation: LT Capsule: UNIRR Fluence: n/cm^2



Charpy V-Notch Data

Temperature	Input L.E.	Computed L.E.	Differential
-40.00	8.00	6.81	1.19
-40.00	9.00	6.81	2.19
-40.00	10.00	6.81	3.19
10.00	21.00	22.13	-1.13
10.00	24.00	22.13	1.87
10.00	18.00	22.13	-4.13
40.00	24.00	39.38	-15.38
40.00	54.00	39.38	14.62
40.00	40.00	39.38	.62

US PLATE D-3802-1 - L. E. - LT - UNIRRADIATED

Page 2
Plant: Palisades Material: SA302BM Heat: C-1279-2
Orientation: LT Capsule: UNIRR Fluence: n/cm^2

Charpy V-Notch Data

Temperature	Input L.E.	Computed L.E.	Differential
110.00	72.00	76.46	-4.46
110.00	75.00	76.46	-1.46
110.00	85.00	76.46	8.54
160.00	88.00	85.30	2.70
160.00	86.00	85.30	.70
160.00	80.00	85.30	-5.30

Correlation Coefficient = .978

A.2 US PLATE D-3802-2 (HEAT # C-1308)

US PLATE D-3802-2 - IMPACT ENERGY - LT - UNIRRADIATED

CVGRAPH 5.3 Hyperbolic Tangent Curve Printed on 05/11/2011 08:03 AM

Page 1

Coefficients of Curve 1

A = 57.6 B = 55.4 C = 79.47 T0 = 69.48 D = 0.00E+00

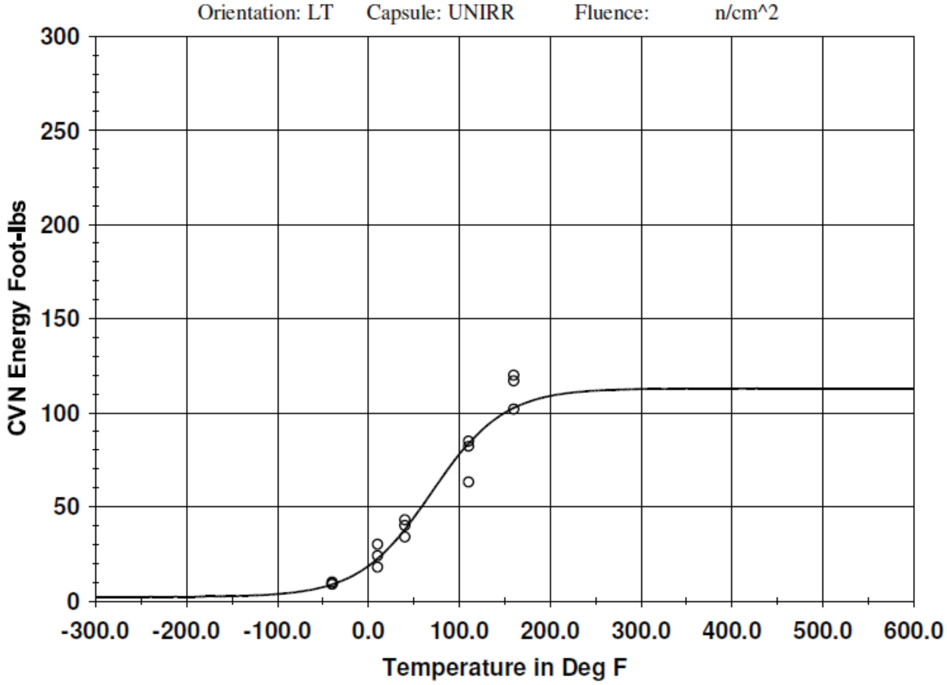
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=113.0(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=26.1 Deg F Temp@50 ft-lbs=58.6 Deg F

Plant: Palisades Material: SA302BM Heat: C-1308-2

Orientation: LT Capsule: UNIRR Fluence: n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-40.00	10.00	8.82	1.18
-40.00	9.00	8.82	.18
-40.00	9.00	8.82	.18
10.00	18.00	22.46	-4.46
10.00	24.00	22.46	1.54
10.00	30.00	22.46	7.54
40.00	34.00	37.94	-3.94
40.00	43.00	37.94	5.06
40.00	40.00	37.94	2.06

US PLATE D-3802-2 - IMPACT ENERGY - LT - UNIRRADIATED

Page 2
Plant: Palisades Material: SA302BM Heat: C-1308-2
Orientation: LT Capsule: UNIRR Fluence: n/cm^2

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
110.00	85.00	83.63	1.37
110.00	63.00	83.63	-20.63
110.00	82.00	83.63	-1.63
160.00	120.00	102.70	17.30
160.00	117.00	102.70	14.30
160.00	102.00	102.70	-.70

Correlation Coefficient = .977

US PLATE D-3802-2 - L. E. - LT - UNIRRADIATED

CVGRAPH 5.3 Hyperbolic Tangent Curve Printed on 05/11/2011 08:03 AM

Page 1

Coefficients of Curve 1

A = 52.69 B = 51.69 C = 113. T0 = 81.31 D = 0.00E+00

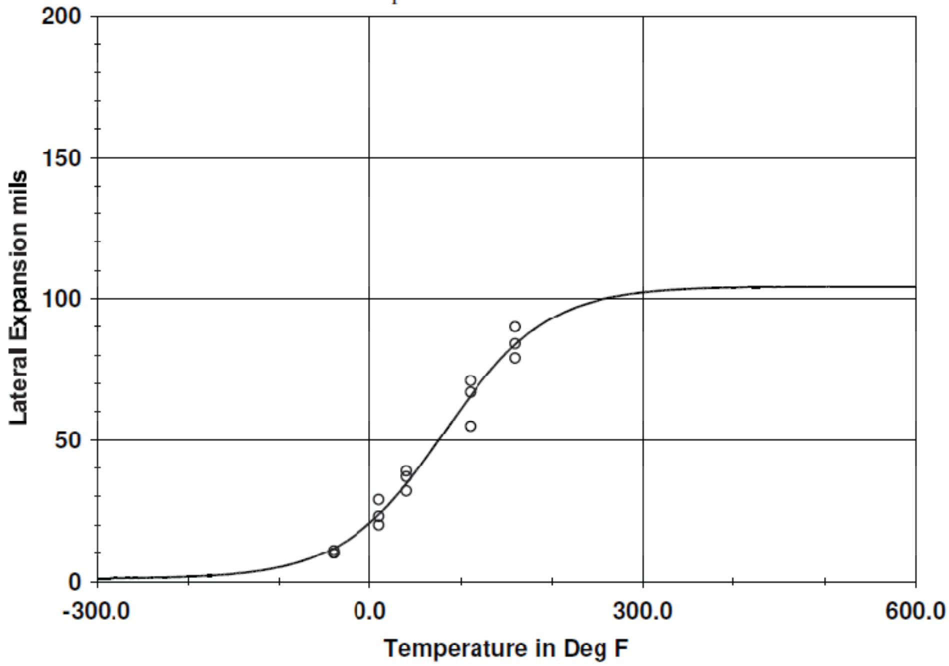
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf L.E.=104.4 Lower Shelf L.E.=1.0(Fixed)

Temp.@L.E. 35 mils=41.1 Deg F

Plant: Palisades Material: SA302BM Heat: C 1308 2

Orientation: LT Capsule: UNIRR Fluence: n/cm^2



Charpy V-Notch Data

Temperature	Input L.E.	Computed L.E.	Differential
-40.00	10.00	11.82	-1.82
-40.00	11.00	11.82	-.82
-40.00	10.00	11.82	-1.82
10.00	20.00	23.81	-3.81
10.00	23.00	23.81	-.81
10.00	29.00	23.81	5.19
40.00	32.00	34.60	-2.60
40.00	39.00	34.60	4.40
40.00	37.00	34.60	2.40

US PLATE D-3802-2 - L. E. - LT - UNIRRADIATED

Page 2
Plant: Palisades Material: SA302BM Heat: C-1308-2
Orientation: LT Capsule: UNIRR Fluence: n/cm^2

Charpy V-Notch Data

Temperature	Input L.E.	Computed L.E.	Differential
110.00	71.00	65.55	5.45
110.00	55.00	65.55	-10.55
110.00	67.00	65.55	1.45
160.00	90.00	83.82	6.18
160.00	79.00	83.82	-4.82
160.00	84.00	83.82	.18

Correlation Coefficient = .987

A.3 US PLATE D-3802-3 (HEAT # C-1281)

US PLATE D-3802-3 - IMPACT ENERGY - LT - UNIRRADIATED

CVGRAPH 5.3 Hyperbolic Tangent Curve Printed on 05/11/2011 09:44 AM

Page 1

Coefficients of Curve 1

A = 46.6 B = 44.4 C = 73.63 T0 = 40.51 D = 0.00E+00

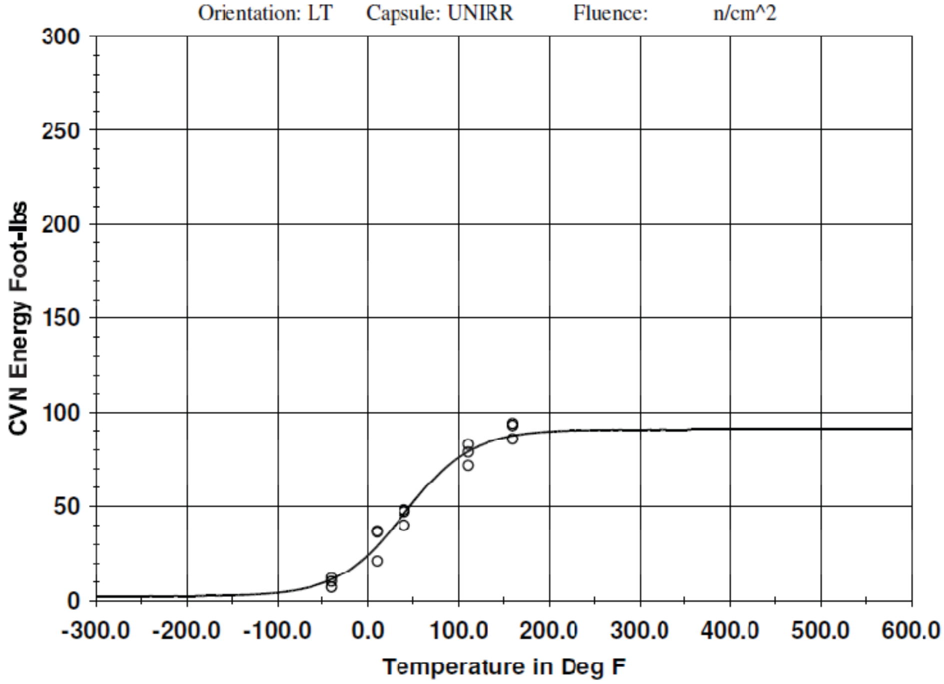
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=91.0(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=11.6 Deg F Temp@50 ft-lbs=46.2 Deg F

Plant: Palisades Material: SA302BM Heat: C-1281-1

Orientation: LT Capsule: UNIRR Fluence: n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-40.00	7.00	11.16	-4.16
-40.00	10.00	11.16	-1.16
-40.00	12.00	11.16	.84
10.00	21.00	29.19	-8.19
10.00	36.50	29.19	7.31
10.00	37.00	29.19	7.81
40.00	48.00	46.29	1.71
40.00	40.00	46.29	-6.29
40.00	47.00	46.29	.71

US PLATE D-3802-3 - IMPACT ENERGY - LT - UNIRRADIATED

Page 2
Plant: Palisades Material: SA302BM Heat: C-1281-1
Orientation: LT Capsule: UNIRR Fluence: n/cm^2

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
110.00	79.00	79.32	-.32
110.00	83.00	79.32	3.68
110.00	72.00	79.32	-7.32
160.00	86.00	87.67	-1.67
160.00	94.00	87.67	6.33
160.00	93.00	87.67	5.33

Correlation Coefficient = .986

US PLATE D-3802-3 - L. E. - LT - UNIRRADIATED

CVGRAPH 5.3 Hyperbolic Tangent Curve Printed on 05/11/2011 09:44 AM
Page 1
Coefficients of Curve 1

A = 41.99 B = 40.99 C = 96.52 T0 = 49.4 D = 0.00E+00

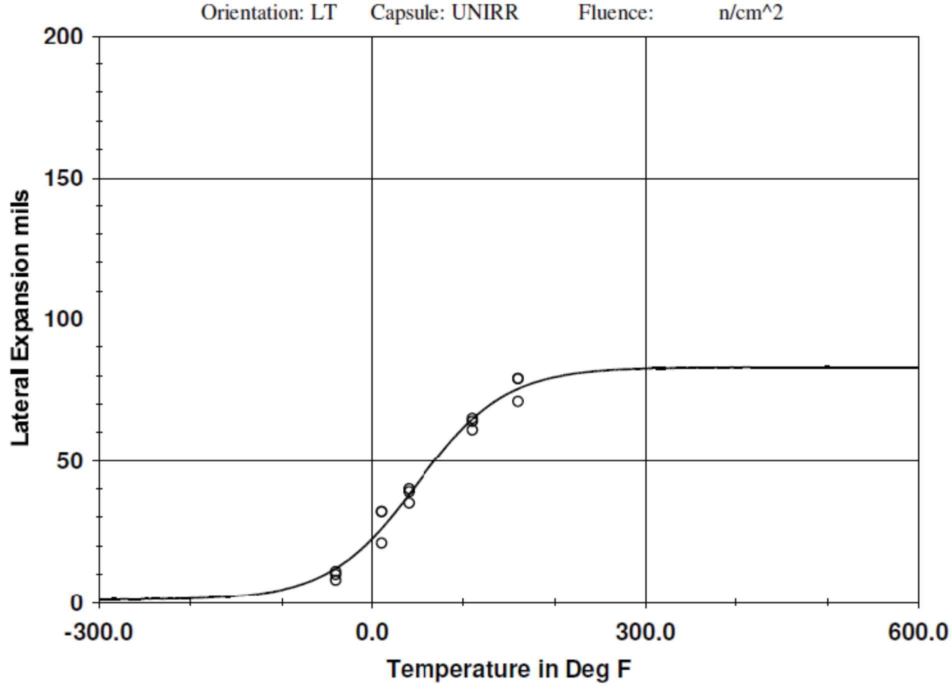
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf L.E.=83.0 Lower Shelf L.E.=1.0(Fixed)

Temp.@L.E. 35 mils=32.8 Deg F

Plant: Palisades Material: SA302BM Heat: C-1281-1

Orientation: LT Capsule: UNIRR Fluence: n/cm^2



Charpy V-Notch Data

Temperature	Input L.E.	Computed L.E.	Differential
-40.00	8.00	12.11	-4.11
-40.00	10.00	12.11	-2.11
-40.00	11.00	12.11	-1.11
10.00	21.00	26.13	-5.13
10.00	32.00	26.13	5.87
10.00	32.00	26.13	5.87
40.00	40.00	38.01	1.99
40.00	35.00	38.01	-3.01
40.00	39.00	38.01	.99

US PLATE D-3802-3 - L. E. - LT - UNIRRADIATED

Page 2
Plant: Palisades Material: SA302BM Heat: C-1281-1
Orientation: LT Capsule: UNIRR Fluence: n/cm^2

Charpy V-Notch Data

Temperature	Input L.E.	Computed L.E.	Differential
110.00	65.00	64.80	.20
110.00	64.00	64.80	-.80
110.00	61.00	64.80	-3.80
160.00	71.00	75.45	-4.45
160.00	79.00	75.45	3.55
160.00	79.00	75.45	3.55

Correlation Coefficient = .989

Attachment 4

Westinghouse, WCAP-15353 – Supplement 2 – NP
Revision 0

Palisades Reactor Pressure Vessel Fluence Evaluation

37 pages follow

WCAP-15353 – Supplement 2 – NP July 2011
Revision 0

Palisades Reactor Pressure Vessel Fluence Evaluation

Westinghouse

WCAP-15353 – Supplement 2 - NP, Revision 0

Palisades Reactor Pressure Vessel Fluence Evaluation

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Radiation Engineering & Analysis

July 2011

Reviewed:
J. Sejvar*, Consultant
Radiation Engineering & Analysis

Approved:
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WESTINGHOUSE ELECTRIC COMPANY LLC
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Cranberry Township, PA 16066

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EXECUTIVE SUMMARY

In early 2000, WCAP-15353, Revision 0[1] describing the methodology used in the fluence evaluations for the Palisades plant was submitted to the NRC staff for review. Subsequent to that review and a further exchange of information documented in Reference 2, the methodology described in WCAP-15353, Revision 0 was approved for application to the Palisades reactor pressure vessel[3]. Subsequent to that approval, additional submittals[4,5] in support of the benchmarking of this fluence methodology were reviewed and approved by the NRC Staff as being in compliance with the requirements of Regulatory Guide 1.190[6].

The fluence analysis described in WCAP-15353, Revision 0[1] included cycle specific evaluations through fuel Cycle 14 (the then current operating cycle). In mid 2010, Supplement 1 to WCAP-15353, Revision 0 was issued to provide an updated neutron fluence assessment for the Palisades pressure vessel that included cycle specific analysis for additional operating cycles for which the design had been finalized (Cycles 15 through 21). This prior supplement included projections for future operation through approximately 44 effective full power years (EFPY). The results of the evaluation documented in Supplement 1 were used as input to vessel materials studies that included updates to surveillance capsule credibility analysis, material chemistry factor determination, Pressurized Thermal Shock (PTS) evaluation, and generation of Pressure-Temperature (PT) limit curves.

The fluence analysis documented in Supplement 1 of WCAP-15353, Revision 0 was limited to an axial range that extended approximately one foot above and below the active fuel stack. This model did not include all of the pressure vessel materials that could potentially exceed the 1.0E+17 n/cm^2 (E > 1.0 MeV) fluence threshold defined in 10 CFR 50 Appendix H[7]. The purpose of this second supplement is to define which materials in the Palisades pressure vessel are projected to exceed the 1.0E+17 n/cm^2 threshold neutron fluence before the end of the license renewal period (EOLE); and, to project the neutron fluence for each of these specific materials. This additional fluence information will be used to perform necessary material evaluations for those materials in the extended beltline region that are projected to exceed the 1.0E+17 n/cm^2 threshold.

The results of the neutron exposure calculations for the extended beltline region of the Palisades pressure vessel are provided in Tables E-1 and E-2. In Table E-1, the axial locations of the maximum neutron exposure location are listed for each of the traditional and extended beltline materials. The axial span of each material is indexed to Z = 0.0 cm at the midplane of the active fuel stack modeled in the neutronic calculations. In Table E-2, the maximum projected neutron fluence for each of the beltline materials is provided. Projected fluence values are listed for EOC21 (23.4 EFPY), EOLE (42.1 EFPY), and EOC36 (44.0 EFPY).

It should be noted that Supplement 2 of this report was generated simply to address the neutron fluence experienced by materials located in the extended beltline regions above and below the reactor core that were not included in either Revision 0 of WCAP-15353 or in Supplement 1 of that report. None of the fluence information that was included in Supplement 1 has been changed in Supplement 2.

Table E-1
Palisades Pressure Vessel Material Locations in the Traditional and Extended Beltline Regions

Material	Axial Location[a,b] [cm]	Notes
Outlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	260.02	Extended Beltline
Inlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	277.72	Extended Beltline
Upper Shell Plates D-3802	204.93	Extended Beltline
Upper Shell Longitudinal Welds 1-112	204.93	Extended Beltline
Upper Shell to Intermediate Shell Circumferential Weld 8-112	204.93	Extended Beltline
Intermediate Shell Plates D-3803	-42.24	Traditional Beltline
Intermediate Shell Longitudinal Welds 2-112	-42.24	Traditional Beltline
Intermediate Shell to Lower Shell Circumferential Weld 9-112	-42.24	Traditional Beltline
Lower Shell Plates D-3804	-42.24	Traditional Beltline
Lower Shell Longitudinal Welds 3-112	-42.24	Traditional Beltline
Lower Shell to Lower Vessel Head Circumferential Weld 10-112	-279.43	Extended Beltline

[a] Axial elevations are indexed to Z = 0.0 at the midplane of the active fuel stack.

[b] Elevations listed represent the location of maximum neutron exposure for the material.

Table E-2
Palisades Maximum Fast Neutron (E > 1.0 MeV) Fluence Experienced by Materials in the Traditional and Extended Beltline

Material	Neutron Fluence [n/cm^2]		
	23.4 EFPY	42.1 EFPY	44 EFPY
Outlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	<1.0E+17	<1.0E+17	<1.0E+17
Inlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	<1.0E+17	<1.0E+17	<1.0E+17
Upper Shell Plates D-3802	9.895E+17	1.529E+18	1.584E+18
Upper Shell Longitudinal Welds 1-112	6.782E+17	9.707E+17	1.001E+18
Upper Shell to Intermediate Shell Circumferential Weld 8-112	9.895E+17	1.529E+18	1.584E+18
Intermediate Shell Plates D-3803	2.157E+19	3.428E+19	3.558E+19
Intermediate Shell Longitudinal Welds 2-112	1.472E+19	2.161E+19	2.232E+19
Intermediate Shell to Lower Shell Circumferential Weld 9-112	2.157E+19	3.428E+19	3.558E+19
Lower Shell Plates D-3804	2.157E+19	3.428E+19	3.558E+19
Lower Shell Longitudinal Welds 3-112	1.472E+19	2.161E+19	2.232E+19
Lower Shell to Lower Vessel Head Circumferential Weld 10-112	<1.0E+17	<1.0E+17	<1.0E+17

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SECTION 1.0

INTRODUCTION

In the assessment of the state of embrittlement of light water reactor (LWR) pressure vessels, an accurate evaluation of the neutron exposure of each of the materials comprising the beltline region of the vessel is required. In Section II F of 10 CFR 50[7] Appendix G, the beltline region is defined as:

> *"the region of the reactor vessel shell material (including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the reactor core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage"*.

In Section II A of 10 CFR 50 Appendix H, the lower limit of neutron exposure for consideration of radiation induced material damage is specified by a neutron fluence (E > 1.0 MeV) threshold of 1.0E+17 n/cm^2. Each of the materials that is anticipated to experience a neutron exposure that exceeds this fluence threshold must be considered in the overall embrittlement assessments for the pressure vessel.

Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [6], describes state-of-the-art calculation and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence. Also included in Regulatory Guide 1.190 is a discussion of the steps required to qualify and validate the methodology used to determine the neutron exposure of the pressure vessel wall. One important step in the validation process is the comparison of plant-specific neutron calculations with available measurements.

In early 2000, WCAP-15353, Revision 0[1] describing the methodology used in the fluence evaluations for the Palisades plant was submitted to the NRC staff for review. Subsequent to that review and a further exchange of information documented in Reference 2, the methodology described in WCAP-15353, Revision 0 was approved for application to the Palisades reactor pressure vessel[3]. Subsequent to that approval additional submittals[4,5] in support of the benchmarking of this fluence methodology were reviewed and approved by the NRC Staff as being in compliance with the requirements of Regulatory Guide 1.190[6].

The fluence analysis described in WCAP-15353, Revision 0[1] included cycle specific evaluations through fuel Cycle 14 (the then current operating cycle). In mid 2010, Supplement 1 to WCAP-15353, Revision 0 was issued to provide an updated neutron fluence assessment for the Palisades pressure vessel that included cycle specific analysis for additional operating

cycles for which the design had been finalized (Cycles 15 through 21). This supplement included projections for future operation through approximately 44 effective full power years (EFPY). The results of the evaluation documented in Supplement 1 were used as input to vessel materials studies that included updates to surveillance capsule credibility analysis, material chemistry factor determination, Pressurized Thermal Shock (PTS) evaluation, and generation of Pressure-Temperature (PT) limit curves.

Since the PTS screening criterion determination for the Palisades pressure vessel requires the evaluation of all weld heat W5214 surveillance capsule data from Palisades and other PWR's, Supplement 1 also included a compilation of the latest fluence evaluation from withdrawn surveillance capsules containing the W5214 material from sister plants. That compilation of capsule fluence values provided a data set based on the same Regulatory Guide 1.190 compliant methodology described in this report.

The fluence analysis documented in Supplement 1 of WCAP-15353, Revision 0 was limited to an axial range that extended approximately one foot above and below the active fuel stack. This model did not include all the pressure vessel materials that could potentially exceed the 1.0E+17 n/cm^2 (E > 1.0 MeV) fluence threshold defined in 10 CFR 50 Appendix H[7]. The purpose of this second supplement is to define which materials in the Palisades pressure vessel are projected to exceed the 1.0E+17 n/cm^2 threshold neutron fluence before the end of the license renewal period (EOLE); and, to project the neutron fluence for each of these specific materials. The additional fluence information will be used to perform necessary material evaluations for those materials in the extended beltline region that are projected to exceed the 1.0E+17 n/cm^2 threshold.

In subsequent sections of this supplement, the methodologies used to perform neutron transport calculations and dosimetry evaluations are described in some detail and the results of the plant specific transport calculations are given for each of the materials located in the traditional and extended beltline regions of the Palisades pressure vessel. For completeness, comparisons of calculations and measurements demonstrating that the transport calculations meet the requirements of Regulatory Guide 1.190 that were previously discribed in Reference 1 are also included in this supplement. These comparisons demonstrate the adequacy of the methodology for use in the fluence determinations provided in this report. Finally, a listing of updated neutron fluence values, based on the use of an approved Regulatory Guide 1.190 compliant fluence methodology, for several previously withdrawn surveillance capsules that contain some of the Palisades vessel materials is provided for use in data correlation studies.

SECTION 2.0

NEUTRON TRANSPORT CALCULATIONS

As noted in Section 1.0 of this report, the exposure of the Palisades pressure vessel was developed based on a series of fuel cycle-specific neutron transport calculations validated by comparison with plant-specific measurements. Measurement data used in the validation process were obtained from both in-vessel and ex-vessel capsule irradiations. In this section, the neutron transport methodology is discussed in some detail, and the calculated results applicable to the in-vessel surveillance capsules and the pressure vessel beltline materials are presented. A discussion of the Palisades dosimetry evaluations and measurement to calculation comparisons is included in Section 3.0 of this supplement. The data comparisons included in Section 3.0 cover a wide range of both in-vessel and ex-vessel locations. These comparisons along with the benchmarking information described in References 4 and 5 demonstrate that the transport methodology provides results that meet the requirements of Regulatory Guide 1.190 for the pressure vessel and surveillance capsule locations considered in this report.

2.1 – Method of Analysis

In performing the fast neutron exposure evaluations for the Palisades reactor, plant-specific forward transport calculations were carried out using the three-dimensional flux synthesis technique described in Section 1.3.4 of Regulatory Guide 1.190. In particular, the following single channel synthesis approach was employed for all fuel cycles:

$$\Phi(r,\theta,z) = \Phi(r,\theta) * \frac{\Phi(r,z)}{\Phi(r)}$$

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution, $\phi(r,\theta)$ is the transport solution in r,θ geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source-per-unit height as that used in the r,θ two-dimensional calculation.

For the Palisades analysis, all of the transport calculations were carried out using the DORT two-dimensional discrete ordinates code Version 3.2[8] and the BUGLE-96 cross section library[9]. The BUGLE-96 library provides a 67-group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering was treated with a P_5 legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature. Energy and space dependent core power distributions as well as system operating conditions were treated on a fuel cycle specific basis.

The geometry used for the Palisades transport analysis is discussed in some detail in Reference 3 and the geometric model established for Cycle 15 and beyond was also used for the evaluations documented in Supplement 1 to Reference 1. A plan view of the r,θ model of the reactor geometry at the core midplane is shown in Figure 2.1-1. This model depicts a single quadrant of the reactor. A section view of the r,z model of the Palisades reactor is shown in Figure 2.1-2. The r,z model extended radially from the centerline of the reactor core out to a location interior to the primary biological shield and over an axial span from an elevation approximately four feet below the active fuel to an axial elevation approximately four feet above the active fuel. The one-dimensional radial model used in the synthesis procedure consisted of the same radial mesh intervals included in the r,z model. Thus, radial synthesis factors could be determined on a meshwise basis throughout the entire geometry.

These geometric models formed the basis for the extended beltline evaluation described in this supplement. In completing the current analysis, the two-dimensional r,θ and the one-dimensional r transport calculations from the prior analyses were retained as is and the two-dimensional r,z transport calculations were re-run with expanded models designed to encompass all axial elevations that were anticipated to experience a neutron fluence (E > 1.0 MeV) greater than 1.0E+17 n/cm^2. The expanded r,z calculations were then coupled with the existing r,θ and r calculations to provide the final synthesized three-dimensional neutron fluence solution, $\Phi(r,\theta,z)$, from which the maximum neutron exposure of each of the traditional and extended beltline materials was extracted.

The geometric models used in the fluence analysis were developed from nominal design dimensions for the reactor core and the reactor internals components. However, for the pressure vessel inner radius and thickness, as-built dimensions were available from fabrication records and additional data were subsequently obtained during plant operation. These as-built dimensions were used in the analytical models.

The as-built value of the pressure vessel base metal inner radius was derived from available diameter measurements of the shell of the vessel prior to the addition of cladding. Based on an evaluation of the maximum and minimum vessel measurements, the nominal radius of the pressure vessel base metal was determined to be:

Base Metal IR = 86.35 inches

Based on a nominal cladding thickness of 0.25 inches, the cladding inner radius was determined to be:

Cladding IR = 86.10 inches

The as-built thickness of the pressure vessel base metal was based on a statistical evaluation of 151 thickness measurements taken during the 1995 in-service inspection. The measurement points were taken around the circumference and over the height of the beltline region. Based on the evaluation of these measurement points the total thickness of clad plus base metal was determined to be:

Total Thickness = 9.042 ± 0.088 inches

Based on a nominal cladding thickness of 0.25 inches, the thickness of the pressure vessel base metal shell was determined to be:

Base Metal Thickness = 8.792 inches

These as-built pressure vessel dimensions were submitted to the NRC staff in the form of a reply to a Request for Additional Information (RAI)[2] and were subsequently approved for use in the Palisades fluence analyses.

The core power distributions used in the plant-specific transport analysis for the reactor were provided by Entergy[16]. The data used in the source generation included fuel assembly-specific initial enrichments, beginning-of-cycle burnups and end-of-cycle burnups. Appropriate axial burrnup distributions were also used.

For each fuel cycle of operation, the fuel assembly-specific enrichment and burnup data were used to generate the spatially-dependent neutron source throughout the reactor core. This source description included the spatial variation of isotope dependent (U-235, U-238, Pu-239, Pu-240, Pu-241, and Pu-242) fission spectra, neutron emission rate per fission, and energy release per fission based on the burnup history of individual fuel assemblies. These fuel assembly-specific neutron source strengths derived from the detailed isotopics were then converted from fuel pin cartesian coordinates to the [r,θ], [r,z], and [r] spatial mesh arrays used in the DORT discrete ordinates calculations.

This same qualified methodology was used along with reactor specific input in the determination of the surveillance capsule fluence values discussed in Section 4.0 of this report. It should be noted that these additional surveillance capsule fluence evaluations are not directly related to the extended beltline materials in the Palisades pressure vessel.

Figure 2.1-1

Palisades r,θ Reactor Geometry

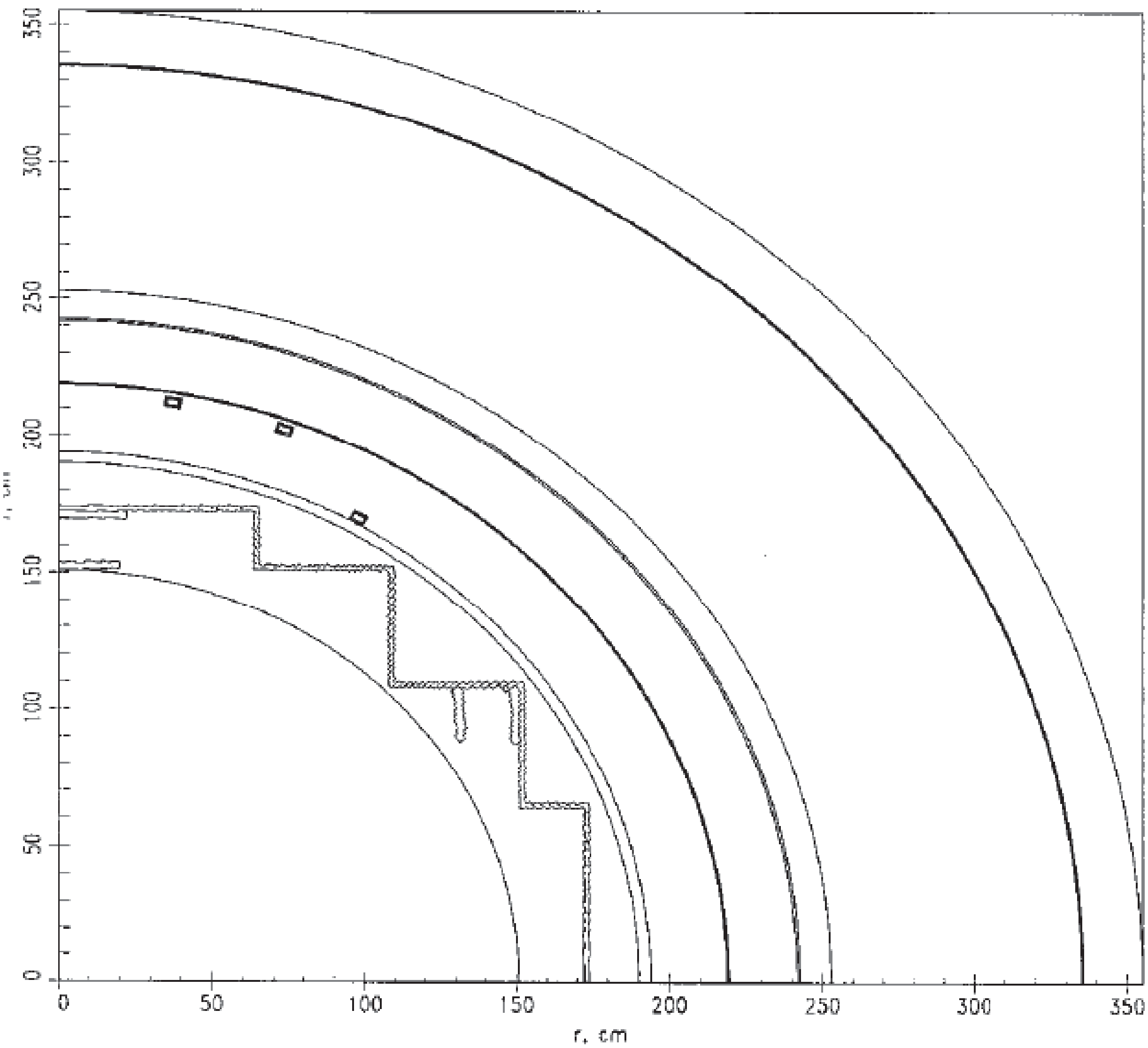
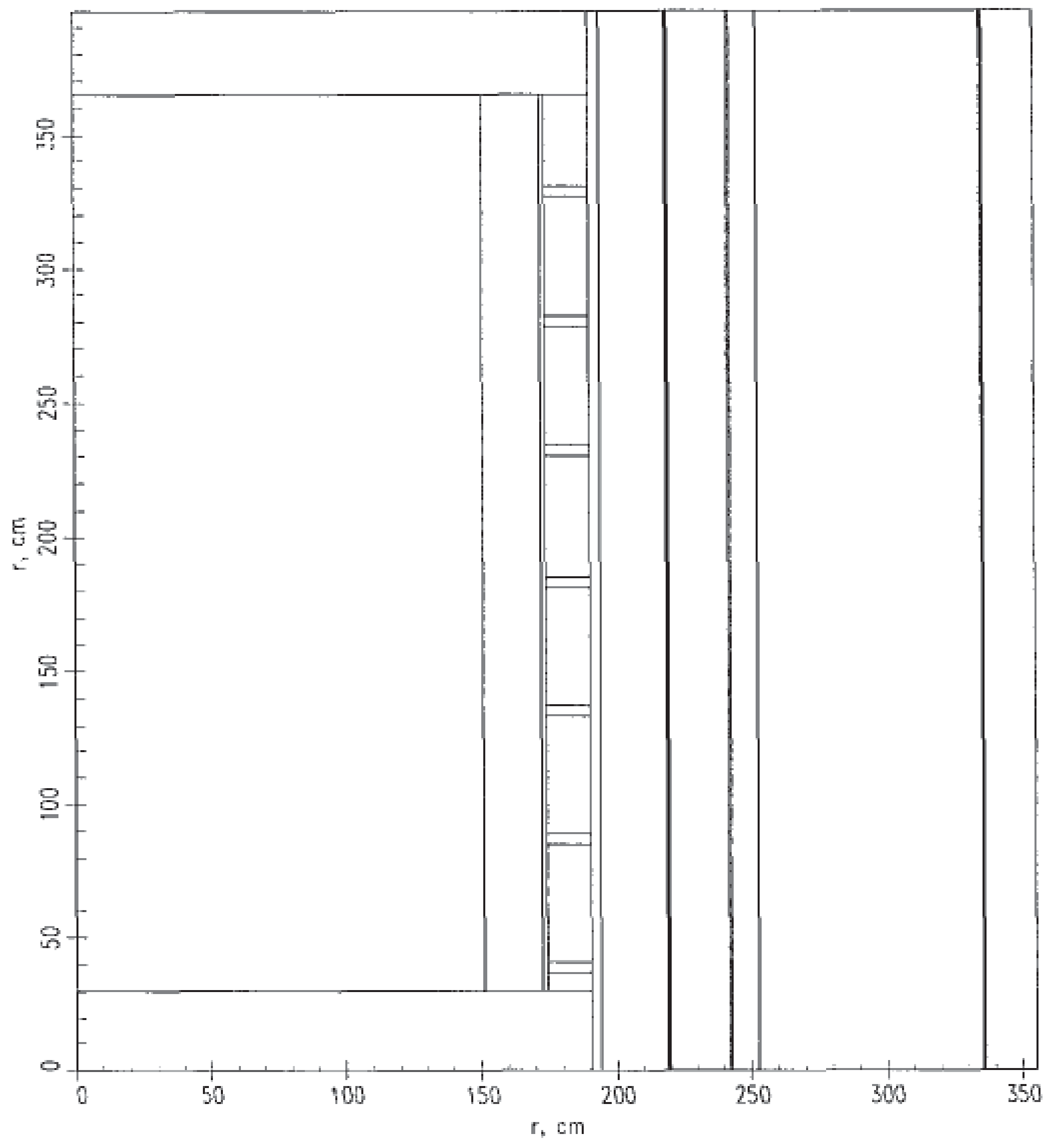


Figure 2.1-2

Palisades r,z Reactor Geometry



2.2 – Calculated Neutron Exposure of Pressure Vessel Beltline Materials

The results of the neutron exposure calculations for both the traditional and extended beltline regions of the Palisades pressure vessel are provided in Tables 2.2-1 and 2.2-2. In Table 2.2-1, the axial location of the maximum neutron exposure of each of the traditional and extended beltline materials is listed. The axial location of each material is indexed to Z = 0.0 cm at the midplane of the active fuel stack modeled in the neutronic calculation. In Table 2.2-2, the maximum projected neutron fluence for each of the beltline materials is provided. Projected fluence values are listed for EOC21 (23.4 EFPY), EOLE (42.1 EFPY), and EOC36 (44.0 EFPY).

From Table 2.2-2, it is noted that, although the upper shell course and the associated upper shell to intermediate shell circumferential weld are projected to exceed the 1.0E+17 n/cm^2 fluence (E > 1.0 MeV) threshold, the nozzles themselves as well as the nozzle to nozzle shell welds remain below 1.0E+17 n/cm^2 through 44 EFPY. Likewise, the lower shell to lower head circumferential weld will remain below the threshold of 1.0E+17 n/cm^2 through 44 EFPY of operation. The current analysis demonstrates that, for 44 EFPY of reactor operation, the axial span of the pressure vessel that is projected to exceed the neutron fluence (E > 1.0 MeV) threshold of 1.0E+17 n/cm^2 is limited to a zone extending from approximately 245 cm below the midplane of the active fuel stack to approximately 245 cm above the midplane of the active fuel stack.

The azimuthal distribution of fuel cycle-specific calculated fast neutron (E > 1.0 MeV) flux and fluence experienced by the materials comprising the beltline region of the Palisades pressure vessel are given in Tables 2.2-3 through 2.2-6, respectively, for plant operation through the conclusion of the twenty-first fuel cycle. Cycle 21 represents the last fuel cycle for which final fuel loading patterns have been designed. As presented, the data in Tables 2.2-3 through 2.2-6 represent the maximum neutron exposures at the pressure vessel clad base metal interface at azimuthal angles of 0°, 15°, 30°, 45°, 60°, 75°, and 90° relative to the core major axes. The limiting weld material for the Palisades pressure vessel occurs along the 60° azimuth (Heat W5214, Weld IDs 2-112A/C and 3-112A/C). All of the data provided in Tables 2.2-3 through 2.2-6 were taken at the axial location of the maximum exposure experienced at each azimuth based on the results of the three-dimensional synthesized neutron fluence evaluations.

In Tables 2.2-7 and 2.2-8, projections of neutron (E > 1.0 MeV) fluence beyond the end of Cycle 21 are provided for the traditional and extended beltline materials, respectively. These projections were based on assumed future operating conditions provided by Entergy. In particular the following assumptions were applied to the analysis:

1 - For Cycle 22, the nominal calculated neutron flux based on the average of the prior uprated fuel cycles (18 through 21) was used. This approach is a realistic representation of the neutron flux that would be expected based on existing preliminary designs for Cycle 22.

2 - For Cycles 23 and beyond, the Cycle 21 neutron flux distribution was applied for all fuel cycles. This is a conservative assumption in that, considering Cycles 15 through 21, the Cycle 21 power distribution results in the highest calculated flux at the location of the critical pressure vessel weld material(60°).

3 - Projected fuel cycle lengths were provided by Entergy as follows:

	Design	**95% Capacity**
Cycle 22	525 EFPD	499 EFPD
Cycle 23	525 EFPD	499 EFPD
Cycle 24	502 EFPD	477 EFPD
Cycles 25+	530 EFPD	504 EFPD

Fuel cycles were assumed to operate with a breaker to breaker capacity factor of 95%.

In completing the projections beyond the end of Cycle 21, operation was assumed to a total of 44 EFPY. Given the assumed operating scenario, this would cover a calendar time period extending to 2033.

In regard to the fluence data provided in Tables 2.2-2, 2.2-5, 2.2-6, 2.2-7 and 2.2-8, it should be noted that the critical longitudinal welds (2-112A, 2-112C, 3-112A, and 3-112C) are exposed to the neutron flux characteristic of the 60° azimuthal location. The beltline circumferential weld 9-112 and all plate materials are exposed to the maximum neutron exposure characteristic of the 75° azimuthal location.

Table 2.2-1
Palisades Pressure Vessel Material Locations in the
Traditional and Extended Beltline Regions

Material	Axial Location[a,b] [cm]	Notes
Outlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	260.02	Extended Beltline
Inlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	277.72	Extended Beltline
Upper Shell Plates D-3802	204.93	Extended Beltline
Upper Shell Longitudinal Welds 1-112	204.93	Extended Beltline
Upper Shell to Intermediate Shell Circumferential Weld 8-112	204.93	Extended Beltline
Intermediate Shell Plates D-3803	-42.24	Traditional Beltline
Intermediate Shell Longitudinal Welds 2-112	-42.24	Traditional Beltline
Intermediate Shell to Lower Shell Circumferential Weld 9-112	-42.24	Traditional Beltline
Lower Shell Plates D-3804	-42.24	Traditional Beltline
Lower Shell Longitudinal Welds 3-112	-42.24	Traditional Beltline
Lower Shell to Lower Vessel Head Circumferential Weld 10-112	-279.43	Extended Beltline

[a] Axial elevations are indexed to Z = 0.0 at the midplane of the active fuel stack.

[b] Elevations listed represent the location of maximum neutron exposure for the material.

Table 2.2-2
Palisades Maximum Fast Neutron (E > 1.0 MeV) Fluence Experienced by Materials in the Tradional and Extended Beltline

Material	Neutron Fluence [n/cm^2]		
	23.4 EFPY	**42.1 EFPY**	**44 EFPY**
Outlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	<1.0E+17	<1.0E+17	<1.0E+17
Inlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	<1.0E+17	<1.0E+17	<1.0E+17
Upper Shell Plates D-3802	9.895E+17	1.529E+18	1.584E+18
Upper Shell Longitudinal Welds 1-112	6.782E+17	9.707E+17	1.001E+18
Upper Shell to Intermediate Shell Circumferential Weld 8-112	9.895E+17	1.529E+17	1.584E+17
Intermediate Shell Plates D-3803	2.157E+19	3.428E+19	3.558E+19
Intermediate Shell Longitudinal Welds 2-112	1.472E+19	2.161E+19	2.232E+19
Intermediate Shell to Lower Shell Circumferential Weld 9-112	2.157E+19	3.428E+19	3.558E+19
Lower Shell Plates D-3804	2.157E+19	3.428E+19	3.558E+19
Lower Shell Longitudinal Welds 3-112	1.472E+19	2.161E+19	2.232E+19
Lower Shell to Lower Vessel Head Circumferential Weld 10-112	<1.0E+17	<1.0E+17	<1.0E+17

Table 2.2-3
Summary of Calculated Maximum Pressure Vessel Neutron Flux (E > 1.0 MeV)
For Cycles 15 through 21 and for Future Projection
Traditional Beltline Materials

Fuel Cycle	Cycle Time (EFPY)	Neutron (E > 1.0 MeV) Flux (n/cm^2-s)			
		0 Deg.	**15 Deg.**	**30 Deg.**	**45 Deg.**
15	1.1	9.671E+09	1.572E+10	1.277E+10	7.924E+09
16	1.2	1.068E+10	1.614E+10	1.330E+10	7.797E+09
17	1.3	1.080E+10	1.875E+10	1.332E+10	7.613E+09
18	1.3	1.292E+10	2.102E+10	1.352E+10	7.337E+09
19	1.3	1.059E+10	1.940E+10	1.445E+10	7.037E+09
20	1.4	1.123E+10	2.021E+10	1.517E+10	8.143E+09
21	1.4	1.138E+10	2.032E+10	1.501E+10	8.506E+09
22	Projected	1.153E+10	2.024E+10	1.454E+10	7.756E+09
23+	Projected	1.138E+10	2.032E+10	1.501E+10	8.506E+09

Fuel Cycle	Cycle Time (EFPY)	Neutron (E > 1.0 MeV) Flux (n/cm^2-s)			
		60 Deg.	**75 Deg.**	**90 Deg.**	
15	1.1	1.105E+10	1.679E+10	1.257E+10	
16	1.2	1.135E+10	1.761E+10	1.401E+10	
17	1.3	9.781E+09	1.968E+10	1.539E+10	
18	1.3	1.088E+10	2.236E+10	1.664E+10	
19	1.3	1.090E+10	2.232E+10	1.743E+10	
20	1.4	1.161E+10	2.197E+10	1.650E+10	
21	1.4	1.172E+10	2.152E+10	1.618E+10	
22	Projected	1.128E+10	2.204E+10	1.669E+10	
23+	Projected	1.172E+10	2.152E+10	1.618E+10	

Data Applicability

Lower Shell Plates D-3804
Lower Shell Longitudinal Welds 3-112
Intermediate Shell Plates D-3803
Intermediate Shell Longitudinal Welds 2-112
Intermediate-Lower Shell Circumferential Weld 2-112

Table 2.2-4
Summary of Calculated Maximum Pressure Vessel Neutron Flux (E > 1.0 MeV)
For Cycles 15 through 21 and for Future Projection
Extended Beltline Materials

Fuel Cycle	Cycle Time (EFPY)	Neutron (E > 1.0 MeV) Flux (n/cm^2-s)			
		0 Deg.	**15 Deg.**	**30 Deg.**	**45 Deg.**
15	1.1	4.105E+08	6.613E+08	5.420E+08	3.363E+08
16	1.2	4.495E+08	6.751E+08	5.597E+08	3.281E+08
17	1.3	4.639E+08	7.990E+08	5.722E+08	3.270E+08
18	1.3	5.553E+08	9.001E+08	5.811E+08	3.154E+08
19	1.3	4.652E+08	8.452E+08	6.347E+08	3.091E+08
20	1.4	4.956E+08	8.844E+08	6.695E+08	3.594E+08
21	1.4	4.830E+08	8.625E+08	6.370E+08	3.610E+08
22	Projected	4.913E+08	8.624E+08	6.200E+08	3.315E+08
23+	Projected	4.830E+08	8.625E+08	6.370E+08	3.610E+08

Fuel Cycle	Cycle Time (EFPY)	Neutron (E > 1.0 MeV) Flux (n/cm^2-s)			
		60 Deg.	**75 Deg.**	**90 Deg.**	
15	1.1	4.690E+08	7.135E+08	5.336E+08	
16	1.2	4.777E+08	7.415E+08	5.896E+08	
17	1.3	4.201E+08	8.449E+08	6.611E+08	
18	1.3	4.677E+08	9.607E+08	7.152E+08	
19	1.3	4.788E+08	9.796E+08	7.657E+08	
20	1.4	5.124E+08	9.700E+08	7.282E+08	
21	1.4	4.975E+08	9.136E+08	6.868E+08	
22	Projected	4.811E+08	9.384E+08	7.102E+08	
23+	Projected	4.975E+08	9.136E+08	6.868E+08	

Data Applicability

Upper Shell Plates D-3802

Upper Shell Longitudinal Welds 1-112

Upper Shell-Intermediate Shell Circumferential Weld 8-112

Table 2.2-5
Summary of Calculated Maximum Pressure Vessel Neutron Exposure
Through the Conclusion of Cycle 21
Traditional Beltline Materials

Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			0 Deg.	15 Deg.	30 Deg.	45 Deg.
1-14	14.4	14.4	1.132E+19	1.576E+19	1.192E+19	7.467E+18
15	1.1	15.5	1.165E+19	1.631E+19	1.237E+19	7.742E+18
16	1.2	16.7	1.206E+19	1.693E+19	1.288E+19	8.041E+18
17	1.3	18.0	1.252E+19	1.773E+19	1.344E+19	8.365E+18
18	1.3	19.3	1.305E+19	1.858E+19	1.400E+19	8.665E+18
19	1.3	20.6	1.347E+19	1.935E+19	1.457E+19	8.944E+18
20	1.4	22.0	1.395E+19	2.023E+19	1.522E+19	9.295E+18
21	1.4	23.4	1.447E+19	2.114E+19	1.590E+19	9.677E+18

Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			60 Deg.	75 Deg.	90 Deg.	
1-14	14.4	14.4	1.158E+19	1.576E+19	1.132E+19	
15	1.1	15.5	1.196E+19	1.635E+19	1.175E+19	
16	1.2	16.7	1.240E+19	1.702E+19	1.229E+19	
17	1.3	18.0	1.282E+19	1.786E+19	1.295E+19	
18	1.3	19.3	1.326E+19	1.877E+19	1.363E+19	
19	1.3	20.6	1.369E+19	1.966E+19	1.432E+19	
20	1.4	22.0	1.419E+19	2.060E+19	1.503E+19	
21	1.4	23.4	1.472E+19	2.157E+19	1.576E+19	

Data Applicability

Lower Shell Plates D-3804
Lower Shell Longitudinal Welds 3-112
Intermediate Shell Plates D-3803
Intermediate Shell Longitudinal Welds 2-112
Intermediate-Lower Shell Circumferential Weld 9-112

Table 2.2-6
Summary of Calculated Maximum Pressure Vessel Neutron Exposure
Through the Conclusion of Cycle 21
Extended Beltline Materials

Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			0 Deg.	**15 Deg.**	**30 Deg.**	**45 Deg.**
1-14	14.4	14.4	5.308E+17	7.394E+17	5.593E+17	3.503E+17
15	1.1	15.5	5.450E+17	7.623E+17	5.780E+17	3.620E+17
16	1.2	16.7	5.623E+17	7.882E+17	5.996E+17	3.746E+17
17	1.3	18.0	5.821E+17	8.223E+17	6.240E+17	3.885E+17
18	1.3	19.3	6.047E+17	8.590E+17	6.477E+17	4.013E+17
19	1.3	20.6	6.232E+17	8.925E+17	6.728E+17	4.136E+17
20	1.4	22.0	6.446E+17	9.307E+17	7.017E+17	4.291E+17
21	1.4	23.4	6.663E+17	9.694E+17	7.303E+17	4.453E+17

Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			60 Deg.	**75 Deg.**	**90 Deg.**	
1-14	14.4	14.4	5.432E+17	7.394E+17	5.308E+17	
15	1.1	15.5	5.595E+17	7.641E+17	5.493E+17	
16	1.2	16.7	5.778E+17	7.926E+17	5.719E+17	
17	1.3	18.0	5.957E+17	8.286E+17	6.001E+17	
18	1.3	19.3	6.148E+17	8.678E+17	6.293E+17	
19	1.3	20.6	6.338E+17	9.066E+17	6.596E+17	
20	1.4	22.0	6.559E+17	9.485E+17	6.911E+17	
21	1.4	23.4	6.782E+17	9.895E+17	7.219E+17	

Data Applicability
Upper Shell Plates D-3802
Upper Shell Longitudinal Welds 1-112
Upper Shell-Intermediate Shell Circumferential Weld 8-112

Table 2.2-7
Projections of Calculated Maximum Neutron Exposure of Pressure Vessel Beltline Materials
Lower Shell Longitudinal Welds, Lower Shell, Intermediate Shell Longitudinal Welds,
Intermediate Shell, Intermediate-Lower Shell Circumferential Weld

End of Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			0 Deg.	15 Deg.	30 Deg.	45 Deg.
21	1.4	23.4	1.447E+19	2.114E+19	1.590E+19	9.677E+18
22	1.4	24.7	1.496E+19	2.201E+19	1.652E+19	1.001E+19
23	1.4	26.1	1.545E+19	2.289E+19	1.717E+19	1.038E+19
24	1.3	27.4	1.592E+19	2.372E+19	1.779E+19	1.073E+19
25	1.4	28.8	1.642E+19	2.461E+19	1.844E+19	1.110E+19
26	1.4	30.2	1.691E+19	2.549E+19	1.910E+19	1.147E+19
27	1.4	31.6	1.741E+19	2.638E+19	1.975E+19	1.184E+19
28	1.4	32.9	1.790E+19	2.726E+19	2.040E+19	1.221E+19
29	1.4	34.3	1.840E+19	2.815E+19	2.106E+19	1.258E+19
30	1.4	35.7	1.889E+19	2.903E+19	2.171E+19	1.295E+19
31	1.4	37.1	1.939E+19	2.992E+19	2.236E+19	1.332E+19
32	1.4	38.5	1.989E+19	3.080E+19	2.302E+19	1.369E+19
33	1.4	39.8	2.038E+19	3.169E+19	2.367E+19	1.406E+19
34	1.4	41.2	2.088E+19	3.257E+19	2.432E+19	1.443E+19
EOLE		42.1	2.118E+19	3.311E+19	2.472E+19	1.466E+19
35	1.4	42.6	2.137E+19	3.346E+19	2.498E+19	1.480E+19
36	1.4	44.0	2.187E+19	3.434E+19	2.563E+19	1.517E+19

End of Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			60 Deg.	75 Deg.	90 Deg.	
21	1.4	23.4	1.472E+19	2.157E+19	1.576E+19	
22	1.4	24.7	1.520E+19	2.252E+19	1.647E+19	
23	1.4	26.1	1.571E+19	2.345E+19	1.717E+19	
24	1.3	27.4	1.619E+19	2.433E+19	1.784E+19	
25	1.4	28.8	1.670E+19	2.527E+19	1.854E+19	
26	1.4	30.2	1.721E+19	2.621E+19	1.925E+19	
27	1.4	31.6	1.772E+19	2.715E+19	1.995E+19	
28	1.4	32.9	1.823E+19	2.808E+19	2.066E+19	
29	1.4	34.3	1.874E+19	2.902E+19	2.136E+19	
30	1.4	35.7	1.925E+19	2.996E+19	2.207E+19	
31	1.4	37.1	1.976E+19	3.090E+19	2.277E+19	
32	1.4	38.5	2.028E+19	3.183E+19	2.347E+19	
33	1.4	39.8	2.079E+19	3.277E+19	2.418E+19	
34	1.4	41.2	2.130E+19	3.371E+19	2.488E+19	
EOLE		42.1	2.161E+19	3.428E+19	2.531E+19	
35	1.4	42.6	2.181E+19	3.464E+19	2.559E+19	
36	1.4	44.0	2.232E+19	3.558E+19	2.629E+19	

Table 2.2-8
Projections of Calculated Maximum Neutron Exposure of Pressure Vessel Extended Beltline Materials Upper Shell, Upper Shell Longitudinal Welds, Intermediate-Upper Shell Circumferential Weld

End of Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			0 Deg.	15 Deg.	30 Deg.	45 Deg.
21	1.4	23.4	6.663E+17	9.694E+17	7.303E+17	4.453E+17
22	1.4	24.7	6.874E+17	1.007E+18	7.570E+17	4.596E+17
23	1.4	26.1	7.082E+17	1.044E+18	7.845E+17	4.752E+17
24	1.3	27.4	7.281E+17	1.079E+18	8.107E+17	4.900E+17
25	1.4	28.8	7.492E+17	1.117E+18	8.385E+17	5.058E+17
26	1.4	30.2	7.702E+17	1.154E+18	8.662E+17	5.215E+17
27	1.4	31.5	7.912E+17	1.192E+18	8.939E+17	5.372E+17
28	1.4	32.9	8.123E+17	1.230E+18	9.217E+17	5.529E+17
29	1.4	34.3	8.333E+17	1.267E+18	9.494E+17	5.686E+17
30	1.4	35.7	8.543E+17	1.305E+18	9.772E+17	5.844E+17
31	1.4	37.1	8.754E+17	1.342E+18	1.005E+18	6.001E+17
32	1.4	38.4	8.964E+17	1.380E+18	1.033E+18	6.158E+17
33	1.4	39.8	9.174E+17	1.417E+18	1.060E+18	6.315E+17
34	1.4	41.2	9.385E+17	1.455E+18	1.088E+18	6.472E+17
EOLE		42.1	9.513E+17	1.478E+18	1.105E+18	6.568E+17
35	1.4	42.6	9.595E+17	1.492E+18	1.116E+18	6.630E+17
36	1.4	44.0	9.805E+17	1.530E+18	1.144E+18	6.787E+17

End of Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			60 Deg.	75 Deg.	90 Deg.	
21	1.4	23.4	6.782E+17	9.895E+17	7.219E+17	
22	1.4	24.7	6.989E+17	1.030E+18	7.525E+17	
23	1.4	26.1	7.204E+17	1.069E+18	7.821E+17	
24	1.3	27.4	7.409E+17	1.107E+18	8.104E+17	
25	1.4	28.8	7.625E+17	1.147E+18	8.403E+17	
26	1.4	30.2	7.842E+17	1.187E+18	8.702E+17	
27	1.4	31.5	8.059E+17	1.226E+18	9.001E+17	
28	1.4	32.9	8.275E+17	1.266E+18	9.300E+17	
29	1.4	34.3	8.492E+17	1.306E+18	9.599E+17	
30	1.4	35.7	8.709E+17	1.346E+18	9.898E+17	
31	1.4	37.1	8.925E+17	1.385E+18	1.020E+18	
32	1.4	38.4	9.142E+17	1.425E+18	1.050E+18	
33	1.4	39.8	9.359E+17	1.465E+18	1.080E+18	
34	1.4	41.2	9.575E+17	1.505E+18	1.109E+18	
EOLE		42.1	9.707E+17	1.529E+18	1.128E+18	
35	1.4	42.6	9.792E+17	1.545E+18	1.139E+18	
36	1.4	44.0	1.001E+18	1.584E+18	1.169E+18	

SECTION 3.0

NEUTRON DOSIMETRY EVALUATIONS

During the first 14 operating fuel cycles at the Palisades plant, five sets of in-vessel surveillance capsule dosimetry and three sets of ex-vessel dosimetry were irradiated, withdrawn, and analyzed. The results of these dosimetry evaluations provide a measurement data base that can be used to demonstrate that the neutron fluence calculations completed for the Palisades reactor meet the uncertainty requirements described in Regulatory Guide 1.190.[6] That is, the calculations and measurements should agree within 20% at the 1σ level.

These calculation/measurement comparisons were previously completed and documented in Reference 1. However, for completeness, a brief description of the measurement program, dosimetry evaluation procedure, and final results are also included in this supplement to Reference 1.

In addition to the Palisades dosimetry evaluations, this general methodology was also used in the determination of capsule exposures from the other PWR's included in Section 4.0 of this report.

3.1 – Method of Analysis

Evaluations of neutron sensor sets contained in the in-vessel and ex-vessel dosimetry capsules withdrawn to date from the Palisades reactor were completed using current state-of-the art least-squares methodology that meet the requirements of Regulatory Guide 1.190[6].

These least-squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculations resulting in a best estimate neutron energy spectrum with associated uncertainties. Best estimates for key exposure parameters such as $\phi(E > 1.0\ \text{MeV})$ and iron atom displacement rate (dpa/s) along with their uncertainties are then easily obtained from the adjusted spectrum. In general, the least-squares methods, as applied to reactor dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross sections, and the calculated neutron energy spectrum within their respective uncertainties.

For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\phi_g \pm \delta_{\phi_g})$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, ϕ_g, through the multigroup dosimeter reaction cross section, σ_{ig}, each with an uncertainty δ. The primary

objective of the least-squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least-squares evaluation of the Palisades dosimetry, the NRC approved methodology based on the use of the FERRET adjustment code[5] was employed to combine the results of the plant-specific neutron transport calculations and sensor set reaction rate measurements to determine best estimate values of exposure parameters along with associated uncertainties at the measurement locations.

The application of the least-squares methodology requires the following input.

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rate and associated uncertainty for each sensor contained in the multiple foil set.

3. The energy-dependent dosimetry reaction cross sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the Palisades application, the calculated neutron spectrum at each measurement location was obtained from the results of plant-specific neutron transport calculations based on the methodology described in Section 2.0 of this report. The calculated spectrum at each sensor set location was input to the adjustment procedure in an absolute sense (rather than as simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements. The sensor reaction rates were derived from the measured specific activities of each sensor set and the operating history of the respective fuel cycles. The dosimetry reaction cross sections were obtained from the SNLRML dosimetry cross-section library [10].

In addition to the magnitude of the calculated neutron spectra, the measured sensor set reaction rates, and the dosimeter set reaction cross sections, the least-squares procedure requires uncertainty estimates for each of these input parameters. The following provides a summary of the uncertainties associated with the least-squares evaluation of the Palisades dosimetry.

Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, the irradiation history corrections, and the corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM national consensus standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input to the least-squares evaluation:

Reaction	Uncertainty
$Cu^{63}(n,\alpha)Co^{60}$	5%
$Ti^{46}(n,p)Sc^{46}$	5%
$Fe^{54}(n,p)Mn^{54}$	5%
$Ni^{58}(n,p)Co^{58}$	5%
$U^{238}(n,f)Cs^{137}$	10%
$Nb^{93}(n,n')Nb^{93m}$	5%
$Np^{237}(n,f)Cs^{137}$	10%
$Co^{59}(n,\gamma)Co^{60}$	5%

These uncertainties are given at the 1σ level.

Dosimetry Cross-Section Uncertainties

As noted above, the reaction rate cross sections used in the least-squares evaluations were taken from the SNLRML library. This data library provides reaction cross sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross sections and uncertainties are provided in a fine multigroup structure for use in least-squares adjustment applications. These cross sections were compiled from the most recent cross-section evaluations and they have been tested with respect to their accuracy and consistency for least-squares evaluations. Further, the library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources. Detailed discussions of the contents of the SNLRML library along with the evaluation process for each of the sensors is provided in Reference 10.

For sensors included in the Palisades dosimetry sets, the following uncertainties in the fission spectrum-averaged cross sections are provided in the SNLRML documentation package:

Reaction	Uncertainty
$Cu^{63}(n,\alpha)Co^{60}$	4.08-4.16%
$Ti^{46}(n,p)Sc^{46}$	4.50-4.87%
$Fe^{54}(n,p)Mn^{54}$	3.05-3.11%
$Ni^{58}(n,p)Co^{58}$	4.49-4.56%
$U^{238}(n,f)FP$	0.54-0.64%
$Nb^{93}(n,n')Nb^{93m}$	6.96-7.23%
$Np^{237}(n,f)FP$	10.32-10.97%
$Co^{59}(n,\gamma)Co^{60}$	0.79-3.59%

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with the sensor sets used in LWR irradiations.

Calculated Neutron Spectrum Uncertainties
While the uncertainties associated with the reaction rates were obtained from the measurement procedures and counting benchmarks, and the dosimetry cross-section uncertainties were supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$M_{gg'} = R_n^2 + R_g * R_{g'} * P_{gg'}$$

where R_n specifies an overall fractional normalization uncertainty, and the fractional uncertainties $R_{g'}$ and R_g specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

$$P_{gg'} = [1 - \theta]\delta_{gg'} + \theta\, e^{-H}$$

where

$$H = \frac{(g - g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1.0 when g = g' and 0.0 otherwise.

The set of parameters defining the input covariance matrix for the Palisades calculated spectra was as follows:

Flux Normalization Uncertainty (R_n)	15%
Flux Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	29%
(E < 0.68 eV)	52%
Short-Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5

Flux Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

These uncertainty assignments are consistent with an industry consensus uncertainty of 15-20% (1σ) for the fast neutron portion of the spectrum and provide for a reasonable increase in the uncertainty for neutrons in the intermediate and thermal energy ranges.

3.2 – Dosimetry Evaluations

In this section, comparisons of the measurement results from the Palisades surveillance capsule and reactor cavity dosimetry with corresponding analytical predictions at the measurement locations are presented. These comparisons are provided on two levels. In the first instance, calculations of individual sensor reaction rates are compared directly with the measured reaction rates derived from the counting data obtained from the radiochemical laboratories. In the second case, the calculated values of neutron exposure expressed in terms of ϕ(E > 1.0 MeV), ϕ(E > 0.1 MeV), and iron atom displacements (dpa) are compared with the results of the least squares adjustment procedure described in Section 3.1. It is shown that these two levels of comparison yield consistent and similar results which demonstrate that the transport calculations for Palisades reactor produce neutron exposure results that meet the requirements of Regulatory Guide 1.190.[6]

In Table 3.2-1, measurement/calculation (M/C) ratios for each fast neutron sensor reaction from surveillance capsule and reactor cavity irradiations are listed. This comparison provides a direct comparison, on an absolute basis, of calculation and measurement prior to the application of the least squares adjustment procedure. In Table 3.2-2, comparisons of measured and adjusted neutron exposures are given in terms of adjusted/calculated ratios for the five surveillance capsule dosimetry sets withdrawn to date as well as for the three cycles of reactor cavity midplane dosimetry sets irradiated during Cycles 8, 9, and 10/11.

Table 3.2-1

Comparison of Measured and Calculated Threshold Foil Reaction Rates

Capsule	M/C Ratio					
	^{63}Cu(n,α)	^{46}Ti(n,p)	^{54}Fe(n,p)	^{58}Ni(n,p)	^{238}U(n,f)	^{237}Np(n,f)
A240	1.09	1.21	1.02	0.95		
W290	1.15	1.11	0.99	1.00	0.98	
W290-9	1.12	1.16	0.96	0.98	0.96	0.92
W110	1.17	1.17	1.02	1.01		
SA60-1	1.13	1.19	1.05	1.07	1.15	
84° Cavity						
Cycle 9	1.11	1.10	1.08	1.03	1.13	1.21
Cycle 10/11	1.15	1.11	1.10	1.08	1.32	1.11
74° Cavity						
Cycle 8	1.09	1.14	1.08	1.07	1.06	1.40
Cycle 9	1.03	1.07	1.01	1.01	0.93	1.13
Cycle 10/11	1.08	1.05	1.02	1.03	1.07	1.08
64° Cavity						
Cycle 8	1.09	1.15	1.08	1.06	1.04	1.32
Cycle 9	1.05	1.08	1.01	1.03	1.09	1.24
Cycle 10/11	1.07	1.10	1.05	1.03	1.10	1.12
54° Cavity						
Cycle 10/11	1.09	1.05	1.00		1.06	1.04
39° Cavity						
Cycle 8	1.08	1.21	1.14	1.11	1.06	1.32
Cycle 9	1.06	1.06	0.99	1.00	0.87	0.98
Cycle 10/11	1.03	1.12	1.05	1.05	1.06	1.06
24° Cavity						
Cycle 10/11	1.03	1.08	1.03	1.04	1.19	0.96
Average	**1.09**	**1.12**	**1.04**	**1.03**	**1.07**	**1.14**
% std dev	**3.9**	**4.7**	**4.4**	**3.8**	**10.0**	**12.8**

Reaction	Average M/C	% Standard Deviation
^{63}Cu(n,α)	1.09	3.9
^{46}Ti(n,p)	1.12	4.7
^{54}Fe(n,p)	1.04	4.4
^{58}Ni(n,p)	1.03	3.8
^{238}U(n,f)	1.07	10.0
^{237}Np(n,f)	1.14	12.8
Linear Average	**1.08**	**7.9**

Table 3.2-2

Comparison of Adjusted and Calculated Exposure Parameters

Capsule	Adjusted/Calculated (A/C) Ratio		
	φ(E > 1.0 MeV)	**φ(E > 0.1 MeV)**	**dpa**
A240	0.983	0.972	0.988
W290	0.988	0.981	0.997
W290-9	0.955	0.937	0.966
W110	1.011	1.001	1.020
SA60-1	1.078	1.067	1.077
84° Cavity			
Cycle 9	1.091	1.083	1.084
Cycle 10/11	1.142	1.133	1.134
74° Cavity			
Cycle 8	1.108	1.120	1.116
Cycle 9	0.999	0.993	0.996
Cycle 10/11	1.044	1.058	1.055
64° Cavity			
Cycle 8	1.086	1.096	1.092
Cycle 9	1.055	1.033	1.038
Cycle 10/11	1.065	1.078	1.075
54° Cavity			
Cycle 10/11	1.026	1.039	1.036
39° Cavity			
Cycle 8	1.116	1.139	1.135
Cycle 9	0.949	0.956	0.957
Cycle 10/11	1.058	1.060	1.060
24° Cavity			
Cycle 10/11	1.062	1.050	1.053
Average	**1.05**	**1.04**	**1.05**
% std dev	**5.3**	**5.8**	**5.1**

SECTION 4.0

SURVEILLANCE CAPSULE NEUTRON FLUENCE

In support of embrittlement evaluations for the Palisades reactor pressure vessel, a compilation of calculated neutron fluence (E > 1.0 MeV) values for a series of materials surveillance capsules that contain test samples that apply to the Palisades plant is provided in this section. The compilation, encompassing a total of 18 surveillance capsules irradiated at the Palisades, Indian Point Unit 2, H. B. Robinson Unit 2, and Indian Point Unit 3 reactors is provided in Table 4-1.

For each surveillance capsule listed in Table 4-1, the reported fluence value was calculated using an NRC approved methodology that meets the requirements of Regulatory Guide 1.190[6]. Therefore, this tabulation represents a consistent set of fluence values for use in data correlations. Details of the analysis methodology as applied to each of the four host reactors are given in References 1, 11, 12, 13, 14, and 15.

In providing the data listed in Table 4-1, no new fluence calculations were performed. The data were obtained either from proprietary Palisades specific documents[11, 12] or from non-proprietary public domain documents[1, 13, 14, 15] that have been submitted to the NRC and are available on the ADAMS document system. It should be noted that, relative to the Palisades data listed in Table 4-1, References 1, 11, and 12 did not explicitly report fluence (E > 1.0 MeV) values for the individual capsules. Rather, the irradiation environment was reported in terms of irradiation time and calculated neutron flux (E > 1.0 MeV) averaged over the irradiation period. The fluence values listed in Table 4-1 were computed as the product of the irradiation time and the average neutron flux reported in these documents.

Relative to the data in Table 4-1 and the listed references, it should also be noted that, in addition to the Reg. Guide 1.190 derived fluence values for Indian Point Unit 2, Table 3 of Reference 13 also lists fluence values for H. B. Robinson Unit 2 and Indian Point Unit 3 that were extracted from older references. These older values have been updated and superseded by the fluence values documented in References 14 and 15, respectively. All of these updated fluence values reflect the application of a fluence methodology that meets the requirements of Reg. Guide 1.190.

Table 4-1

Summary of Neutron Fluence (E > 1.0 MeV) Derived from the Application of Methodology Meeting the Requirements of Regulatory Guide 1.190

Reactor	Surveillance Capsule Designation	Fluence (E > 1.0 Mev) [n/cm^2]	Reference
Palisades	A240	4.09e+19	WCAP-15353, R0 (Ref. 1)
Palisades	W290	9.38e+18	WCAP-15353, R0 (Ref. 1)
Palisades	W110	1.64e+19	WCAP-15353, R0 (Ref. 1)
Palisades	SA60-1	1.50e+19	WCAP-15353, R0 (Ref. 1)
Palisades	SA240-1	2.38e+19	CPAL-01-009 (Ref. 11)
Palisades	W100	2.09e+19	CPAL-04-8 (Ref. 12)
Indian Point 2	T	2.53e+18	WCAP-15629, R1 (Table 3) (Ref. 13)
Indian Point 2	Y*	4.55e+18	WCAP-15629, R1 (Table 3) (Ref. 13)
Indian Point 2	Z	1.02e+19	WCAP-15629, R1 (Table 3) (Ref. 13)
Indian Point 2	V*	4.92e+18	WCAP-15629, R1 (Table 3) (Ref. 13)
H. B. Robinson	S	4.79e+18	WCAP-15805, R0 (Table 5-10) (Ref. 14)
H. B. Robinson	V*	5.30e+18	WCAP-15805, R0 (Table 5-10) (Ref. 14)
H. B. Robinson	T*	3.87e+19	WCAP-15805, R0 (Table 5-10) (Ref. 14)
H. B. Robinson	X*	4.49e+19	WCAP-15805, R0 (Table 5-10) (Ref. 14)
Indian Point 3	T*	2.63e+18	WCAP-16251-NP, R0 (Table 5-10) (Ref. 15)
Indian Point 3	Y*	6.92e+18	WCAP-16251-NP, R0 (Table 5-10) (Ref. 15)
Indian Point 3	Z*	1.04e+19	WCAP-16251-NP, R0 (Table 5-10) (Ref. 15)
Indian Point 3	X*	8.74e+18	WCAP-16251-NP, R0 (Table 5-10) (Ref. 15)

Notes:

1 - Relative to the Palisades data, References 1, 11, and 12 did not explicitly report fluence values for the listed capsules. Rather, the irradiation environment was reported in terms of irradiation time and neutron flux averaged over the irradiation period. The fluence values listed in Table 4-1 were computed as the product of the irradiation time and the average neutron flux (E > 1.0 MeV) reported in those documents.

2 - In addition to the Reg. Guide 1.190 derived fluence values for Indian Point Unit 2, Table 3 of Reference 13 also lists fluence values for H. B. Robinson and Indian Point Unit 3 that were taken from older references. These values have been updated and superseded by the fluence values documented in References 14 and 15 that are based on a methodology that meets the requirements of Reg. Guide 1.190.

* Indicates Capsules in other plants that contain W5214 weld material.

SECTION 5.0

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Attachment 5

Westinghouse WCAP-17651-NP
Revision 0

Palisades Nuclear Power Plant Reactor Vessel Equivalent Margins Analysis

44 pages follow

Westinghouse Non-Proprietary Class 3

WCAP-17651-NP
Revision 0

February 2013

Palisades Nuclear Power Plant Reactor Vessel Equivalent Margins Analysis

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WCAP-17651-NP
Revision 0

Palisades Nuclear Power Plant Reactor Vessel Equivalent Margins Analysis

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February 2013

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ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
CD	cooldown
CE	Combustion Engineering
CEOG	Combustion Engineering Owners Group
CFR	Code of Federal Regulations
CVN	Charpy V-notch
EFPY	effective full-power years
EMA	equivalent margins analysis
EOLE	end-of-license extension
FSAR	Final Safety Analysis Report
HU	heatup
IS	intermediate shell
J-R	fracture toughness resistance
LS	lower shell
L-T	lateral-transverse
MnS	manganese-sulfide
NRC	U.S. Nuclear Regulatory Commission
RG	Regulatory Guide
RRVCH	replacement reactor vessel closure head
SF	structural factor
SIF	stress intensity factor
T-L	transverse-lateral
US	upper shell
USE	upper-shelf energy

EXECUTIVE SUMMARY

This report presents the methodology and results of the upper-shelf equivalent margins analysis (EMA) for the three Palisades Nuclear Power Plant reactor vessel materials with end-of-license-extension (EOLE) upper-shelf energy (USE) levels below the 50 ft-lb limit of 10 CFR 50, Appendix G. Materials with EOLE USE levels below 50 ft-lb are required to be evaluated, per paragraph IV.A.1.a of 10 CFR 50, Appendix G, for equivalent margins of safety specified in ASME Code Section XI, Appendix K.

The two Palisades reactor vessel beltline materials and one extended beltline material that drop below the 50 ft-lb limit were identified in WCAP-17341-NP, Revision 0 and WCAP-17403-NP, Revision 1, respectively. These reports concluded that upper shell plate D-3802-3 in the extended beltline, and lower shell plate D-3804-1 and intermediate to lower shell circumferential weld 9-112 (Heat #27204) in the traditional beltline, are predicted to drop below the 50 ft-lb limit required per 10 CFR 50, Appendix G at EOLE, which corresponds to 42.1 effective full-power years (EFPY). In WCAP-17403-NP, Revision 1, a methodology was proposed that could demonstrate acceptance to the 10 CFR 50, Appendix G 50 ft-lb limit for upper shell plate D-3802-3. However, Palisades has elected to perform the EMA on this material due to the risk that it may fall below the 50 ft-lb limit if future operation includes higher flux levels.

All three Palisades reactor vessel beltline and extended beltline regions with predicted Charpy upper-shelf energy levels falling below 50 ft-lb at EOLE were found to be acceptable for equivalent margins of safety per the ASME Code Section XI.

Service Level A and B Transients

- Intermediate to lower shell circumferential weld 9-112 (Heat #27204) is governing for EOLE USE margin at the 1/4-thickness location for normal Level A and B load conditions, based on the Regulatory Guide 1.161 fracture toughness methodology.

- This limiting material passed the flaw extension and stability criteria of ASME Section XI Appendix K.

- The equivalent margins analysis for the plate materials are acceptable and bounded by the conservative test data reported in NUREG/CR-5265 in all cases for Service Level A and B transients.

Service Level C Condition

- Intermediate to lower shell circumferential weld 9-112 (Heat #27204) is governing for EOLE USE margin at the 1/10-thickness location for the service Level C load condition, based on the Regulatory Guide 1.161 fracture toughness methodology.

- This limiting material passed the flaw extension and stability criteria of ASME Section XI Appendix K.

- The equivalent margins analysis for the plate materials are acceptable and bounded by the conservative test data reported in NUREG/CR-5265 in all cases for the Service Level C transient.

Service Level D Condition

- Intermediate to lower shell circumferential weld 9-112 (Heat #27204) is governing for EOLE USE margin at the 1/10-thickness location for the service Level D load condition, based on the Regulatory Guide 1.161 fracture toughness methodology.

- This limiting material passed the flaw extension and stability criterion of ASME Section XI, Appendix K.

- The equivalent margins analysis for the plate materials is acceptable and bounded by the conservative test data reported in NUREG/CR-5265 in all cases for the Service Level D transient.

1 INTRODUCTION

An upper-shelf energy (USE) evaluation was performed for the Palisades reactor vessel (RV) beltline materials in WCAP-17341-NP, Revision 0 (Reference 1) and for the extended beltline materials in WCAP-17403-NP, Revision 1 (Reference 2). These reports concluded that materials in three locations – upper shell plate D-3802-3 in the extended beltline, and lower shell plate D-3804-1 and intermediate to lower shell circumferential weld 9-112 (Heat #27204) in the traditional beltline – are predicted to drop below the 50 ft-lb limit required per 10 CFR 50, Appendix G (Reference 3) at end-of-license extension (EOLE), which corresponds to 42.1 effective full-power years (EFPY). In WCAP-17403-NP, Revision 1, a methodology was proposed that could demonstrate acceptance to the 10 CFR 50, Appendix G 50 ft-lb limit for upper shell plate D-3802-3. However, Palisades has elected to perform the equivalent margins analysis (EMA) on this material due to the risk that it may fall below the 50 ft-lb limit if future operation includes higher flux levels.

Reactor vessel materials with USE levels below 50 ft-lb are required to be evaluated, per paragraph IV.A.1.a of 10 CFR 50, Appendix G (References 3 and 4), for equivalent margins of safety specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI, Appendix K (Reference 5). This summary report provides the methodology and results of the upper-shelf EMA of the Palisades reactor vessel limiting materials for 42.1 EFPY. Figure 1-1 shows the locations of interest for this analysis in the Palisades reactor vessel.

Section 2 of this report discusses the methodologies used to complete the Palisades EMA. Section 3 identifies the acceptance criteria for the EMA with consideration of the various service loadings including Levels A, B, C, and D. Section 4 provides the inputs necessary to complete the EMA, while Section 5 documents the EMA evaluations. The conclusions of this report are documented in Section 6.

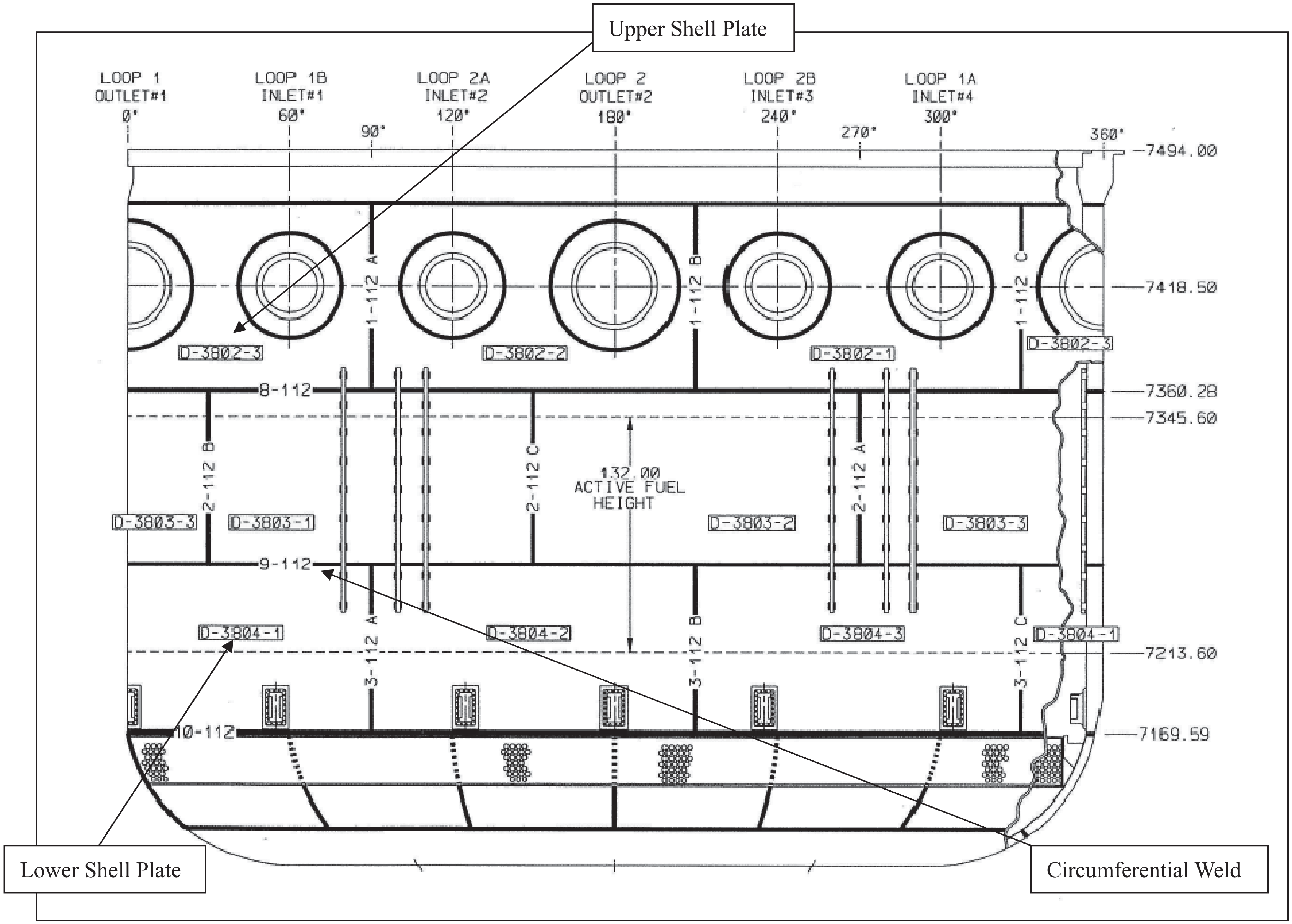

Figure 1-1 Palisades Reactor Vessel with Locations for EMA

2 METHOD DISCUSSION

2.1 ASME SECTION XI, APPENDIX K METHODOLOGY

ASME Section XI, Appendix K (Reference 5) specifies the methodology to be used to evaluate the equivalent margins for low upper-shelf materials. Reference 5 contains different postulated flaw depths, locations, and orientations, as well as the applied J-integral and stability criteria. These are briefly described here.

Applied SIF Calculation for Axial and Circumferential Flaws with Pressure Loading

For an axial flaw of depth a, the stress intensity factor (SIF) due to internal pressure is calculated with a structural factor (SF) on pressure using procedure in Article K-4000 in Reference 5.

Applied SIF for Axial and Circumferential Flaws with Thermal Loading

For an axial or circumferential flaw of depth a, the SIFs due to radial thermal gradients for cooldown rates up to 100°F/hr are calculated using procedure in Article K-4000 in Reference 5.

The SIFs for all other thermal design transients are computed using the stress distributions from the actual design transients analyzed in this EMA. The procedure from ASME Section XI, Appendix A (Reference 6) employing the cubic polynomial coefficients are also used.

Effective Flaw Depth and Applied J-Integrals

The effective flaw depth for small-scale yielding (a_e) is used and the applied J-integrals are calculated again using the procedure in Reference 5.

The calculation of a J-integral due to applied loads accounts for the material's elastic-plastic behavior of a stress-strain curve.

Postulated Flaws for Level A and B Service Loadings

The postulated flaw is an interior semi-elliptical surface with a depth of one-quarter of the vessel wall thickness and an aspect ratio (length over depth) of 6:1. Orientation of the flaw is assumed to be as follows:

$$a_0 = \frac{t_{base}}{4}$$

where,

a_0	=	postulated initial flaw depth, inches
t_{base}	=	thickness of the base or weld material, inches

For weld materials, the major axis of the flaw is to be oriented along the weld line.

For base materials, both axial and circumferential orientations are to be considered.

All the postulated flaws are oriented in the radial direction.

Postulated Flaws for Levels C and D Service Loadings

The postulated flaw is an interior semi-elliptical surface with a depth of 1/10 of the base metal wall thickness plus the cladding thickness (with a total depth not exceeding 1 inch), a surface length of six times the depth, and the flaw plane oriented in the radial direction.

$$a_0 = \frac{t_{base}}{10} + t_{clad}$$

where,

t_{clad} = thickness of the cladding, inches

For weld materials, the adequacy of the upper-shelf toughness with a flaw's major axis oriented along the weld of concern is evaluated.

For the base materials, the adequacy of the upper-shelf toughness with a flaw's major axis oriented along axial and circumferential directions is evaluated. The toughness properties for the corresponding orientations are used.

Flaws of various depths, ranging up to the maximum postulated depth, shall be analyzed to determine the most limiting flaw depth.

2.2 REGULATORY GUIDE 1.161 METHODOLOGY

Material Fracture Toughness Property

The material J-integral resistance property as a function of flaw extension is a conservative representation for the RV material beltline region. As the actual beltline J-integral fracture resistance material properties for the Palisades RV are not available, U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.161 (Reference 7, Section 3) is used. Regulatory Guide 1.161 has been developed to provide comprehensive guidance acceptable to the NRC staff for evaluating reactor pressure vessels when the Charpy USE falls below the 50 ft-lb limit of Appendix G to 10 CFR Part 50. The analysis methods in the regulatory position are based on methods developed for the ASME Code, Section XI, Appendix K (Reference 5). The NRC staff has reviewed the analysis methods in Appendix K and finds that they are technically acceptable but are not complete, because Appendix K does not provide information on the selection of transients and gives very little detail on the selection of material properties. In RG 1.161, specific guidance is provided on selecting transients for consideration and on appropriate material properties to be used in the analyses. The material fracture toughness J-resistance is provided in Reference 7 and is expressed as:

$$J_R = (MF)C_1(\Delta a)^{C_2}\, exp[C_3(\Delta a)^{C_4}]$$

where,

J_R	=	J-integral fracture resistance for the material, in-lb/in^2
MF	=	margin factor (see Table 4-4)
Δa	=	amount of ductile flaw extension, inches
C_1, C_2, C_3, C_4,	=	material constants used to describe the power-law fit to the J-integral resistance curve for the material

<u>Base Metal</u>

$$C_1 = exp[-2.44 + 1.13 * \ln CVN - 0.00277 * T]$$

$$C_2 = 0.077 + 0.116 * \ln C_1$$

$$C_3 = -0.0812 - 0.0092 * \ln C_1$$

$$C_4 = -0.409$$

<u>Weld Metal</u>

$$C_1 = exp[-4.12 + 1.49 * \ln CVN - 0.00249 * T]$$

$$C_2 = 0.077 + 0.116 * \ln C_1$$

$$C_3 = -0.0812 - 0.0092 * \ln C_1$$

$$C_4 = -0.5$$

Per RG 1.161, the Charpy v-notch upper-shelf energy (CVN) value should be matched to the proper orientation of the plate material (see Figure 2-1). Therefore, for axial flaws, the CVN value for the lateral-transverse (L-T) “strong” orientation in the vessel wall should be used. Similarly, for circumferential flaws, the CVN value for the transverse-lateral (T-L) “weak” orientation should be used. See Section 5.1 for additional details.

Also, with consideration of plate materials, the J-R model described in this section is developed for materials with high fracture toughness. For plate material with sulfur content less than 0.018 wt. %, the J-R model may be used. For plate material with sulfur content greater than 0.018 wt. %, the model may be used if it can be justified as conservative or a material-specific justification can be made based on other data. See Section 5.2 for additional details.

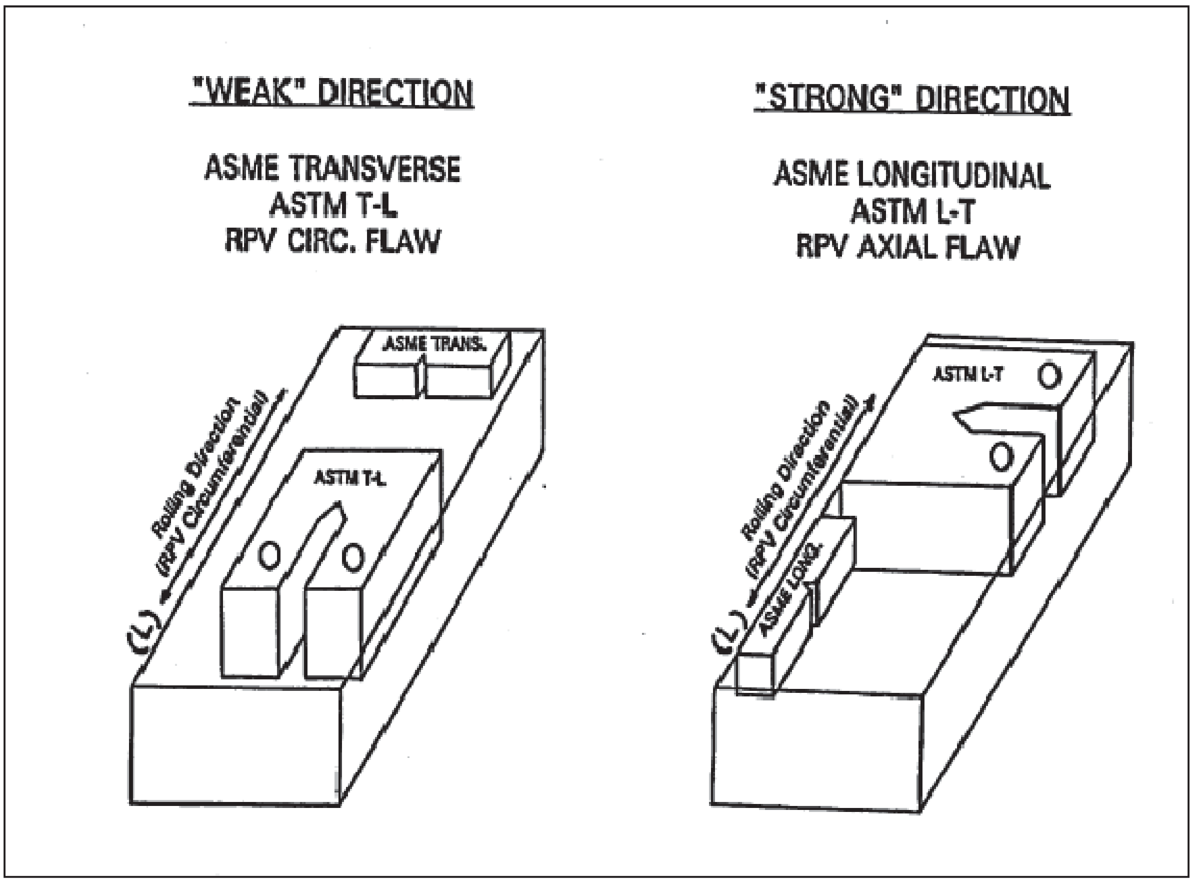

Figure 2-1 Definition of ASME Orientations

3 ACCEPTANCE CRITERIA

The ASME Code forms the basis for the requirements of Appendix G to 10 CFR Part 50. The acceptance criteria for the low-USE locations in the RV beltline materials are established in the ASME Code Section XI, Appendix K, Article K-2000 (Reference 5) and are summarized here.

3.1 LEVEL A AND B SERVICE LOADINGS

Flaw Extension Criterion

The applied J-integral evaluated at a pressure 1.15 times the accumulation pressure (as defined in the plant-specific overpressure protection report), with a structural factor of 1 on thermal loading for the plant-specific heatup (HU) and cooldown (CD) conditions, shall be less than the J-integral of the material at a ductile flaw extension of 0.1 inch.

$$J_1(a_1, 1.15P_a, CD\ or\ HU) < J_{0.1}$$

where J_1 is the applied J-integral with:

$$a_1 = \frac{t}{4} + 0.1\ in$$

where,

P_a	=	accumulation pressure as defined in the plant-specific overpressure protection report, but not exceeding 1.1 times the design pressure, ksi
HU	=	heatup
CD	=	cooldown
$J_{0.1}$	=	the J-integral resistance at a ductile flaw extension of 0.1 inch, in-lb/in^2

Flaw Stability Criterion

Flaw extensions at pressures up to 1.25 times the accumulation pressure shall be ductile and stable, using a structural factor of 1 on thermal loading for the plant-specific HU and CD conditions.

$$J(1.25\ P_a, CD\ or\ HU)\ should\ be\ in\ ductile\ tearing\ mode\ and\ stable$$

The flaw stability criterion is evaluated using:

- The J-integral due to applied loads for the postulated flaw in the vessel should satisfy the equilibrium equation for the stable flaw extension:

$$J = J_R$$

In the preceding equation:

J	=	J-integral due to applied loads
J_R	=	J-integral resistance to ductile tearing for the material

- The applied J-integral should satisfy the stability criterion for the following ductile tearing equation. Under increasing load, stable flaw extension will continue as long as $\frac{\partial J}{\partial a}$ remains less than $\frac{\partial J_R}{\partial a}$.

$$\frac{\partial J}{\partial a} \leq \frac{\partial J_R}{\partial a}$$

In the preceding equation:

$\frac{\partial J}{\partial a}$	=	partial derivative of applied J-integral with respect to flaw depth, a, with constant load
$\frac{\partial J_R}{\partial a}$	=	slope of the J-resistance curve

The above requirements for flaw extension and stability for Level A and B service loadings are satisfied as discussed in Section 5 of this report.

3.2 LEVEL C SERVICE LOADINGS

Flaw Extension Criterion

The applied J-integral, with a structural factor of 1 on loading, shall be less than the J-integral of the material at a ductile flaw extension of 0.1 inch.

$$J_1(a_1, P_a, CD\ or\ HU) < J_{0.1}$$

where J_1 is the applied J-integral with:

$$a_1 = a_0 + 0.1\ in$$

Flaw Stability Criterion

Flaw extensions shall be ductile and stable, similar to Level A and B service loadings, with a structural factor of 1 on loading.

The preceding requirements for flaw extension and stability for Level C service loadings are satisfied as discussed in Section 5 of this report.

3.3 LEVEL D SERVICE LOADINGS

Flaw Stability Criterion

The total flaw depth after stable flaw extension shall be less than or equal to 25 percent of the vessel wall thickness and the remaining ligament shall not be subject to tensile instability.

This requirement for flaw stability for Level D service loadings is satisfied as discussed in Section 5 of this report.

Tensile instability occurs only when the applied J-integral slope exceeds that of the material curve and the flaw continually grows per RG 1.161 and Appendix K of the ASME B&PV Code. As shown in Section 5 of this report for the Palisades EMA, flaws are stable with adequate margins. Therefore, crack growth or stability is not an issue.

4 EQUIVALENT MARGINS ANALYSIS INPUTS

The following material property inputs, vessel design data, and transient data were used in the EMA of the three Palisades reactor vessel materials with predicted EOLE USE values below 50 ft-lb. The two materials in the traditional beltline region, lower shell plate D-3804-1 and intermediate to lower shell circumferential weld 9-112 (Heat #27204) have EOLE USE values below 50 ft-lb. Though it can be shown by the methodology documented in WCAP-17403-NP that the extended beltline region material, upper shell plate D-3802-3, remains above 50 ft-lb at EOLE, Palisades has elected to perform the EMA on this material with consideration of the possibility of future operation at higher flux levels. Table 4-1 documents the Palisades reactor vessel geometry. Table 4-2 contains the unirradiated, EOLE 1/10T, and EOLE 1/4T USE for the three Palisades reactor vessel materials. Unit pressure load through-wall stress profiles for axial and hoop stresses were used in all the pressure SIF calculations. Design transients for all Level A and B transients were considered in addition to the 100°F/hr CD transient specified in Reference 5; see Table 4-3. ASME Code Section D material properties for yield strength and modulus of elasticity are from Reference 8.

Level A and B transients with a 100°F/hr cooldown rate, Level C transients with a 400°F/hr cooldown rate, and Level D transients with a 600°F/hr cooldown rate were used in the EMA. These transient definition points are also listed in Tables 4-5 through 4-7. Finally, cladding effects for the Levels C and D load levels were conservatively ignored because the temperatures at evaluation points are above the cladding stress free temperature of 400°F (Reference 9).

The Palisades Final Safety Analysis Report (FSAR) subsection 4.2.2 lists and describes RV design basis transients; however, further information is needed to conduct a transient stress analysis. Therefore, to accommodate this need, the Design Specification for the Replacement Reactor Vessel Closure Head (RRVCH) for Palisades Nuclear Generation Station, DS-ME-04-10, Revision 3 (Reference 10) was used to obtain temperature vs. time and pressure vs. time RV design basis transient data. Design Specification DS-ME-04-10 contains one additional transient (steam line rupture) that was not included in the original design basis transients for the reactor vessel. The steam line rupture transient is conservatively included in the EMA. Additional transients from RG 1.161 are also evaluated.

Table 4-1 Palisades RV Beltline Geometry and Design

Parameter	Value[(1)]
Base Metal Inside Diameter (Di)	172.7 in
Base Metal Inner Radius	86.35 in
Base Metal Wall Thickness (t)	8.79 in
Cladding Thickness	0.25 in[(2)]
Material Specification	SA-302 Gr. B Modified Plate
Accumulation Pressure (p_{acc})	2.750 ksi

Notes:
1. Reactor vessel beltline geometry values were obtained from WCAP-15353 – Supplement 2 – NP (Reference 11).
2. Cladding and cladding effects were conservatively ignored in the various stress analyses performed for Palisades as part of the EMA.

Table 4-2 Palisades RV Beltline Predicted Upper-Shelf Energy at 42.1 EFPY

Reactor Vessel Material[1]			Unirradiated USE[1] (ft-lb)	Projected EOLE USE[2]	
Location		**Heat Number**		**At 1/10t (ft-lb)**	**At 1/4t (ft-lb)**
LS Plate D-3804-1		C-1308-1[3]	72	46.1	48.2
US Plate D-3802-3[4]	Using CVGraph Refitted Initial USE	C-1281	62.2	47.5	50.1
	Using 95% Shear Initial USE	C-1281	59	46.6	47.5
IS to LS Circumferential Weld 9-112		27204	84	47.9	49.6

Notes:
1. Reactor vessel material information, heat numbers, and unirradiated initial USE values were taken from WCAP-17341-NP (Reference 1) for LS plate D-3804-1 and IS to LS circumferential weld 9-112 and from WCAP-17403-NP (Reference 2) for US plate D-3802-3. This information is consistent with P-PENG-ER-006 (Reference 12).
2. The projected EOLE USE values at 1/4t were taken from WCAP-17341-NP for LS Plate D-3804-1 and IS to LS circumferential weld 9-112 and from WCAP-17403-NP for US plate D-3802-3. The projected EOLE USE values at 1/10t were calculated for the EMA using the methodology described in RG 1.99, Revision 2 (Reference 13), which is equivalent to the methodology used in the previous reports.
3. The heat number for LS plate D-3804-1 has also been reported as C-1308A.
4. Using the methodology proposed in WCAP-17403-NP, it can be demonstrated that US Plate D-3802-3 meets the 50 ft-lb limit of 10 CFR 50, Appendix G. However, Palisades has elected to perform the EMA on this material due to the risk that it may fall below the 50 ft-lb limit if future operation includes higher flux levels.

Table 4-3 List of Transients Evaluated in the EMA

Number	Transient Description	Load Level
1	Plant HU at 100°F/hr	A
2	Plant CD at 100°F/hr	A
3	Plant Loading Change, 5% Full Load/Minimum	A
4	Plant Unloading Change, 5% Full Load/Minimum	A
5	Plant Load Change, 10% Full Load Step, Step Increase, T_{cold}	A
6	Plant Load Change, 10% Full Load Step, Step Decrease, T_{cold}	A
7	Plant Load Change, 10% Full Load Step, Step Increase, T_{hot}	A
8	Plant Load Change, 10% Full Load Step, Step Decrease, T_{hot}	A
9	Plant Loading Change, 15% Full Load/Min	A
10	Plant Unloading Change, 15% Full Load/Min	A
11	Loss of Primary Coolant Flow, T_{cold}	B
12	Loss of Primary Coolant Flow, T_{hot}	B
13	Reactor Trip or Loss of Load, T_{cold}	B
14	Reactor Trip or Loss of Load, T_{hot}	B

Table 4-3 List of Transients Evaluated in the EMA (cont.)

Number	Transient Description	Load Level
15	Reactor Trip, Loss of Load, or Loss of Primary Coolant Flow, $T_{surgeflow}$	B
16	Safety Valve Operation, T_{inlet}	B
17	Safety Valve Operation, T_{outlet}	B
18	Steam Line Rupture[(1)]	D
19	RG 1.161 Cooldown at 100°F/hr	B
20	RG 1.161 Cooldown at 400°F/hr	C
21	RG 1.161 Cooldown at 600°F/hr	D

Notes:
1. This design transient is conservatively bounded by the Regulatory Guide (Reference 7) transient for Level D loads.

Table 4-4 Fracture Toughness Margin Factors from Reference 7

Metal	Levels A, B, and C	Level D
Base	0.749	1
Welds	0.629	1

Table 4-5 Level A and B 100°F/hr Cooldown Transient

Time (sec)	Pressure (ksi)	Fluid T_{fluid} (°F)
0	2.75	533
2,800	2.75	456
3,600	2.75	433
5,400	2.75	383
7,200	2.75	333
9,000	2.75	283
10,800	2.75	233

Table 4-6 Level C 400°F/hr Cooldown Transient		
Time (sec)	**Pressure (ksi)**	**Fluid T_{fluid} (°F)**
0	2.25	533
1,197	1.3	400

Table 4-7 Level D 600°F/hr Cooldown Transient		
Time (sec)	**Pressure (ksi)**	**Fluid T_{fluid} (°F)**
0	2.25	533
798	1.3	400

5 EQUIVALENT MARGINS ANALYSIS EVALUATIONS

5.1 APPLIED J-INTEGRAL CALCULATIONS

For the Level A and B service load conditions, the Palisades EMA has considered a total of 17 design transients from the Palisades FSAR, along with the 100°F/hr, 400°F/hr and 600°F/hr cooldown rate transients provided in Reference 7. The typical through-wall thermal stress, shown in Figure 5-1, was computed analytically at inside surface, mid-wall, and outside surface locations. Typical axial through-wall stress distributions for the vessel during a heatup transient, shown in Figure 5-2, were used in this EMA. The associated vessel wall metal temperatures, required for the applied J-integral evaluation and the material fracture toughness resistance (J-R), were also used.

The applied J-integral values for the circumferential flaws for all Level A and B service level conditions are shown in Figure 5-1. These figures show the peak J-integral values during each transient as a function of crack extension starting from the 1/4-thickness flaw. These calculations used a structural margin of 1.25 for pressure loading and 1 for thermal loading, as required by Reference 5.

Figure 5-2 shows the applied J-integral values at 1/10-thickness flaws, with a structural margin of 1 for pressure and thermal loadings, for circumferential flaws under Level C and D conditions.

All applied J-integral values shown in Figure 5-1 and Figure 5-2 are applicable for both the weld and base metals because flaws are considered circumferential. Only circumferential base metal flaws are considered in this analysis, because only the "weak" orientation USE is projected to drop below 50 ft-lbs as described below.

The measured initial USE value for the Palisades Nuclear Power Plant LS plate D3804-1 is 110 ft-lb in the longitudinal direction. Similarly, US plate D3802-3 has an initial USE value of 91 ft-lb in the longitudinal direction per P-PENG-ER-006 (Reference 12). The estimated transverse values for the LS and US plates are 72 and 59 ft-lb, respectively, which were reduced by 35 percent to approximate the transverse direction per NUREG-0800, Revision 2 Branch Technical Position MTEB 5-3 (Reference 14). Table 5-1 documents the calculation of the end-of-license-extension (EOLE) USE with consideration of the Charpy testing direction for Palisades. Data were obtained from WCAP-17341-NP and WCAP-17403-NP for the LS and US plates, respectively.

The table shows that for the longitudinal "strong" direction, both plates exhibit an EOLE USE value per 10 CFR 50, Appendix G above 50 ft-lb. When the initial longitudinal USE value is reduced to 65 percent per MTEB 5-3 to approximate the transverse "weak" direction, both plates drop below the 50 ft-lb limit. Therefore, only circumferential flaws are postulated in the two plates, because the EOLE USE in the longitudinal "strong" direction is above the 10 CFR 50, Appendix G limit. As stated previously in Section 2.2, the CVN value should be matched to the proper orientation of the plate material. Therefore, for axial flaws, the CVN value for the lateral transverse (L-T) "strong" orientation in the vessel wall will be used. Similarly, for circumferential flaws, the CVN value for the transverse-lateral (T-L) "weak" orientation will be used. Therefore, only circumferential base metal flaws are considered in this analysis.

The applied J-integral values shown in Figure 5-1 and Figure 5-2 are used in the flaw evaluations. Table 5-2 summarizes the maximum circumferential applied J-integrals for all design, 100°F/hr, 400°F/hr, and 600°F/hr transients.

5.2 MATERIAL FRACTURE TOUGHNESS PROPERTIES

The estimated base material fracture toughness properties for the 1/4- and 1/10-thickness locations are shown in Figure 5-3 and Figure 5-4, respectively using the high-toughness / low-sulfur model from RG 1.161. The corresponding material toughness values for the weld material are shown in Figures 5-6 and 5-7, also using the model from RG 1.161. These figures show the toughness values at different metal temperatures ranging from 300°F to 600°F, over which the vessel wall metal temperatures vary during the transients. These include the USE levels considered for the materials at the flaw location and a flaw extension of up to 1 inch.

Per P-PENG-ER-006, the sulfur content of US plate D-3802-3 is 0.029 wt. %. Similarly, for LS plate D-3804-1, the sulfur content is 0.024 wt. %. The Palisades plates have a sulfur content greater than the high-toughness model limit of 0.018 wt. % specified in RG 1.161. The J-R model in RG 1.161 has an upper limit in sulfur because J-R data for plates with high sulfur content are scarce and the available data showed low toughness, flat J-R curves, and a size effect. The most data available for a high-sulfur A-302 B plate are for the V-50 plate in NUREG/CR-5265 (Reference 15). This plate has a reported sulfur content of 0.021 and 0.025 wt. % with USE values of 44 to 51 ft-lb, averaging around 48 ft-lb at the 1/4T locations in the T-L (weak) orientation. This USE is comparable to the EOLE projection for the Palisades high-sulfur plates.

The V-50 plate was unusual in that it had a test specimen size effect that has not been observed in other RV material J-R curves and is unique to the V-50 plate. A high content of manganese-sulfide (MnS) inclusions and banded regions of microstructure, are believed to be the causes of the unusual specimen size effect observed. Conservatively, the lowest J-R curve test data from this testing program is plotted in Figure 5-5, which is from a 6T size specimen and is considerably lower than test data for the 1T J-R, which is the standard size specimen typically used. In addition, the manufacturing practices used to produce this extremely low-toughness V-50 plate are not representative of those used in the Palisades RV. The V-50 plate is A-302 B plate with a nickel content of 0.23 wt. % while the Palisades plates are SA-302B Modified, which means that they have at least 0.4 wt. % nickel. Nickel was added to increase toughness. Therefore, the J-R curve test data from the V-50 plate data can be conservatively viewed as the worst possible case and can be compared to the J-applied values from this evaluation. Adjusting the 180°F 6T plate V-50 J-R curve data to 600°F using the ratio of the RG 1.161 correlation, the 600°F data can be approximated as shown in Figure 5-5.

High-sulfur A-302 B Modified plate J-R data are available in NUREG/CR-6426 (Reference 16). However, the weak-direction Charpy USE value is 64 ft-lb, which is above the 10 CFR 50, Appendix G limit of 50 ft-lb. This further validates that the V-50 plate was an anomaly and can be considered a very conservative lower bound of the available high-sulfur A-302 B plate J-R data. The J-applied in the Palisades SA-302 B Modified plate remains below the measured very conservative lower-bound V-50 A-302 B plate J-R data.

5.3 FLAW EVALUATION RESULTS

The flaw stabilities for various material, flaw location, and service load levels are shown in Figure 5-8 through Figure 5-13. Figure 5-8 shows the applied J-integral and material J-resistance for circumferential flaws in base metal at the 1/4-thickness location for Level A and B design transients. The corresponding results for the NRC Regulatory Guide 100°F/hr cooldown transient are shown in Figure 5-9. The J-applied remains below even the conservative temperature-adjusted V-50 plate J-R data. Figure 5-8 compares the J-R data using the high-toughness, low-sulfur model of RG 1.161, along with the measured V-50 plate J-R data, to the J-applied calculated in this analysis.

For the weld metal, Figure 5-10 and Figure 5-11 show the applied J-integral and material J-resistance results for the Level A and B transients for the circumferential flaws, and the corresponding NRC Regulatory Guide 100°F/hr cooldown transient.

For Level C and D loads, circumferential flaw versus crack extension results are shown Figure 5-12 for base metal and in Figure 5-13 for the weld metal. The J-applied curves for Levels C and D are essentially flat, indicating very small flaw extension. The J-applied curves are also well below the J_R curves, indicating stable flaw extension.

Table 5-2 lists the minimum material fracture toughness J-resistance as calculated per RG 1.161 at a peak metal temperature of 610°F, which is observed at the 1/4-thickness locations. All transients that have applied J-integral values with the crack tip at significantly lower temperatures than 610°F are well below the J-resistance listed, indicating that the EMA criteria are met.

Maximum available equivalent margins were computed for the Level A and B governing transient with 100°F/hr cooldown rate at accumulation pressure levels by iteration. The maximum structural margin factors that result in the J-applied values equal to the material J-resistance at 0.1-inch crack extension as calculated per RG 1.161 are listed in Table 5-3. This evaluation indicates that the minimum structural margin available for the base material is 2.874 (with circumferential flaws). For the weld material with the circumferential flaws, the minimum structural margin available is 2.490. All these cases have their structural factors well above the minimum requirement of 1.15 (Reference 5).

The flaw extension figures demonstrate that:

- The NRC Regulatory Guide 100°F/hr cooldown transient with the accumulation pressure levels governs the Level A and B transients.
- All cases considered are acceptable with the applied J-integral values at 0.1-inch crack extensions below the material J-resistance ($J_{0.1}$) required by Reference 5.

Table 5-1 Palisades US and LS Plate EOLE USE Calculation with Consideration of Charpy Test Specimen Orientation

Charpy Orientation	Reactor Vessel Material	Wt.% Cu	1/4T EOLE Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE (ft-lb)	Projected USE Decrease (%)	Projected EOLE USE (ft-lb)
Longitudinal(1)	US Plate D-3802-3	0.25	0.0902	91(1)	19.5	73
	LS Plate D-3804-1	0.19	2.024	110(1)	33	74
Transverse(2)	US Plate D-3802-3	0.25	0.0902	59	19.5	47.5
	LS Plate D-3804-1	0.19	2.024	72	33	48.2

Notes:
1. Measured longitudinal-direction initial USE values from P-PENG-ER-006. All other data are taken from WCAP-17341-NP and WCAP-17403-NP for the LS and US plates, respectively.
2. Data are taken from WCAP-17341-NP and WCAP-17403-NP for the LS and US plates, respectively, and summarized in Table 4-2.

Table 5-2 Applied J-Integral and Material J-Resistance at 0.1-Inch Crack Extension for All Transients

Number	Transient Description	Load Level	a/t	Circumferential $J_{applied}$ (in-lb/in^2)	Base J_R (in-lb/in^2)	Weld J_R (in-lb/in^2)
1	Plant HU at 100°F/hr	A	1/4	49.1	601	462
2	Plant CD at 100°F/hr	A		106.4		
3	Plant Loading Change, 5% Full Load/Minimum	A		47.9		
4	Plant Unloading Change, 5% Full Load/Minimum	A		104.7		
5	Plant Load Change, 10% Full Load Step, Step Increase, T_{cold}	A		55.2		
6	Plant Load Change, 10% Full Load Step, Step Decrease, T_{cold}	A		49.5		
7	Plant Load Change, 10% Full Load Step, Step Increase, T_{hot}	A		53.3		
8	Plant Load Change, 10% Full Load Step, Step Decrease, T_{hot}	A		52.5		
9	Plant Loading Change, 15% Full Load/Min	A		48.2		
10	Plant Unloading Change, 15% Full Load/Min	A		68.1		
11	Loss of Primary Coolant Flow, T_{cold}	B		53.5		
12	Loss of Primary Coolant Flow, T_{hot}	B		90.3		
13	Reactor Trip or Loss of Load, T_{cold}	B		48.1		
14	Reactor Trip or Loss of Load, T_{hot}	B		90.1		
15	Reactor Trip, Loss of Load, or Loss of Primary Coolant Flow, $T_{surgeflow}$	B		86.5		
16	Safety Valve Operation, T_{inlet}	B		69.7		
17	Safety Valve Operation, T_{outlet}	B		96.6		
19	RG 1.161 Cooldown at 100°F/hr	B		181.5		
20	RG 1.161 Cooldown at 400°F/hr	C	1/10	163.8	783	708
21	RG 1.161 Cooldown at 600°F/hr	D		304.8		

Table 5-3 Available Margins on Pressure Load for Level A and B 100°F/hr Cooldown Transient

Time (sec)	Base Material			Weld Material		
	Circumferential Flaw		$J_{0.1}$ Material (in-lb/in^2)	Circumferential Flaw		$J_{0.1}$ Material (in-lb/in^2)
	SF	J-applied (in-lb/in^2)		SF	J-applied (in-lb/in^2)	
0	3.106	699	699	2.760	527	528
2,800	2.874	776	776	2.490	578	580
3,600	2.882	807	807	2.490	600	601
5,400	2.963	885	885	2.549	653	653
7,200	3.102	975	975	2.659	711	711
9,000	3.272	1,076	1,076	2.797	776	777
10,800	31.69	1,188	1,188	27.05	847	848
Minimum SF	2.874			2.490		

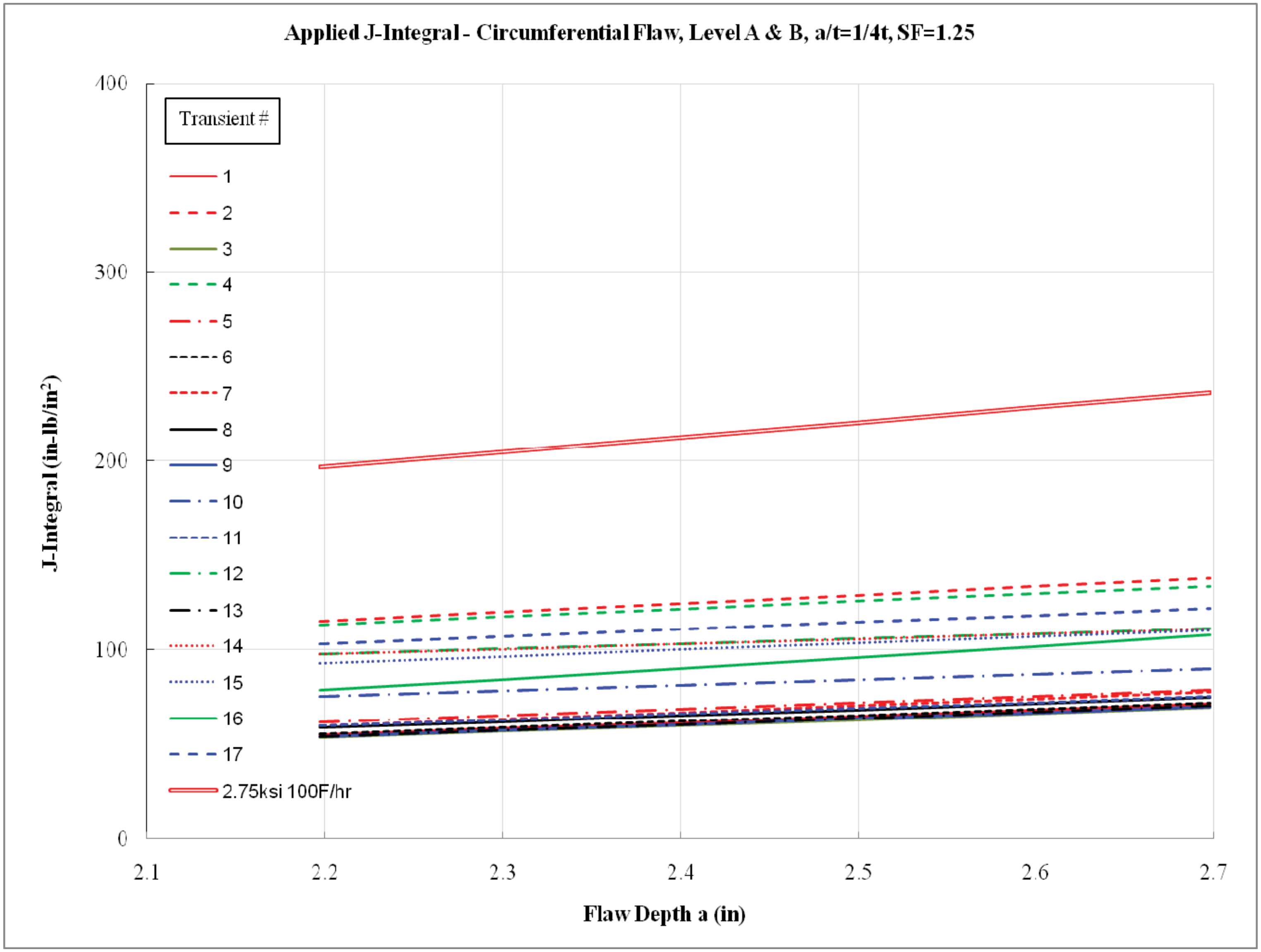

Figure 5-1 Applied J-Integral versus Crack Extension for Circumferential Flaw – 1/4t, Level A and B

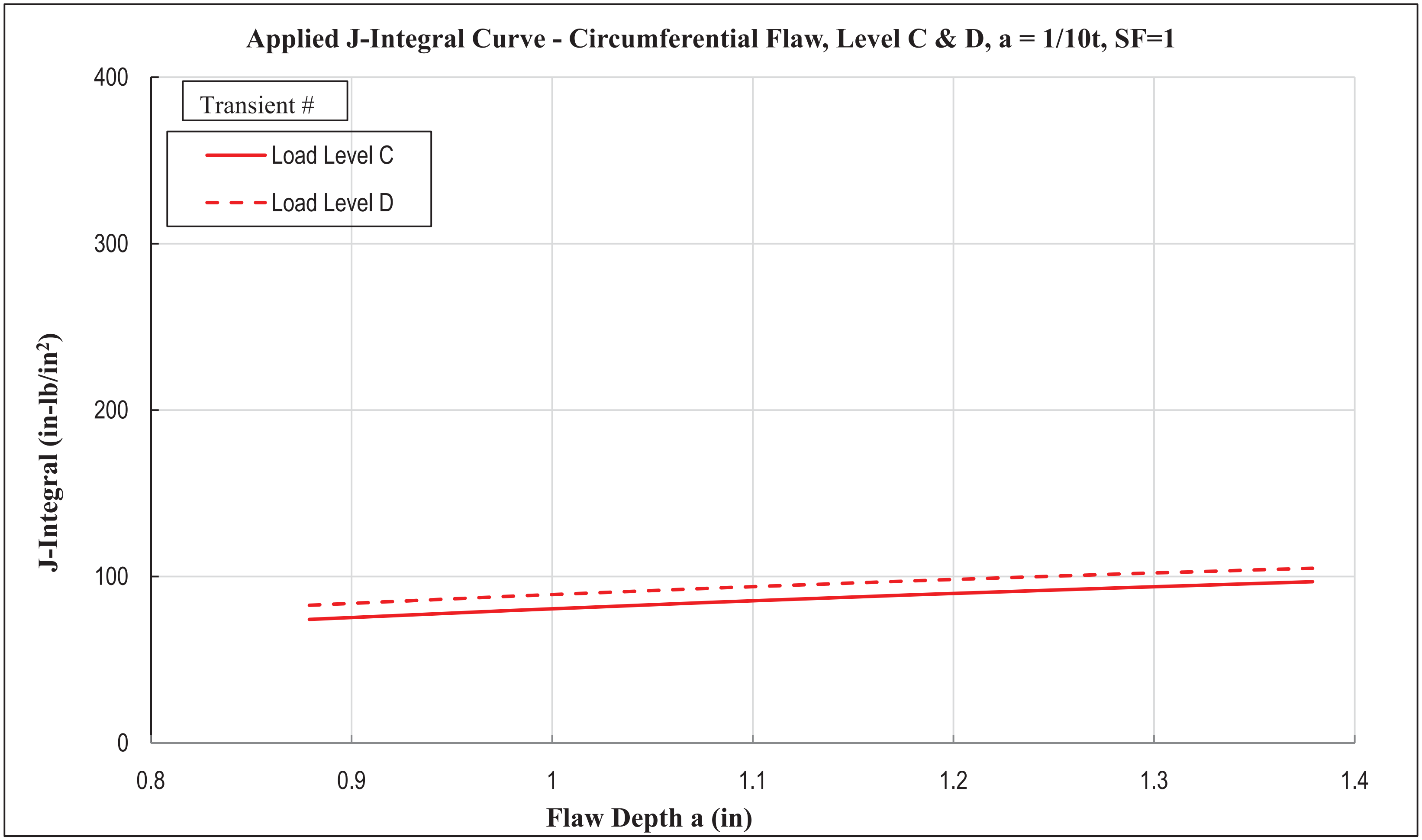

Figure 5-2 Applied J-Integral versus Crack Extension for Circumferential 1/10t Flaw, Levels C and D

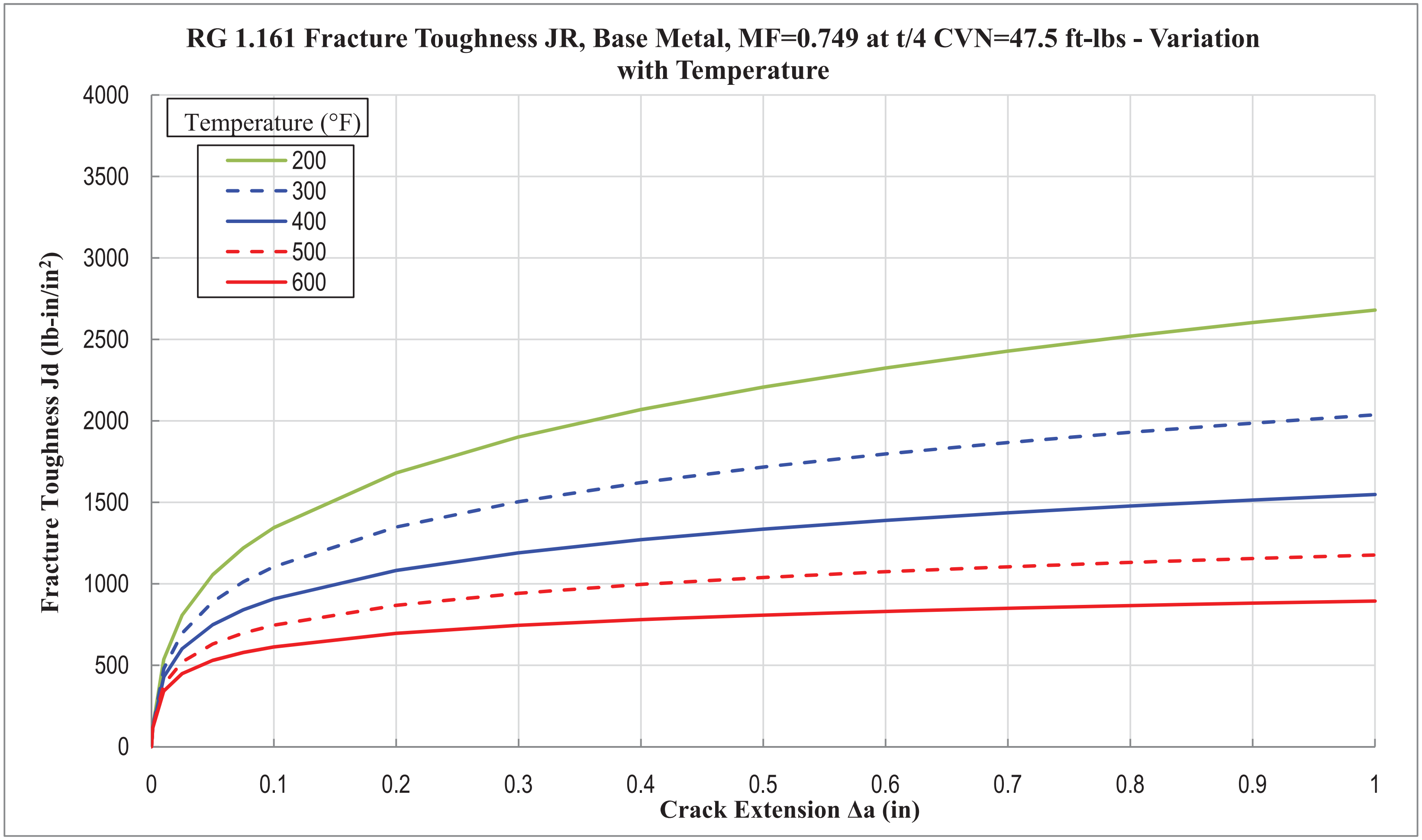

Figure 5-3 Base Metal Fracture Toughness at t/4 CVN = 47.5 ft-lb – Variation with Temperature

Note: JR = J-resistance, CVN = Charpy V-notch

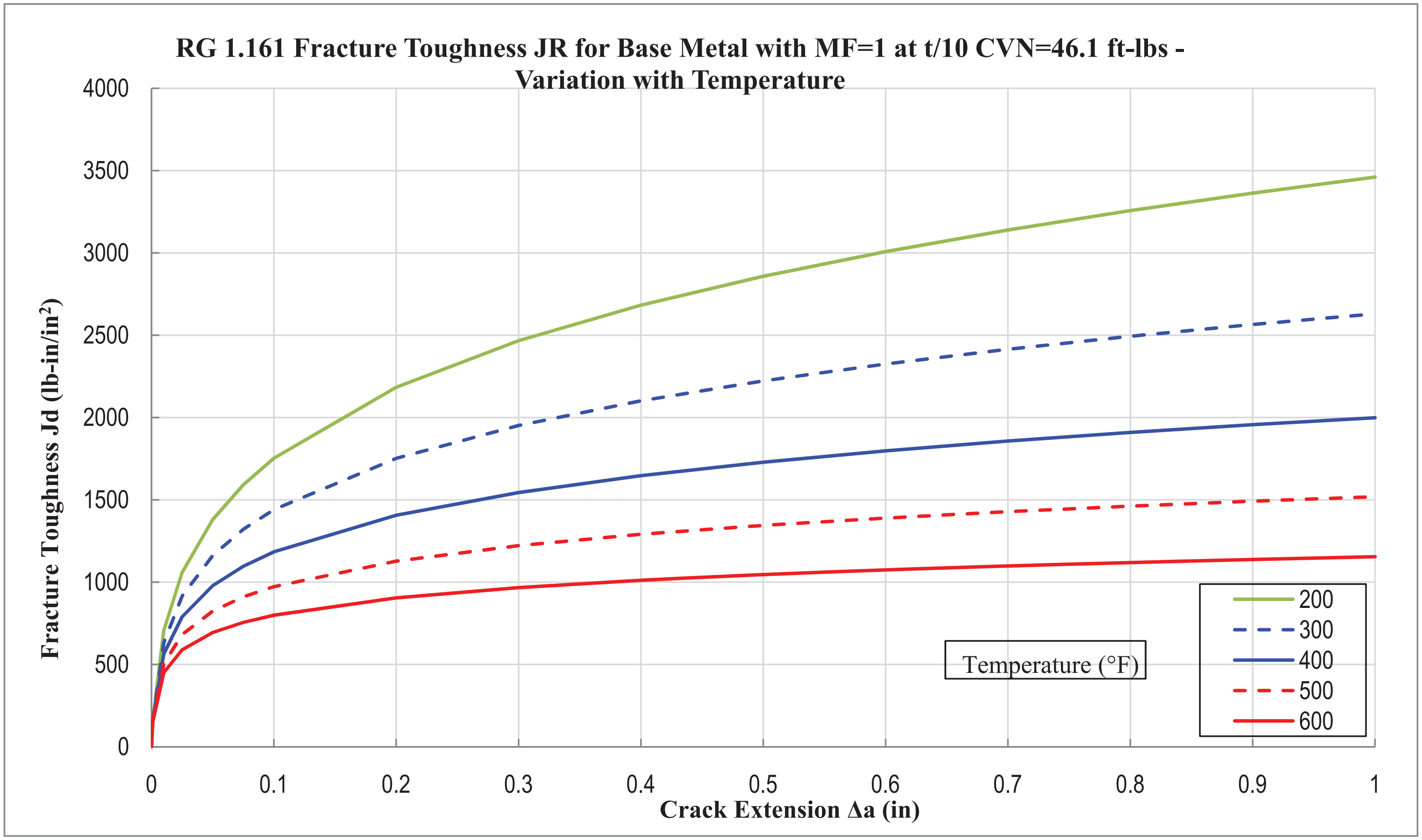

Figure 5-4 Base Metal Fracture Toughness at t/10 CVN = 46.1 ft-lb – Variation with Temperature

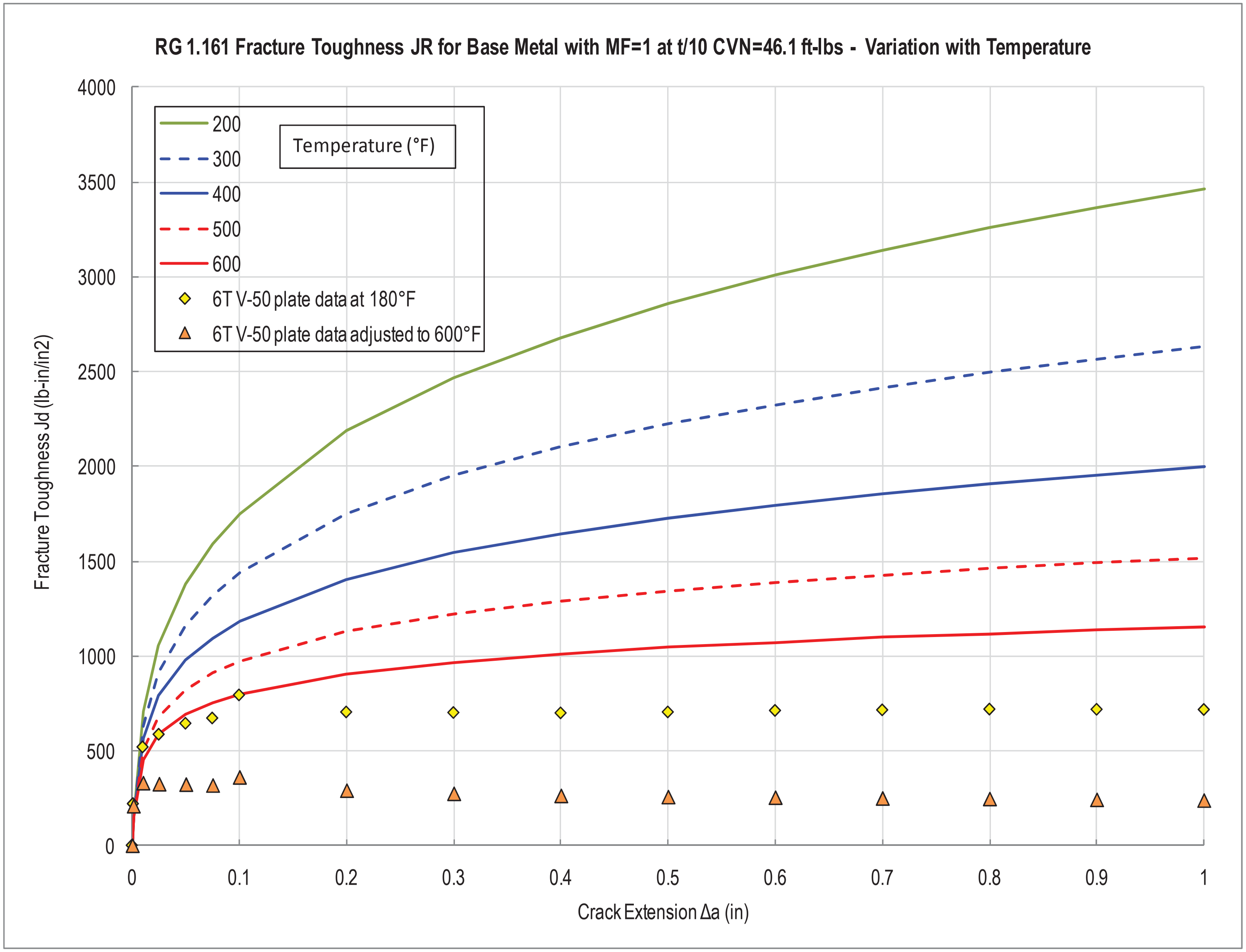

Figure 5-5 Base Metal Fracture Toughness at t/10 CVN = 46.1 ft-lb vs. Measured High-Sulfur V-50 Plate Data – Variation with Temperature

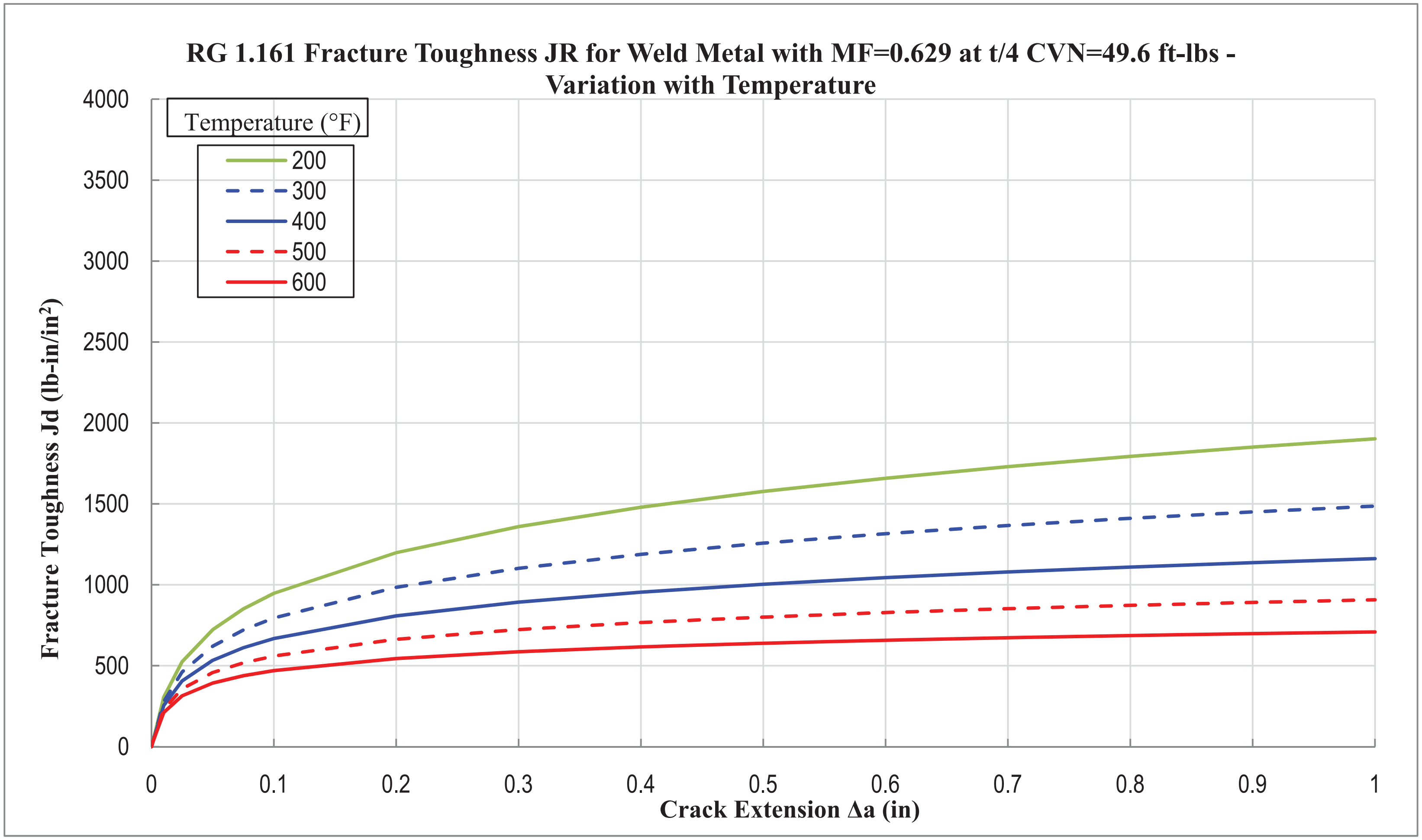

Figure 5-6 Weld Metal Fracture Toughness at t/4 CVN = 49.6 ft-lb – Variation with Temperature

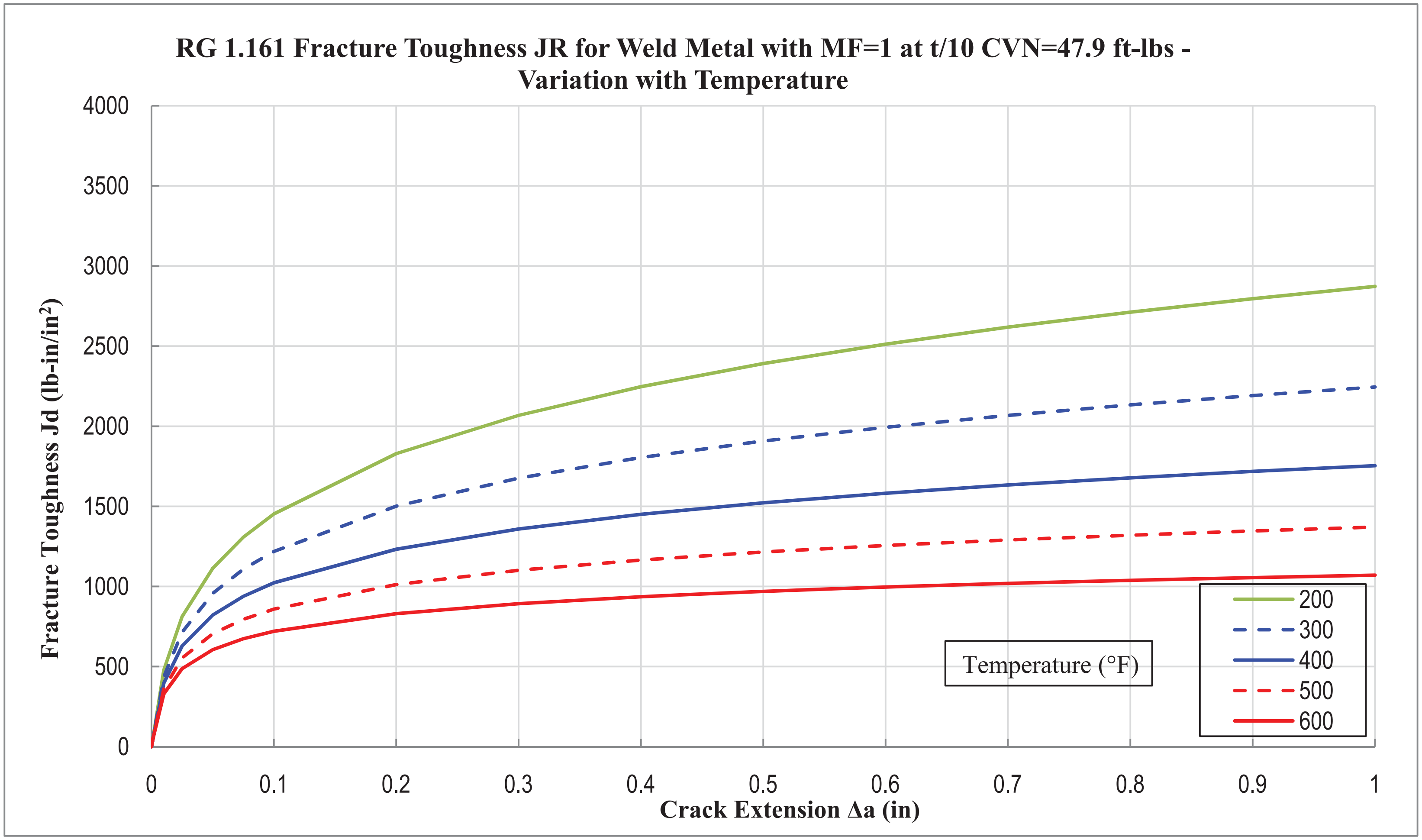

Figure 5-7 Weld Metal Fracture Toughness at t/10 CVN = 47.9 ft-lb – Variation with Temperature

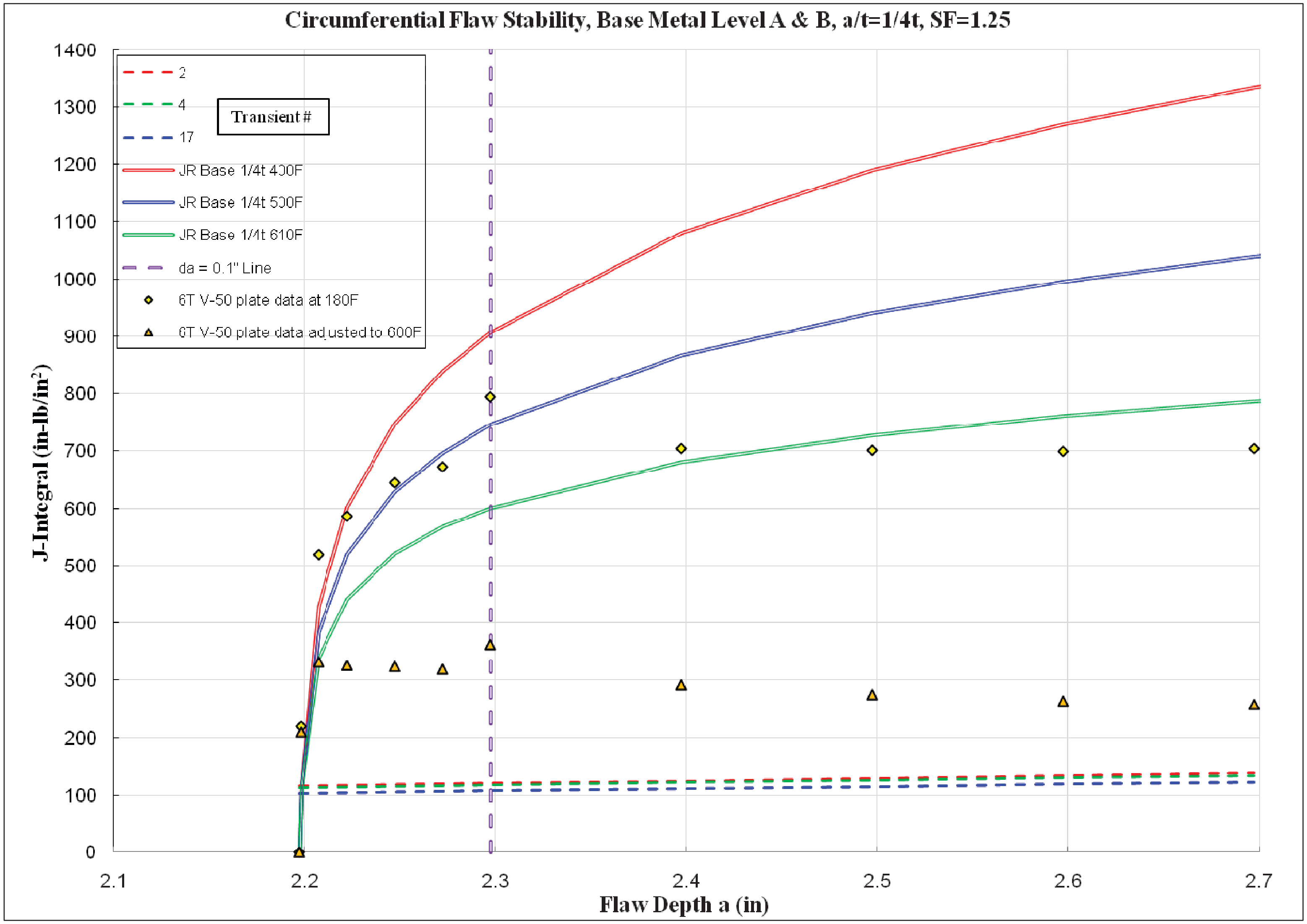

Figure 5-8 Circumferential Flaw J-Integral versus Crack Extension – t/4, Level A and B, Base Material with Comparison of the Measured High-Sulfur V-50 Plate Data

Note: The limiting transients 2, 4, and 17 are shown.

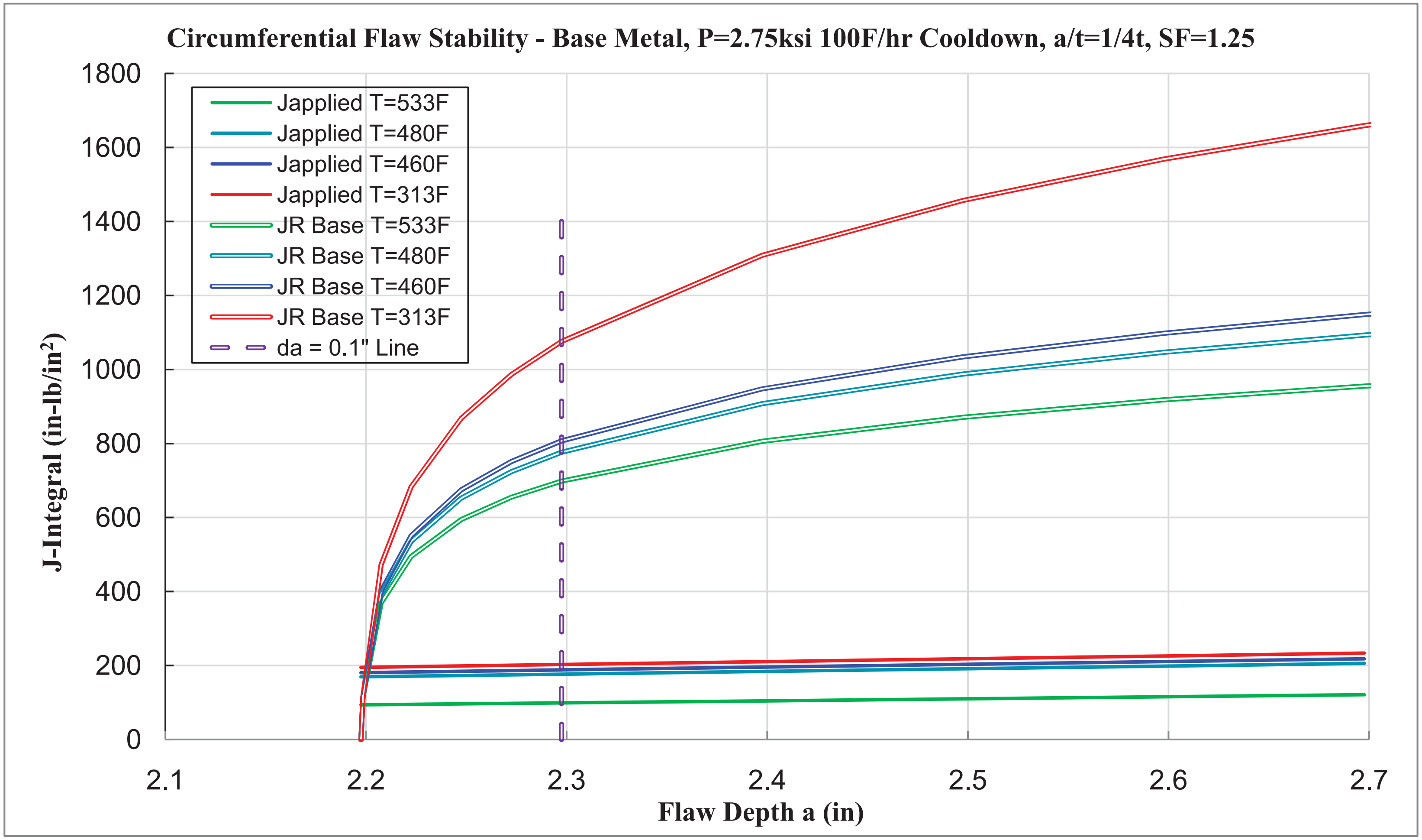

Figure 5-9 Circumferential Flaw J-Integral versus Crack Extension – t/4, P=2.75 ksi 100°F/hr Cooldown, Base Metal

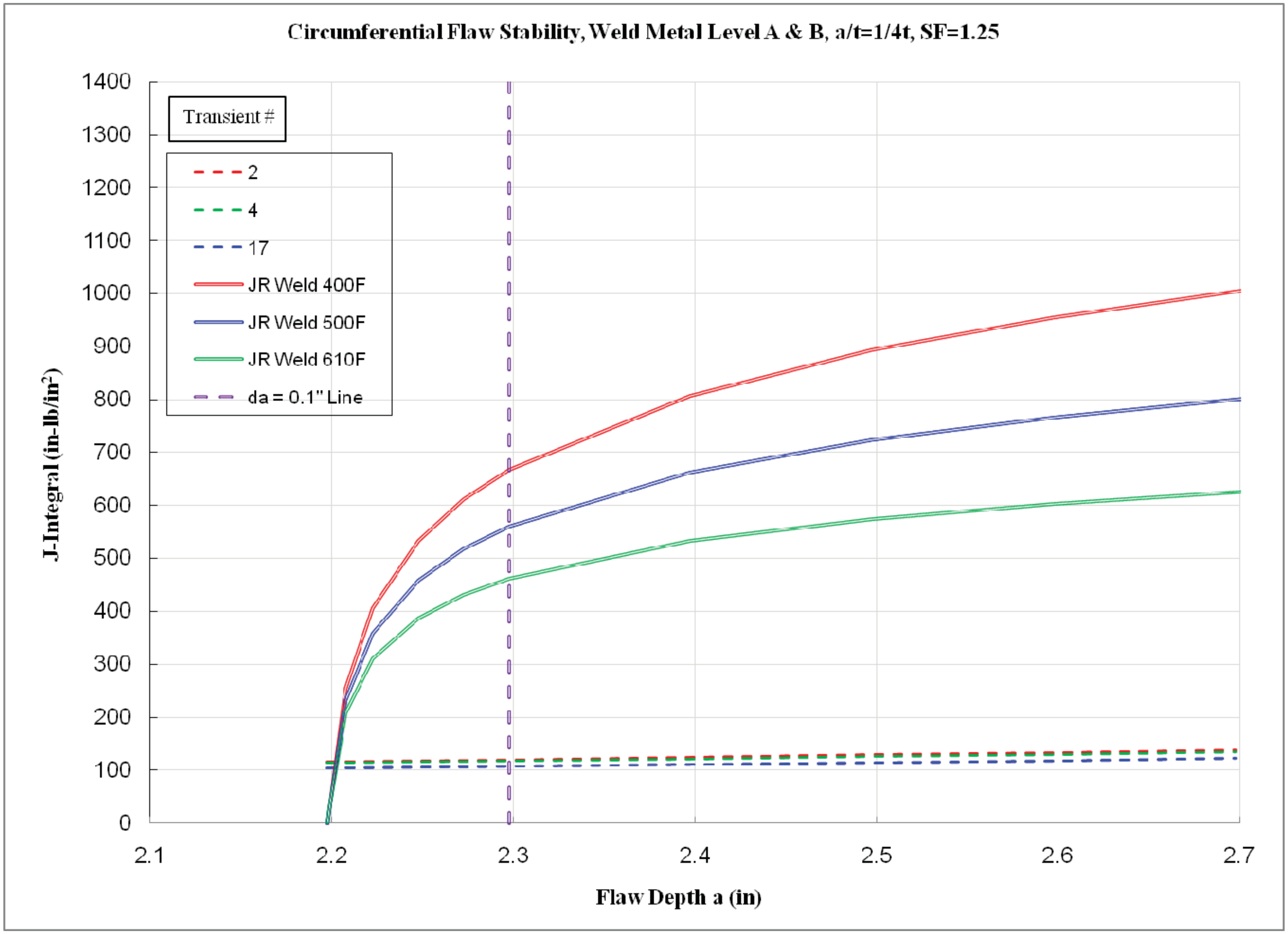

Figure 5-10 Circumferential Flaw J-Integral versus Crack Extension – t/4, Level A and B, Weld Material

Note: The limiting transients 2, 4, and 17 are shown.

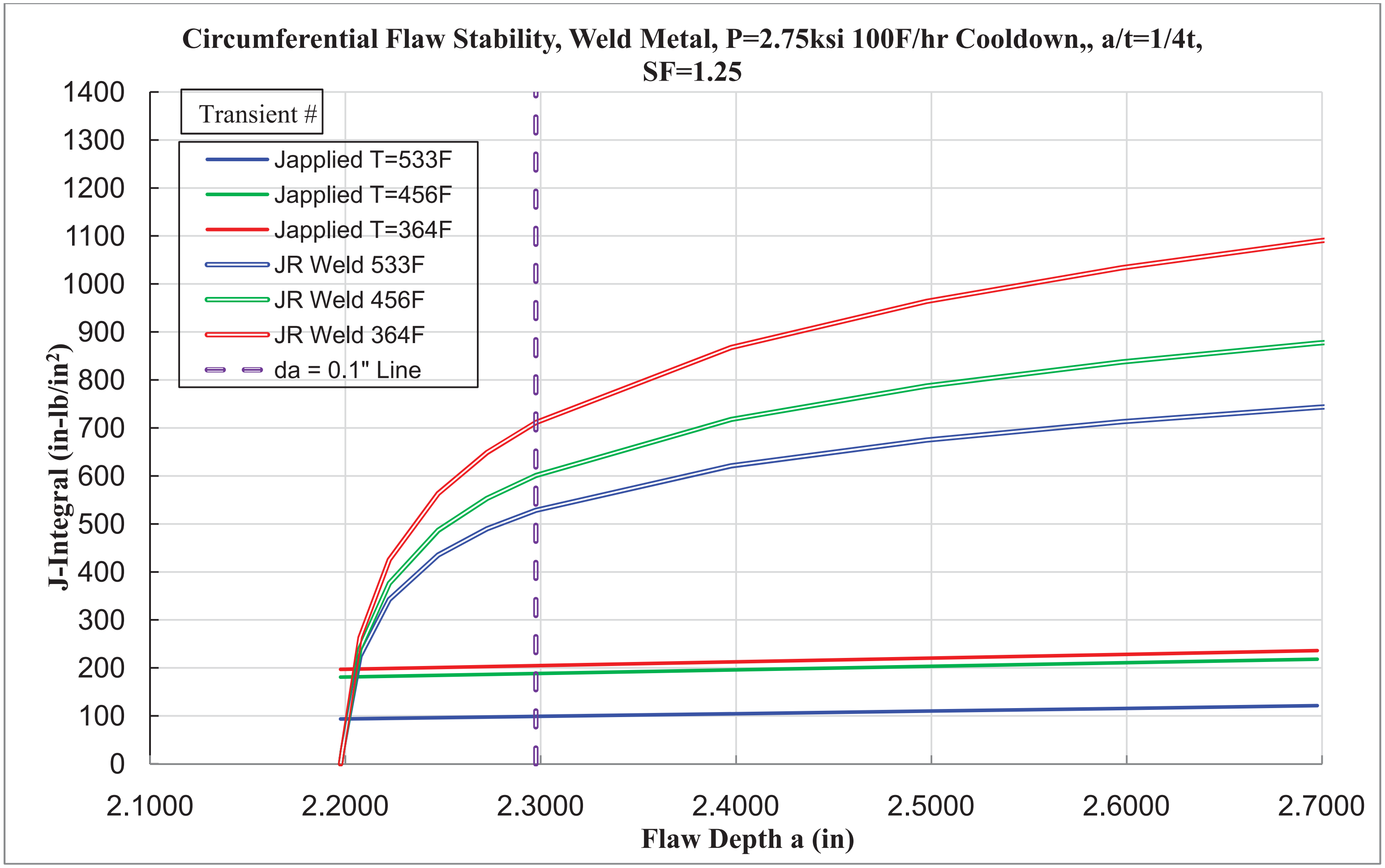

Figure 5-11 Circumferential Flaw J-Integral versus Crack Extension – t/4, P=2.75 ksi 100°F/hr Cooldown, Weld Metal

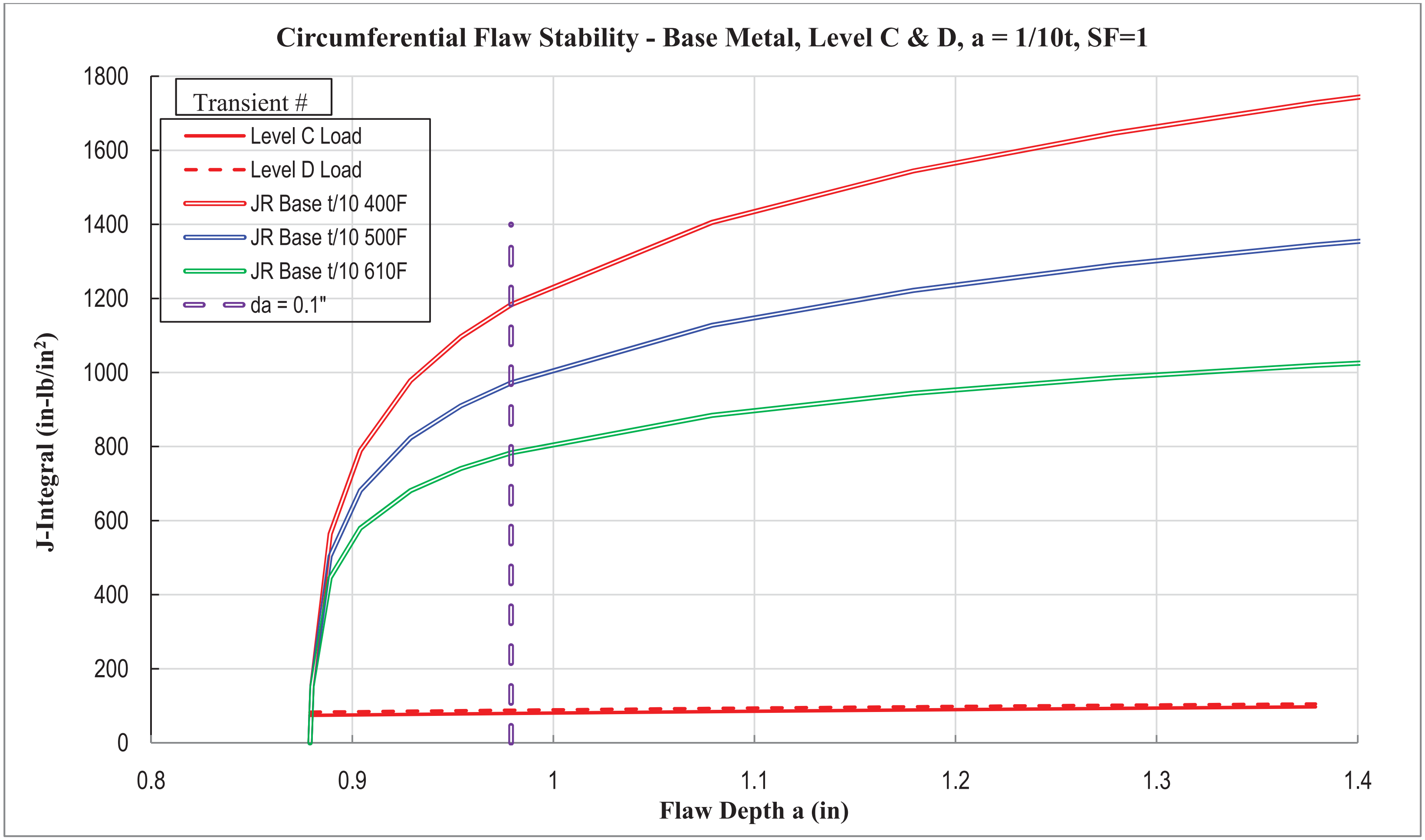

Figure 5-12 Circumferential Flaw J-Integral versus Crack Extension – t/10, Levels C and D, Base Metal

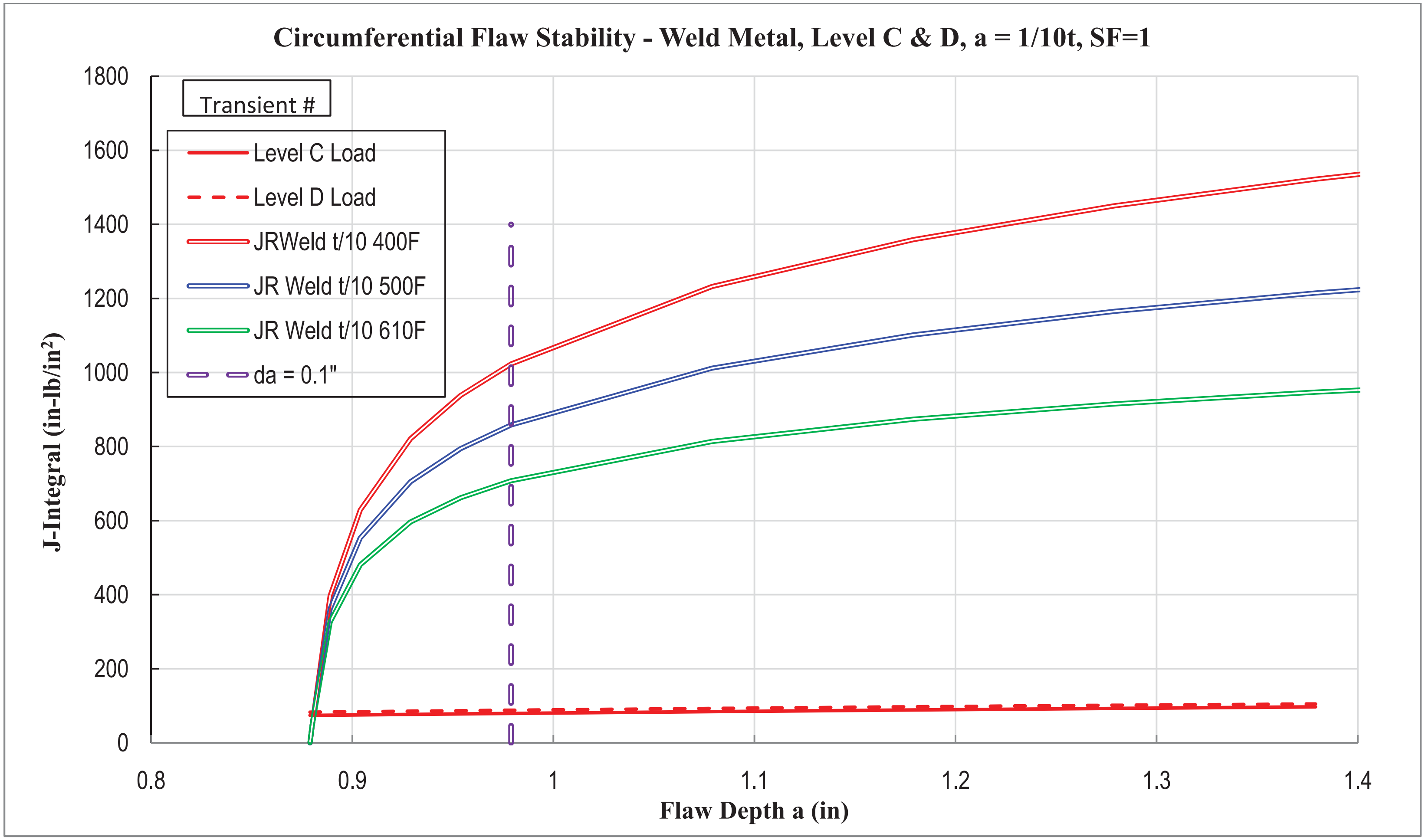

Figure 5-13 Circumferential Flaw J-Integral versus Crack Extension – t/10, Levels C and D Loads, Weld Metal

6 CONCLUSIONS

The Palisades reactor vessel beltline and extended beltline regions with predicted Charpy upper-shelf energy levels falling below 50 ft-lb at the EOLE period were evaluated for equivalent margins of safety per the ASME Code Section XI (References 5, 6, and 18) and found to be acceptable. The minimum structural margin available for the limiting reactor vessel material (intermediate to lower shell circumferential weld 9-112 [Heat #27204]) of 2.490 (circumferential flaws) occurs during a Service Level A and B transient using the toughness model of RG 1.161. The equivalent margins analyses for the plate materials, lower shell plate D-3804-1 and upper shell plate D-3802-3, are bounded by the conservative test data reported in NUREG/CR-5265. Use of the conservative V-50 plate data from NUREG/CR-5265 for the Palisades plate materials with sulfur content greater than the 0.018 wt. % limit specified in Regulatory Guide 1.161 shows that the applied J-integral values are acceptable.

Palisades Plant-Specific EMA Comparison to CE-NPSD-993

CE-NPSD-993, Revision 0 (Reference 17) is a generic Combustion Engineering (CE) EMA that was completed in May 1995 for the Combustion Engineering Owners Group (CEOG). This report summarized that all CE reactor vessel materials with end-of-life USE values below 50 ft-lb would satisfy equivalent margins of safety to ASME Code Section III, Appendix G with consideration of the generic design transients and reactor vessel geometries assumed in that report. Therefore, based on the results of this plant-specific EMA, the results of the CEOG report with consideration of the design transients and vessel geometries for Palisades are confirmed.

Per CE-NPSD-993, the minimum acceptable USE value for plate materials in the longitudinal direction was 30 ft-lb for Level A and B transients. Likewise, for the transverse direction, a value of 19.5 ft-lb was concluded to be the acceptable value for plate materials. On a generic basis, weld materials need to exhibit at least 34 ft-lb for longitudinal welds and 19.5 ft-lb for circumferential welds to provide equivalent margins of safety for Level A and B transients. The minimum acceptable USE values for Level C and D transients were generically determined to be 30 and 19.5 ft-lb for plate materials in the longitudinal and transverse directions, respectively. Weld materials need to exhibit at least 30 ft-lb for longitudinal and circumferential welds to provide equivalent margins of safety for Level C and D transients. For Palisades, the predicted USE in the transverse “weak” direction at EOLE was 47.5 ft-lb for US Plate D-3802-3. The predicted USE does not drop below 50 ft-lb for the longitudinal “strong” direction plate data at EOLE. The predicted USE for intermediate to lower shell circumferential weld 9-112 (Heat #27204) material at EOLE was 49.6 ft-lb.

Service Level A and B Transients

- Intermediate to lower shell circumferential weld 9-112 (Heat #27204) is governing for EOLE USE margin at the 1/4-thickness location for normal Level A and B load conditions, based on the Regulatory Guide 1.161 fracture toughness methodology.

- The applied J-integral values for the assumed 1/4-thickness inside-surface circumferential flaws in the base metal and circumferential flaws in the weld metal with a safety margin of 1.15 on pressure loading are within the material fracture toughness J-resistance at 0.1-inch crack extension.

- The assumed flaw is ductile and stable with crack extension with a safety margin of 1.25 on pressure loading.

- The equivalent margins analyses for the plate materials are acceptable and bounded by the conservative test data reported in NUREG/CR-5265 in all cases for Service Level A and B transients.

Service Level C Condition with 400°F/hr Cooldown Transient

- Intermediate to lower shell circumferential weld 9-112 (Heat #27204) is governing for EOLE USE margin at the 1/10-thickness location for the Service Level C load condition, based on the Regulatory Guide 1.161 fracture toughness methodology.

- The applied J-integral values for the assumed 1/10 base metal thickness inside-surface circumferential flaws in the base metal and circumferential flaws in the weld metal with a safety margin of 1.00 on loading are within the material fracture toughness J-resistance at 0.1-inch crack extension.

- The assumed flaw is ductile and stable with crack extension with a safety margin of 1 on pressure loading.

- The equivalent margins analyses for the plate materials are acceptable and bounded by the conservative test data reported in NUREG/CR-5265 in all cases for the Service Level C transient.

Service Level D Condition with 600°F/hr Cooldown Transient

- Intermediate to lower shell circumferential weld 9-112 (Heat #27204) is governing for EOLE USE margin at the 1/10-thickness location for the Service Level D load condition, based on the Regulatory Guide 1.161 fracture toughness methodology.

- The applied J-integral values for the assumed 1/10 base metal thickness inside-surface circumferential flaws in the base metal and circumferential flaws in the weld metal with a safety margin of 1.00 on loading are within the material fracture toughness J-resistance at 0.1-inch crack extension.

- The total flaw depth after a stable flaw extension is well within 75 percent of the vessel wall thickness, with the remaining ligament stable for crack propagation.

- The equivalent margins analyses for the plate materials are acceptable and bounded by the conservative test data reported in NUREG/CR-5265 in all cases for the Service Level D transient.

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