Entergy

Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043
Tel 269 764 2000

Paula K. Anderson
Licensing Manager

June 2, 2010

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

SUBJECT: Documentation for Pressurized Thermal Shock Evaluation Meeting

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

Dear Sir or Madam:

On February 11, 2010, Entergy Nuclear Operations, Inc. (ENO) and the Nuclear Regulatory Commission (NRC) had a conference call to discuss ENO plans to revise the Palisades Nuclear Plant (PNP) pressurized thermal shock (PTS) analysis. During the conference call, a future meeting at the NRC was proposed to discuss in more detail ENO plans to revise the PTS analysis. Subsequently, a meeting was scheduled between ENO and the NRC on June 22, 2010.

The attached documentation is provided in support of the meeting. Attached you will find (1) a PNP reactor vessel fluence evaluation, (2) a preliminary PTS screening criteria assessment, (3) a documented review of the PTS screening criteria assessment, (4) a list of ENO representatives planning to attend the meeting, and (5) a meeting agenda. The attachments contain no proprietary information.

This letter contains no new commitments and no revisions to existing commitments.

Sincerely,

pka/jse

Attachment(s):
1. Background Information
2. Reactor Pressure Vessel Fluence Evaluation
3. Preliminary PTS Screening Criteria Assessment
4. Westinghouse Review of PTS Screening Criteria Assessment
5. List of ENO Representatives Planning to Attend Meeting
6. Planned Meeting Agenda

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

ATTACHMENT 1

Background Information

3 pages follow

Background Information

On March 22, 2005 (ADAMS accession no. ML050940446), Nuclear Management Company, LLC (former plant operator) submitted a license renewal application (LRA) to the Nuclear Regulatory Commission (NRC) to renew the operating license for the Palisades Nuclear Plant (PNP). The LRA included discussion of a time limiting aging analysis (TLAA) for pressurized thermal shock (PTS) of the PNP reactor pressure vessel (RPV).

NRC issued a safety evaluation report (SER) related to the license renewal of the PNP on September 28, 2006 (ML062710068). In January 2007, NRC issued NUREG-1871, "Safety Evaluation Report Related to the License Renewal of Palisades Nuclear Plant" (ML070600578).

In the SER, it is noted that the RPV is projected to exceed the PTS screening criteria in 2014. The axial welds at the 60° locations, fabricated with weld wires of heat W5214, are the limiting materials in the beltline region of the PNP RPV.

The SER states that if an RPV is projected to exceed the PTS screening criteria, 10 CFR 50.61(b)(3) requires flux reduction be implemented reasonably practicable to avoid exceeding the PTS screening criteria. If the flux reduction program does not prevent the RPV from exceeding the PTS screening criteria before the end of the operating license, the applicant, in order to meet PTS requirements, can choose in 10 CFR 50.61 between two options:

1. Submittal of a plant-specific safety analysis, pursuant to 10 CFR 50.61 (b)(4), to determine what, if any, modifications to equipment, systems, and plant operation are necessary to prevent failure of the RPV from a postulated PTS event. This analysis must be submitted at least three years before RT_{PTS} is projected to exceed the PTS screening criteria.

2. Perform a thermal annealing treatment of the RPV, pursuant 10 CFR 50.61(b)(7), to recover fracture toughness.

However, recognizing that the RPV welds are expected to exceed 10 CFR 50.61 PTS screening criteria during the period of extended operation, Entergy Nuclear Operations (ENO)(current plant owner) has chosen the 10 CFR 54.21(c)(1)(iii) option for managing the PTS TLAA. Per Section 4.2.2.2 of the SER, this option involved the following activities:

a) An assessment of the current licensing basis TLAA for PTS,
b) A discussion of the flux reduction program implemented in accordance with 10 CFR 50.61(b)(3), and
c) An identification of viable options for managing the aging effect in the future ("Pressurized Thermal Shock Analyses for Renewal of Certain Nuclear Power

Plant Operating Licenses," Executive Director Memo to Commissioners, dated May 27, 2004, ML041190564).

The license renewal SER states that, prior to exceeding the PTS screening criteria limit, PNP will select the best option to manage PTS in accordance with NRC regulations and make required submittals to obtain NRC review and approval.

Subsequent to NRC approval of the LRA, the NRC issued 10 CFR 50.61a to provide an alternative method for evaluating PTS for plants projected to exceed the screening criteria limit. ENO intends to use this alternative method for evaluating PTS for the PNP RPV at the appropriate time to meet the regulation.

The purpose of the scheduled June 22, 2010, meeting with the NRC is to discuss the status of activities recently completed and planned in the near future, under 10 CFR 54.21(c)(1)(iii), involving the PTS analysis. Activities recently completed include:

1. An update of Westinghouse Report, WCAP-15353, "Palisades Reactor Pressure Vessel Neutron Fluence Evaluation," Revision 0, January, 2000, to account for actual capacity factors and core loading patterns since that time.

   The update is documented in WCAP-15353-Supplement 1-NP, "Palisades Reactor Pressure Vessel Fluence Evaluation," Revision 0, April 2010 (Attachment 2).

2. An assessment of embrittlement for the limiting axial weld heat number W5214 relative to the PNP RPV.

   A new preliminary assessment for the RPV projecting RT_{PTS} and the revised date when weld heat W5214 will exceed the PTS screening criteria of 270°F is contained in Structural Integrity Associates (SIA) Report No. 0901132.401, "Evaluation of Surveillance Data for Weld Heat No. W5214 for Application to Palisades PTS Analysis," Revision 0, April 2010 (Attachment 3).

   This new preliminary assessment uses the updated fluence evaluation performed by Westinghouse and all available surveillance data relevant to the PNP RPV limiting axial weld heat number W5214 as permitted by 10 CFR 50.61(c)(2).

   A review of this report was performed by Westinghouse (Attachment 4).

The preliminary SIA PTS screening assessment report indicates that, based on the new information, the revised projected date for the PNP RPV to reach the PTS screening criteria limit of 270°F would be approximately April 2017 or later.

10 CFR 50.61(c)(3) states that any information believed to improve the accuracy of the RT_{PTS} value significantly must be reported to the NRC and any value or RT_{PTS} that has been modified using the procedures of 10 CFR 50.61(c)(2) is subject to NRC approval.

In accordance with 10 CFR 50.61(c)(3), at a later date, ENO plans to report new information that improves the accuracy of the RT_{PTS} value to the NRC and to submit the updated fluence evaluation and the new PTS screening assessment report for NRC review and approval.

Based upon this new projection when the PNP RPV will reach the PTS screening criteria, ENO has identified the need for the following future activities:

1. Reschedule of the volumetric inspection for the PNP RPV from the fall 2010 refueling outage to the spring 2012 refueling outage.

2. Update and submit the PNP RPV heatup and cooldown curves located in the PNP Technical Specifications for operation through spring 2017.

3. Following inspection of the PNP RPV in spring 2012, perform an assessment of the PNP RPV in accordance with 10 CFR 50.61a.

4. Following NRC review and approval of the PTS assessment of the PNP RPV under 10 CFR 50.61a, update and transmit the PNP RPV heatup and cooldown curves located in the PNP Technical Specifications for operation through the balance of the extended licensed life of the plant.

The ultrasonic inspection of the PNP RPV planned in the spring of 2012 and subsequent evaluation and submittal in accordance with 10 CFR 50.61a will satisfy the PTS regulatory requirements through the remaining period of extended operation.

ATTACHMENT 2

Reactor Pressure Vessel Fluence Evaluation

WCAP-15353-NP,
"Palisades Reactor Pressure Vessel Fluence Evaluation,"
Revision 0 - Supplement 1, May 2010

29 pages follow

WCAP-15353 – Supplement 1-NP
Revision 0

May 2010

Palisades Reactor Pressure Vessel Fluence Evaluation

Westinghouse

WCAP-15353 – Supplement 1-NP, Revision 0

Palisades Reactor Pressure Vessel Fluence Evaluation

S. L. Anderson*, Fellow Engineer
Radiation Engineering & Analysis

May 2010

Reviewed:
P. M. Song*, Principal Engineer
Radiation Engineering & Analysis

Approved:
A. R. Dulloo*, Manager
Radiation Engineering & Analysis

Work Performed Under Shop Order 450
Purchase Order No. 66941
Prepared by Westinghouse for

WESTINGHOUSE ELECTRIC COMPANY LLC
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

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

EXECUTIVE SUMMARY

Calculations of the neutron exposure of the Palisades reactor pressure vessel were previously completed and documented in WCAP-15353, Revision 0, "Palisades Reactor Vessel Neutron Fluence Evaluation," January 2000.[3] This evaluation was submitted for review by the NRC Staff and, after consideration of RAI's addressed in Reference 4, the fluence methodology as well as the final results were approved by the Staff.

The fluence analysis described in WCAP-15353, Revision 0[3] included cycle specific evaluations through Cycle 14 (the then current operating cycle). This supplement to WCAP-15353 provides an updated neutron fluence assessment for the Palisades pressure vessel beltline region that includes cycle specific analysis for additional operating cycles for which the design has been finalized (Cycles 15 through 21) and includes projections for future operation through approximately 44 effective full power years (EFPY). Updated evaluations of surveillance capsule credibility analysis and determination of material chemistry factors are being completed in parallel with this fluence calculation and will be documented elsewhere.

Based on the cycle specific analysis through Cycle 21 (approximately 23.4 EFPY) and the projection scenario for future operation provided by Entergy, the maximum neutron exposure of the pressure vessel beltline materials through 44 EFPY is summarized as follows.

End of Fuel Cycle	Estimated Calendar Date	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			0 Deg.	15 Deg.	30 Deg.	45 Deg.
21	10/2010	23.4	1.447E+19	2.114E+19	1.590E+19	9.677E+18
22	4/2012	24.7	1.496E+19	2.201E+19	1.652E+19	1.001E+19
23	10/2013	26.1	1.545E+19	2.288E+19	1.717E+19	1.038E+19
24	4/5/2015	27.4	1.592E+19	2.372E+19	1.779E+19	1.073E+19
25	10/2016	28.8	1.642E+19	2.461E+19	1.844E+19	1.110E+19
26	4/2018	30.2	1.691E+19	2.549E+19	1.909E+19	1.147E+19
27	10/2019	31.5	1.741E+19	2.637E+19	1.975E+19	1.184E+19
28	4/2021	32.9	1.790E+19	2.726E+19	2.040E+19	1.221E+19
29	10/2022	34.3	1.840E+19	2.814E+19	2.105E+19	1.258E+19
30	4/2024	35.7	1.889E+19	2.903E+19	2.170E+19	1.295E+19
31	10/2025	37.1	1.939E+19	2.991E+19	2.236E+19	1.332E+19
32	4/2027	38.4	1.988E+19	3.079E+19	2.301E+19	1.369E+19
33	10/2028	39.8	2.038E+19	3.168E+19	2.366E+19	1.406E+19
34	4/2030	41.2	2.087E+19	3.256E+19	2.432E+19	1.443E+19
35	10/2031	42.6	2.137E+19	3.344E+19	2.497E+19	1.480E+19
36	4/2033	44.0	2.186E+19	3.433E+19	2.562E+19	1.517E+19

End of Fuel Cycle	Estimated Calendar Date	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			60 Deg.	75 Deg.	90 Deg.	
21	10/2010	23.4	1.472E+19	2.157E+19	1.575E+19	
22	4/2012	24.7	1.520E+19	2.252E+19	1.647E+19	
23	10/2013	26.1	1.571E+19	2.345E+19	1.717E+19	
24	4/5/2015	27.4	1.619E+19	2.433E+19	1.784E+19	
25	10/2016	28.8	1.670E+19	2.527E+19	1.854E+19	
26	4/2018	30.2	1.721E+19	2.621E+19	1.925E+19	
27	10/2019	31.5	1.772E+19	2.714E+19	1.995E+19	
28	4/2021	32.9	1.823E+19	2.808E+19	2.065E+19	
29	10/2022	34.3	1.874E+19	2.902E+19	2.136E+19	
30	4/2024	35.7	1.925E+19	2.995E+19	2.206E+19	
31	10/2025	37.1	1.976E+19	3.089E+19	2.277E+19	
32	4/2027	38.4	2.027E+19	3.182E+19	2.347E+19	
33	10/2028	39.8	2.078E+19	3.276E+19	2.417E+19	
34	4/2030	41.2	2.129E+19	3.370E+19	2.488E+19	
35	10/2031	42.6	2.180E+19	3.463E+19	2.558E+19	
36	4/2033	44.0	2.231E+19	3.557E+19	2.628E+19	

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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 Appendix G to 10 CFR 50[1], 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"*. Each of the materials comprising the beltline region must be considered in the overall embrittlement assessments for the pressure vessel. Therefore, plant-specific exposure assessments must include evaluations as a function of position over the beltline region.

Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [2], 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[3] 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 4, the methodology described in WCAP-15353, Revision 0 was approved for application to the Palisades reactor pressure vessel. Subsequent to that approval additional submittals[7,8] in support of the benchmarking of this fluence methodology were reviewed and approved by the NRC Staff.

The fluence analysis described in WCAP-15353, Revision 0[3] included cycle specific evaluations through Cycle 14 (the then current operating cycle). This supplement to WCAP-15353 provides an updated neutron fluence assessment for the Palisades pressure vessel beltline region that includes cycle specific analysis for additional operating cycles for which the design has been finalized (Cycles 15 through 21) and includes projections for future operation through approximately 44 effective full power years (EFPY). The results of this evaluation are intended for use as input to vessel materials analyses (to be documented elsewhere) that include updates to surveillance capsule credibility analysis and material chemistry factor determination.

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, this report also includes the latest fluence evaluation from capsules containing the W5214 material. This compilation of capsule fluence values is based on the same fluence methodology described in this report.

In subsequent sections of this supplement, the methodologies used to perform neutron transport calculations and dosimetry evaluations are described in some detail, the updated results of the plant specific transport calculations are given for the beltline region of the Palisades pressure vessel. Comparisons of calculations and measurements demonstrating that the transport calculations meet the requirements of Regulatory Guide 1.190 that were previously included in Reference 3 are also included in this supplement for completeness. 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 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.

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[5] and the BUGLE-96 cross section library[6]. 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.

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 current evaluations. A plan views 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 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 one foot below the active fuel to an axial elevation one foot 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.

The core power distributions used in the plant-specific transport analysis for the reactor were provided by Entergy.[15] The data used in the source generation included fuel assembly-specific initial enrichments, beginning-of-cycle burnups and end-of-cycle burnups. Appropriate axial distributions were also obtained.

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.

Figure 2.1-1

Palisades r,θ Reactor Geometry

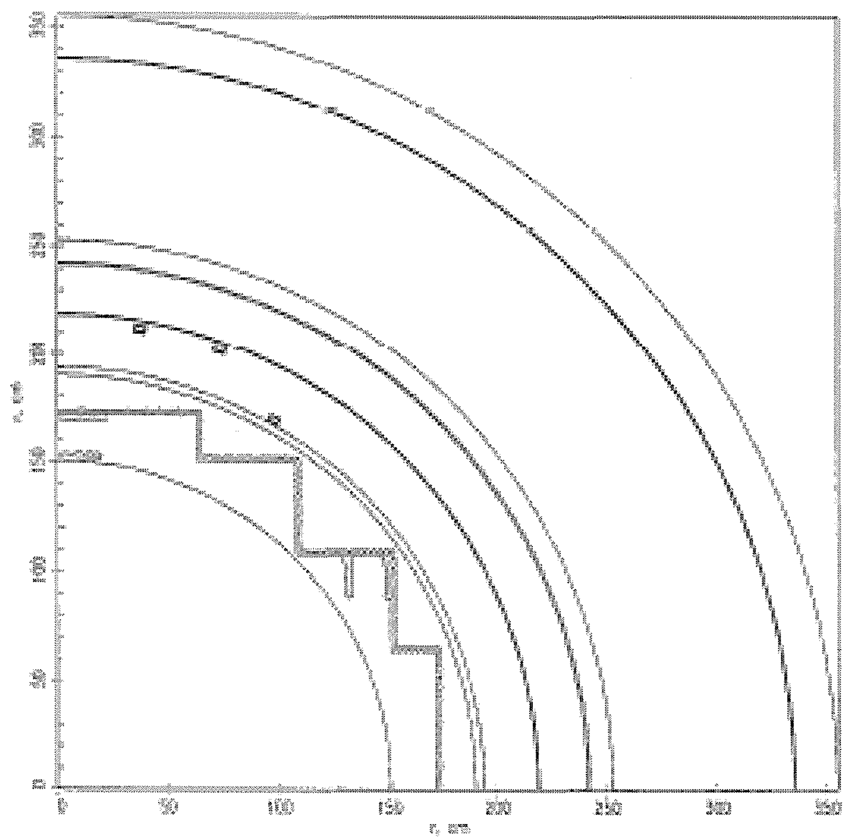
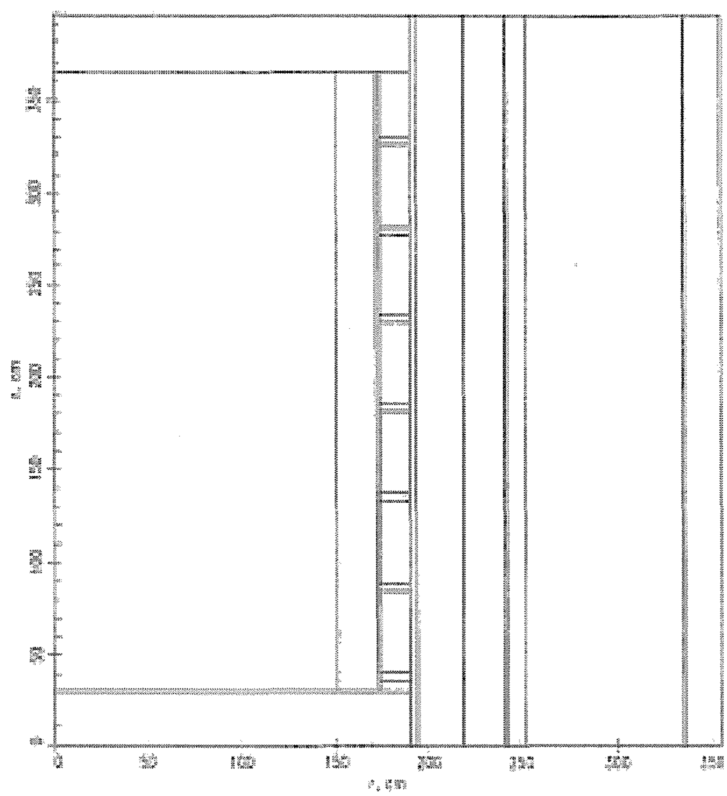


Figure 2.1-2

Palisades r,z Reactor Geometry



2.2 – Calculated Neutron Exposure of Pressure Vessel Beltline Materials

The plant- and 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 is given in Tables 2.2-1 and 2.2-2, 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 2.2-1 and 2.2-2 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 2.2-1 and 2.2-2 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 Table 2.2-3, projections of neutron (E > 1.0 MeV) fluence beyond the end of Cycle 21 are provided. 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 (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 2.2-1, 2.2-2, and 2.2-3, 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 is exposed to the maximum neutron exposure characteristic of the 75° azimuthal location.

Table 2.2-1

Summary of Calculated Maximum Pressure Vessel Neutron Flux (E > 1.0 MeV)
For Cycles 15 Through 21 and for Future Projection

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	1.558E+10	1.277E+10	7.924E+09
16	1.2	1.068E+10	1.604E+10	1.330E+10	7.797E+09
17	1.3	1.080E+10	1.860E+10	1.332E+10	7.613E+09
18	1.3	1.292E+10	2.094E+10	1.352E+10	7.337E+09
19	1.3	1.059E+10	1.924E+10	1.445E+10	7.037E+09
20	1.4	1.123E+10	2.004E+10	1.517E+10	8.143E+09
21	1.4	1.138E+10	2.016E+10	1.501E+10	8.506E+09
22	Proj.	1.153E+10	2.024E+10	1.454E+10	7.756E+09
23+	Proj.	1.138E+10	2.016E+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	1.681E+10	1.257E+10	
16	1.2	1.135E+10	1.762E+10	1.401E+10	
17	1.3	9.781E+09	1.967E+10	1.539E+10	
18	1.3	1.088E+10	2.235E+10	1.664E+10	
19	1.3	1.090E+10	2.230E+10	1.743E+10	
20	1.4	1.161E+10	2.198E+10	1.650E+10	
21	1.4	1.172E+10	2.151E+10	1.618E+10	
22	Proj.	1.128E+10	2.204E+10	1.669E+10	
23+	Proj.	1.172E+10	2.151E+10	1.618E+10	

Table 2.2-2

Summary of Calculated Maximum Pressure Vessel Neutron Exposure
Through the Conclusion of Cycle 21

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	8.366E+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	9.296E+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	1.575E+19	

Table 2.2-3

Projections of Calculated Maximum Pressure Vessel Neutron Exposure

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	2.288E+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	1.909E+19	1.147E+19
27	1.4	31.5	1.741E+19	2.637E+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	2.814E+19	2.105E+19	1.258E+19
30	1.4	35.7	1.889E+19	2.903E+19	2.170E+19	1.295E+19
31	1.4	37.1	1.939E+19	2.991E+19	2.236E+19	1.332E+19
32	1.4	38.4	1.988E+19	3.079E+19	2.301E+19	1.369E+19
33	1.4	39.8	2.038E+19	3.168E+19	2.366E+19	1.406E+19
34	1.4	41.2	2.087E+19	3.256E+19	2.432E+19	1.443E+19
35	1.4	42.6	2.137E+19	3.344E+19	2.497E+19	1.480E+19
36	1.4	44.0	2.186E+19	3.433E+19	2.562E+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	1.575E+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	31.5	1.772E+19	2.714E+19	1.995E+19	
28	1.4	32.9	1.823E+19	2.808E+19	2.065E+19	
29	1.4	34.3	1.874E+19	2.902E+19	2.136E+19	
30	1.4	35.7	1.925E+19	2.995E+19	2.206E+19	
31	1.4	37.1	1.976E+19	3.089E+19	2.277E+19	
32	1.4	38.4	2.027E+19	3.182E+19	2.347E+19	
33	1.4	39.8	2.078E+19	3.276E+19	2.417E+19	
34	1.4	41.2	2.129E+19	3.370E+19	2.488E+19	
35	1.4	42.6	2.180E+19	3.463E+19	2.558E+19	
36	1.4	44.0	2.231E+19	3.557E+19	2.628E+19	

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.[2] 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 3. However, for completeness, a brief description of the measurement program, dosimetry evaluation procedure, and final results are also included in this supplement to Reference 3.

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[8].

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[8] 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. [9]

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 9.

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.[2]

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[2]. 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 3, 12, 13, and 14.

In providing the data listed in Table 4-1, no new fluence calculations were performed. The data were obtained either from Palisades specific documents[10, 11] or from public domain documents[3, 12, 13, 14] 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 3, 10, and 11 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 12 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 13 and 14, 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. 3)
Palisades	W290	9.38e+18	WCAP-15353, R0 (Ref. 3)
Palisades	W100-1	1.64e+19	WCAP-15353, R0 (Ref. 3)
Palisades	SA60-1	1.50e+19	WCAP-15353, R0 (Ref. 3)
Palisades	SA240-1	2.38e+19	CPAL-01-009 (Ref. 10)
Palisades	W100-2	2.09e+19	CPAL-04-8 (Ref. 11)
Indian Point 2	T	2.53e+18	WCAP-15629, R1 (Table 3) (Ref. 12)
Indian Point 2	Y*	4.55e+18	WCAP-15629, R1 (Table 3) (Ref. 12)
Indian Point 2	Z	1.02e+19	WCAP-15629, R1 (Table 3) (Ref. 12)
Indian Point 2	V*	4.92e+18	WCAP-15629, R1 (Table 3) (Ref. 12)
H. B. Robinson	S	4.79e+18	WCAP-15805, R0 (Table 5-10) (Ref. 13)
H. B. Robinson	V*	5.30e+18	WCAP-15805, R0 (Table 5-10) (Ref. 13)
H. B. Robinson	T*	3.87e+19	WCAP-15805, R0 (Table 5-10) (Ref. 13)
H. B. Robinson	X*	4.49e+19	WCAP-15805, R0 (Table 5-10) (Ref. 13)
Indian Point 3	T*	2.63e+18	WCAP-16251-NP, R0 (Table 5-10) (Ref. 14)
Indian Point 3	Y*	6.92e+18	WCAP-16251-NP, R0 (Table 5-10) (Ref. 14)
Indian Point 3	Z*	1.04e+19	WCAP-16251-NP, R0 (Table 5-10) (Ref. 14)
Indian Point 3	X*	8.74e+18	WCAP-16251-NP, R0 (Table 5-10) (Ref. 14)

Notes:

1 - Relative to the Palisades data, References 1, 10, and 11 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 12 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 13 and 14 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

REFERENCES

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ATTACHMENT 3

Preliminary PTS Screening Criteria Assessment

Structural Integrity Associates (SIA) Report No. 0901132.401, "Evaluation of Surveillance Data for Weld Heat No. W5214 for Application to Palisades PTS Analysis," Revision 0, April 2010

178 pages follow

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Revision 0
Project No. 0901132
April 2010

Evaluation of Surveillance Data for Weld Heat No. W5214 for Application to Palisades PTS Analysis

Prepared for:

Entergy Nuclear Corp.
Palisades Nuclear Power Plant

Prepared by:

Structural Integrity Associates, Inc.
San Jose, California

Prepared by: Timothy J. Griesbach — Date: 4/20/2010

Prepared by: Vikram Marthandam — Date: 4/20/2010

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Section	Pages	Revision	Date	Comments
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EXECUTIVE SUMMARY

This evaluation was performed as part of a review of the Palisades Pressurized Thermal Shock (PTS) re-evaluation. A previous analysis performed for the Palisades vessel in 2000 determined that the PTS screening criteria limit of 270°F for weld heat No. W5214 would not be reached until January 2014. That evaluation was based on the fluence projections and weld material chemistry for weld heat No. W5214 available at that time; no credit was given for surveillance data to improve the RT_{PTS} projection. In the fall of 2009 it became apparent to Entergy that new information was available that could affect the RT_{NDT} of the limiting Palisades vessel beltline material. The new data included revised fluence calculations and a total of eleven irradiated surveillance capsules that contain Charpy V-notch data for weld heat No. W5214. This report examines the updated fluence calculations performed by Westinghouse and all the available surveillance data relevant to the Palisades reactor pressure vessel weld heat No. W5214. Using the revised fluences and chemistry factors based on the refitted surveillance data for this weld heat, this re-evaluation shows that the projected date to reach the PTS screening criteria limit using the surveillance weld data would be approximately April 2017 or later.

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EVALUATION OF SURVEILLANCE DATA FOR WELD HEAT NO. W5214 FOR APPLICATION TO PALISADES PTS ANALYSIS

1.0 INTRODUCTION

The Palisades Nuclear Plant submitted to NRC a Pressurized Thermal Shock (PTS) evaluation in 2000 that projected the value for RT_{PTS}, or maximum Adjusted Reference Temperature (ART) of the limiting vessel weld or plate, based on the calculated fluences and material properties available at that time [1]. The limiting vessel beltline material was determined to be weld heat No. W5214, and the projected ART value was based on the method in the PTS Rule given in 10CFR50.61, Paragraph (c)(1) [2] using the best estimate chemistry for this weld, the corresponding chemistry factor, CF, and the fluence values from the vessel fluence evaluation in WCAP-15353 [3]. These inputs to the PTS Rule equations were used to calculate RT_{PTS}, and the Palisades vessel was projected to reach the screening criterion limit of 270°F for the limiting weld in January 2014.

Since the time that the previous PTS evaluation was performed for the Palisades vessel, ten years have passed and more data and information are available now to update the projected RT_{NDT} value for the limiting Palisades vessel beltline material. In particular, this evaluation considers all the available surveillance data for weld heat No. W5214 that can be used to refine the projected RT_{NDT} in accordance with 10CFR50.61, Paragraph (c)(2) [2]. This evaluation is being updated now because there is new information that changes the projected values of RT_{PTS} for the Palisades vessel.

The new information was generated by performing a survey of all relevant surveillance data for weld heat No. W5214. Eleven irradiated surveillance capsule reports were found containing this weld, and the Charpy data contained in these capsules was compiled and refitted consistently using the CVGRAPH hyperbolic tangent curve-fitting methodology [4]. Also, the eleven capsule fluence values have been updated (over time) by Westinghouse using their NRC approved fluence methodology for implementing the Regulatory Guide 1.190 benchmarking procedure [5]. All data was evaluated in accordance with the PTS Rule approach to determine the shift values, fitted CF from the surveillance data, scatter from the mean predicted shift, and credibility of the data. It was determined that this new evaluation provides the most technically complete and sound assessment for the Palisades weld heat No. W5214 and allows for more accurate life projections of the limiting material in the Palisades vessel. The latest W5214 life projection provides improvement (i.e., more time) from the previous prediction to reach the PTS screening criterion limit.

2.0 APPLICABILITY

This evaluation is applicable to the Palisades Reactor vessel PTS analysis relative to intermediate shell axial welds 2-112A/B/C fabricated from weld heat No. W5214 [1]. The results of this evaluation are used to revise the RT_{PTS} projection to determine the maximum fluence, or

equivalent date, to reach the PTS screening criteron of 270°F for the limiting axial weld at the 60° azimuthal location. Evaluation of other beltline region materials was not performed at this time because they are not projected or expected to exceed the PTS screening criterion over the next several years. A complete PTS analysis for all of the vessel beltline materials will be performed at a later time per 10CFR50.61.

3.0 METHODOLOGY

3.1 Charpy TANH Curve-Fitting Method

All Charpy data has been re-evaluated to assure that the Charpy curve fits and the 30 ft-lb shift values from the surveillance capsule reports are performed in a consistent manner. The general shape of Charpy test data (energy versus temperature, or lateral expansion versus temperature) is that of an "S", generally with definable lower and upper shelves and a connecting region between the shelves called the transition region. The hyperbolic tangent (TANH) function has been used for some time as a simple statistical curve-fit tool to describe this "S"-shaped response [25]. Other functional relationships could have been used to produce a similar shape (e.g., an error function), but the benefit of the TANH function is that the curve fit parameters defining the "S" shape have physical meaning relative to what is generally evaluated from the test results. As a result, the hyperbolic TANH curve fitting of Charpy V-notch (CV) impact energy data has been a standard practice within the industry.

The TANH model used for modeling Charpy V-notch curves is given by Equation (1) [25]:

$$C_V = A + B \tanh [(T - T_o) / C] \qquad (1)$$

where,

C_V = Charpy V-notch impact energy
T = test temperature
A = the mean energy level between the upper and lower shelves
B = the + or – deviation from the mean energy level
T_o = a parameter that represents the mid-energy transition temperature
C = the + or – deviation of the intercepts of the tangent to the transition of T_o and the upper and lower shelves

The lower shelf (A – B) was fixed at 2.2 ft-lb, and the upper shelf (A + B) was fixed in accordance with standard practice for applying hyperbolic tangent fits to Charpy V-notch data [24].

3.2 Fluence Analysis Method

The surveillance capsule fluence values were re-evaluated by Westinghouse using the DORT neutron transport calculation method which has been benchmarked to meet the criteria in NRC Regulatory Guide 1.190 [5], shows close agreement between calculations and neutron dosimetry measurements, and has been approved for use by NRC. The capsule fluence values were provided as design inputs by Westinghouse [18].

3.3 10CFR50.61 (PTS Rule) Embrittlement Prediction Methods

The PTS Rule in 10CFR50.61 [2] provides two methods for determining the reference temperature. The first method considers only the copper and nickel chemistry, fluence, and initial RT_{NDT} of the weld, plate, or forging material in the reactor vessel beltline. For those beltline materials, Equation (2) is used to determine the adjusted RT_{NDT} for comparison to the PTS screening criteria limits.

$$RT_{NDT} = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \tag{2}$$

where ΔRT_{NDT} is the mean value of the transition temperature due to irradiation, and must be calculated using the Equation (3):

$$\Delta RT_{NDT} = (CF)\, f^{(0.28-0.10 \log f)} \tag{3}$$

where CF (°F) is the chemistry factor, which is a function of the copper and nickel content. CF is determined by using Table 1 (from 10CFR50.61) for welds and by using Table 2 (from 10CFR50.61) for base metals (plates and forgings). "Wt % copper" and "Wt % nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging. For a weld, the best estimate values will normally be the mean of the measured values for a weld deposit made using the same weld wire heat number as the critical vessel weld. For weld heat number W5214, the best-estimate chemistry, as determined by the industry best-estimate results from the CEOG report [20], is Cu = 0.213 wt%, Ni = 1.007 wt%. The best-estimate chemistry values for the C-E fabricated welds are shown in Appendix A. The corresponding chemistry factor for this weld heat is CF = 230.73°F. This value for best-estimate nickel content varies slightly from the value used previously in the PTS submittal (Ni = 1.01%) [1]. It is noted that the basis for the best estimate Cu and Ni values came from the CEOG report [20] which is considered to be the industry standard for the C-E fabricated welds, but Palisades chose to round up the nickel content from 1.007% to 1.01% in the 1998 RAI response to Generic Letter 92-01 [19]. For the current analysis we have also used the CEOG determined actual nickel best estimate chemistry which gives a CF value of 230.73°F for comparison to a CF value of 231.08°F for the rounded up nickel content.

The Initial RT_{NDT} for weld heat No. W5214 is determined from the generic value of -56°F for C-E fabricated Linde 1092 flux type welds [31], and the margin term is determined from Equation 4:

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \tag{4}$$

where σ_I is the standard deviation for the initial RT_{NDT}. If the generic mean Initial RT_{NDT} value for a Linde 1092 weld is used, then σ_I = 17°F [2]. The (1-sigma) standard deviation for ΔRT_{NDT}, σ_Δ, is 28°F for welds, so the margin term for this case is 65.5°F.

Using this approach to determine the RT_{PTS} at the screening criteria limit for axial welds (i.e., 270°F) yields a maximum allowable fluence at the 60° azimuthal weld location of:

$$RT_{PTS} = 270°F = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} = -56 + \Delta RT_{NDT} + 65.5°F \tag{5}$$

$$\Delta RT_{NDT} = 260.5°F = (CF)\, f^{(0.28\text{-}0.10 \log f)} = (230.73)\, f^{(0.28\text{-}0.10 \log f)} \tag{6}$$

and $f = 1.595\text{x}10^{19}\ \text{n/cm}^2$ (7)

The second method for determining the chemistry factor and the RT_{NDT} states that, "To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results." Surveillance program results means any data that demonstrates the embrittlement trends for the limiting beltline material, including but not limited to data from test reactors or from surveillance programs at other plants with or without surveillance program integrated per 10 CFR Part 50, appendix H. This is the case for Palisades; eleven previously tested surveillance capsules are now available that contain the limiting vessel weld heat No. W5214. The axial weld in the Palisades vessel made from weld heat No. W5214 was determined to be the limiting vessel beltline material [1].

Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible as judged by the following criteria [2]:

(A) The materials in the surveillance capsules must be those which are the controlling materials with regard to radiation embrittlement,

(B) Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions must be small enough to permit the determination of the 30-foot-pound temperature unambiguously,

(C) Where there are two or more sets of surveillance data from one reactor, the scatter of RT_{NDT} values must be less than 28°F for welds and 17°F for base metal. Even if the range

in the capsule fluences is large (two or more orders of magnitude), the scatter may not exceed twice those values (i.e., 56°F),

(D) The irradiation temperature of the Charpy specimens in the capsule must equal the vessel wall temperature at the cladding/base metal interface within 25°F, and

(E) The surveillance data for the correlation monitor material in the capsule, if present, must fall within the scatter band of the data base for the material.

Surveillance data deemed credible according to these criteria must be used to determine a material-specific value of CF for use in Equation (2). A material-specific value of CF is determined from Equation (8) [19].

$$CF = \frac{\sum_{i=1}^{n} \left[A_i \times f_i^{(0.28\ -\ 0.10 \log f_i)} \right]}{\sum_{i=1}^{n} \left[f_i^{(0.56\ -\ 0.20 \log f_i)} \right]} \tag{8}$$

where "n" is the number of surveillance data points,
"A_i" is the measured value of ΔT_{30} from the Charpy specimens, and
"f_i" is the fluence for each surveillance capsule data point.

If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld (i.e. differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld), the measured values of ΔT_{30} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld [2].

For cases in which the results from a credible plant-specific surveillance program are used, the value of σ_Δ to be used in the margin term of Equation (4) is 14°F for welds; this is called a reduced margin term. The value of σ_Δ need not exceed one-half of ΔRT_{NDT} [2].

The use of results from the plant-specific surveillance program method may result in a RT_{NDT} that is higher or lower than that determined from the first (chemistry table) method. If the resulting RT_{NDT} from using the surveillance data gives a higher value it must be used. If the resulting RT_{NDT} from using the surveillance data gives a lower value it may be used.

NRC provided additional guidance for evaluation and use of surveillance data in Reference 19. The guidance provides examples of Case 4 that may be used for evaluating the Palisades related surveillance data, as shown in Appendix B. Two cases are considered, Case 4a considers surveillance data from the plant of interest, and Case 4b for use of surveillance capsule data from both the plant of interest and also from its sister plants containing the same weld heat.

Also, if there is clear evidence that the copper or nickel content of the surveillance weld differs from that of the vessel weld, i.e., differs from the average of the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of ΔRT_{NDT} should be adjusted by multiplying them by the ratio of the chemistry factor of the vessel weld to that of the surveillance weld using the equation [19]:

$$Ratio\ Adjusted\ \Delta RT_{NDT} = \left(\frac{Table\ CF_{Vessel\ Chem.}}{Table\ CF_{Surv.\ Chem.}}\right) * Measured\ \Delta RT_{NDT} \tag{9}$$

According to the NRC guidance [19], further adjustment to the ΔRT_{NDT} data from other sources is needed if there is a difference between the capsule temperature from the other plant and the plant of interest. A temperature correction of 1°F/°F is made to the ΔRT_{NDT} values to account for this difference; a positive temperature adjustment is made to capsules exposed to (time-weighted average) temperatures below the mean vessel temperature, and a negative temperature adjustment is made to capsules exposed to (time-weighted average) temperatures above the mean vessel temperature. The mean vessel temperature for the Palisades vessel, using a time-weighted average for the plant operating cycles, is determined to be 535.2°F, as shown in Table 8.

Guidance from Reference 19, Case 4, "Surveillance Data from Plant and Other Sources," and 10CFR50.61, method 2 for inclusion of plant-specific surveillance data, has been applied to evaluation of the W5214 surveillance data as described in Section 5.0.

4.0 DATA EVALUATION RESULTS

In 1998, Consumers Energy provided a response to a Request for Information from NRC regarding pressure vessel integrity for the Palisades vessel [26]. That response evaluated seven surveillance capsules containing weld heat No. W5214 which were available at that time (two capsules from H. B. Robinson 2, two capsules from Indian Point 2, and three capsules from Indian Point 3) and determined those data were not credible and, therefore, the data was not used to improve the projected RT_{PTS} for the Palisades vessel. Since then, four more capsules containing this weld heat can be included in the analysis for use of surveillance data related to the Palisades limiting weld material. An evaluation of these data starts with the original capsule reports.

The new data survey was performed to gather all the unirradiated and irradiated capsule test results for the Palisades limiting weld material. The data from all related surveillance capsules containing weld heat No. W5214 were compiled and the results were reviewed for applicability to the Palisades vessel weld. New data were discovered in the process of compiling these capsule reports. For example, there are two capsules from the Palisades supplemental

surveillance program that were previously unreported (References 14 – 17). In addition, three capsules from the H. B. Robinson 2 surveillance program (References 12), two capsules from the Indian Point 2 surveillance program (References 6 – 8), and four capsules from the Indian Point 3 surveillance program (References 9, 10, 11 and 13) were compiled and the Charpy V-notch test results were reviewed. These reports include surveillance capsule fluences and comparisons between the unirradiated and irradiated Charpy V-notch curves to determine the ΔRT_{NDT} (or ΔT_{30}) shifts. The reported fluence values from these capsule reports, the average of measured surveillance weld copper and nickel chemistries, and the (measured and reported) ΔT_{30} shift results for these eleven capsules are shown in Table 1. The capsule reports are included in Appendix C (Palisades), Appendix D (Indian Point Unit 2), Appendix E (Indian Point Unit 3) and Appendix F (H. B. Robinson Unit 2). These are considered to be the reported data (or original data), with the exception of the Palisades capsule reports that were considered to be supplemental capsule test results.

It is useful to first combine these data without any adjustments for chemistry or irradiation temperature to determine the mean trend in irradiation damage behavior. The mean trend, or average chemistry factor, can be calculated directly from a least squares fit to the data using Equation (8). The least squares fit method was used and a best fit chemistry factor (CF) of 217.67°F was determined from these data, as shown in Table 1. The results are plotted in Figure 1 and are shown here for information only. The scatter in the measured – predicted results show that the scatter exceeds the 28°F (1-sigma) margin for two out of eleven points, but these two points are within the 56°F (2-sigma) margin. The average copper content for these surveillance materials is Cu = 0.243 wt%, and the average nickel content for these surveillance materials is Ni = 0.965 wt%. The predicted (average) chemistry factor for the surveillance specimens based on chemistry (from the PTS Rule Table 1) is CF = 234.37°F. The mean fit to the data shows that the CF value and the fitted trend for these data is well below that predicted by the PTS Rule method 1 (i.e., surveillance data not available).

4.1 Original and Re-evaluated Surveillance Capsule Fluence

Westinghouse recalculated the capsule fluences from Palisades, Indian Point 2, Indian Point 3, and H. B. Robinson 2 using a consistent methodology to establish a common basis for the fluence values. This was an essential step so that all the surveillance capsule data could be evaluated properly for credibility and applicability to the Palisades vessel limiting weld material. The revised fluence values for capsules containing weld heat No. W5214 are shown in Table 2 [18]. It is noted that there were changes in the fluences from the originally calculated fluence values (shown in Table 1), and the new calculated fluence results (shown in Table 2) that were used to re-evaluate all the relevant surveillance data. The same Westinghouse fluence methodology was used to calculate fluence in the wall of the Palisades vessel for prediction of the vessel embrittlement.

4.2 Surveillance Capsule Temperatures

Surveillance capsule temperatures are necessary for the temperature corrections of the surveillance data when applying these data to the plant of interest. Time-weighted average temperatures were determined for the Palisades, Indian Point 2, Indian Point 3, and H. B. Robinson 2 capsules containing weld heat No. W5214. The data and method for determining the time-weighted average temperatures is given in Appendix H. The time-weighted average temperatures for the Indian Point Units 2 & 3 capsules were verified in Reference 33.

4.3 Original and Re-evaluated Charpy V-notch Surveillance Data

The surveillance capsule test results for weld heat No. W5214 from the Palisades supplemental capsules SA-60-1 and SA-240-1 are provided in Appendix C. The supplemental capsules with this weld heat were irradiated for a number of cycles, and removed and tested; capsule SA-60-1 was removed at the end of cycle 13, and capsule SA-240-1 was removed at the end of cycle 14. The specimens containing weld metal inserts were reconstituted to full size Charpy V-notch specimens. The capsule materials were tested by Framatome in 2001 [14, 15]. The unirradiated Charpy energy values for the weld metals are documented in a letter from John R. Kneeland to Matthew J. DeVan dated February 2, 1999 [16]. The baseline (unirradiated) curve and weld metal chemistry data for these specimens are also provided in Appendix C.

The surveillance capsule test results for weld heat No. W5214 from the Indian Point Unit 2 reactor surveillance capsule program are provided in Appendix D. The unirradiated data is contained in a Westinghouse report [8]. Southwest Research Institute tested two irradiated capsules, capsule Y [7] and capsule V [6]. The Indian Point 2 surveillance weld metal chemistry is also contained in these reports.

There are four irradiated surveillance capsules containing weld heat No. W5214 and one baseline test report for the Indian Point Unit 3 plant, as shown in Appendix E. The baseline capsule report from Westinghouse contains the unirradiated data and one chemistry measurement for the surveillance weld [9]. The results for the four irradiated capsules are also given in one Westinghouse (WCAP-16251-NP) report [13]. This WCAP report contains the surveillance weld Charpy V-notch test results and measured chemistry data for the Indian Point Unit 3 plant.

The surveillance capsule test results for weld heat No. W5214 from the H. B. Robinson Unit 2 reactor surveillance capsule program are provided in Appendix F. The unirradiated data is contained in a Westinghouse report [27]. There are three irradiated capsules from H. B. Robinson Unit 2 which contain weld heat No. W5214, capsule T, capsule V, and capsule X. The Charpy V-notch test results from these three capsules are contained in one Westinghouse (WCAP-15805) report [12]. This WCAP report also documents the measured surveillance weld chemistry for H. B. Robinson Unit 2.

These original Charpy V-notch energy data were refitted using the CVGRAPH 5.0 hyperbolic tangent curve-fitting method [4]. The data were carefully fitted to obtain the best TANH fits. The CVGRAPH curve-fit results are shown in Appendix G from Reference 32. The results of the refitted and reanalyzed weld heat No. W5214 data for 30 ft-lb shift (ΔT_{30}) are shown in Table 3. These refitted Charpy data results have been verified for use in the new credibility evaluation [32].

The results presented here are considered to be "new data" because of the updated fluences and refitted ΔT_{30} values and because several additional capsules containing weld heat No. W5214 were uncovered in this survey that had not been previously evaluated together with the other data. The results from these new data were evaluated for applicability to the prediction of the RT_{PTS} value for weld heat No. W5214 per 10CFR50.61 [2] and the NRC guidance shown in Appendix B [19].

An evaluation of the credibility for the use of these data for the Palisades limiting weld is given in Section 5.0.

Table 1.

Results for all W5214 Surveillance Data with Reported Fluence and Vendor Reported Shift Results

Capsule	%Cu[(a)]	%Ni[(a)]	CF (F)	Reported Fluence [(b)] (n/cm^2)	FF	Reported ΔRTndt (F)	Predicted ΔRTndt (F)	Measured - Predicted ΔRTndt (F)
SA-60-1	0.307	1.045	266.5	1.61E+19	1.13	259	246.3	12.7
SA-240-1	0.307	1.045	266.5	2.60E+19	1.26	280.1	273.4	6.7
HB2 T	0.34	0.66	217.7	3.87E+19	1.35	288.15	293.6	-5.5
HB2 V	0.34	0.66	217.7	5.30E+18	0.82	209.32	179.1	30.3
HB2 X	0.34	0.66	217.7	4.49E+19	1.38	265.93	300.5	-34.6
IP2 V	0.20	1.03	226.3	5.59E+18	0.84	204	182.3	21.7
IP2 Y	0.20	1.03	226.3	5.89E+18	0.85	195	185.4	9.6
IP3 T	0.16	1.12	206.2	2.63E+18	0.64	151.6	138.6	13.0
IP3 Y	0.16	1.12	206.2	6.92E+18	0.90	172	195.2	-23.2
IP3 Z	0.16	1.12	206.2	1.04E+19	1.01	229.2	220.1	9.1
IP3 X	0.16	1.12	206.2	8.74E+18	0.96	193.2	209.4	-16.2
Average =	**0.243**	**0.965**	**Table CF =**	**234.37°F**		**Best fit CF =**	**217.67°F**	

(a) Measured capsule weld materials Cu and Ni values obtained from [6, 12, 13, 14, 15, 20, 26]

(b) Reported capsule fluence values from [6, 7, 12, 13, 28]

Table 2.

Summary of Revised Capsule Fluences and Time-Weighted Average Temperatures for Surveillance Capsules Containing Weld Heat No. W5214

Reactor	Surveillance Capsule Designation	Time-Weighted Average Temperature (°F)	Fluence [(a)] (E > 1 MeV) [n/cm^2]
Palisades	SA-60-1	535.0 [from Table 8]	1.50E19
Palisades	SA-240-1	535.7 [from Table 8]	2.38E19
H. B. Robinson 2	T	547 [12]	3.87E19
H. B. Robinson 2	V	547 [12]	5.30E18
H. B. Robinson 2	X	547 [12]	4.49E19
Indian Point 2	V	524 [12]	4.92E18
Indian Point 2	Y	529.1 [from App. H]	4.55E18
Indian Point 3	T	539.4 [from App. H]	2.63E18
Indian Point 3	Y	539.5 [from App. H]	6.92E18
Indian Point 3	Z	538.9 [from App. H]	1.04E19
Indian Point 3	X	539.7 [from App. H]	8.74E18

(a) Revised capsule fluence values from Reference 18.

Table 3.
Summary of Revised (Refitted) Surveillance Capsule Results for Weld Heat No. W5214

Capsule	Unirradiated (Refitted) T_{30} (F)[a]	Irradiated (Refitted) T_{30} (F)[a]	Revised (Refitted) ΔT_{30} (F)	Upper Shelf Energy (ft-lbs)[a]
SA-60-1	-60.1	198.9	259	54.5
SA-240-1	-60.1	220	280.1	52.5
HB2 T	-85.8	203.3	289.1	60.5
HB2 V	-85.8	123	208.8	70.5
HB2 X	-85.8	179.8	265.6	79.8
IP2 V	-65.4	132.1	197.5	76
IP2 Y	-65.4	128.5	193.9	66.5
IP3 T	-63.8	86	149.8	90.5
IP3 Y	-63.8	107.3	171.1	69
IP3 Z	-63.8	164.5	228.3	76
IP3 X	-63.8	128.7	192.5	75

(a) Charpy TANH curve-fit parameters, T_{30} values and plots are shown in Appendix G [32]

5.0 DATA CREDIBILITY ASSESSMENT AND FLUENCE EVALUATION

The purpose of this evaluation is to apply the credibility requirements in 10CFR50.61 to the Palisades, H.B Robinson Unit 2, Indian Point Unit 2, and Indian Point Unit 3 surveillance capsule data and to determine if the surveillance capsule data is credible and can be used to improve the RT_{NDT} predictions for the limiting vessel weld heat No. W5214.

10CFR50.61 describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of low-alloy steels currently used for light-water-cooled reactor vessels. 10CFR50.61 provides two methods for calculating the adjusted reference temperature of the reactor vessel beltline materials. The first method is described in paragraph (c)(1). The second method is described in paragraphs (c)(2) and (c)(3). The procedures in paragraphs (c)(2) and (c)(3) can only be applied when two or more credible surveillance data sets become available. These tests of surveillance data credibility are also stated in Section 3.3.

NRC provided additional guidance for evaluation and use of surveillance data in Attachment 3 of Reference 19. The evaluation presented herein is organized like Case 4 from this guidance document, the case for plants with surveillance data for their plant and from other sources.

5.1 Credibility Evaluation:

Criterion 1: The materials in the surveillance capsules must be those which are the controlling materials with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR 50, "Fracture Toughness Requirements" as follows:

> "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 and regard to radiation damage."

The Palisades reactor vessel consists of the following beltline region materials [1, 31]:

- Intermediate Shell, Axial Welds 2-112 A/B/C, material heat No. W5214,
- Lower Shell, Axial Welds 3-112 A/B/C, material heat No. W5214 and 34B009,
- Intermediate to Lower Shell, Circumferential Weld 9-112, material heat No. 27204,
- Intermediate Shell, Plate D-3803-1, material heat No. C-1279,
- Intermediate Shell, Plate D-3803-2, material heat No. A-0313,
- Intermediate Shell, Plate D-3803-3, material heat No. C-1279,

- Lower Shell, Plate D-3804-1, material heat No. C-1308A,
- Lower Shell, Plate D-3804-2, material heat No. C-1308B,
- Lower Shell, Plate D-3804-3, material heat No. B-5294.

The Palisades reactor vessel was designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition, including all addenda through Winter 1965 [21]. The Palisades reactor vessel surveillance program was originally developed with the intent to comply, where possible, with the guidance of ASTM E185-66, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors" [22]. At the time that the Palisades surveillance capsules were built, 10 CFR50 Appendices G and H did not exist.

5.1.1 Description of Original Palisades Surveillance Capsule Program

ASTM E 185-66 [22] describes the requirements for test specimens. ASTM E 185-66 requires the base metal specimen be from "...one heat with the highest initial ductile-brittle transition temperature", also known as the nil-ductility transition temperature (NDTT). Drop weight tests of Palisade's beltline samples identified five of the plates in contention for the highest initial NDTT at –30°F. The base material from shell plate D-3803-1 was selected over the other base metal specimens for the capsule base metal because it had the highest initial RT_{NDT} temperature [31]. ASTM E 185-66 requires a sample to represent one vessel weld if a weld occurs in the irradiated region. The original Palisades surveillance weld specimens were fabricated with the same procedure used to fabricate the reactor vessel axial welds, and were fabricated with a similar filler wire and fluxes as the reactor vessel beltline welds. However, the original Palisades surveillance capsules did not contain limiting axial weld heat No. W5214.

5.1.2 Description of Supplemental Surveillance Capsules SA-60-1 and SA-240-1

At the end of Cycle 11, the Palisades surveillance capsule program was augmented to contain two supplemental surveillance capsules, designated as SA-60-1 and SA-240-1, installed in the capsule holders located on the core support barrel. The new surveillance capsules, SA-60-1 and SA-240-1, included welds fabricated with weld wires of identical heats to those of the Palisades reactor vessel beltline welds. Surveillance capsule SA-60-1 and SA-240-1 contained test specimens from the following material heat No.'s: W5214, 34B009, 27204, and standard reference material HSST-02. All of these materials are the same heats as the materials used to fabricate portions of the reactor vessel that surround the active core and adjacent regions of the reactor vessel.

Table 4 provides a tabulation of the specimens included in the Palisades supplemental surveillance capsules SA-60-1 and SA-240-1.

Table 4.
Test Specimens Contained in Palisades Capsules SA-60-1 and SA-240-1

Material Description	Tension	Standard Charpy V-Notch Impact	18 mm Charpy V-Notch Inserts
Weld Metal W5214	---	---	42 (39)*
Weld Metal 34B009	---	---	36 (39)*
Weld Metal 27204	3	12	36
Correlation Monitor Material, HSST Plate 02 (Heat No. A1195-1)	---	12	---

* number of specimens in SA-60-1 capsule [15]

Capsules SA-60-1 and SA-240-1 were removed from the Palisades reactor vessel at the end of cycles 13 and 14, respectively. Twelve Charpy V-notch specimens made from weld heat No. W5214 were tested in capsule SA-60-1 [15], and twelve Charpy specimens from heat No. W5214 were tested in capsule SA-240-1 [14]. Twelve Charpy V-notch specimens made from the HSST-02 correlation monitor material were tested from capsule SA-240-1 [14].

Because weld heat No. W5214 in the supplemental capsules matches the limiting axial welds, Criterion 1 is met for the Palisades reactor vessel.

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper shelf energy unambiguously.

Criterion 2 is satisfied if the Charpy energy data for the surveillance capsules containing weld heat No. W5214 can be fitted to determine the 30 ft-lb temperature (T_{30}) and upper shelf energy (USE) unambiguously. An accurate determination of the 30 ft-lb shift (ΔT_{30}) values is the reason these data were re-evaluated. The TANH curve fit method provides an accurate and reproducible determination of these values and can be used to establish the T_{30} and USE values for a given Charpy data set [24]. Unirradiated and irradiated Charpy energy versus temperature data for the weld metal were fitted and plotted using the CVGRAPH hyperbolic tangent curve fitting program [4]. The Charpy energy fitted results for the eleven surveillance capsules, including the calculated 30 ft-lb temperatures and upper shelf energy values, are shown in the Appendix G and summarized in Table 3. Based on engineering judgment by looking at the fitting parameters and the plots, the scatter in the data is small enough, and the correlation coefficients are high enough, to permit the determination of the 30 ft-lb temperature and upper shelf energy of the surveillance weld materials unambiguously. Hence, Criterion 2 is met for all the surveillance capsules evaluated here which contain weld metal heat No. W5214.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Position 2 (surveillance data available) normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fails this criterion for use in shift calculations, they may be credible for determining decrease in upper shelf energy if the upper shelf can be clearly determined, following the definition in ASTM E185.

The functional form of the least squares method as described in paragraph (c)(2) of 10CFR 50.61 will be utilized. A best-fit line is generated for this data to determine if the scatter of the ΔRT_{NDT} values about this line is less than 28°F for weld metal heat No. W5214.

The Palisades limiting weld metal will be evaluated for credibility. This weld is made from weld heat No. W5214. This weld metal is also contained in the Indian Point Unit 2, Indian Point Unit 3, and H. B. Robinson Unit 2 surveillance programs. Since the welds in question utilized data from other surveillance programs, the recommended NRC methods for determining creditability will be followed. Of the recommended methods, Case 4 most closely represents the situation listed above for the Palisades surveillance weld metal.

Case 4a Credibility Assessment – Palisades W5214 Data Only

The data most representative for the Palisades limiting vessel weld are the supplemental surveillance capsules containing weld heat No. W5214 since the irradiation environment of the surveillance capsules and the reactor vessel are the same. The data requires the least adjustment. An adjustment can be made for the difference between the chemistry of the capsule specimens (CF = 266.5°F) and the best estimate chemistry of the vessel (CF = 230.73°F) using the ratio procedure. The updated fluence and ratio adjusted shift values were used to calculate a new least-squares fitted chemistry factor. The Palisades capsule data are shown in Table 5 along with the fitted solution (i.e., mean shift prediction) result, and the comparison of the measured – predicted scatter from the fitted CF of 198.8°F. A plot of the measured ΔT_{30} vs. fluence results for the Palisades supplemental capsule weld (W5214) is shown in Figure 2 along with the +/- 1σ bounds for credible data scatter. The data clearly fall within the 1-sigma scatter band for credible surveillance data and the margin term can be reduced when using credible data.

Based on criterion 3, the Palisades surveillance data is credible since the scatter is less than 28°F for both of these surveillance capsules.

Table 5.
Evaluation of Palisades Surveillance Data Results for Weld Heat No. W5214

Weld Heat No. W5214 Surveillance Data

Palisades Capsule Number	Cu (wt%)	Ni (wt%)	Table CF (F)	Revised Fluence (n/cm^2)	FF	Refitted ΔRTndt (F)	ΔRTndt (F) Adjusted to Vessel CF (F)	Predicted ΔRTndt (F)	Measured - Predicted ΔRTndt (F)
SA-60-1	0.307	1.045	266.5	1.50E+19	1.11	259.0	224.2	221.13	3.11
SA-240-1	0.307	1.045	266.5	2.38E+19	1.23	280.1	242.5	245.30	-2.80
								Fitted CF =	198.8°F

Case 4b Credibility Assessment - All W5214 Surveillance Capsule Data

Following the guidance in Case 4 [19], the data from all sources should also be considered. For weld heat No. W5214 there are a total of eleven surveillance capsules from Palisades, Indian Point Unit 2, Indian Point Unit 3, and H.B. Robinson Unit 2. Since data are from multiple sources, the data must be adjusted first for chemical composition differences and then for irradiation temperature differences before determining the least-squares fit.

For a credibility determination, the measured and refitted T_{30} shift data for all the relevant plant data was normalized to the mean chemistry factor of the vessel (230.73°F) using the ratio procedure and then to the mean operating temperature (535.2°F) for the Palisades vessel (see Table H-7). The fitted CF value, shown in Table 6, is determined to be 227.74°F for this case. The results for (measured – predicted) scatter for all the W5214 surveillance data results are shown in Table 7. The results for all the surveillance capsule data are plotted in Figure 3 along with the +/- 2σ scatter bands. The scatter in the measured – predicted values exceeds 28°F (1-sigma) for a few points. Four of the measured - predicted ΔRT_{NDT} values are outside the 1-sigma band of 28°F, but all data points are within the 56°F (2-sigma) scatter band for welds. According to 10CFR50.61 paragraph (c)(2)(iv), the use of results from the plant-specific surveillance program may result in an RT_{NDT} that is higher or lower than that determined from the chemistry of the weld and a chemistry factor using the tables. If the CF value is higher, it must be used for vessel RT_{PTS} predictions, if the CF value is lower, it may be used.

The chemistry factor from paragraph (c)(1) is 230.73°F, and the adjusted chemistry factor using the Palisades surveillance capsule data is 227.74°F. It is noted that per NRC guidance that it is possible to use a lower value of chemistry factor based upon all sources of surveillance capsule data with a full margin term (i.e., 56°F) if the data is credible in all other ways but the scatter.

In summary, the (measured – predicted) scatter for all the W5214 weld data is within the acceptable range of 56°F for a wide range of fluence. For this case, the surveillance capsule fluence ranges between 2.63×10^{18} n/cm^2 to 4.49×10^{19} n/cm^2. Therefore, the weld data meets this criterion, and the Palisades surveillance program weld metal chemistry factor to be used for determining RT_{PTS} and RT_{NDT} is 227.74°F in combination with a full (2-sigma) margin term.

Table 6.
Evaluation of all Surveillance Capsule Results Containing Weld Heat No. W5214

Capsule	%Cu	%Ni	Table CF (F)	Revised Fluence (n/cm^2)	Fluence Factor FF	Irrad. Temp. Ti (F)	Measured (Refitted) ΔRTndt (F)	Ratio Adjusted ΔRTndt (F)	Chem. & Temp. Adj. ΔRTndt (F)	FF^2	ΔT30 x FF
SA-60-1	0.307	1.045	266.5	1.50E+19	1.11	535.0	259	224.2	224.0	1.237	249.186
SA-240-1	0.307	1.045	266.5	2.38E+19	1.23	535.7	280.1	242.5	243.0	1.522	299.830
HB2 T	0.34	0.66	217.7	3.87E+19	1.35	547	289.1	306.4	318.2	1.820	429.263
HB2 V	0.34	0.66	217.7	5.30E+18	0.82	547	208.8	221.3	233.1	0.677	191.749
HB2 X	0.34	0.66	217.7	4.49E+19	1.38	547	265.6	281.5	293.3	1.906	404.943
IP2 V	0.20	1.03	226.3	4.92E+18	0.80	524.0	197.5	201.4	190.2	0.643	152.544
IP2 Y	0.20	1.03	226.3	4.55E+18	0.78	529.1	193.9	197.7	191.6	0.610	149.601
IP3 T	0.16	1.12	206.2	2.63E+18	0.64	539.4	149.8	167.6	171.8	0.405	109.400
IP3 Y	0.16	1.12	206.2	6.92E+18	0.90	539.5	171.1	191.5	195.8	0.804	175.543
IP3 Z	0.16	1.12	206.2	1.04E+19	1.01	538.9	228.3	255.5	259.2	1.022	262.003
IP3 X	0.16	1.12	206.2	8.74E+18	0.96	539.7	192.5	215.4	219.9	0.926	211.596
									SUM	11.573	2635.658
Vessel Best Estimate CF =			230.73°F		Mean Vessel T = 535.2°F						
								Least Squares Fitted CF = 227.74°F			

(a) Measured capsule weld materials Cu and Ni values obtained from [6, 12, 13, 14, 15, 20, 26]
(b) Fluence values obtained from Reference 18
(c) Time-weighted average temperatures obtained from References 23 and 33 and Appendix H
(d) Refitted Charpy V-notch shift data obtained from Reference 32 and Appendix G

Table 7.
Scatter in Fit to all Surveillance Capsule Results Containing Weld Heat No. W5214

Capsule	Irrad. Temp. Ti (F)	Revised Fluence (n/cm^2)	Fluence Factor FF	Adjusted ΔRTndt (F)	Predicted ΔRTndt (F)	Adjusted - Predicted (F)
SA-60-1	535	1.50E+19	1.11	224.0	253.31	**-29.27**
SA-240-1	535.7	2.38E+19	1.23	243.0	281.00	**-38.00**
HB2 T	547	3.87E+19	1.35	318.2	307.23	10.97
HB2 V	547	5.30E+18	0.82	233.1	187.34	**45.75**
HB2 X	547	4.49E+19	1.38	293.3	314.44	-21.14
IP2 V	524	4.92E+18	0.80	190.2	182.69	7.48
IP2 Y	529.1	4.55E+18	0.78	191.6	177.83	13.77
IP3 T	539.4	2.63E+18	0.64	171.8	145.01	26.81
IP3 Y	539.5	6.92E+18	0.90	195.8	204.23	-8.48
IP3 Z	538.9	1.04E+19	1.01	259.2	230.24	**28.92**
IP3 X	539.7	8.74E+18	0.96	219.9	219.14	0.76

Note: four of the eleven (measured – predicted) data points exceed the 1 standard deviation of 28°F for credible data for welds. All eleven (measured – predicted) data points fall within 2 standard deviations of 56°F for welds.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The Palisades supplemental surveillance capsules SA-60-1 and SA-240-1 were located in the reactor vessel between the core barrel and the vessel wall opposite the center of the core. These supplemental surveillance capsules were installed in the capsule holders located on the core support barrel. Table 8 provides a history of the time-weighted temperature for the Palisades supplemental surveillance capsules and reactor vessel wall.

Table 8.

History of Time-Weighted Operating Temperature for Palisades

Operating Cycle Number	Cycle Length[(a)] (EFPD)	Cycle Average Vessel Temp.[(b)] (°F)	Surveillance Capsule Removed	Time Weighted Capsule Avg. T (°F)	Time Weighted Vessel Avg. T (°F)
1	371.7	523			
2	440.1	529			
3	342.5	534			
4	321.0	536			
5	386.7	536			
6	326.7	536			
7	362.5	536			
8	366.1	537			
9	292.5	534			
10	349.7	534			
11	421.9	533			
12	399.3	534			
13	419.6	536	SA-60-1	535.0	
14	449.3	537	SA-240-1	535.7	
15	401.3	537			
16	444.3	537			
17	493.1	537			
18	472	537			
19	459.2	537			
20	499.8	537			
21	519.2	537			
22	498.8	537			535.2

(a) Cycle length (EFPD) values obtained from Reference 23

(b) Cycle average vessel temperatures obtained from Reference 28

The location of the specimens with respect to the reactor vessel beltline assured that the reactor vessel wall and the specimens have experienced equivalent operating conditions such that the temperatures did not differ by more than 25°F. Therefore, this criterion is satisfied for the Palisades capsules.

The Indian Point Unit 2 and Indian Point Unit 3 average surveillance capsule temperatures have been also reviewed and updated. The H. B. Robinson Unit 2 average capsule temperature was confirmed by the utility. The time-weighted average temperature values for these capsules are listed in Table 2, and the method for calculating these temperatures is given in Appendix H.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band for that material.

The Palisades supplemental surveillance capsules, SA-60-1 and SA-240-1, both contain standard reference material HSST02 plate. Plots of the Charpy energy versus temperature for the irradiated condition of correlation monitoring material (HSST Plate 02, Heat A1195-1) from SA-60-1 and SA-240-1 are documented in BAW-2341 Rev 2 [15] and BAW-2398 [14], respectively. Charpy energy versus temperature for the unirradiated correlation monitoring material (HSST Plate 02, Heat A1195-1) is taken from NUREG/CR-6413, ORNL/TM-13133 [24]. Tables 9 and 10 provide the updated calculation of (measured – predicted) scatter versus fast fluence in the correlation monitor material (HSST 02) data. Figure 4 (from Reference 24) shows that the measured scatter band for the correlation monitor materials is 50°F.

Table 9.
Correlation Monitor Material HSST Plate 02
Calculation of Fitted CF

Capsule	Fluence ($\times 10^{19}$) (a)	Fluence Factor (FF) (b)	ΔRT_{NDT} (c) (°F)	FF * ΔRT_{NDT}	FF^2
SA-60-1	1.5	1.112	113.7	126.4344	1.2365
SA-240-1	2.38	1.234	140.9	173.871	1.5223
			Sum	300.305	2.7588
CF Surveillance weld = Σ (FF x RT_{NDT}) / Σ (FF^2)= 300.305/2.7588 = 108.853					
Slope of best fit line is 108.853					

Notes:

(a) Calculated fluence ($\times 10^{19}$ n/cm^2, E>1.0 MeV)

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$

(c) Irradiated values of 30 ft-lb Transition Temperature From BAW-2341 Rev 2 and BAW-2398 [15, 14]

Table 10.
Correlation Monitor Material HSST Plate 02
Calculation of Measured – Predicted Scatter

Capsule	Fluence (x 10^{19}) (a)	Fluence Factor (FF) (b)	ΔRT_{NDT} (c)	Predicted ΔRT_{NDT}	(Measured – Predicted) ΔRT_{NDT}
SA-60-1	1.5	1.112	113.7	121.044	-7.344
SA-240-1	2.38	1.234	140.9	134.324	6.575
Where predicted ΔRT_{NDT} = (slope $_{best fit}$)*(Fluence Factor) Slope of best fit line is 108.853					
Notes: (a) Calculated fluence (x 10^{19} n/cm^2, E>1.0 MeV) (b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$ (c) Irradiated values of 30 ft-lb Transition Temperature From BAW-2341 Rev 2 and BAW-2398 [15, 14]					

Table 10 shows that the scatter in these data is less than 50°F, which is the allowable scatter in NUREG/CR-6413, ORNL/TM-13133 [24]. Thus, criterion 5 is satisfied for the correlation monitor materials.

5.2 Palisades Vessel Fluence Evaluation

Fluence in the Palisades vessel beltline has been tracked to manage the PTS issue. The fluence projections are important to be able to predict the future levels of embrittlement in the vessel beltline materials. Calculations of the neutron exposure of the Palisades reactor pressure vessel were previously completed and documented in WCAP-15353, Revision 0 [3]. That evaluation, along with the benchmarking method, was submitted for review by the NRC Staff and the methodology and the final results were approved as part of the PTS evaluation in 2000 [1]. The previous evaluation determined that the peak fluence at the clad-to-base-metal interface at the 60° limiting axial weld was 1.158×10^{19} n/cm^2 (E > 1 MeV) at the end of Cycle 14 (i.e., October 1999) [34]. Since then, ten more years of plant operation has occurred and, as a result, the vessel has accumulated additional fluence. Recently, Westinghouse provided an updated fluence assessment for the Palisades vessel beltline region that includes cycle specific analysis for additional operating cycles for which the design has been finalized and operations are known (Cycles 15 through 21) and projections for future operation based on the best available knowledge as a function of EFPY and estimated calendar dates [18]. The calculated and projected neutron fluence values for the limiting 60° weld location are given in Table 11. Note: the cycle specific projections for the designs of Cycles 21 and beyond were provided by Entergy and include an assumed load factor of 95% for future plant operation [18].

Table 11.

Calculated and Projected Fluence Values at 60° Weld Location [18]

End of Fuel Cycle	Estimated Calendar Date	Cumulative Time (EFPY)	Neutron Fluence @ 60° n/cm^2 (E > 1 MeV)
14	October 1999	14.4	1.158E+19
15	March 2001	15.5	1.196E+19
16	March 2003	16.7	1.240E+19
17	September 2004	18.0	1.282E+19
18	April 2006	19.3	1.326E+19
19	September 2007	20.6	1.369E+19
20	March 2009	22.0	1.419E+19
21	October 2010	23.4	1.472E+19
22	April 2012	24.7	1.520E+19
23	October 2013	26.1	1.571E+19
24	April 2015	27.4	1.619E+19
25	October 2016	28.8	1.670E+19
26	April 2018	30.2	1.721E+19

6.0 DISCUSSION

The results for the surveillance capsules containing weld heat No. W5214 have been re-evaluated for applicability to the Palisades vessel. The refitted Charpy data results have been incorporated along with updates to the average capsule irradiation temperatures, corrections to account for chemistry differences, and results of the revised fluence calculations for the capsules and for the Palisades vessel have been included. Four cases are considered for prediction of the date to reach the PTS screening criteria limit.

The first case is Position 1 of the PTS Rule using the current licensing basis method and considering the revised fluence calculations and projections, as shown in Table 11. The chemistry factor for the weld heat No. W5214 was based on best-estimate Cu = .213%, Ni = 1.01%, and a CF = 231.08°F. The maximum fluence limit was calculated to be 1.584×10^{19} n/cm^2 according to the embrittlement prediction method when surveillance data is not available. That prediction of RT_{NDT} shift is shown in Table 12, and the corresponding date to reach the PTS screening criteria limit of 270°F for axial welds is March 2014 using the fluence interpolation given in Table 16. The current licensing basis date to reach the PTS screening criteria date is

January 2014 as given in Reference 1. It has been determined that the Palisades plant is still operating within that licensing basis.

The second case follows Position 1 of the PTS Rule but considers that the actual best-estimate chemistry for weld heat No. W5214 has slightly lower nickel as determined by the CEOG report in 1998 [20]. This evaluation was performed after the initial PTS submittal in 1995 and, using this information, the revised Ni = 1.007% which gives a new value of CF = 230.73°F. It is permitted under the PTS Rule in 10CFR50.61 to use the best-estimate values for Cu and Ni, however there is only a slight difference in the maximum fluence to reach the PTS screening criteria limit (i.e., 1.595×10^{19} n/cm^2) as shown in Table 13. The projected date to reach that limit using the interpolated fluence projections is July 2014 as shown in Table 16. Although this case does not show much additional margin from the January 2014 date, it is provided here to show that there is still slightly more time to be gained within the Position 1 approach of the PTS Rule method before the vessel reaches the 270°F screening criteria limit.

The third case considered plant-specific surveillance data from the Palisades supplemental capsules containing weld heat No. W5214. This case is labeled as Case 4a per the guidance document for use of surveillance data [19]. The Case 4a credibility assessment calculated a fitted chemistry factor of 198.8°F from the two Palisades capsule data points., The data were deemed to be credible based on meeting all the credibility criteria including scatter within the 1-sigma (i.e., 28°F) scatter bounds. The limiting fluence for the vessel for this case is shown in Table 14. Using these results and a reduced margin term to account for credible data, the projected date to reach the PTS screening criteria limit would be beyond 2034 for the limiting vessel weld heat No. W5214, as shown in Table 18. Note: It is likely that some other beltline material would become limiting if this case was used for weld heat No. W5214. However, this case demonstrates that surveillance data can provide significant improvement in determining the effects of embrittlement on the limiting vessel beltline weld material.

The fourth case, permitted under the PTS Rule, is to use all sources of surveillance data that match the limiting weld heat No. W5214. This is designated as Case 4b, and the credibility assessment determined the fitted CF = 227.74°F. The data meet credibility criteria 1, 2, 4, & 5, and the scatter of the (measured – predicted) data was within 2-sigma (i.e., 56°F) such that it can be considered to be credible data for the chemistry factor, however the margin term, σ_Δ, cannot be reduced in half. Use of Case 4b for the Palisades vessel is acceptable because the (measured – predicted) scatter in the weld data is within the acceptable range of 56°F for a wide range of fluence. The limiting fluence for Case 4b is 1.685×10^{19} n/cm^2 (E > 1 MeV) as shown in Table 15. Table 17 interpolates the vessel fluence and shows the projected date to reach the screening criteria limit is April 2017, a difference of three years compared to the first case using the current licensing basis and Position 1 approach. A summary of the four cases considered in this analysis is given in Table 18.

Table 12.

Limiting Fluence Determination for Current Licensing Basis (Case 1)

FLUENCE=	1.584E+19	n/cm^2
f =	1.584	
f FACTOR=	f^(0.28-0.1*@LOG(f))	
=	1.1270	
CHEM FACTOR =	231.08	°F
ΔRTNDT =	260.5	°F
RTNDT0=	-56.0	°F
MARGIN=	65.5	°F
TOTAL RTNDT =	270.0	°F

Table 13.

Limiting Fluence Determination for Revised Best-Estimate CF Value (Case 2)

FLUENCE=	1.595E+19	n/cm^2
f =	1.595	
f FACTOR=	f^(0.28-0.1*@LOG(f))	
=	1.1289	
CHEM FACTOR =	230.73	°F
ΔRTNDT =	260.5	°F
RTNDT0=	-56.0	°F
MARGIN=	65.5	°F
TOTAL RTNDT =	270.0	°F

Table 14.

Limiting Fluence Determination for Case 4a

FLUENCE=	5.438E+19	n/cm^2
f =	5.438	
f FACTOR=	f^(0.28-0.1*@LOG(f))	
=	1.4185	
CHEM FACTOR =	198.8	°F
ΔRTNDT =	282.0	°F
RTNDT0=	-56.0	°F
MARGIN=	44.0*	°F
TOTAL RTNDT =	270.0	°F

* reduced margin term based on credible surveillance data

Table 15.

Limiting Fluence Determination for Case 4b

FLUENCE=	1.685E+19	n/cm^2
f =	1.685	
f FACTOR=	f^(0.28-0.1*@LOG(f))	
=	1.1437	
CHEM FACTOR =	227.74	°F
ΔRTNDT =	260.5	°F
RTNDT0=	-56.0	°F
MARGIN=	65.5	°F
TOTAL RTNDT =	270.0	°F

Table 16.

Interpolation of PTS Limit Date Based on Current Licensing Basis and Revised Fluence

Date	Neutron Fluence @ 60° n/cm^2 (E > 1 MeV)
November 2013	1.571E+19
December 2013	1.574E+19
January 2014	1.577E+19
February 2014	1.579E+19
March 2014	1.582E+19*
April 2014	1.585E+19
May 2014	1.588E+19
June 2014	1.591E+19
July 2014	1.594E+19**
August 2014	1.596E+19
September 2014	1.599E+19
October 2014	1.602E+19
November 2014	1.605E+19
December 2014	1.608E+19
January 2015	1.611E+19
February 2015	1.613E+19
March 2015	1.616E+19
April 2015	1.619E+19

* Maximum fluence limit = 1.584×10^{19} n/cm^2 for current licensing basis material case, CF = 231.08°F

**Maximum fluence limit = 1.595×10^{19} n/cm^2 for revised best-estimate weld, CF = 230.73°F

Table 17.

Interpolation of PTS Limit Date Based on Limiting Fluence for Case 4b

Date	Neutron Fluence @ 60° n/cm^2 (E > 1 MeV)
November 2016	1.670E+19
December 2016	1.673E+19
January 2017	1.676E+19
February 2017	1.679E+19
March 2017	1.682E+19
April 2017	1.685E+19*
May 2017	1.688E+19
June 2017	1.691E+19
July 2017	1.694E+19
August 2017	1.697E+19
September 2017	1.700E+19
October 2017	1.703E+19
November 2017	1.706E+19
December 2017	1.709E+19
January 2018	1.712E+19
February 2018	1.715E+19
March 2018	1.718E+19
April 2018	1.721E+19

* Maximum fluence limit = 1.685×10^{19} n/cm^2 for Case 4b using revised fluence and W5214 surveillance data CF = 227.74°F and full (2-sigma) margin term

Table 18.
Projected Maximum Fluence and Estimated PTS Limit Dates for Palisades Weld W5214

Case No.	CF	IRTNDT	Fluence	FF	□RT_{NDT}	Margin	RTPTS	Est. PTS Date
	(°F)	(°F)	(10^{19} n/cm^2)		(°F)	(°F)	(°F)	
(1) Current LB w/revised fluence	231.08	-56	1.584	1.1270	260.5	65.5	270	March 2014
(2) Current LB w/revised fluence and revised CF value	230.73	-56	1.595	1.1289	260.5	65.5	270	July 2014
4a	198.80	-56	5.438	1.4185	282	44	270	> 2034*
4b	227.74	-56	1.685	1.1437	260.5	65.5	270	April 2017

* Other beltline materials will likely become more limiting and will affect this date

Case 1 – Current licensing basis CF value for W5214 weld and revised fluence calculation
Case 2 – CEOG best estimate chemistry and CF value for W5214 weld and revised fluence calculation
Case 4a – Use of credible Palisades W5214 surveillance data and revised fluence with reduced margin term
Case 4b – Use of all W5214 surveillance data and revised fluence with full margin term

7.0 SUMMARY AND CONCLUSIONS

The results for all available surveillance capsules containing weld heat No. W5214 have been evaluated for applicability to the Palisades limiting vessel weld. Updates to the surveillance capsule fluences and the projected fluence in the Palisades vessel were also reviewed and included in these analyses. The methods of 10CFR50.61 were applied including options for considering the effects of surveillance data on the projected RT_{NDT} values. Using Position 1 of the PTS Rule (without the use of surveillance data) shows a projected date to reach the PTS screening criteria limit as late as July 2014. However, use of the weld heat No. W214 surveillance data can improve the projections of embrittlement and significantly changes the date to reach the screening criteria limit. Since weld heat no. W5214 is currently identified as the limiting material, the projections using Case 4a with the credible Palisades supplemental surveillance data show that the PTS screening criteria limit of 270°F would not be reached until after 2034; however, other vessel beltline materials would become limiting and that would change that date. For Case 4b, the surveillance data for weld heat No. W5214 were shown to be credible for determination of the CF value, but the scatter in the data would not permit a reduction in the margin term. However, use of the fitted chemistry factor for Case 4b with the revised fluence projections and the full margin term provides a better determination of the vessel embrittlement prediction for the limiting vessel weld. Using all the available weld heat No. W5214 surveillance data, a CF value of 227.74°F was determined for Case 4b and a projected date to reach the screening criteria limit of approximately April 2017 was estimated using the updated fluence projections from Westinghouse.

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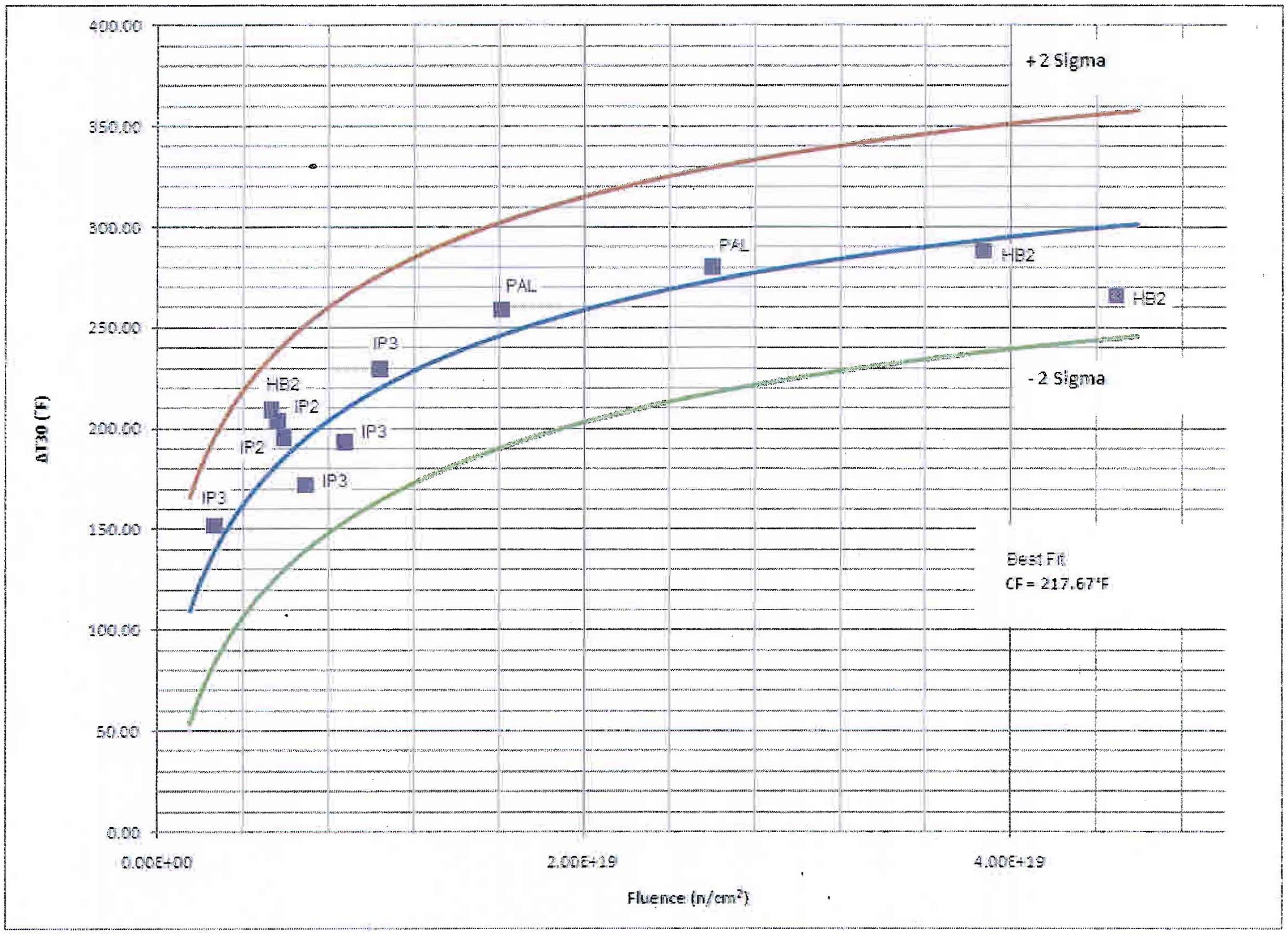


Figure 1. Best Fit to Data for all W5214 Surveillance Data with Reported Fluence and Vendor Shift Values

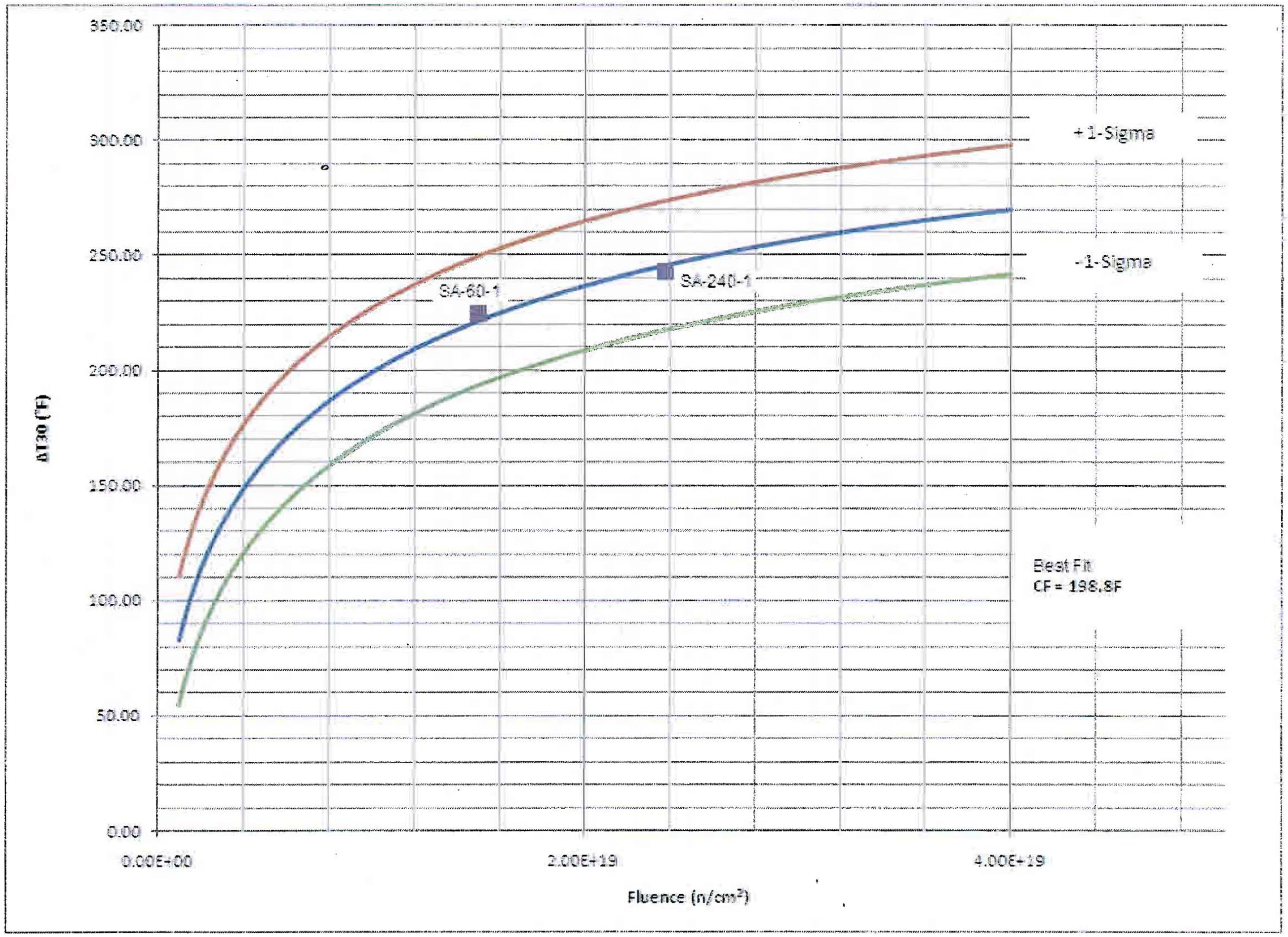


Figure 2. Palisades Supplemental Surveillance Data (W5214) with Revised Fluence and Refitted Shift (Case 4a)

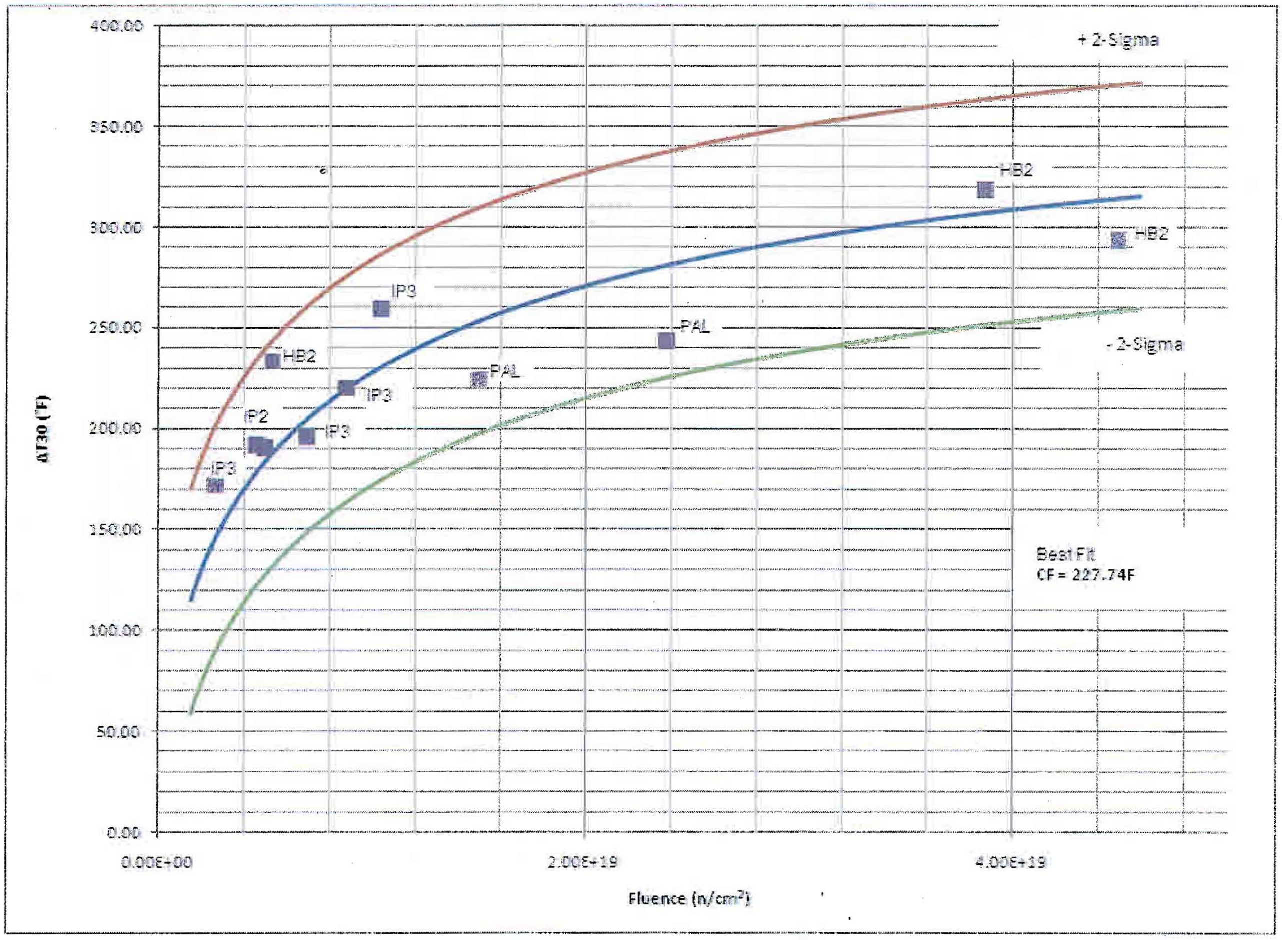


Figure 3. Best Fit for all W5214 Surveillance Data with Revised Fluence and Refitted Shift (Case 4b)

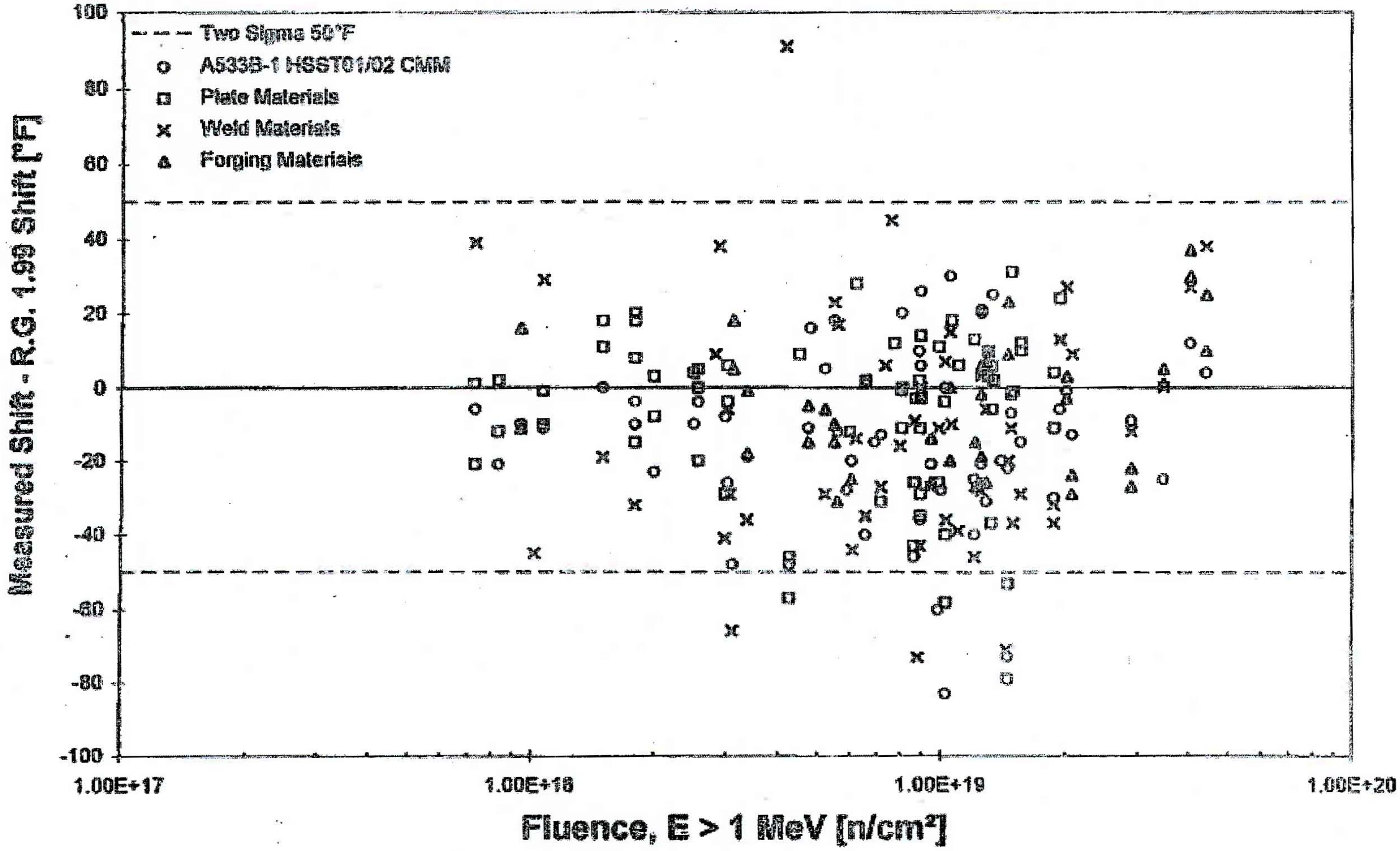


Figure 4. Plot of Residual vs. Fast Fluence for A533B-1 HSST-01/HSST-02 CMM with Companion Materials, the Overall 2-Sigma Scatter is 50°F [24].

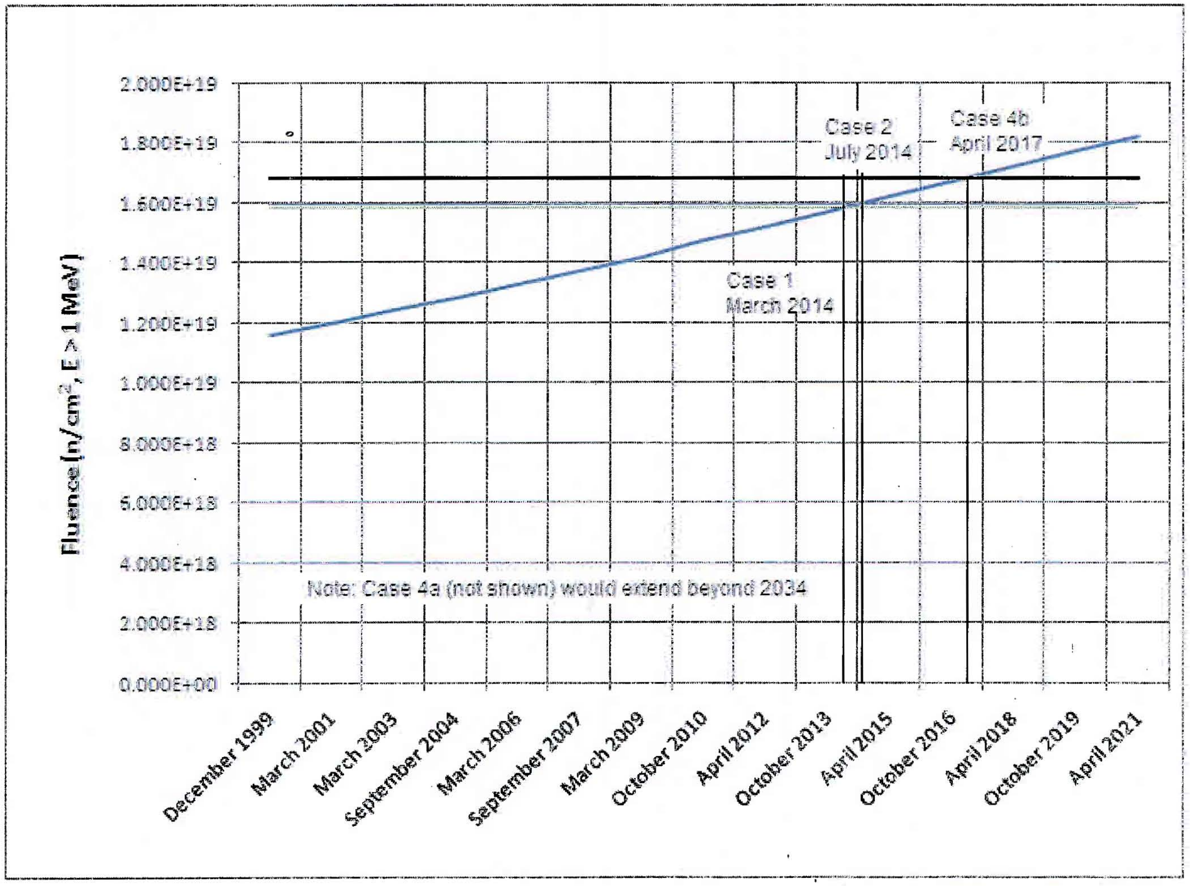


Figure 5. Projected Peak Fluence at 60° Weld Location (from [18]) and Revised RT_{PTS} Limit Dates

APPENDIX A

CEOG DETERMINATION OF BEST-ESTIMATE CHEMISTRY FOR WELD HEAT NUMBER W5214 (from CE NPSD-1119, Rev. 1 [20])

Table 5 (continued)
Best Estimate Copper and Nickel by Heat Number

Heat Number	Copper, %	Nickel, %	Basis
51912	0.156	0.059	weighted mean
51989	0.170	0.165	simple mean
5P5622	0.153	0.077	simple mean
6329637	0.205	0.105	simple Cu, bare wire Ni
83637	0.048	0.066	weighted mean
83640	0.051	0.096	simple mean
83642	0.046	0.086	simple mean
83648	0.042	0.136	simple mean
83650	0.045	0.087	weighted mean
83653	0.042	0.102	weighted mean
86054B	0.214	0.046	simple mean
86054B, 9565	0.213	0.052	mean of indeterminate data
87005	0.054	0.151	weighted mean
88112	0.045	0.200	simple mean
88114	0.043	0.189	simple mean
89476	0.022	0.071	simple mean
89833	0.046	0.059	simple mean
90069	0.040	0.075	weighted mean
90071	0.035	0.079	simple mean
90077	0.036	0.057	simple mean
90099	0.197	0.060	simple Cu, bare wire Ni
90130	0.044	0.133	simple mean
90136	0 269	0.070	weighted mean
90144	0.042	0.075	simple mean
90146	0.039	0.082	weighted mean
90209	0.044	0.126	simple mean
9565	0.213	0.052	Ht. 86045B/9565 data
A8746	0.150	0.13	Avg. Cu with bare wire data: generic Ni
BOLA	0.027	0.913	simple mean
HODA	0.027	0.947	simple mean
W5214 w/Ni200	0.213	1.007	coil wgt'd Cu, Ni200 BE

* See Report CE NPSD-1039, Rev. 02 for method of determination.

APPENDIX B

EXCERPT FROM GENERIC LETTER 92-01 AND RPV INTEGRITY ASSESSMENT NRC/INDUSTRY WORKSHOP ON RPV INTEGRITY ISSUES

REQUIREMENTS AND RECOMMENDATIONS IN 10 CFR 50.61 & RG 1.99 REV. 2

Per 10 CFR 50.61(c)(2):

"To verify that RT_{NDT} for each vessel beltline material is a ***bounding value*** for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the ***reactor vessel operating temperature*** and any ***related surveillance program[5] results***."

Per Footnote 5:

[5] Surveillance program results means any data that demon-strates the embrittlement trends for the limiting beltline material, including but not limited to data from test reactors or from surveillance programs at other plants with or without surveil-lance program integrated per 10 CFR Part 50 , Appendix H.

Per RG 1.99 Rev. 2 Position 2.1 and 10 CFR 50.61(c)(2)(ii)(B):

"if there is ***clear evidence that the copper or nickel content of the surveillance weld differs from that of the vessel weld***, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, ***the measured values of ΔRT_{NDT} should be adjusted by multiplying them by the ratio of the chemistry factor for the vessel weld to that for the surveillance weld***"

$$Ratio\ Adjusted\ \Delta RT_{NDT} = \left(\frac{Table\ CF_{Vessel\ Chem.}}{Table\ CF_{Surv.\ Chem.}}\right) * Measured\ \Delta RT_{NDT}$$

IRRADIATION ENVIRONMENT ADJUSTMENTS

- Irradiation temperature and fluence are first order environmental variables in assessing irradiation damage
  - Other variables are believed to be less significant contributors
  - Must account for differences in temperature between surveillance specimens and vessel
- Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}

RECENT ISSUES ON USE OF SURVEILLANCE DATA

- "Best-fit line" through surveillance data (plot of ΔRT_{NDT} vs. fluence) must go through origin
- Using a CF determined from non-credible surveillance data
- Correcting for chemical composition (ratio procedure)
- Correcting for irradiation environment (temperature)
- Appropriate chemical composition for multiple surveillance capsules from a single source (i.e., mean value for all capsules from that source)
- Appropriate normalizing parameters for surveillance data when assessing credibility (i.e., mean of surveillance data) and determining CF (i.e., best estimate of vessel)

CASE 4: SURVEILLANCE DATA FROM PLANT AND OTHER SOURCES

Surveillance data (Weld metal)

Capsule	NSSS Vendor	Cu	Ni	Irrad. Temp. (°F)	Fluence (10^{19} n/cm^2)	Measured ΔRT_{NDT} (°F)	Adjusted ΔRT_{NDT} [using Ratio and Temperature (550°F)] (°F)
Plant A - 1	B&W	0.37	0.70	556.0	0.779	214.0	196.0
Plant B - 1	B&W	0.33	0.67	556.0	0.107	124.0	126.0
Plant B - 2	B&W	0.33	0.67	556.0	0.966	203.0	202.5
Plant C - 1	B&W	0.33	0.67	556.0	0.830	182.0	182.2
Plant C - 2	B&W	0.33	0.67	556.0	0.968	222.0	221.0
Plant X - 1	West.	0.24	0.66	536.0	0.281	165.0	172.1
Plant X - 2	West.	0.24	0.66	536.0	1.940	240.0	257.6

Vessel being analyzed is Plant X

Best estimate chemistry for heat (Weld metal)

0.34% Cu, 0.68% Ni ⇒ Table $CF_{Vessel\ Chem.}$ = 220.6°F

Credibility assessment - Using Plant "X" data only

No temperature adjustment needed

Determine Surveillance CF for Plant X data only (214.8°F)

Capsule	Cu	Ni	Irrad. Temp. (°F)	Fluence (10^{19} n/cm^2)	Measured ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	(Measured - Predicted) ΔRT_{NDT} (°F)
Plant x - 1	0.24	0.66	536.0	0.281	165.0	140.3	24.7
Plant x - 2	0.24	0.66	536.0	1.940	240.0	253.6	-13.6

Data are credible since scatter is less than σ_{Δ} (28°F) for all surveillance specimens

CASE 4: SURVEILLANCE DATA FROM PLANT AND OTHER SOURCES (cont'd)

Determination of CF - Plant "X" data only

No temperature adjustments needed

Adjust measured ΔRT_{NDT} to chemical composition of VESSEL

Table $CF_{Surv.\ chem.}$ = 182.9°F

Determine Surveillance CF

No temperature adjustment needed

Surveillance CF = 259.0°F

Final Result: assuming $RT_{NDT(U)}$ = -7.0°F; M = 49.8; F = 0.8745

RT_{NDT} = -7.0 + 49.8 + (259.0 * 0.8745) = 269.2°F

CASE 4: SURVEILLANCE DATA FROM PLANT AND OTHER SOURCES (cont'd)

Credibility assessment - All data

Data adjusted to mean chemical comp. of surveillance capsules

Cu = 0.31%
Ni = 0.67%

Data adjusted to mean temperature of surveillance capsules

Temp. = 550°F

Determine Surveillance CF (218.4°F)

Capsule	Cu	Ni	Irrad. Temp. (°F)	Fluence (10^{19} n/cm^2)	Adjusted ΔRT_{NDT} [using Ratio and Temperature (550°F)] (°F)	Predicted ΔRT_{NDT} (°F)	(Adjusted - Predicted) ΔRT_{NDT} (°F)
Plant A - 1	0.37	0.70	556.0	0.779	196.0	203.1	-7.1
Plant B - 1	0.33	0.67	556.0	0.107	126.0	94.1	31.9
Plant B - 2	0.33	0.67	556.0	0.866	202.5	209.6	-7.1
Plant C - 1	0.33	0.67	556.0	0.830	182.2	207.0	-24.8
Plant C - 2	0.33	0.67	556.0	0.968	221.0	216.4	4.6
Plant X - 1	0.24	0.66	536.0	0.281	172.1	142.8	29.4
Plant X - 2	0.24	0.66	536.0	1.940	257.6	258.0	-0.4

Data are not credible since scatter is greater than σ_Δ (28°F) for several surveillance specimens

APPENDIX C

PALISADES SUPPLEMENTAL MATERIALS SURVEILLANCE PROGRAM RESULTS FOR WELD NO. W5214

BAW-2398
May 2001

Test Results of Capsule SA-240-1
Consumers Energy
Palisades Nuclear Plant

-- Reactor Vessel Material Surveillance Program --

by

M. J. DeVan

FTI Document No. 77-2398-00
(See Section 7 for document signatures.)

Prepared for

Consumers Energy

Prepared by

Framatome ANP, Inc.
3315 Old Forest Road
P. O. Box 10935
Lynchburg, Virginia 24506-0935

FRAMATOME ANP

Executive Summary

This report describes the results of the tests performed on the specimens contained in the second supplemental reactor vessel surveillance capsule (Capsule SA-240-1) from the Consumers Energy Palisades Nuclear Plant. The objective of the program is to monitor the effects of neutron irradiation on the mechanical properties of the reactor vessel materials by testing and evaluation of Charpy impact specimens.

Supplemental Capsule SA-240-1 was removed from the Palisades reactor vessel at the end-of-cycle 14 (EOC-14) for testing and evaluation. The test specimens included modified 18mm Charpy V-notch inserts for three weld metals fabricated with weld wire heats W5214, 34B009, and 27204 and standard Charpy V-notch specimens fabricated from the correlation monitor plate material, HSST Plate 02. The weld metal Charpy inserts were reconstituted to full size Charpy V-notch specimens. The reconstituted weld metals along with HSST Plate 02 material were Charpy impact tested. The results of these tests are presented in this document.

Table 3-2. Chemical Composition of Palisades Capsule SA-240-1 Surveillance Materials

Element	Chemical Composition, wt%			
	Weld Metal W5214(a)	Weld Metal 34B009(a)	Weld Metal 27204(b)	Correlation Monitor Plate Heat No. A1195-1(c)
C	0.094	0.110	0.142	0.23
Mn	1.161	1.269	1.281	1.39
P	0.009	0.012	0.009	0.013
S	0.012	0.016	0.008	0.013
Si	0.252	0.181	0.217	0.21
Ni	1.045(b)	1.121(b)	1.067	0.64
Cr	0.040	0.040	0.071	---
Mo	0.510	0.543	0.525	0.50
Cu	0.307(b)	0.185(b)	0.194	0.17

(a) AEA Technology analysis.[9]
(b) Analysis provided by Consumers Energy.[10]
(c) ORNL analysis.[11]

Table 4-3. Charpy Impact Results for Palisades Capsule SA-240-1 Irradiated Weld Metal W5214

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, mil	Shear Fracture, %
2AL1	70	14	9	0
2AH3	125	15.5	6	20
AW5	175	24.5	15	10
2AJ1	200	13	10	40
AU4	200	26.5	15	35
2AL3	225	25	11	50
AP1	250	40	26	65
AU5	300	54.5	47	95
2AE5	350	49	42	95
2AK5	400	50.5*	35	100
AP5	450	52.5*	45	100
AS2	500	54.5*	43	100

* Value used to determine upper-shelf energy (USE) in accordance with ASTM Standard E 185-82.[17]

Table 4-6. Charpy Impact Results for Palisades Capsule SA-240-1 Irradiated Correlation Monitor Plate Material (HSST Plate 02) Heat No. A1195-1

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, mil	Shear Fracture, %
O2D2-10	70	6.5	4	0
O2D2-13	125	15.5	10	20
O2D2-23	175	27.5	24	30
O2D2-17	200	26	19	35
O2D2-2	200	44.5	29	55
OCD2-22	225	44.5	30	55
O2D2-19	240	54	40	70
O2D2-8	250	70	50	80
O2D2-5	300	83*	66	100
O2D2-15	350	82.5*	72	100
O2D2-24	400	89.5*	67	100
O2D2-11	500	82.5*	65	100

* Value used to determine upper-shelf energy (USE) in accordance with ASTM Standard E 185-82.[17]

Table 4-11. **Hyperbolic Tangent Curve Fit Coefficients for the Palisades Capsule SA-240-1 Surveillance Materials**

Material Description	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Weld Metal W5214	A: 27.4 B: 25.2 C: 111.6 T0: 208.1	A: 22.8 B: 21.8 C: 83.5 T0: 231.7	A: 50.0 B: 50.0 C: 72.5 T0: 223.2
Weld Metal 34B009	A: 29.8 B: 27.6 C: 111.7 T0: 176.6	A: 22.9 B: 21.9 C: 88.0 T0: 184.3	A: 50.0 B: 50.0 C: 109.8 T0: 192.6
Weld Metal 27204	A: 28.0 B: 25.8 C: 145.7 T0: 215.3	A: 25.6 B: 24.6 C: 169.2 T0: 225.9	A: 50.0 B: 50.0 C: 118.4 T0: 210.1
Correlation Monitor Plate, HSST Plate 02 (Heat No. A1195-1)	A: 43.3 B: 41.1 C: 75.3 T0: 211.8	A: 35.8 B: 34.8 C: 83.1 T0: 222.2	A: 50.0 B: 50.0 C: 75.9 T0: 206.5

Table 4-12. Summary of Charpy Impact Test Results for the Palisades Capsule SA-240-1 Surveillance Materials

Material Description	30 ft-lb Transition Temperature, °F			50 ft-lb Transition Temperature, °F			35 mil Lateral Expansion Transition Temperature, °F			Upper-Shelf Energy, ft-lb		
	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	Decrease
Weld Metal W5214	-60.2(a)	219.9	280.1	-17.4(a)	372.7	390.1	-29.6(a)	284.3	313.9	102.7(a)	52.5	50.2
Weld Metal 34B009	-82.0(a)	177.4	259.4	-45.0(a)	280.8	325.8	-51.6(a)	238.6	290.2	113.9(a)	57.4	56.5
Weld Metal 27204	-41.2(b)	226.6	267.8	-6.1(b)	399.7	405.8	Not available.	293.7	---	108.4(b)	53.8	54.6
HSST Plate 02 Heat No A1195-1	45.7(c)	186.6	140.9	78.3(c)	224.2	145.9	Not available.	220.3	---	120.3(c)	84.4	35.9

(a) Data reported in AEA Technology Report AEA-TSD-0774.[9]

(b) Data reported in CE Report No. TR-MCC-189.[18]

(c) Data reported in NUREG/CR-6413.[11]

Figure 4-2. Palisades Capsule SA-240-1 Charpy Impact Data for Irradiated Weld Metal W5214

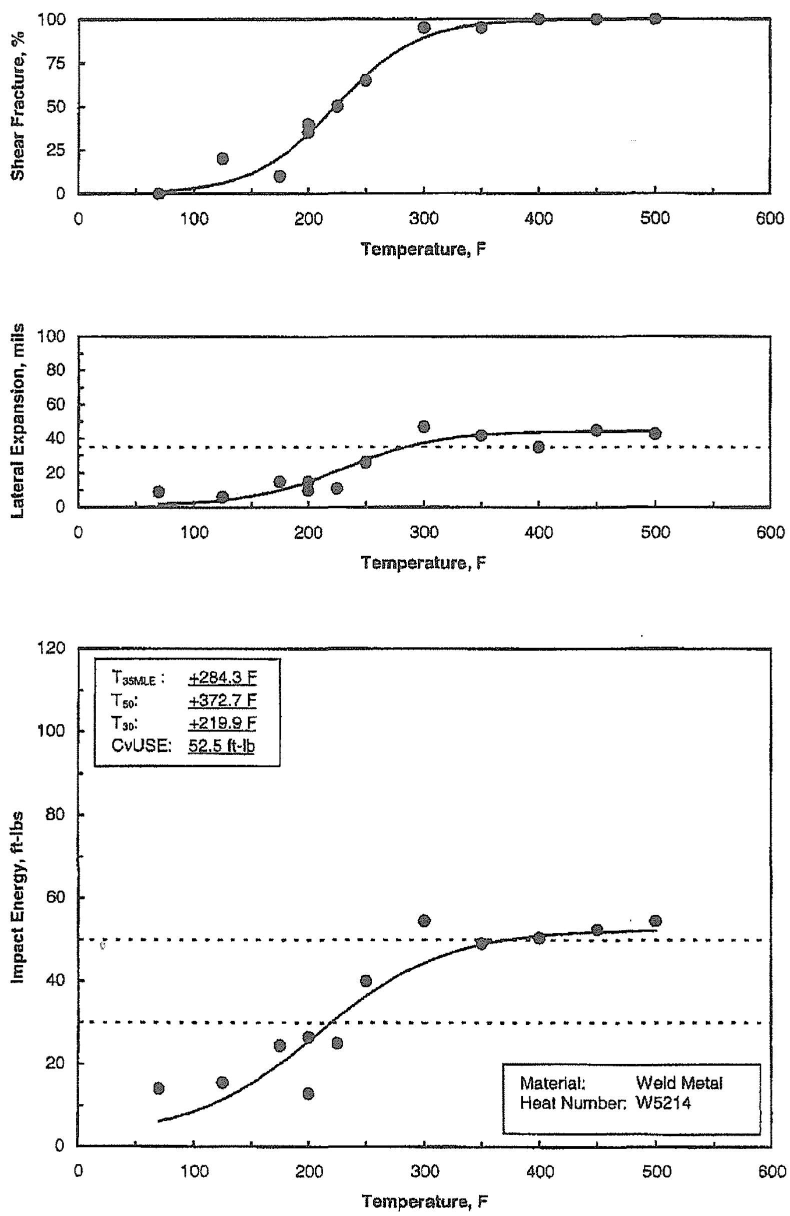
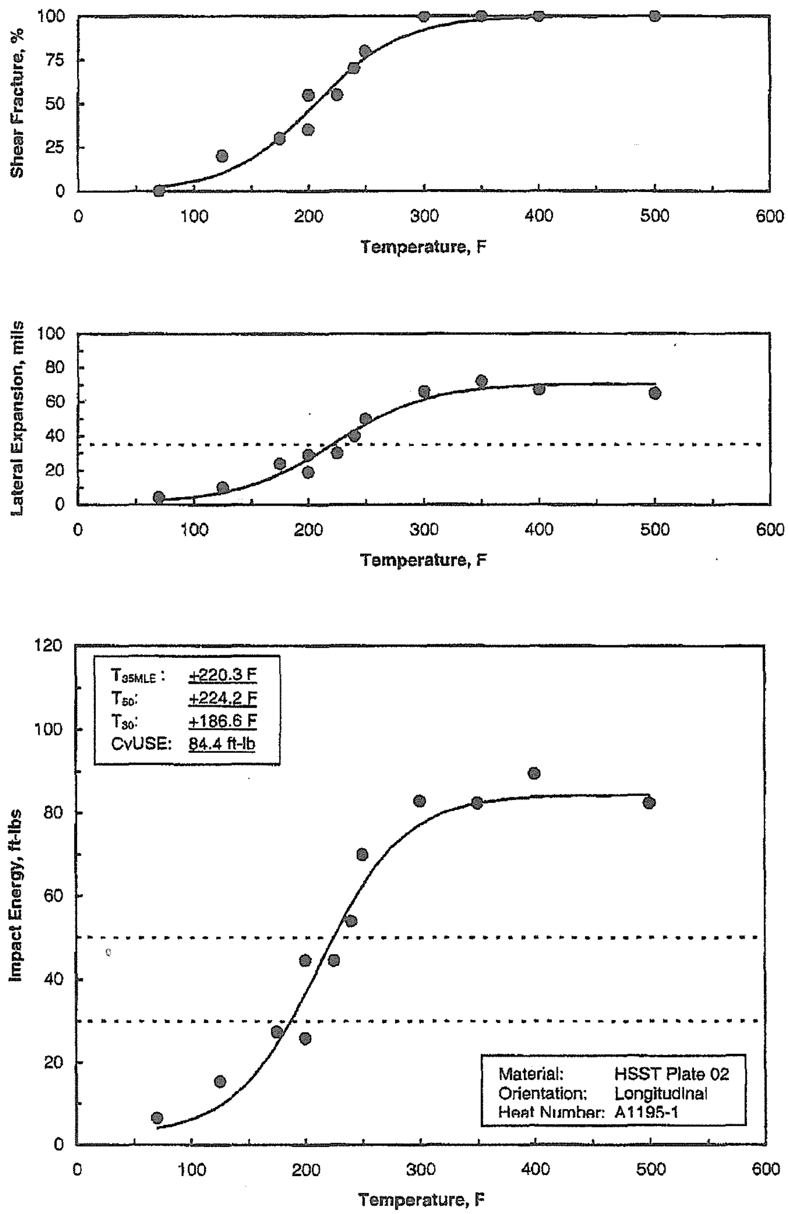


Figure 4-5. Palisades Capsule SA-240-1 Charpy Impact Data for Irradiated Correlation Monitor Plate Material (HSST Plate 02), Heat No. A1195-1



6.0 Summary of Results

The investigation of the post-irradiation test results of the materials contained in the second supplemental surveillance capsule, Capsule SA-240-1 removed from the Consumers Power Company Palisades reactor vessel, led to the following conclusions:

1. Observation of the Capsule SA-240-1 thermal monitors indicated that the irradiated test specimens were exposed to a maximum irradiation temperature less than 558°F.

2. Thirty-six pre-machined irradiated 18mm Charpy inserts were successfully reconstituted and machined to Type A Charpy impact specimens. The reconstituted Charpy specimens were subsequently impact tested.

3. The 30 ft-lb and 50 ft-lb transition temperatures for the weld metal W5214 increased 280.1°F and 390.1°F, respectively. In addition, the C_vUSE for this material decreased 48.9%.

4. The 30 ft-lb and 50 ft-lb transition temperatures for the weld metal 34B009 increased 259.4°F and 325.8°F, respectively. In addition, the C_vUSE for this material decreased 49.6%.

5. The 30 ft-lb and 50 ft-lb transition temperatures for the weld metal 27204 increased 267.8°F and 399.7°F, respectively. In addition, the C_vUSE for this material decreased 50.4%.

6. The correlation monitor plate demonstrated similar behavior with an increase in the 30 ft-lb and 50 ft-lb transition temperatures of 140.9°F and 145.9°F, respectively. The percent decrease in the C_vUSE for this material is 29.8%.

BAW-2341, Revision 2
May 2001

Test Results of Capsule SA-60-1
Consumers Energy
Palisades Nuclear Plant

-- Reactor Vessel Material Surveillance Program --

by

M. J. DeVan

FTI Document No. 77-2341-02
(See Section 7 for document signatures.)

Prepared for

Consumers Energy

Prepared by

Framatome ANP, Inc.
3315 Old Forest Road
P. O. Box 10935
Lynchburg, Virginia 24506-0935

FRAMATOME ANP

Executive Summary

This report describes the results of the test specimens from the first supplemental capsule (Capsule SA-60-1) of the Consumers Energy Palisades Nuclear Plant as part of their reactor vessel surveillance program. The objective of the program is to monitor the effects of neutron irradiation on the mechanical properties of the reactor vessel materials by testing and evaluation of Charpy impact specimens.

Supplemental Capsule SA-60-1 was removed from the Palisades reactor vessel at the end-of-cycle 13 (EOC-13) for testing and evaluation. The capsule contents were removed from Capsule SA-60-1 for testing and examination. The test specimens included modified 18mm Charpy V-notch inserts for three weld metals fabricated with weld wire heats W5214, 34B009, and 27204 and standard Charpy V-notch specimens fabricated from the correlation monitor plate material, HSST Plate 02. The weld metal Charpy inserts were reconstituted to full size Charpy V-notch specimens. The reconstituted weld metals along with HSST Plate 02 material were Charpy impact tested.

Following the initial Charpy V-notch impact testing, the laboratory performed a calibration of the temperature indicator used in the Palisades Capsule SA-60-1 testing. The results of the laboratory calibration indicated the instrument was out-of-tolerance. Based on the results of this calibration test, the laboratory revised the Charpy impact test temperatures accordingly. Revision 1 corrects the test temperatures for the Supplemental Capsule SA-60-1 reconstituted weld metal Charpy V-notch impact specimens and the HSST Plate 02 Charpy V-notch impact specimens.

Revision 2 provides an update to the hyperbolic tangent curve fits of the Charpy impact curves by restraining the upper-shelf energy. For these curve fits, the lower-shelf energy was fixed at 2.2 ft-lbs for all cases, and for each materials the upper-shelf energy was fixed at the average of all test energies exhibiting 100% shear, consistent with ASTM Standard E 185-82. 2

Table 3-2. Chemical Composition of Palisades Capsule SA-60-1 Surveillance Materials

Element	Chemical Composition, wt%			
	Weld Metal W5214[a]	Weld Metal 34B009[a]	Weld Metal 27204[b]	Correlation Monitor Plate Heat No. A1195-1[c]
C	0.094	0.110	0.142	0.23
Mn	1.161	1.269	1.281	1.39
P	0.009	0.012	0.009	0.013
S	0.012	0.016	0.008	0.013
Si	0.252	0.181	0.217	0.21
Ni	1.045[b]	1.121[b]	1.067	0.64
Cr	0.040	0.040	0.071	---
Mo	0.510	0.543	0.525	0.50
Cu	0.307[b]	0.185[b]	0.194	0.17

(a) AEA Technology analysis.[8]
(b) Analysis provided by Consumers Energy.[9]
(c) ORNL analysis.[10]

Table 4-3. Charpy Impact Results for Palisades Capsule SA-60-1 Irradiated Weld Metal W5214

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, mil	Shear Fracture, %
AA2	74	10	4	0
AW1	129	24	15	5
AW2	154	23.5	15	15
2AF5	204	30	16	50
AL3	229	33.5	23	65
AA4	254	28	19	60
2AL6	279	43.5	35	80
2AE2	279	48.5	38	90
2AH6	329	47.5	35	90
AR94	404	51.5*	43	100
2AH1	454	55*	47	100
AV4	479	57*	46	100

* Value used to determine upper-shelf energy (USE) in accordance with ASTM Standard E 185-82.[15]

Table 4-6. Charpy Impact Results for Palisades Capsule SA-60-1 Irradiated Correlation Monitor Plate Material (HSST Plate 02) Heat No. A1195-1

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, mil	Shear Fracture, %
O2D2-9	74	8	3	5
O2D2-3	104	20.5	16	10
O2D2-1	129	24.5	18	20
O2D2-7	154	26.5	23	40
O2D2-18	179	35.5	28	45
OCD2-21	204	48.5	40	70
O2D2-14	229	53.5	43	65
O2D2-16	229	51.5	43	70
O2D2-4	254	73.5	65	80
O2D2-6	279	85*	70	100
O2D2-12	329	87.5*	74	100
O2D2-20	404	86.5*	77	100

* Value used to determine upper-shelf energy (USE) in accordance with ASTM Standard E 185-82.[15]

Table 4-11. Hyperbolic Tangent Curve Fit Coefficients for the Palisades Capsule SA-60-1 Surveillance Materials

Material Description	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Weld Metal W5214	A: 28.4 B: 26.2 C: 158.1 T0: 188.8	A: 25.0 B: 24.0 C: 160.0 T0: 239.6	A: 50.0 B: 50.0 C: 80.5 T0: 214.9
Weld Metal 34B009	A: 28.7 B: 26.5 C: 123.8 T0: 161.8	A: 25.3 B: 24.3 C: 97.6 T0: 196.4	A: 50.0 B: 50.0 C: 89.6 T0: 179.6
Weld Metal 27204	A: 27.6 B: 25.4 C: 111.4 T0: 201.4	A: 25.9 B: 24.9 C: 101.8 T0: 214.4	A: 50.0 B: 50.0 C: 92.1 T0: 187.1
Correlation Monitor Plate, HSST Plate 02 (Heat No. A1195-1)	A: 44.3 B: 42.1 C: 95.1 T0: 193.0	A: 41.3 B: 40.3 C: 104.9 T0: 208.6	A: 50.0 B: 50.0 C: 85.2 T0: 183.7

2

Table 4-12. Summary of Charpy Impact Test Results for the Palisades Capsule SA-60-1 Surveillance Materials

Material Description	30 ft-lb Transition Temperature, °F			50 ft-lb Transition Temperature, °F			35 mil Lateral Expansion Transition Temperature, °F			Upper-Shelf Energy, ft-lb		
	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	Decrease
Weld Metal W5214	-60.2[a]	198.8	259.0	-17.4[a]	375.6	393.0	-29.6[a]	310.1	339.7	102.7[a]	54.5	48.2
Weld Metal 34B009	-82.0[a]	167.8	249.8	-45.0[a]	298.6	343.6	-51.6[a]	237.5	289.1	113.9[a]	55.25	58.65
Weld Metal 27204	-41.2[b]	211.9	253.1	-6.1[b]	355.6	361.7	Not available.	249.4	---	108.4[b]	53.0	55.4
HSST Plate 02 Heat No A1195-1	45.7[c]	159.4	113.7	78.3[c]	206.0	127.7	Not available.	187.9	---	120.3[c]	86.3	34.0

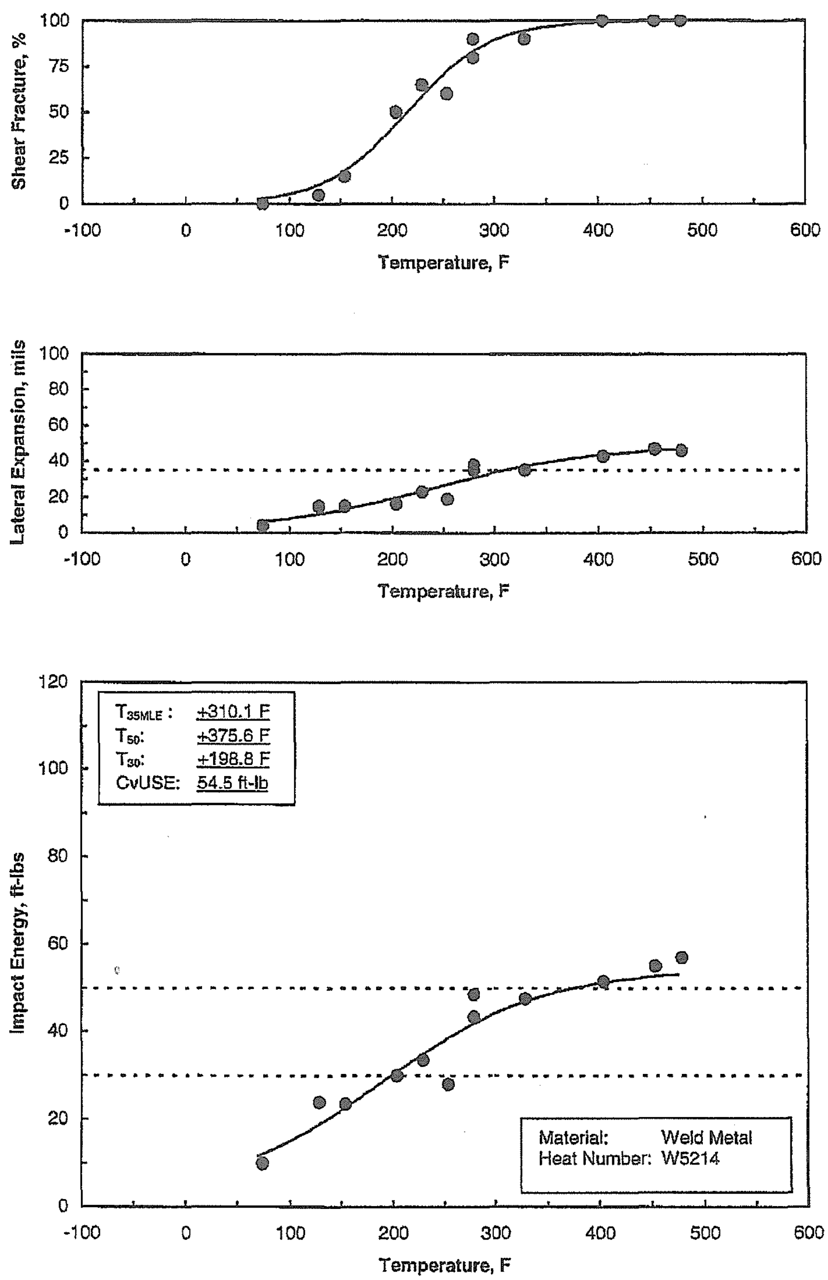
2

(a) Data reported in AEA Technology Report AEA-TSD-0774.[8]

(b) Data reported in CE Report No. TR-MCC-189.[16]

(c) Data reported in NUREG/CR-6413.[10]

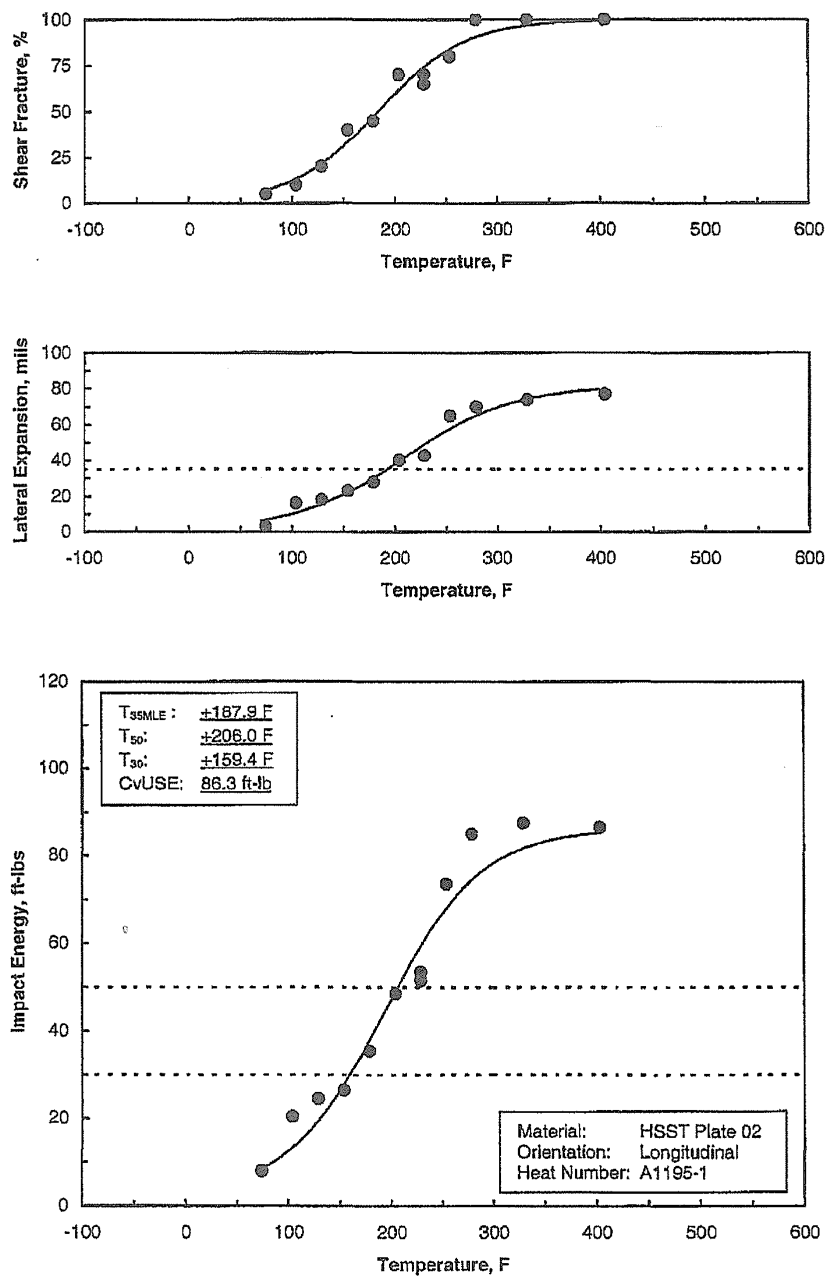
Figure 4-2. Charpy Impact Data for Irradiated Weld Metal W5214



2

FRAMATOME ANP

Figure 4-5. Charpy Impact Data for Irradiated Correlation Monitor Plate Material (HSST Plate 02), Heat No. A1195-1



2

FRAMATOME ANP

Palisades Nuclear Plant - Weld/W5214 (Unirr)

CVGRAPH 4.0 Hyperbolic Tangent Curve Printed at 15:08:08 on 12-01-1995

Coefficients of Curve 1

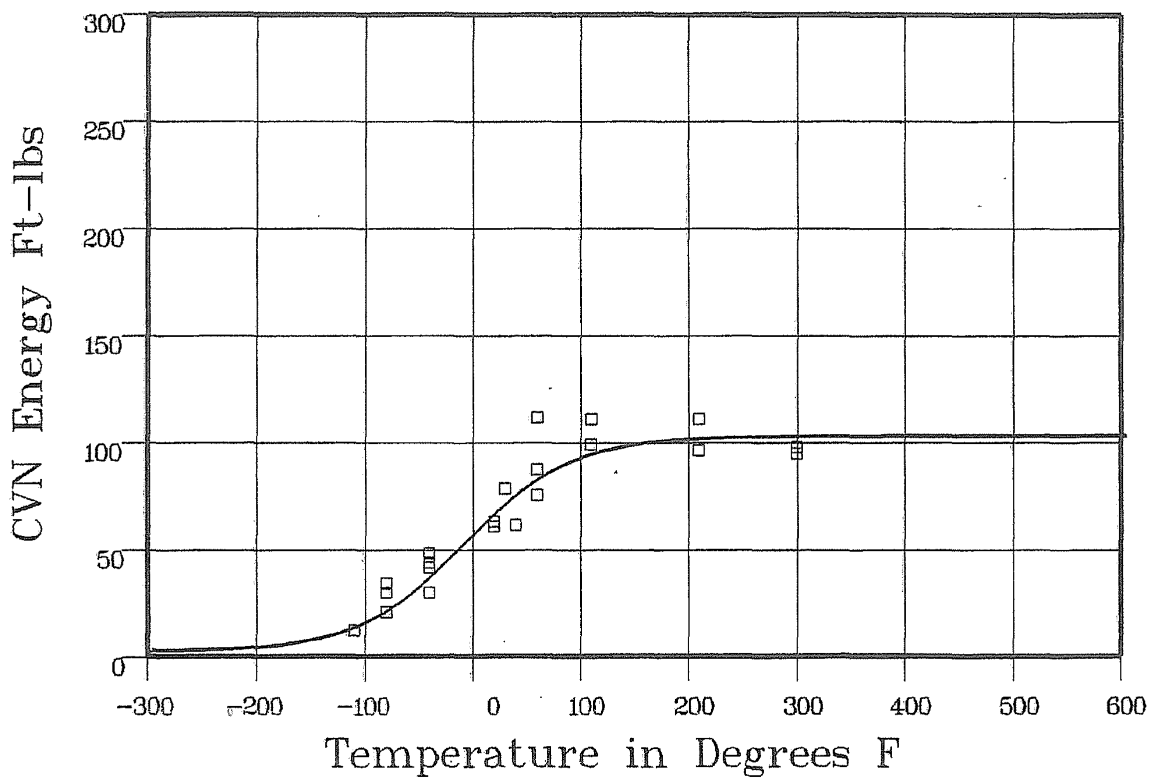
A = 52.4	B = 50.29	C = 99.19	T0 = -12.69

Equation is: CVN = A + B * [tanh((T - T0)/C)]

Upper Shelf Energy: 102.7 Fixed   Temp. at 30 ft-lbs: -60.2   Temp. at 50 ft-lbs: -17.4   Lower Shelf Energy: 2.1 Fixed

Material: WELD   Heat Number: W5214   Orientation: TL

Capsule: Unirr   Total Fluence: 0.0



Data Set(s) Plotted

Plant: PAL   Cap.: Unirr   Material: WELD   Ori.: TL   Heat #: W5214

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-110	11.8	14.5	-2.7
-110	11.8	14.5	-2.7
-80	33.9	22.69	11.2
-80	20.6	22.69	-2.09
-80	29.5	22.69	6.8
-40	47.9	38.89	9
-40	43.5	38.89	4.6
-40	29.5	38.89	-9.39
-40	41.29	38.89	2.4

**** Data continued on next page ****

Palisades Nuclear Plant - Weld/W5214 (Unirr)

Material: WELD   Heat Number: W5214   Orientation: TL

Capsule: Unirr   Total Fluence: 0.0

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
0	64.19	58.8	5.39
0	39.09	58.8	-19.7
20	62.7	68.4	-5.7
20	60.5	68.4	-7.9
30	78.19	72.8	5.39
40	61.2	76.86	-15.66
60	87	83.82	3.17
60	75.19	83.82	-8.62
60	111.4	83.82	27.57
110	110.59	94.88	15.71
110	98.8	94.88	3.91
210	110.59	101.58	9.01
210	95.9	101.58	-5.68
300	97.4	102.51	-5.11
300	94.4	102.51	-8.11

SUM of RESIDUALS = 10.77

APPENDIX D

INDIAN POINT 2 REACTOR VESSEL MATERIALS SURVEILLANCE PROGRAM RESULTS FOR WELD NO. W5214

WCAP 7323

c. 1

CONSOLIDATED EDISON CO.
INDIAN POINT UNIT NO. 2 REACTOR VESSEL
RADIATION SURVEILLANCE PROGRAM

By

S. E. Yanichko

May 1969

IPP 106

Approved: C Landerman
E. Landerman

Westinghouse Electric Corporation
Nuclear Energy Systems Division
Box 355
Pittsburgh, Pa. 15230

WCAP 7323

SECTION 4

POST-IRRADIATION TESTING

Specimen capsules will be removed from the reactor only during normal refueling periods. The recommended schedule for removal of capsules is as follows:

Capsule Type	Capsule Identification	Exposure Time
I	T	(Replacement of 1st Region)
II	S	(Replacement of 2nd Region)
I	Z	(Replacement of 4th Region)
II	V	10 years
I	U	Extra capsules for complementary testing or additional exposure
I	W	
I	X	
II	Y	

Each specimen capsule upon removal after radiation exposure will be transferred to a post-irradiation test facility for disassembly of the capsule and testing of all specimens.

4.1 CHARPY V-NOTCH IMPACT TESTS

The testing of the eight Charpy impact specimens from each of the IPP vessel plates, the weld and HAZ metal, and the correlation monitor material in the capsules can be done singulary, making possible the performance of Charpy impact tests at five different temperatures, with three extra specimens to provide an optimum curve for each plate. The initial Charpy specimen from the first capsule removed should be tested at room temperature. The impact energy value for this test temperature should be compared with pre-irradiation test data. The testing temperatures for the remaining specimen should then be appropriately raised or lowered. The test temperatures of

specimens from capsules exposed to longer irradiation periods should be determined by the test results for the previous capsule.

4.2 TENSILE TESTS

The two tensile specimens per plate or weld from each of the capsules should be tested at room temperature and the approximate operating temperature of the reactor (550°F).

4.3 WEDGE OPENING LOADING TESTS

The WOL specimens from each individual capsule should be tested at a temperature based on the transition temperature shift obtained from the associated Charpy impact specimens. A mean temperature of -200 °F plus the transition temperature shift should be the initial test temperature.

4.4 POST-IRRADIATION TEST EQUIPMENT

1. Milling machine or special cut-off wheel for opening capsules and dosimeter blocks.
2. Hot-cell tensile testing machine with:
   a. pin-type adapter for pulling tensile tests
   b. clevis and extensometer for pulling WOL specimens.
3. Hot-cell Charpy impact testing machine.
4. NaI scintillation detector and pulse height analyzer for gamma counting of the specific activities of the dosimeters.

WCAP 7323

TABLE 7

PRE-IRRADIATION TENSILE PROPERTIES FOR THE INDIAN POINT UNIT NO. 2 PRESSURE VESSEL PLATE MATERIAL AND WELD METAL

Plate No.	Test Temp., °F	0.2% Yield Strength, psi	Tensile Strength, psi	Total Elongation %	Reduction In Area, %
B2002-1	Room	68,500	89,000	25.1	67.8
B2002-1	Room	65,850	87,800	25.3	67.4
B2002-1	200	61,550	79,900	24.1	68.6
B2002-1	200	67,950	89,400	23.8	67.6
B2002-1	400	57,900	79,900	23.1	64.7
B2002-1	400	59,800	82,200	22.2	67.8
B2002-1	600	56,750	80,550	21.9	64.3
B2002-1	600	57,750	85,700	22.9	64.2
B2002-2	Room	62,350	83,800	27.1	70.0
B2002-2	Room	66,750	90,500	28.2	69.6
B2002-2	200	63,650	84,450	24.8	70.5
B2002-2	200	63,200	83,800	25.5	67.3
B2002-2	400	53,800	77,900	23.1	68.5
B2002-2	400	52,650	73,150	22.4	67.6
B2002-2	600	53,500	78,800	22.7	64.4
B2002-2	600	54,700	81,450	24.7	64.4
B2002-3	Room	65,650	87,300	27.6	67.3
B2002-3	Room	65,000	87,350	24.8	66.7
B2002-3	200	67,800	88,900	23.4	68.6
B2002-3	200	67,700	89,150	22.1	64.9
B2002-3	400	57,950	79,550	22.3	68.7
B2002-3	400	55,350	77,100	23.2	64.9
B2002-3	600	57,750	83,850	24.9	68.2
B2002-3	600	58,350	86,500	24.9	64.7
Weld	Room	64,500	80,700	28.5	73.9
Weld	Room	65,000	81,000	26.9	71.5
Weld	200	63,450	76,100	28.4	72.9
Weld	200	61,050	75,200	25.2	73.0
Weld	400	57,550	75,000	22.9	68.1
Weld	400	58,300	75,800	22.6	69.6
Weld	600	56,650	79,800	24.4	62.0
Weld	600	56,650	79,200	24.0	66.9

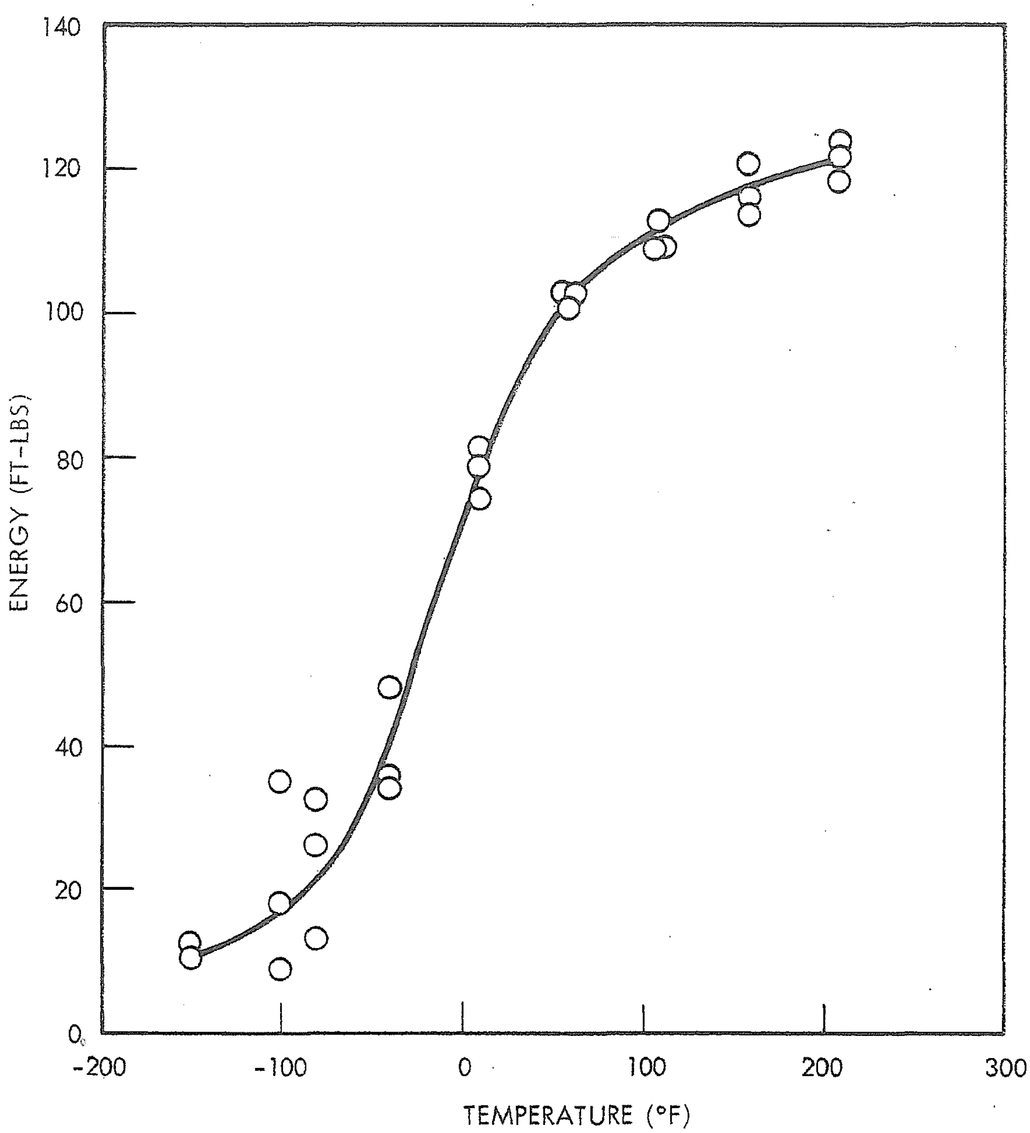


Figure 10. Pre-Irradiation Charpy V-Notch Impact Energy for the Indian Point Unit #2 Reactor Pressure Vessel Weld Metal

SOUTHWEST RESEARCH INSTITUTE
Post Office Drawer 28510, 6220 Culebra Road
San Antonio, Texas 78284

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM FOR INDIAN POINT UNIT NO. 2 ANALYSIS OF CAPSULE Y

by

E. B. Norris

FINAL REPORT
SwRI Project No. 02-5212

Prepared for
Consolidated Edison Company of New York, Inc.
4 Irving Place
New York, New York 10003

November 1980

Approved:

U. S. Lindholm, Director
Department of Materials Sciences

TABLE VII

CHARPY V-NOTCH IMPACT DATA
INDIAN POINT UNIT NO. 2 PRESSURE VESSEL WELD METAL

Condition	Spec. No.	Temp. (°F)	Energy (ft-lbs)	Shear (%)	Lateral Expansion (mils)
Baseline	(a)	-150	12.5	10	10
		-150	10.5	15	11
		-100	35.0	25	29
		-100	9.0	20	9
		-100	18.0	30	19
		-80	13.0	20	12
		-80	32.5	20	27
		-80	26.0	20	23
		-40	34.0	30	30
		-40	35.5	35	31
		-40	48.0	35	40
		10	78.5	60	64
		10	74.0	60	60
		10	81.0	70	68
		60	102.5	80	78
		60	10[illegible].0	85	82
		60	100.0	85	80
		110	112.5	99	88
		110	108.5	90	87
		110	108.5	98	88
		160	115.5	100	90
		160	113.0	100	92
		160	120.0	100	93
		210	121.0	100	92
		210	123.5	100	91
		210	117.5	100	92
Capsule Y	W-17	74	17.5	nil	14
	W-19	110	23.0	5	19
	W-20	160	40.0	25	34
	W-21	190	47.0	50	43
	W-23	210	55.0	60	53
	W-24	260	71.5	100	51
	W-18(b)	300	61.0	100	45
	W-22	350	67.0	100	52

(a) Not reported.
(b) Specimen number stamped on impact side.

TABLE VIII

CHARPY V-NOTCH IMPACT DATA
CORRELATION MONITOR MATERIAL (SUPPLIED BY U. S. STEEL)

Condition	Spec. No.	Temp. (°F)	Energy (ft-lbs)	Shear (%)	Lateral Expansion (mils)
Baseline	(a)	-80	4	2	6
		-80	4	2	6
		-60	8	3	6
		-60	6	3	6
		-40	12	10	14
		-40	10	5	10
		-40	6	5	7
		-20	14	15	14
		-20	13	15	14
		0	22	30	22
		0	18	25	18
		20	29	35	28
		20	23	35	23
		40	36	45	33
		40	26	45	26
		60	36	50	40
		60	33	45	35
		80	67	100	60
		80	50	70	48
		100	68	98	60
		100	62	85	58
Capsule Y	R-60	40	5.0	nil	4
	R-57	74	26.0	5	22
	R-62	90	30.5	10	26
	R-58	110	28.0	15	26
	R-59	135	36.0	20	32
	R-63	160	51.5	40	43
	R-64	210	60.0	90	53
	R-61	260	68.5	100	58

(a) Not reported.

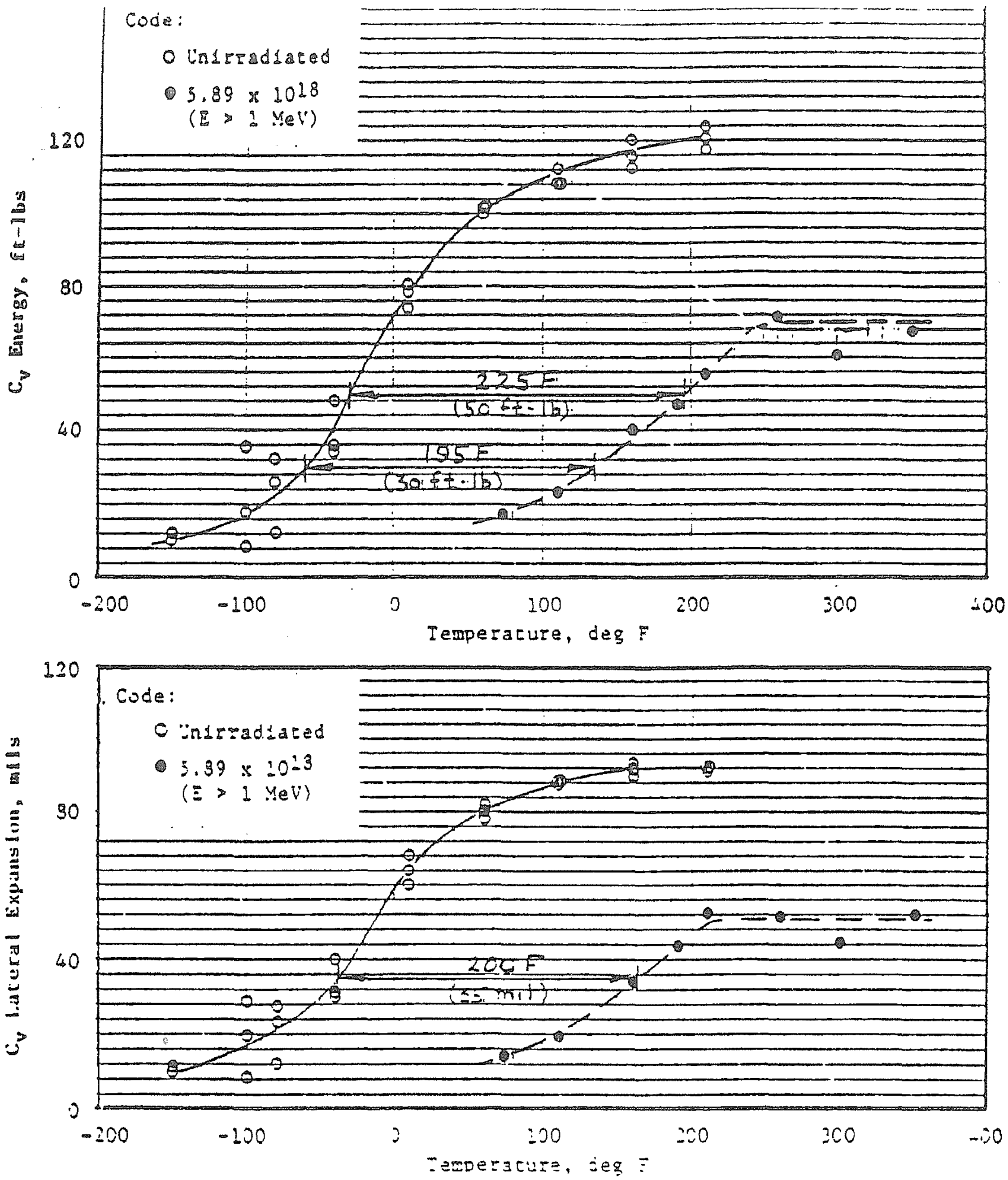


FIGURE 6. EFFECT OF IRRADIATION ON C_v IMPACT PROPERTIES OF INDIAN POINT UNIT NO. 2 WELD METAL

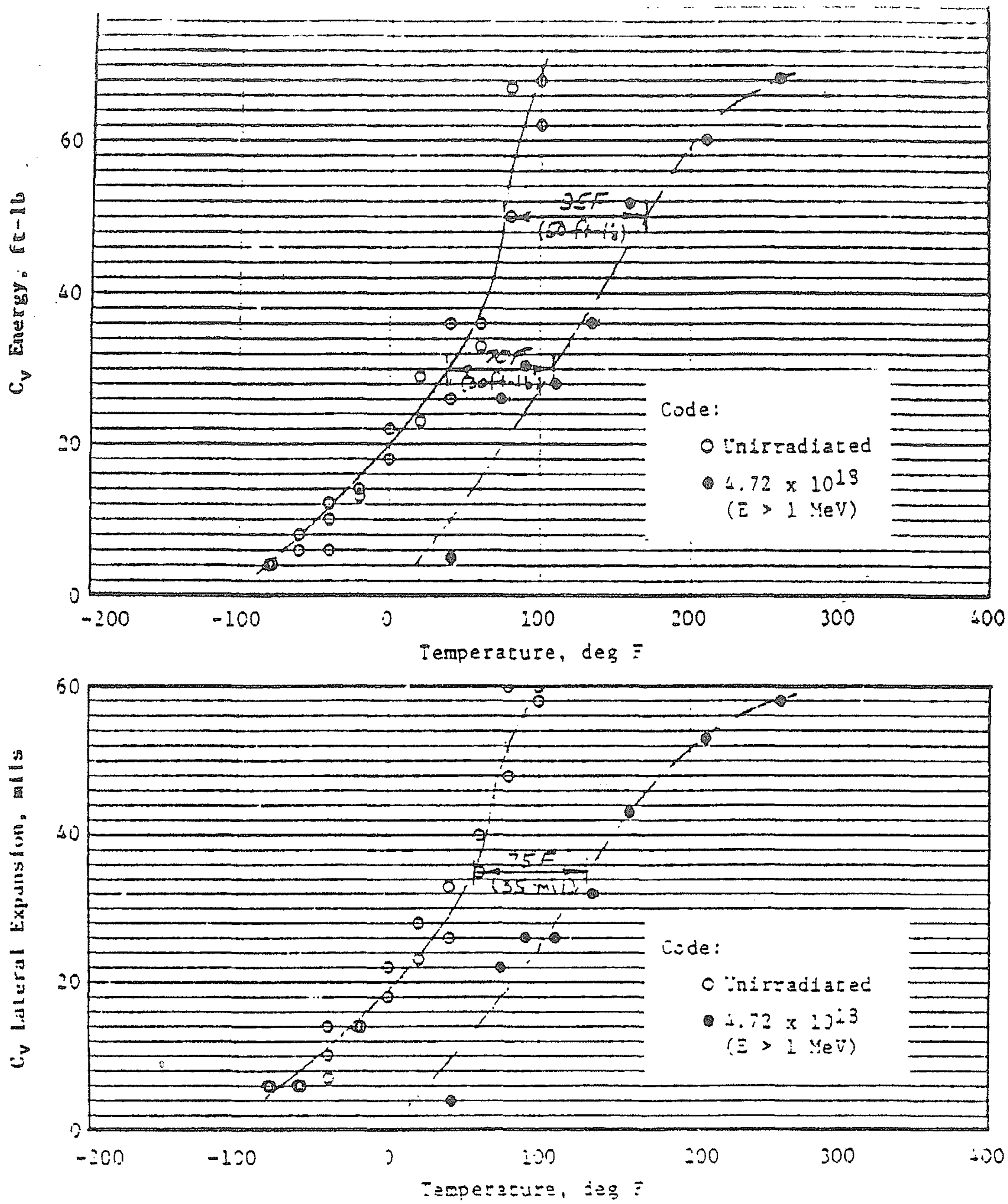


FIGURE 7. EFFECT OF IRRADIATION ON C_v IMPACT PROPERTIES OF INDIAN POINT UNIT NO. 2 CORRELATION MONITOR MATERIAL

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM FOR INDIAN POINT UNIT NO. 2 ANALYSIS OF CAPSULE V

IPP

FINAL REPORT
SwRI Project No. 17-2108

Prepared for

Consolidated Edison Company of New York, Inc.
4 Irving Place
New York, New York 10003

October 1988

SOUTHWEST RESEARCH INSTITUTE
SAN ANTONIO HOUSTON

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM FOR INDIAN POINT UNIT NO. 2 ANALYSIS OF CAPSULE V

FINAL REPORT
SwRI Project No. 17-2108

Prepared for

Consolidated Edison Company of New York, Inc.
4 Irving Place
New York, New York 10003

October 1988

Written by

Frank A. Iddings

F. A. Iddings
D. G. Cadena
Mark Williams (Consultant)

Approved by

B. T. Cross

B. T. Cross
Director
Department of NDE Science and Research
Southwest Research Institute

I. SUMMARY OF RESULTS AND CONCLUSIONS

The analysis of the fourth material surveillance capsule removed from the Indian Point Unit No. 2 reactor pressure vessel led to the following conclusions:

(1) Based on a calculated neutron spectral distribution, Capsule V received a fast fluence of 5.3×10^{18} n/cm^2 (E > 1 MeV) at its radial center line.

(2) The surveillance specimens of the core beltline plate materials experienced shifts in RT_{NDT} (50 ft-lb. values) over the range of 79°F (Plate B2002-2) to 239°F (Weld) as a result of fast neutron exposure up to the 1987 refueling outage.

(3) The core beltline weld exhibited the largest shift in RT_{NDT} and is projected to control the heatup and cooldown limitations throughout the design lifetime of the pressure vessel.

(4) From the previous capsule, Z, the estimated maximum neutron fluence of 3.33×10^{18}* neutrons/cm^2 (E > 1 MeV) was received by the vessel wall in 5.17 effective full power years (EFPY) through Cycle 5, which is equal to a fluence rate of 6.44×10^{17}* per EFPY. At the end of Cycle 8 (8.6 EFPY) the neutron fluence was 4.45×10^{18} n/cm^2 giving 3.26×10^{17} n/cm^2 per EFPY for Cycles 6 through 8. This calculated value for the decrease in fluence per EFPY agrees well with the experimental value for the decrease in fluence rate; i.e., 50.6% vs. 48.9%. The use of a low leakage core loading pattern beginning with Cycle 6 did significantly reduce the fluence rate on the pressure vessel wall.

*Revised from Capsule Z report using the latest plant specific lead factors.

TABLE IV-5

CHARPY IMPACT DATA WITH PHOTOS OF FRACTURE FACES

MATERIAL - (WELD)

Date June 2, 1988

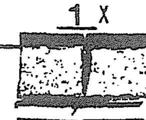
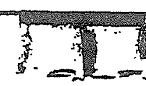
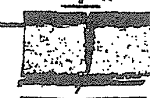
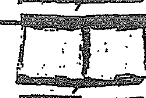
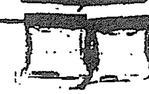
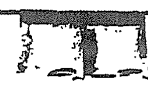
SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH 1 X
W- 9	74°F	24.0	.019	0	
W-10	+130	26.5	.023	20	
W-11	+180	40.5	.035	40	
W-12	+220	53.0	.048	65	
W-13	+260	62.5	.054	95	
W-14	+300	76.0	.064	95	
W-16	+325	72.5	.065	95	
W-15	+350	76.0	.067	100	

TABLE IV-5

CHARPY IMPACT DATA WITH PHOTOS OF FRACTURE FACES

MATERIAL - (WELD)

Date June 2, 1988

SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH 1 X
W- 9	74°F	24.0	.019	0	
W-10	+130	26.5	.023	20	
W-11	+180	40.5	.035	40	
W-12	+220	53.0	.048	65	
W-13	+260	62.5	.054	95	
W-14	+300	76.0	.064	95	
W-16	+325	72.5	.065	95	
W-15	+350	76.0	.067	100	

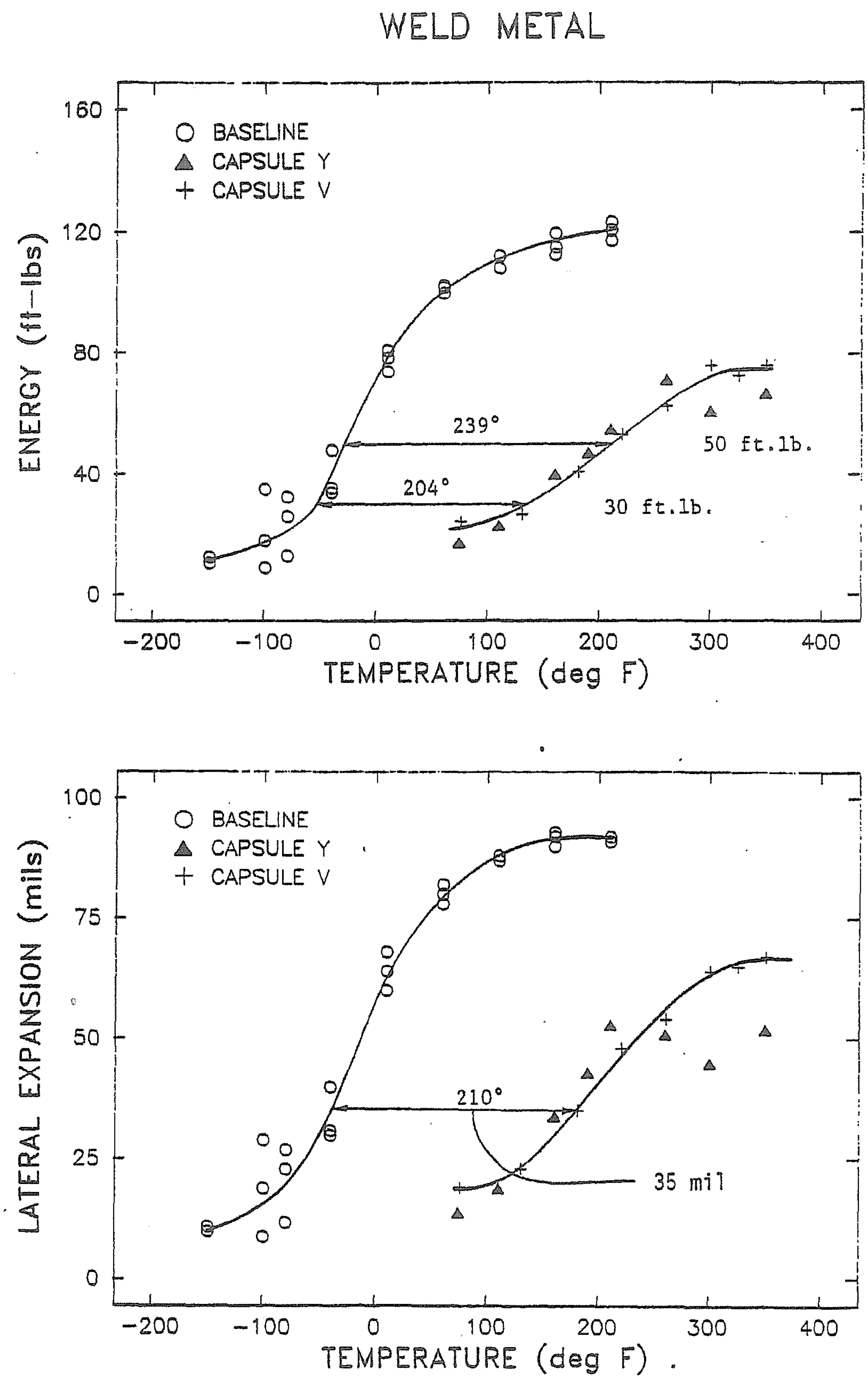


Figure IV-3. Radiation Response of Indian Point Unit No. 2 Weld Metal

Table IV-9

SUMMARY OF RT_{NDT} SHIFTS AND UPPER SHELF ENERGY REDUCTION (C_V) FOR MATERIALS IN CAPSULE V

A. Summary of Fluence and Measured ΔRT_{NDT} Values for Test Specimens in Capsule V

Type of Material	Fluence Neutron cm²	Measured ΔRT_{NDT} (°F) 50 Ft-Lbs	30 Ft-Lbs	35 mils*
Weld	5.59E18	239	204	230
		77 Ft-Lbs	46 Ft-Lbs	
Plate B2002-2	4.57E18	85	80	97
HAZ	5.59E18	190	162	184
Correlation Monitor	4.57E18	NA**	104	108

B. Decrease in Upper Shelf Energy (C_V)

Material	Initial Shelf Ft-lb	Capsule V*** Ft-lb	C_V Ft-lb	% Decrease
B2002-2	117	111	6	5
Weld Metal	118	75	43	36
HAZ	100	98	2 (nil)	2
Correlation Monitor	118	70	48	41

*35 mil + 20°F included in table.
**The upper shelf energy for this capsule was below 77 ft lbs.
*** Average of 3 Charpy measurements at ≈ 100% ductile failure.

TABLE IV-11(Cont'd)

SUMMARY OF CHEMISTRY VALUES FOR INDIAN POINT UNIT NO. 2 MATERIALS

Material	Source of Data	Cu W%	Ni W%
Plate B2002-3	WCAP 7328	(.14)*	(.57)*
	Capsule-Z: C_V Specimen 3-33	.30	.64
	Capsule-Z: C_V Specimen 3-38	.27	.59
	Capsule-Z: Tensile Specimen 3-5	.23	.58
	Capsule-Y: C_V Specimen 3-41	.21	--
	Capsule-Y: C_V Specimen 3-45	.22	--
	Capsule-Y: Tensile Specimen 3-6	(.11)*	--
	Capsule-Y: Tensile Specimen 3-7	(.10)*	--
	Capsule-T: C_V Specimen 3-2	.27	--
	Capsule-T: C_V Specimen 3-3	.23	--
	Capsule-T: Tensile Specimen 3-1	(.09)*	--
	Average	.25	.60
HAZ	Capsule-V: C_V Specimen H-16	.08	1.2
	Capsule-V: C_V Specimen H-12	.06	1.2
	Capsule-Y: C_V Specimen H-21	.15	--
	Capsule-Y: C_V Specimen H-23	.20	--
	Average	.12	1.2
Weld	Capsule-V: C_V Specimen W-13	.23	1.02
	Capsule-V: C_V Specimen W-12	.20	1.06
	Capsule-V: Tensile Specimen W-3	.20	(.69)*
	Capsule-V: C_V Tensile Specimen W-4	(.12)*	1.00
	Capsule-Y: C_V Specimen W-17	.19	--
	Capsule-Y: C_V Specimen W-19	.22	--
	Capsule-Y: Tensile Specimen W-5	.18	--
	Capsule-Y: Tensile Specimen W-6	.20	--
	Average	.20	1.03
Correlation Monitor	Capsule-V: C_V Specimen R-56	.20	.18
	Capsule-V: C_V Specimen R-52	.18	.27
	Capsule-Z: C_V Specimen R-33	.35	.28
	Capsule-Z: C_V Specimen R-36	.31	.27
	Capsule-Z: C_V Specimen R-40	.21	.21
	Capsule-Y: C_V Specimen R-60	.17	--
	Capsule-Y: C_V Specimen R-62	.19	--
	Capsule-T: C_V Specimen R-2	.25	--
	Average	.23	.24

*Values in parentheses discarded because of excessive deviation or were WCAP values. Surveillance specimen WCAP values not used since chemical analyses were available.

APPENDIX E

INDIAN POINT 3 REACTOR VESSEL MATERIALS SURVEILLANCE PROGRAM RESULTS FOR WELD NO. W5214

Westinghouse Class 3

WCAP-8475

CONSOLIDATED EDISON CO. OF NEW YORK INDIAN POINT UNIT NO. 3 REACTOR VESSEL RADIATION SURVEILLANCE PROGRAM

S. E. Yanichko
J. A. Davidson

January 1975

APPROVED: J. N. Chirigos

Work Performed Under INT 106

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems Division
P. O. Box 355
Pittsburgh, Pennsylvania 15230

WCAP 8475

SECTION 3
PREIRRADIATION TESTING

3-1. CHARPY V-NOTCH TESTS

Charpy V-Notch impact tests were performed on the vessel plates at various temperatures from -100 to 210°F to obtain full Charpy V-Notch transition curves (refer to tables 3-1 and 3-2, and to figures 3-1 thru 3-5). Charpy V-Notch impact tests were performed on weld metal and HAZ metal at various temperatures ranging from -150 to 160°F. The results are reported in table 3-3 and in figures 3-6 and 3-7, respectively. The Charpy V-Notch impact data for the correlation monitor material are shown in table 3-4 and figure 3-8.

3-2. TENSILE TESTS

Tensile tests were performed on the shell plates and on weld material at room temperature, 300 and 600°F, respectively. The results are shown in table 3-5 and in figures 3-9 through 3-14.

3-3. DROPWEIGHT NDTT TESTS

Dropweight NDTT tests (ASTM E208) were performed on each plate by the fabricator. The NDTT obtained on each plate follows:

Plate	Temperature
B2802-1	-50°F
B2802-2	-50°F
B2802-3	-40°F
B2803-3	-10°F

TABLE 3-3

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE INDIAN POINT UNIT NO. 3 REACTOR PRESSURE VESSEL WELD METAL AND WELD HEAT AFFECTED ZONE MATERIAL

	Weld Metal			Weld Heat-Affected Zone			
Test Temp °F	Energy FT-LBS	Shear %	Lateral Expansion Mils	Test Temp. °F	Energy FT-LBS	Shear %	Lateral Expansion Mils
-150	5.0	5	2	-150	4.0	0	2
-150	2.0	5	2	-150	6.0	5	1
-150	4.5	9	4	-150	14.0	9	11
-100	29.0	20	22	-125	16.0	9	14
-100	18.0	18	16	-125	7.0	5	8
-100	25.5	23	23	-125	34.0	14	28
-50	35.0	40	34	-100	48.5	20	35
-50	33.0	47	30	-100	59.0	25	40
-50	32.5	40	30	-100	30.0	18	20
-35	78.0	64	66	-50	30.0	29	22
-35	69.5	67	56	-50	62.5	40	44
-35	54.5	40	47	-50	60.0	36	44
-20	87.0	77	69	10	111.0	85	79
-20	82.0	77	63	10	51.5	47	48
-20	89.0	81	74	10	83.0	62	62
10	100.0	81	78	60	142.0	100	90
10	105.0	82	81	60	127.0	100	82
10	113.5	100	85	60	121.5	100	91
60	115.0	100	89	160	111.0	100	81
60	119.0	100	84	160	125.0	100	85
60	121.5	100	90	160	143.0	100	88
160	124.0	100	88				
160	125.0	100	89				
160	112.0	100	90				

8328-10

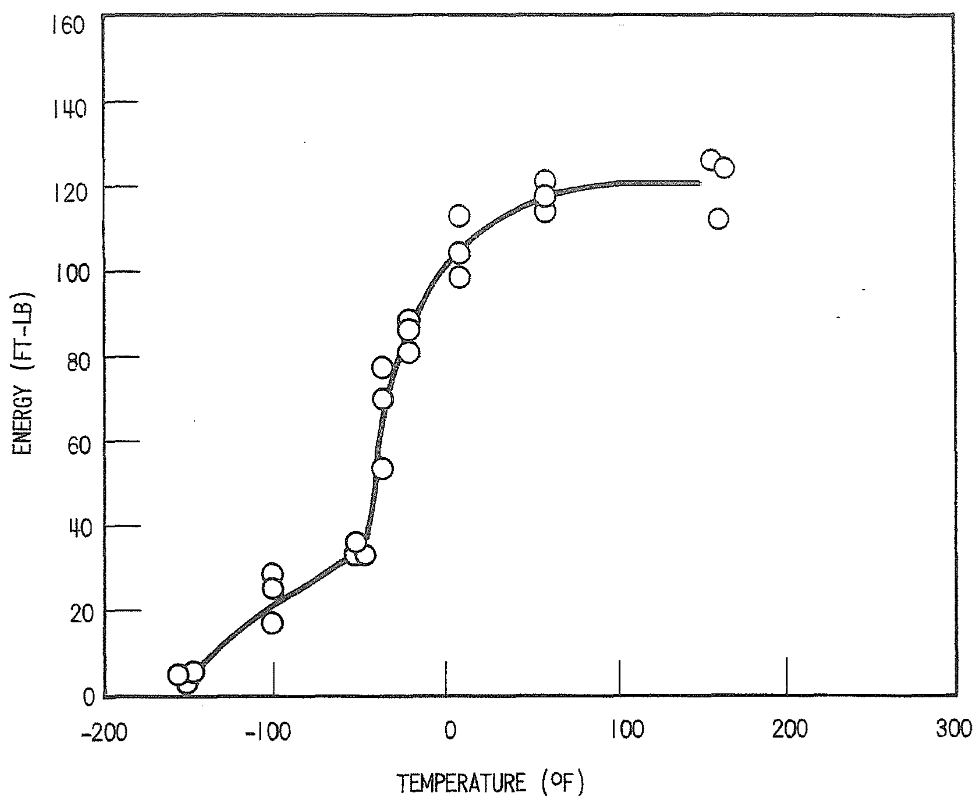


Figure 3-6. Preirradiation Charpy V-Notch Impact Energy for the Indian Point No. 3 Reactor Pressure Vessel Weld Metal

WCAP 8475

TABLE A-1

CHEMICAL COMPOSITION OF THE INDIAN POINT UNIT NO. 3
REACTOR VESSEL MATERIALS

Chemical Composition (wt-%)

Element	Intermediate Shell Course Plate B2802-1	Intermediate Shell Course Plate B2802-2	Intermediate Shell Course Plate B2802-3	Lower Shell Course Plate B2803-3	As-Deposited Weld Metal
C	0.22	0.19	0.20	0.22	0.08
Mn	1.41	1.33	1.32	1.30	1.18
P	0.010	0.015	0.011	0.012	0.019
S	0.023	0.019	0.025	0.024	0.016
Si	0.28	0.21	0.26	0.28	0.17
Ni	0.50	0.53	0.49	0.52	1.02
Cr	0.08	0.09	0.08	0.08	0.04
Mo	0.46	0.48	0.50	0.45	0.53
Cu	0.18	0.20	0.19	0.24	0.15
A1	0.036	0.027	0.042	0.03	<0.01
V	<0.01	<0.01	<0.01	<0.01	<0.01
Sn	0.014	0.017	0.014	<0.01	0.007
Cb	<0.01	<0.01	<0.01	<0.01	<0.01
Zr	<0.01	<0.01	<0.01	<0.01	<0.01
Ti	<0.01	<0.01	<0.01	<0.01	<0.01

All other elements (except Fe) were <0.01%.

Westinghouse Non-Proprietary Class 3

WCAP-16251-NP
Revision 0

July 2004

Analysis of Capsule X from Entergy's Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-16251-NP, Revision 0

Analysis of Capsule X from Entergy's Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program

T.J. Laubham
J. Conermann
S.L. Anderson

July 2004

Approved: J. Ghergurovich
J. Ghergurovich, Manager
Reactor Component Design & Analysis

Westinghouse Electric Company LLC
Energy Systems
P.O. Box 355
Pittsburgh, PA 15230-0355

EXECUTIVE SUMMARY

The purpose of this report is to document the results of the testing of surveillance Capsule X from Indian Point Unit 3. Capsule X was removed at 15.5 EFPY and post irradiation mechanical tests of the Charpy V-notch and tensile specimens were performed. A fluence evaluation utilizing the recently released neutron transport and dosimetry cross-section libraries was derived from the ENDF/B-VI data-base. Capsule X received a fluence of 0.874×10^{19} n/cm^2 after irradiation to 15.5 EFPY. The peak clad/base metal interface vessel fluence after 15.5 EFPY of plant operation was 5.86×10^{18} n/cm^2.

This evaluation lead to the following conclusions: 1) The measured 30 ft-lb shift in transition temperature values of the lower shell plate B2803-3 contained in capsule X (longitudinal & transverse) are greater than the Regulatory Guide 1.99, Revision 2, predictions. However, the shift values are less than the two sigma allowance by Regulatory Guide 1.99, Revision 2. 2) The measured 30 ft-lb shift in transition temperature value of the weld metal contained in capsule X is less than the Regulatory Guide 1.99, Revision 2, prediction. 3) The measured 30 ft-lb shift in transition temperature value of the intermediate shell plate B2802-2 contained in capsule X (longitudinal) is greater than the Regulatory Guide 1.99, Revision 2, prediction. However, the shift value is less than the two sigma allowance by Regulatory Guide 1.99, Revision 2. 4) The measured percent decrease in upper shelf energy for all the surveillance materials of Capsules X contained in the Indian Point Unit 3 surveillance program are in good agreement with the Regulatory Guide 1.99, Revision 2 predictions. 5) All beltline materials exhibit a more than adequate upper shelf energy level for continued safe plant operation and are predicted to maintain an upper shelf energy greater than 50 ft-lb throughout the life of the vessel (27.1 EFPY) as required by 10CFR50, Appendix G [2]. 6) The Indian Point Unit 3 surveillance data from the lower shell plate B2803-3 was found to be credible. This evaluation can be found in Appendix D.

Lastly, a brief summary of the Charpy V-notch testing can be found in Section 1. All Charpy V-notch data was plotted using a symmetric hyperbolic tangent curve fitting program.

1 SUMMARY OF RESULTS

The analysis of the reactor vessel materials contained in surveillance Capsule X, the fourth capsule removed and tested from the Indian Point Unit 3 reactor pressure vessel, led to the following conclusions:

- The Charpy V-notch data presented in WCAP-8475[3], WCAP-9491[4], WCAP-10300[5], and WCAP-11815[6] were based on hand-fit Charpy curves using engineering judgment. However, the results presented in this report are based on a re-plot of all applicable capsule data using CVGRAPH, Version 5.0.2, which is a hyperbolic tangent curve-fitting program. Appendix C presents the CVGRAPH, Version 5.02, Charpy V-notch plots and the program input data.

- Capsule X received an average fast neutron fluence (E> 1.0 MeV) of 0.874 x 10^{19} n/cm^2 after 15.5 effective full power years (EFPY) of plant operation.

- Irradiation of the reactor vessel lower shell plate B2803-3 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction (longitudinal orientation), resulted in an irradiated 30 ft-lb transition temperature of 191.6°F and an irradiated 50 ft-lb transition temperature of 223.8°F. This results in a 30 ft-lb transition temperature increase of 159.6°F and a 50 ft-lb transition temperature increase of 161.7°F for the longitudinal oriented specimens. See Table 5-9.

- Irradiation of the reactor vessel lower shell plate B2803-3 Charpy specimens, oriented with the longitudinal axis of the specimen perpendicular to the major working direction (transverse orientation), resulted in an irradiated 30 ft-lb transition temperature of 216.5°F and an irradiated 50 ft-lb transition temperature of 327.4°F. This results in a 30 ft-lb transition temperature increase of 158.2°F and a 50 ft-lb transition temperature increase of 217.9°F for the longitudinal oriented specimens. See Table 5-9.

- Irradiation of the weld metal (*heat number W5214*) Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of 128.5°F and an irradiated 50 ft-lb transition temperature of 196.8°F. This results in a 30 ft-lb transition temperature increase of 193.2°F and a 50 ft-lb transition temperature increase of 242.8°F. See Table 5-9.

- Irradiation of the reactor vessel intermediate shell plate B2802-2 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction (longitudinal orientation), resulted in an irradiated 30 ft-lb transition temperature of 98.1°F and an irradiated 50 ft-lb transition temperature of 145.0°F. This results in a 30 ft-lb transition temperature increase of 152.6°F and a 50 ft-lb transition temperature increase of 166.5°F for the longitudinal oriented specimens. See Table 5-9.

- The average upper shelf energy of the lower shell plate B2803-3 (longitudinal orientation) resulted in an average energy decrease of 24 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 81 ft-lb for the longitudinal oriented specimens. See Table 5-9.

- The average upper shelf energy of the lower shell plate B2803-3 (transverse orientation) resulted in an average energy decrease of 16 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 52 ft-lb for the longitudinal oriented specimens. See Table 5-9.

- The average upper shelf energy of the weld metal Charpy specimens resulted in an average energy decrease of 46 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 74 ft-lb for the weld metal specimens. See Table 5-9.

- The average upper shelf energy of the intermediate shell plate B2802-2 (longitudinal orientation) resulted in an average energy decrease of 20 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 105 ft-lb for the longitudinal oriented specimens. See Table 5-9.

- A comparison, as presented in Table 5-10, of the Indian Point Unit 3 reactor vessel surveillance material test results with the Regulatory Guide 1.99, Revision 2[1] predictions led to the following conclusions:

  - The measured 30 ft-lb shift in transition temperature values of the lower shell plate B2803-3 contained in capsule X (longitudinal & transverse) are greater than the Regulatory Guide 1.99, Revision 2, predictions. However, each shift value is less than the two sigma allowance by Regulatory Guide 1.99, Revision 2.

  - The measured 30 ft-lb shift in transition temperature value of the weld metal contained in capsule X is less than the Regulatory Guide 1.99, Revision 2, prediction.

  - The measured 30 ft-lb shift in transition temperature values of the intermediate shell plate B2802-2 contained in capsule X (longitudinal) is greater than the Regulatory Guide 1.99, Revision 2, prediction. However, the shift value is less than the two sigma allowance by Regulatory Guide 1.99, Revision 2.

  - The measured percent decrease in upper shelf energy for all the surveillance materials of Capsules X contained in the Indian Point Unit 3 surveillance program are in good agreement with the Regulatory Guide 1.99, Revision 2 predictions.

- All beltline materials exhibit a more than adequate upper shelf energy level for continued safe plant operation and are predicted to maintain an upper shelf energy greater than 50 ft-lb throughout the life of the vessel (27.1 EFPY) as required by 10CFR50, Appendix G [2].

2 INTRODUCTION

This report presents the results of the examination of Capsule X, the fourth capsule removed from the reactor in the continuing surveillance program which monitors the effects of neutron irradiation on the Indian Point Unit 3 reactor pressure vessel materials under actual operating conditions.

The surveillance program for the Indian Point Unit 3 reactor pressure vessel materials was designed and recommended by the Westinghouse Electric Corporation. A description of the surveillance program and the pre-irradiation mechanical properties of the reactor vessel materials are presented in WCAP-8475, "Consolidated Edison Co. of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program"[3]. The surveillance program was planned to cover the 40-year design life of the reactor pressure vessel and was based on ASTM E185-62, "Recommended Practice for Surveillance Tests on Structural Materials for Nuclear Reactors." Capsule X was removed from the reactor after 15.5 EFPY of exposure and shipped to the Westinghouse Science and Technology Department Hot Cell Facility, where the post-irradiation mechanical testing of the Charpy V-notch impact and tensile surveillance specimens was performed.

This report summarizes the testing of and the post-irradiation data obtained from surveillance capsule X removed from the Indian Point Unit 3 reactor vessel and discusses the analysis of the data.

Table 4-1 Chemical Composition (wt%) of the Indian Point Unit 3 Reactor Vessel Surveillance Materials (Unirradiated)[(a)]

	Intermediate Shell Plate			Lower Shell Plate	
Element	**B2802-1**	**B2802-2**	**B2802-3**	**B2803-3**	**Weld Metal [(b)]**
C	0.22	0.19	0.20	0.22	0.08
Mn	1.41	1.33	1.32	1.30	1.18
P	0.010	0.015	0.011	0.012	0.019
S	0.023	0.019	0.025	0.024	0.016
Si	0.28	0.21	0.26	0.28	0.17
Ni	0.50	0.53	0.49	0.52	1.02 (1.21)[(c)]
Cr	0.08	0.09	0.08	0.08	0.04
Mo	0.46	0.48	0.50	0.45	0.53
Cu	0.18	0.20	0.19	0.24	0.15 (0.166)[(c)]
Al	0.036	0.027	0.042	0.03	<0.01
V	<0.01	<0.01	<0.01	<0.01	<0.01
Sn	0.014	0.017	0.014	<0.01	0.007
Cb	<0.01	<0.01	<0.01	<0.01	<0.01
Zr	<0.01	<0.01	<0.01	<0.01	<0.01
Ti	<0.01	<0.01	<0.01	<0.01	<0.01

Notes:

(a) Data obtained from WCAP-11815 and duplicated herein for completeness.

(b) Weld wire Heat Number W5214, Flux Type Linde 1092, and Flux Lot Number 3692. Surveillance weldment has the same heat and flux as the nozzle shell longitudinal weld seams 1-042A, B & C.

(c) Results of chemical analysis performed on irradiated Charpy V-notch Specimen W-15 from Capsule Y.

The average upper shelf energy of the intermediate shell plate B2802-2 (longitudinal orientation) resulted in an average energy decrease of 20 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 105 ft-lb for the longitudinal oriented specimens. See Table 5-9.

A comparison, as presented in Table 5-10, of the Indian Point Unit 3 reactor vessel surveillance material test results with the Regulatory Guide 1.99, Revision 2[1] predictions led to the following conclusions:

- The measured 30 ft-lb shift in transition temperature values of the lower shell plate B2803-3 contained in capsule X (longitudinal & transverse) are greater than the Regulatory Guide 1.99, Revision 2, predictions. However, each shift value is less than the two sigma allowance by Regulatory Guide 1.99, Revision 2.

- The measured 30 ft-lb shift in transition temperature value of the weld metal contained in capsule X is less than the Regulatory Guide 1.99, Revision 2, predictions

- The measured 30 ft-lb shift in transition temperature values of the intermediate shell plate B2802-2 contained in capsule X (longitudinal) is greater than the Regulatory Guide 1.99, Revision 2, prediction. However, the shift value is less than the two sigma allowance by Regulatory Guide 1.99, Revision 2.

- The measured percent decrease in upper shelf energy for all the surveillance materials of Capsules X contained in the Indian Point Unit 3 surveillance program are in good agreement with the Regulatory Guide 1.99, Revision 2 predictions.

All beltline materials exhibit a more than adequate upper shelf energy level for continued safe plant operation and are predicted to maintain an upper shelf energy greater than 50 ft-lb throughout the extended life of the vessel (27.1 EFPY) as required by 10CFR50, Appendix G [2].

The fracture appearance of each irradiated Charpy specimen from the various surveillance Capsule X materials is shown in Figures 5-13 through 5-16 and shows an increasingly ductile or tougher appearance with increasing test temperature.

The load-time records for individual instrumented Charpy specimen tests are shown in Appendix B.

The Charpy V-notch data presented in WCAP-8475[3], WCAP-9491[4], WCAP-10300[5], and WCAP-11815[6] were based on hand-fit Charpy curves using engineering judgment. However, the results presented in this report are based on a re-plot of all applicable capsule data using CVGRAPH, Version 5.0.2, which is a hyperbolic tangent curve-fitting program. This report also shows the composite plots that show the results from the previous capsule. Appendix C presents the CVGRAPH, Version 5.02, Charpy V-notch plots and the program input data.

Table 5-3 Charpy V-notch Data for the Indian Point Unit 3 Surveillance Weld Metal Irradiated to a Fluence of 0.874×10^{19} n/cm^2 (E> 1.0 MeV)							
Sample	**Temperature**		**Impact Energy**		**Lateral Expansion**		**Shear**
Number	**°F**	**°C**	**ft-lbs**	**Joules**	**mils**	**mm**	**%**
W42	75	24	9	12	5	0.13	20
W41	125	52	49	66	36	0.91	50
W43	125	52	24	33	19	0.48	40
W48	150	66	35	47	26	0.66	45
W47	200	93	37	50	30	0.76	70
W44	250	121	67	91	52	1.32	95
W45	300	149	72	98	56	1.42	98
W46	350	177	75	102	57	1.45	100

Table 5-7 Instrumented Charpy Impact Test Results for the Indian Point Unit 3 Surveillance Weld Metal Irradiated to a Fluence of 0.874 x 10^{19} n/cm^2 (E>1.0 MeV)													
Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load P_{GY} (lb)	Time to Yield t_{GY} (msec)	Max. Load P_M (lb)	Time to Max. t_M (msec)	Fast Fract. Load P_F (lb)	Arrest Load P_A (lb)	Yield Stress σ_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_p/A								
W42	75	9	73	36	36	3426	0.14	3696	0.16	3687	0	114	119
W41	125	49	395	226	169	3411	0.15	4363	0.52	4288	617	114	129
W43	125	24	193	68	126	3341	0.14	4109	0.22	4058	1313	111	124
W48	150	35	282	184	98	3416	0.14	4449	0.42	4417	1141	114	131
W47	200	37	298	150	148	3371	0.14	4260	0.37	4222	1713	112	127
W44	250	67	540	227	313	3486	0.14	4432	0.50	4251	2819	116	132
W45	300	72	580	218	362	3329	0.14	4303	0.50	3029	2501	111	127
W46	350	75	604	221	383	3285	0.14	4309	0.51	n/a	n/a	109	126

Table 5-10 Comparison of the Indian Point Unit 3 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

Material	Capsule	Fluence[d] (x 10^{19} n/cm^2, E > 1.0 MeV)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F) [a]	Measured (°F) [b]	Predicted (%) [a]	Measured (%)[c]
Lower Shell Plate	T	0.263	101.9	139.4	24	12
B2803-3	Z	1.04	161.6	167.8	33.5	22
(Longitudinal)	X	0.874	153.9	159.6	32	23
Lower Shell Plate	T	0.263	101.9	105.9	24	16
B2803-3	Y	0.692	143.5	148.9	30	25
	Z	1.04	161.6	157.9	33.5	18
(Transverse)	X	0.874	153.9	158.2	32	24
Surveillance	T	0.263	131.3	151.6	22	30
Program	Y	0.692	185.0	172.0	27	43
Weld Metal	Z	1.04	208.3	229.2	31	37
	X	0.874	198.4	193.2	29	38
Intermediate Shell Plate B2802-2 *(Longitudinal)*	X	0.874	146.2	152.6	30	16

Notes:

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 5.0.2 (See Appendix C)

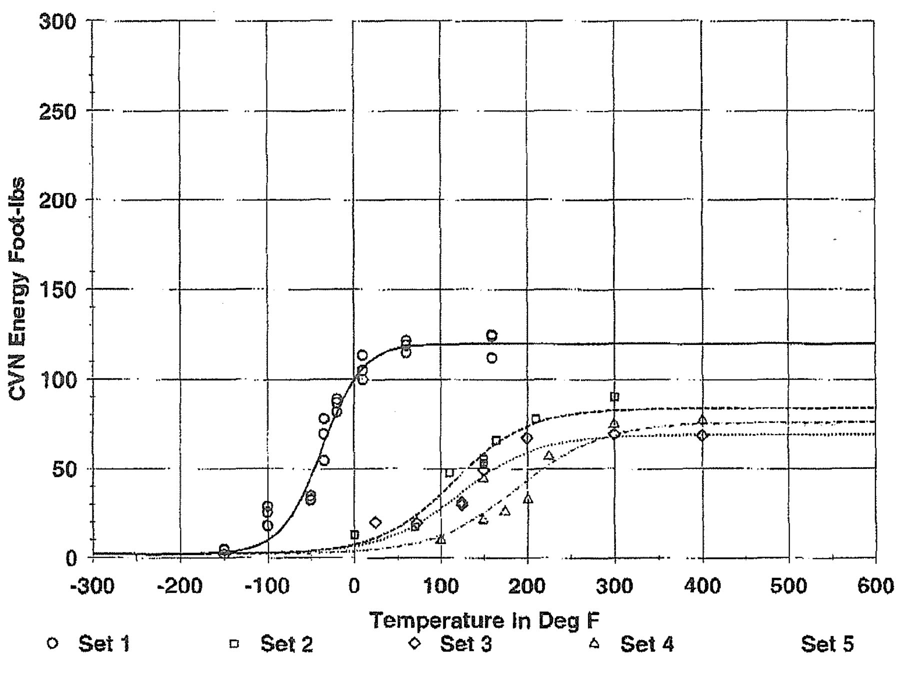
(c) Values are based on the definition of upper shelf energy given in ASTM E185-82.

(d) The fluence values presented here are the calculated values, not the best estimate values.

SURVEILLANCE WELD MATERIAL

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 04/02/2004 02:31 PM
Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Indian Point 3	UNIRR	SAW	NA	W5214
2	Indian Point 3	T	SAW	NA	W5214
3	Indian Point 3	Y	SAW	NA	W5214
4	Indian Point 3	Z	SAW	NA	W5214
5	Indian Point 3	X	SAW	NA	W5214



Results

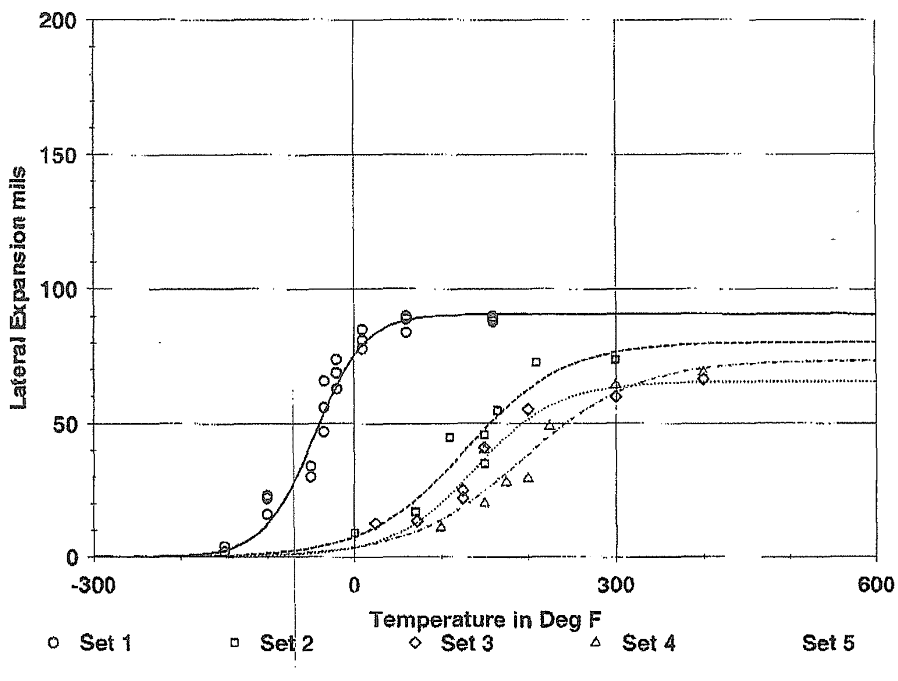
Curve	Fluence	LSE	USE	d-USE	T @30	d-T @30	T @50	d-T @50
1		2.2	120.0	.0	-64.7	.0	-46.0	.0
2		2.2	84.0	-36.0	86.9	151.6	130.6	176.6
3		2.2	69.0	-51.0	107.3	172.0	164.2	210.2
4		2.2	76.0	-44.0	164.5	229.2	218.7	264.7
5		2.2	74.0	-46.0	128.5	193.2	196.8	242.8

Figure 5-7 Charpy V-Notch Impact Energy vs. Temperature for Indian Point Unit 3 Reactor Vessel Weld Metal

SURVEILLANCE WELD MATERIAL

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 04/02/2004 04:11 PM
Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Indian Point 3	UNIRR	SAW	NA	W5214
2	Indian Point 3	T	SAW	NA	W5214
3	Indian Point 3	Y	SAW	NA	W5214
4	Indian Point 3	Z	SAW	NA	W5214
5	Indian Point 3	X	SAW	NA	W5214



Results

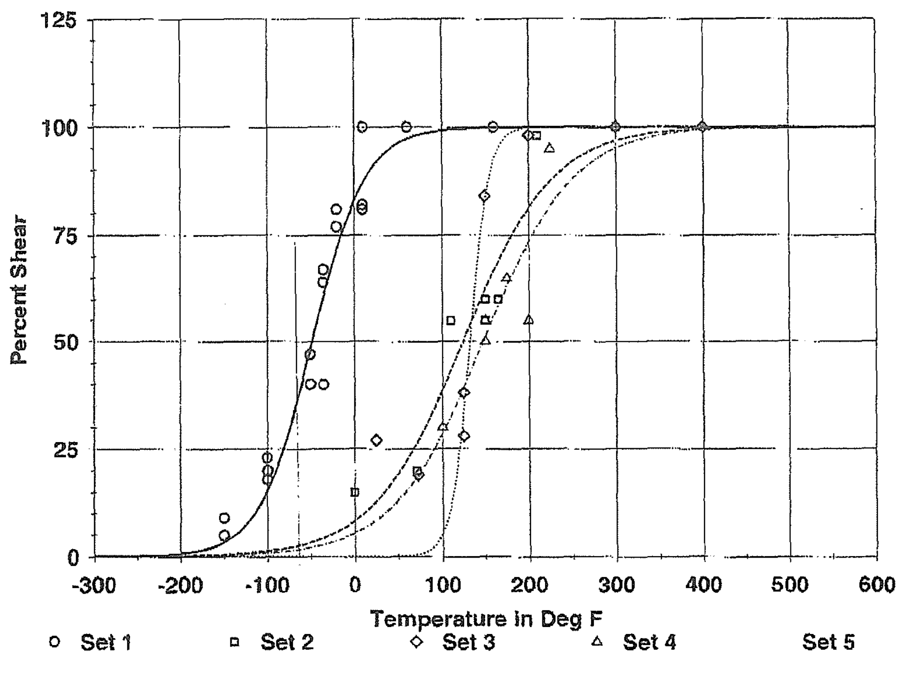
Curve	Fluence	LSE	USE	d-USE	T @35	d-T @35
1		.0	90.8	.0	-59.3	.0
2		.0	80.3	-10.5	113.3	172.6
3		.0	65.6	-25.2	145.0	204.3
4		.0	73.5	-17.3	187.8	247.1
5		.0	62.1	-28.7	184.6	243.9

Figure 5-8 Charpy V-Notch Lateral Expansion vs. Temperature for Indian Point Unit 3 Reactor Vessel Weld Metal

SURVEILLANCE WELD MATERIAL

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 04/02/2004 03:57 PM
Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Indian Point 3	UNIRR	SAW	NA	W5214
2	Indian Point 3	T	SAW	NA	W5214
3	Indian Point 3	Y	SAW	NA	W5214
4	Indian Point 3	Z	SAW	NA	W5214
5	Indian Point 3	X	SAW	NA	W5214



Results

Curve	Fluence	LSE	USE	d-USE	T @50	d-T @50
1		.0	100.0	.0	-47.8	.0
2		.0	100.0	.0	124.0	171.8
3		.0	100.0	.0	132.6	180.4
4		.0	100.0	.0	147.5	195.3
5		.0	100.0	.0	144.5	192.3

Figure 5-9 Charpy V-Notch Percent Shear vs. Temperature for Indian Point Unit 3 Reactor Vessel Weld Metal

APPENDIX C

CHARPY V-NOTCH PLOTS FOR EACH CAPSULE USING SYMMETRIC HYPERBOLIC TANGENT CURVE-FITTING METHOD

UNIRRADIATED (WELD)

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 04/02/2004 02:26 PM
Page 1
Coefficients of Curve 1

A = 61.1 B = 58.9 C = 47.03 T0 = -37.11 D = 0.00E+00

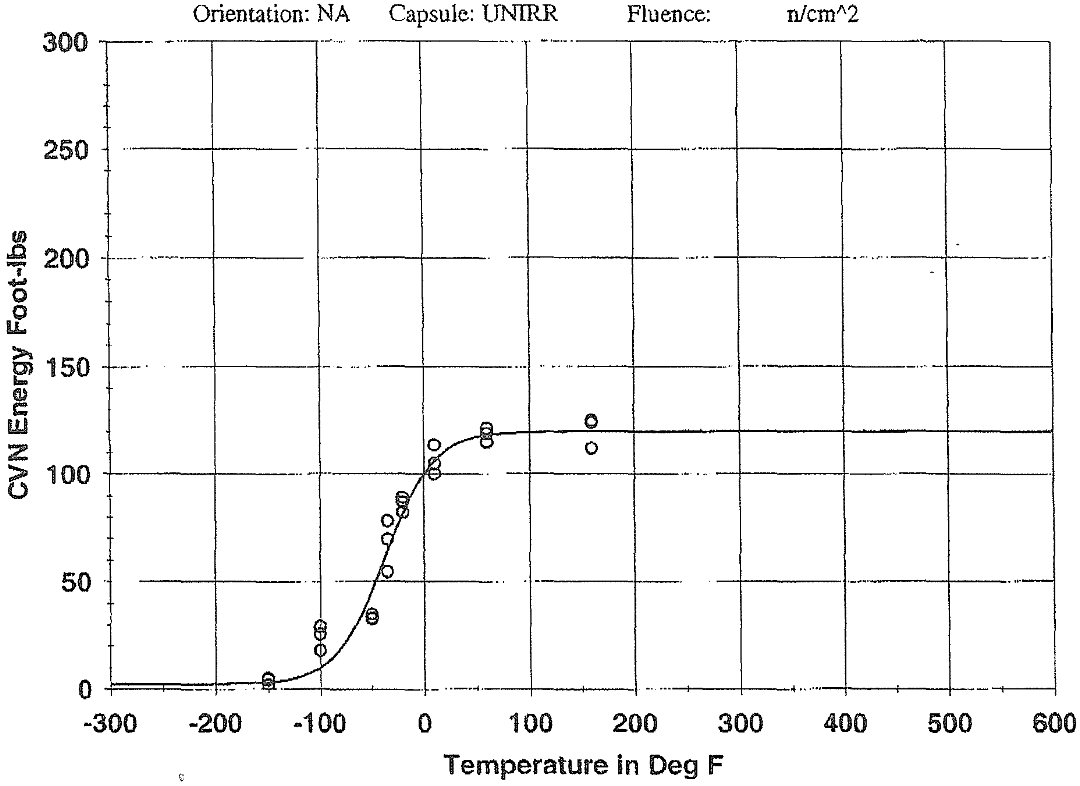
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=120.0(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=-64.7 Deg F Temp@50 ft-lbs=-46.0 Deg F

Plant: Indian Point 3 Material: SAW Heat: W5214

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



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-150.00	5.00	3.16	1.84
-150.00	2.00	3.16	-1.16
-150.00	4.50	3.16	1.34
-100.00	29.00	9.80	19.20
-100.00	18.00	9.80	8.20
-100.00	25.50	9.80	15.70
-50.00	35.00	45.35	-10.35
-50.00	33.00	45.35	-12.35
-50.00	32.50	45.35	-12.85

UNIRRADIATED (WELD)

Page 2
Plant: Indian Point 3 Material: SAW Heat: W5214
Orientation: NA Capsule: UNIRR Fluence: n/cm^2

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-35.00	78.00	63.74	14.26
-35.00	69.50	63.74	5.76
-35.00	54.50	63.74	-9.24
-20.00	87.00	81.63	5.37
-20.00	82.00	81.63	.37
-20.00	89.00	81.63	7.37
10.00	100.00	106.00	-6.00
10.00	105.00	106.00	-1.00
10.00	113.50	106.00	7.50
60.00	115.00	118.14	-3.14
60.00	119.00	118.14	.86
60.00	121.50	118.14	3.36
160.00	124.00	119.97	4.03
160.00	125.00	119.97	5.03
160.00	112.00	119.97	-7.97

Correlation Coefficient = .981

CAPSULE T (WELD)

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 04/02/2004 02:26 PM
Page 1
Coefficients of Curve 2

A = 43.1 B = 40.9 C = 87.09 T0 = 115.75 D = 0.00E+00

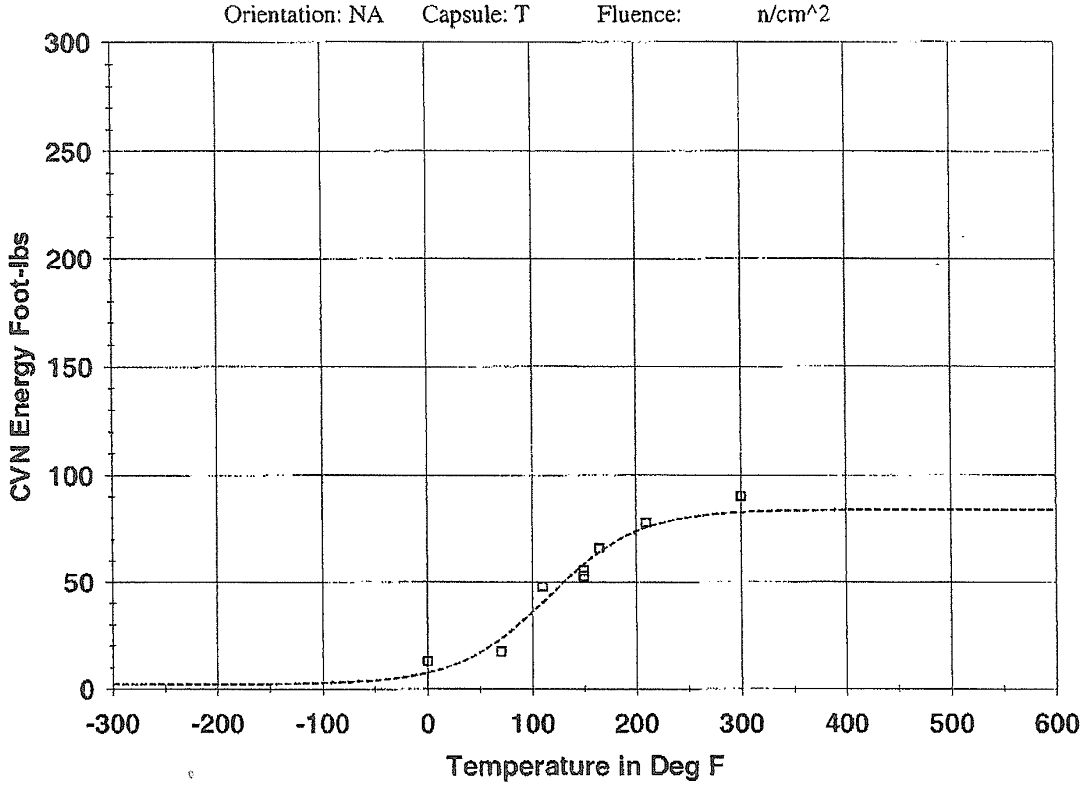
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=84.0(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=86.9 Deg F Temp@50 ft-lbs=130.6 Deg F

Plant: Indian Point 3 Material: SAW Heat: W5214

Orientation: NA Capsule: T Fluence: n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
.00	13.00	7.56	5.44
70.00	17.50	23.39	-5.89
110.00	48.00	40.40	7.60
150.00	55.50	58.40	-2.90
150.00	53.00	58.40	-5.40
165.00	66.00	64.04	1.96
210.00	78.00	75.58	2.42
300.00	90.50	82.83	7.67

Correlation Coefficient = .979

CAPSULE Y (WELD)

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 04/02/2004 02:26 PM

Page 1

Coefficients of Curve 3

A = 35.6 B = 33.4 C = 90.16 T0 = 122.54 D = 0.00E+00

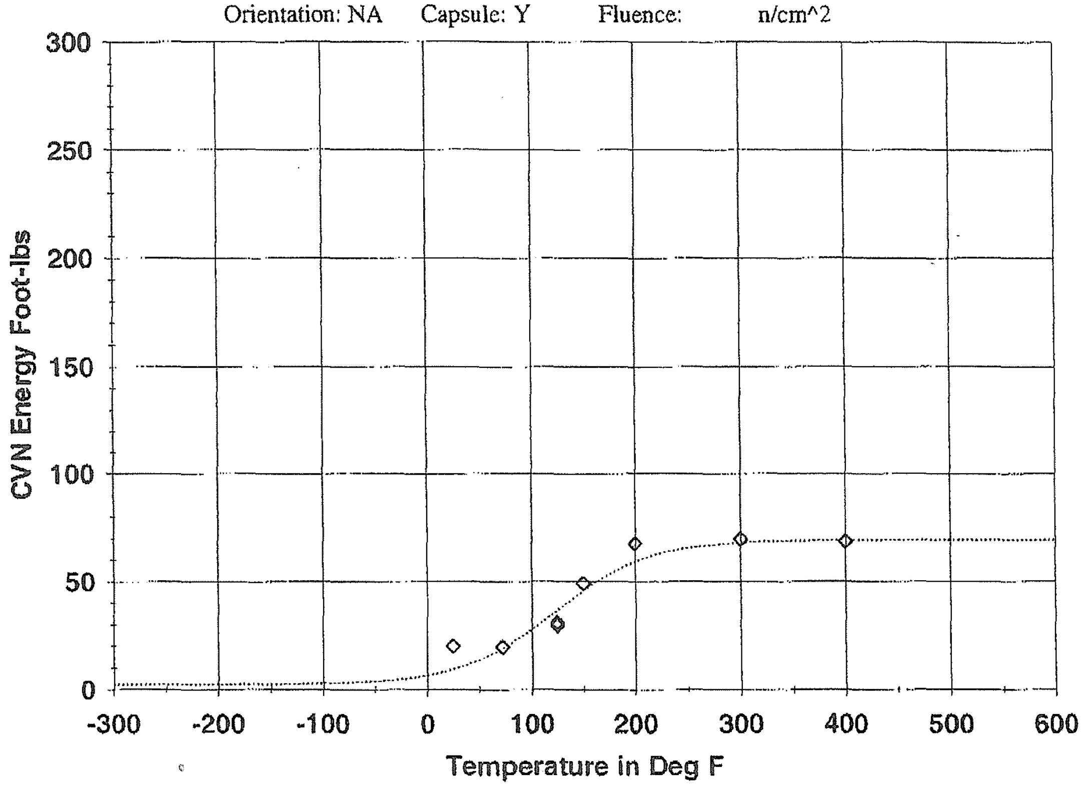
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=69.0(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=107.3 Deg F Temp@50 ft-lbs=164.2 Deg F

Plant: Indian Point 3 Material: SAW Heat: W5214

Orientation: NA Capsule: Y Fluence: n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
25.00	20.00	9.08	10.92
72.00	19.50	18.62	.88
125.00	31.00	36.51	-5.51
125.00	29.50	36.51	-7.01
150.00	49.00	45.47	3.53
200.00	67.50	58.84	8.66
300.00	69.50	67.72	1.78
400.00	68.50	68.86	-.36

Correlation Coefficient = .960

CAPSULE Z (WELD)

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 04/02/2004 02:26 PM

Page 1

Coefficients of Curve 4

A = 39.1 B = 36.9 C = 97.52 T0 = 188.96 D = 0.00E+00

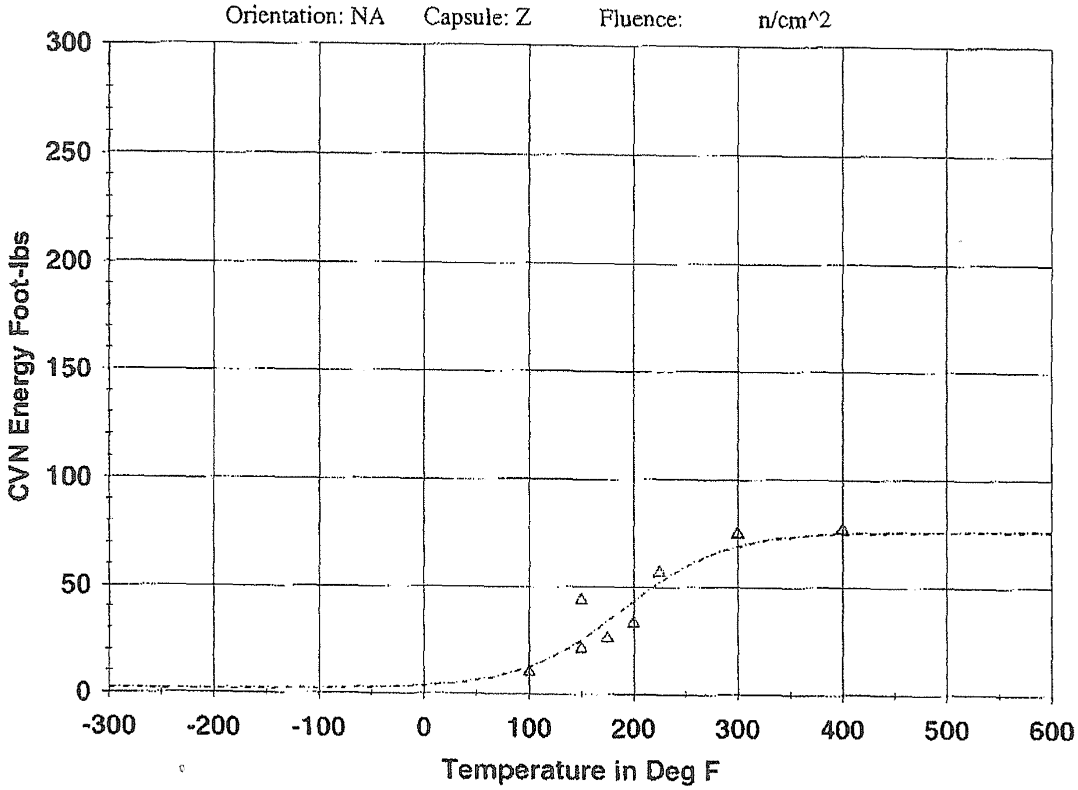
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=76.0(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=164.5 Deg F Temp@50 ft-lbs=218.7 Deg F

Plant: Indian Point 3 Material: SAW Heat: W5214

Orientation: NA Capsule: Z Fluence: n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
100.00	10.00	12.45	-2.45
150.00	21.00	25.10	-4.10
150.00	44.00	25.10	18.90
175.00	26.00	33.85	-7.85
200.00	33.00	43.26	-10.26
225.00	57.00	52.15	4.85
300.00	75.00	69.14	5.86
400.00	77.00	75.04	1.96

Correlation Coefficient = .929

CAPSULE X (WELD)

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 04/02/2004 02:36 PM

Page 1

Coefficients of Curve 5

A = 38.1 B = 35.9 C = 118.98 T0 = 155.76 D = 0.00E+00

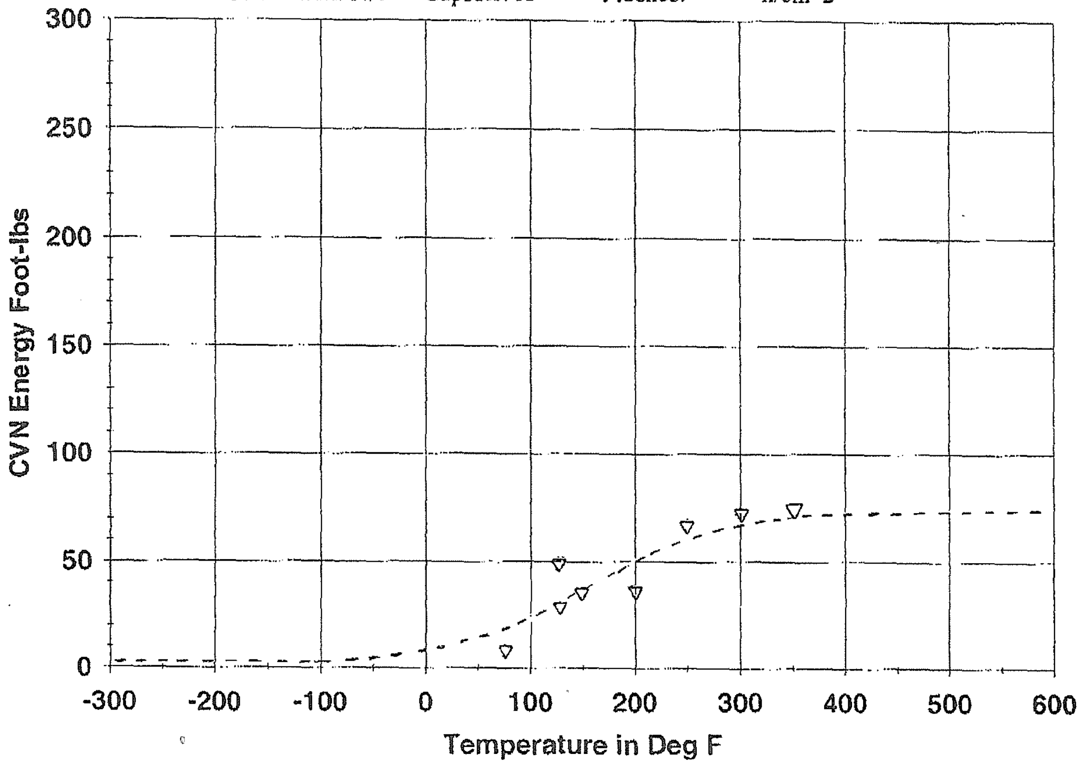
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=74.0(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=128.5 Deg F Temp@50 ft-lbs=196.8 Deg F

Plant: Indian Point 3 Material: SAW Heat: W5214

Orientation: NA Capsule: X Fluence: n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
75.00	9.00	16.89	-7.89
125.00	49.00	29.02	19.98
125.00	24.00	29.02	-5.02
150.00	35.00	36.36	-1.36
200.00	37.00	50.87	-13.87
250.00	67.00	61.78	5.22
300.00	72.00	68.16	3.84
350.00	75.00	71.36	3.64

Correlation Coefficient = .906

APPENDIX F

H. B. ROBINSON 2 REACTOR VESSEL MATERIALS SURVEILLANCE PROGRAM RESULTS FOR WELD NO. W5214

Proprietary C

WCAP - 7373

CAROLINA POWER AND LIGHT CO.

H. B. ROBINSON UNIT NO. 2 REACTOR VESSEL

RADIATION SURVEILLANCE PROGRAM

By

S. E. Yanichko

January 1970

Work Performed Under CPL-106

Approved: E. Landerman

E. Landerman

Westinghouse Electric Corporation
Nuclear Energy Systems
Box 355
Pittsburgh, Pennsylvania 15230

WCAP 7373

TABLE 4

PRE-IRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE CPL H. B. ROBINSON UNIT NO. 2 REACTOR PRESSURE VESSEL WELD METAL AND WELD HEAT AFFECTED ZONE METAL

Weld Metal					Heat Affected Zone				
Specimen No.	Test Temp (°F)	Energy (ft-lb)	Shear (%)	Lateral Expansion (mils)	Specimen No.	Test Temp (°F)	Energy (ft-lb)	Shear (%)	Lateral Expansion (mils)
W25	-150	19.0	29	16	H28	-150	11.0	10	12
W26	-150	10.0	23	9	H29	-150	31.0	18	29
W27	-150	30.0	29	25	H30	-150	34.0	18	30
W34	-150	3.0	9	2					
W35	-150	34.5	26	28	H25	-100	39.5	29	30
W36	-150	2.0	9	2	H26	-100	42.0	37	34
					H27	-100	41.0	34	30
W28	-100	38.0	30	34					
W29	-100	29.0	23	19	H31	- 50	42.5	34	35
W30	-100	25.0	20	22	H32	- 50	60.0	45	50
					H33	- 50	37.5	32	30
W31	- 50	21.0	25	20					
W32	- 50	54.5	36	49	H34	- 20	75.0	42	61
W33	- 50	36.5	30	31	H35	- 20	86.0	61	62
					H36	- 20	45.0	34	37
W37	10	73.5	64	62					
W38	10	68.0	61	58	H37	10	83.0	81	67
W39	10	65.5	59	57	H38	10	119.0	90	81
					H39	10	94.0	77	70
W40	60	97.0	90	80					
W41	60	99.0	91	80	H40	60	116.0	100	88
W42	60	116.0	94	88	H41	60	111.5	100	87
					H42	60	110.0	95	87
W43	110	97.0	95	74					
W44	110	104.0	100	85	H43	110	117.0	100	93
W45	100	107.5	98	89	H44	110	140.0	100	83
					H45	110	119.0	100	86
W46	210	112.0	100	90					
W47	210	111.0	100	91	H46	210	130.0	100	89
W48	210	115.0	100	83	H47	210	134.0	100	86
					H48	210	123.0	100	84

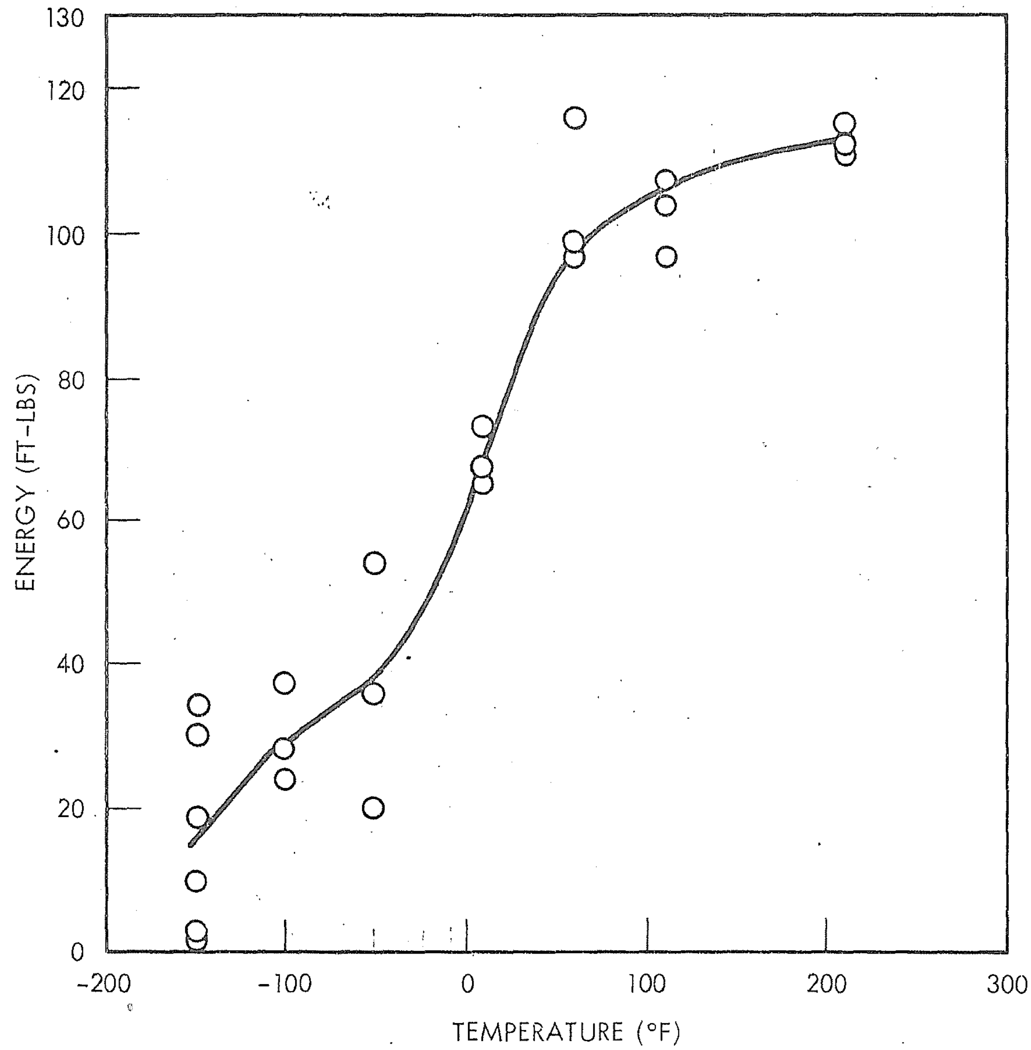


Figure 10. Pre-Irradiation Charpy V-Notch Impact Energy for the CPL H.B. Robinson Unit #2 Reactor Pressure Vessel Weld Metal

05215643

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15805

Analysis of Capsule X from the Carolina Power & Light Company H.B. Robinson Unit 2 Reactor Vessel Radiation Surveillance Program

T. J. Laubham
E. P. Lippincott
J. Conermann

MARCH 2002

Prepared by the Westinghouse Electric Company
for the Carolina Power & Light Company

Approved: C. H. Boyd
C. H . Boyd, Manager
Equipment & Materials Technology

Westinghouse Electric Company LLC
Nuclear Services Division
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

EXECUTIVE SUMMARY

The purpose of this report is to document the results of the testing of surveillance capsule X from H.B. Robinson Unit 2. Capsule X was removed at 20.39 EFPY and post irradiation mechanical tests of the Charpy V-notch and tensile specimens was performed, along with a fluence evaluation based methodology and nuclear data including recently released neutron transport and dosimetry cross-section libraries derived from the ENDF/B-VI database. The calculated peak clad base/metal vessel fluence after 20.39 EFPY of plant operation was 2.76×10^{19} n/cm^2 and the surveillance Capsule X calculated fluence was 4.49×10^{19} n/cm^2. A brief summary of the Charpy V-notch testing results can be found in Section 1 and the updated capsule removal schedule can be found in Section 7. A supplement to this report is a credibility evaluation, which can be found in Appendix D, that shows the H.B. Robinson Unit 2 surveillance weld data, while including all surveillance data for weld heat W5214, is credible. Of the three surveillance plates, only intermediate shell plate W10201-5 was found to be credible.

1 SUMMARY OF RESULTS

The analysis of the reactor vessel materials contained in surveillance capsule X the fourth capsule to be removed from the H.B. Robinson Unit 2 reactor pressure vessel, led to the following conclusions: (General Note: Temperatures are reported to two significant digits only to match CVGraph output.)

- The capsule received an average fast neutron calculated fluence (E > 1.0 MeV) of 4.49 x 10^{19} n/cm^2 after 20.39 effective full power years (EFPY) of plant operation.

- Irradiation of the reactor vessel intermediate shell plate W10201-4 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction of the plate (longitudinal orientation), to 4.49 x 10^{19} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 104.73°F and a 50 ft-lb transition temperature increase of 98.68°F. This results in an irradiated 30 ft-lb transition temperature of 86.55°F and an irradiated 50 ft-lb transition temperature of 116.04°F for the longitudinally oriented specimens

- Irradiation of the weld metal Charpy specimens to 4.49 x 10^{19} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 265.93°F and a 50 ft-lb transition temperature increase of 251.74°F. This results in an irradiated 30 ft-lb transition temperature of 179.64°F and an irradiated 50 ft-lb transition temperature of 211.38°F.

- Irradiation of the weld Heat-Affected-Zone (HAZ) metal Charpy specimens to 4.49 x 10^{19} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature increase of 210.13°F and a 50 ft-lb transition temperature increase of 216.59°F. This results in an irradiated 30 ft-lb transition temperature of 100.47°F and an irradiated 50 ft-lb transition temperature of 150.54°F.

- Irradiation of the correlation monitor material Charpy specimens to 4.49 x 10^{19} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature increase of 125.21°F which resulted in an irradiated 30 ft-lb transition temperature of 188.15°F. The tested specimens did not reach the 50 ft-lb transition temperature.

- The average upper shelf energy of the intermediate shell plate W10201-4 (longitudinal orientation) resulted in an average energy decrease of 1 ft-lb after irradiation to 4.49 x 10^{19} n/cm^2 (E> 1.0 MeV). This results in an irradiated average upper shelf energy of 94 ft-lb for the longitudinally oriented specimens.

- The average upper shelf energy of the weld metal Charpy specimens resulted in an average energy decrease of 33 ft-lb after irradiation to 4.49 x 10^{19} n/cm^2 (E> 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 80 ft-lb for the weld metal specimens.

- The average upper shelf energy of the weld HAZ metal Charpy specimens resulted in an average energy decrease of 24 ft-lb after irradiation to 4.49 x 10^{19} n/cm^2 (E > 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 105 ft-lb for the weld HAZ metal.

Table 4-1 Chemical Composition (wt %) and Heat Treatment of Material for the H.B. Robinson Unit 2 Reactor Vessel Surveillance Material[a]

Chemical Composition					
Element	**Plate W10201-4**	**Plate W10201-5**	**Plate W10201-6**	**Weld Metal**	**Correlation Monitor Material**
C	0.19	0.20	0.19	0.16	0.24
Mn	1.35	1.29	1.32	0.98	1.34
P	0007	0.010	0.010	0.021	0.011
S	0.019	0.021	0.015	0.014	0.023
Si	0.23	0.22	0.19	0.34	0.23
Mo	0.48	0.46	0.49	0.46	0.51
Cu	0.12	0.10	0.09	0.34	0.20
V	---	---	---	0.001	---
Ni	---	---	---	0.66	0.18
Cr	---	---	---	0.024	0.11
Co	---	---	---	---	---

Heat Treatment	
Plate W10201-4,	1550°F to 1600°F, 4 hours, Water Quench
Plate W10201-5, &	1200°F to 1250°F, 4 hours, Air Cooled
Plate W10201-6	1125°F to 1175°F, 15 1/2 hours, Furnace cooled to 600°F
Weld Metal	1125°F to 1175°F, 30 hours, Furnace cooled to 600°F
Correlation Monitor	1650°F, 4 hours, Water Quenched 1200°F – 6 hours, Air Cooled

Notes:

a) The data given in this column (originally) is from WCAP-7373 & WCAP-10304.

5.2 CHARPY V-NOTCH IMPACT TEST RESULTS

The results of the Charpy V-notch impact tests performed on the various materials contained in capsule X, which received a fluence of 4.49 x 10^{19} n/cm^2 (E > 1.0 MeV) in 20.39 EFPY of operation, are presented in Tables 5-1 through 5-8, and are compared with unirradiated results as shown in Figures 5-1 through 5-12.

The transition temperature increases and upper shelf energy decreases for the capsule X materials are summarized in Table 5-9. These results led to the following conclusions:

Irradiation of the reactor vessel intermediate shell plate W10201-4 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction of the plate (longitudinal orientation), to 4.49 x 10^{19} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 104.73°F and a 50 ft-lb transition temperature increase of 98.68°F. This results in an irradiated 30 ft-lb transition temperature of 86.55°F and an irradiated 50 ft-lb transition temperature of 116.04°F for the longitudinally oriented specimens

Irradiation of the weld metal Charpy specimens to 4.49 x 10^{19} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 265.93°F and a 50 ft-lb transition temperature increase of 251.74°F. This results in an irradiated 30 ft-lb transition temperature of 179.64°F and an irradiated 50 ft-lb transition temperature of 211.38°F.

Irradiation of the weld Heat-Affected-Zone (HAZ) metal Charpy specimens to 4.49 x 10^{19} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature increase of 210.13°F and a 50 ft-lb transition temperature increase of 216.59°F. This results in an irradiated 30 ft-lb transition temperature of 100.47°F and an irradiated 50 ft-lb transition temperature of 150.54°F.

Irradiation of the correlation monitor material Charpy specimens to 4.49 x 10^{19} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature increase of 125.21°F which resulted in an irradiated 30 ft-lb transition temperature of 188.15°F. The tested specimens did not reach the 50 ft-lb transition temperature.

The average upper shelf energy of the intermediate shell plate W10201-4 (longitudinal orientation) resulted in an average energy decrease of 1 ft-lb after irradiation to 4.49 x 10^{19} n/cm^2 (E> 1.0 MeV). This results in an irradiated average upper shelf energy of 94 ft-lb for the longitudinally oriented specimens.

The average upper shelf energy of the weld metal Charpy specimens resulted in an average energy decrease of 33 ft-lb after irradiation to 4.49 x 10^{19} n/cm^2 (E> 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 80 ft-lb for the weld metal specimens.

The average upper shelf energy of the weld HAZ metal Charpy specimens resulted in an average energy decrease of 24 ft-lb after irradiation to 4.49 x 10^{19} n/cm^2 (E > 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 105 ft-lb for the weld HAZ metal.

The average upper shelf energy of the correlation monitor material Charpy specimens resulted in no energy decrease after irradiation to 4.49 x 10^{19} n/cm^2 (E > 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 42 ft-lb for the correlation monitor material.

A comparison of the H.B. Robinson Unit 2 reactor vessel beltline material test results with the Regulatory Guide 1.99, Revision 2[1], predictions led to the following conclusions:

- The measured 30 ft-lb shift in transition temperature values of the intermediate shell plate W10201-4 contained in capsule X (longitudinal) is greater than the Regulatory Guide 1.99, Revision 2, predictions. However, the shift value is less than two sigma allowance by Regulatory Guide 1.99, Revision 2.

- The measured 30 ft-lb shift in transition temperature values of the weld metal contained in capsule X (longitudinal) is less than the Regulatory Guide 1.99, Revision 2, predictions.

- The measured percent decrease in upper shelf energy of the Capsule X surveillance material is less than the Regulatory Guide 1.99, Revision 2, predictions.

The fracture appearance of each irradiated Charpy specimen from the various surveillance capsule X materials is shown in Figures 5-13 through 5-16 and show an increasingly ductile or tougher appearance with increasing test temperature.

The load-time records for individual instrumented Charpy specimen tests are shown in Appendix A.

The Charpy V-notch data presented in this report is based on a re-plot of all capsule data using CVGRAPH, Version 4.1, which is a hyperbolic tangent curve-fitting program. Hence, Appendix C contains a comparison of the Charpy V-notch shift results for each surveillance material (hand-fitting versus hyperbolic tangent curve-fitting). Additionally, Appendix B presents the CVGRAPH, Version 4.1, Charpy V-notch plots and the program input data.

Table 5-2 Charpy V-notch Data for the H.B. Robinson Unit 2 Surveillance Weld Metal Irradiated to a Fluence of 4.49×10^{19} n/cm^2 (E> 1.0 MeV)							
Sample	**Temperature**		**Impact Energy**		**Lateral Expansion**		**Shear**
Number	**F**	**C**	**ft-lbs**	**Joules**	**mils**	**mm**	**%**
W3	0	-18	4	5	0	0.00	0
W2	100	38	14	19	4	0.10	15
W6	175	79	28	38	16	0.41	35
W4	200	93	38	52	22	0.56	40
W8	250	121	74	100	49	1.24	100
W7	350	177	78	106	51	1.30	100
W5	375	191	85	115	56	1.42	100
W1	425	218	82	111	54	1.37	100

Table 5-10 Comparison of the H.B. Robinson Unit 2 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

Material	Capsule	Fluence ($x 10^{19}$ n/cm^2)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F) [(a)]	Measured (°F) [(b)]	Predicted (%) [(a)]	Measured (%)[(c)]
Inter. Shell Plate W10201-4 (Longitudinal)	S	0.479	45.39	32.51	18	10
	X	4.49	78.86	104.73	30	1
Surveillance Program Weld Metal	V	0.530	179.17	209.32	39	38
	T	3.87	293.68	288.15	52	46
	X	4.49	300.64	265.93	54	29
Heat Affected Zone Material	V	0.530	--	59.21	---	26
	T	3.87	--	(d)	---	24
	X	4.49	--	210.13	---	19
Correlation Monitor Material	S	0.479	--	72.79	---	3
	V	0.530	--	69.39	---	5
	T	3.87	--	156.83	---	5
	X	4.49	--	125.21	---	0

Notes:

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1 (See Appendix B)

(c) Values are based on the definition of upper shelf energy given in ASTM E185-82.

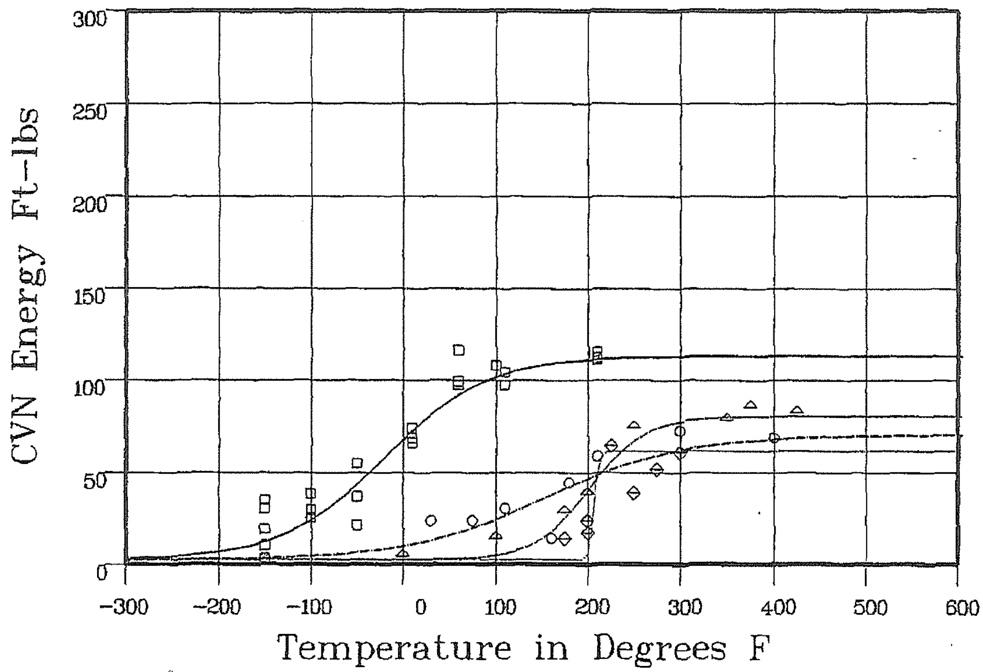
(d) Only 2 specimens were tested from capsule T to confirm the upper shelf energy, thus, there is insufficient data to determine the measured 30 ft-lb shift.

SURVELLIANCE PROGRAM WELD MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:54:57 on 10-24-2001

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	113	0	-86.29	0	-40.35	0
2	0	2.19	0	70	-43	123.02	209.32	214.59	254.94
3	0	2.2	0	61	-52	201.86	288.15	205.94	246.3
4	0	2.19	0	80	-33	179.64	265.93	211.38	251.74



Curve Legend

1 □——— 2 ○-------- 3 ◇——— 4 △———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	HB2	UNIRR	WELD N/A		W5214
2	HB2	V	WELD N/A		W5214
3	HB2	T	WELD N/A		W5214
4	HB2	X	WELD N/A		W5214

Figure 5-4 Charpy V-Notch Impact Energy vs. Temperature for H.B. Robinson Unit 2 Reactor Vessel Surveillance Weld Metal

7 SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following surveillance capsule removal schedule meets the intent of ASTM E185-82 and is recommended for future capsules to be removed from the H.B. Robinson Unit 2 reactor vessel. This recommended removal schedule is applicable to 29 EFPY of operation.

TABLE 7-1

H.B. Robinson Unit 2 Reactor Vessel Surveillance Capsule Withdrawal Schedule

Capsule	Location	Lead Factor[a]	Removal Time (EFPY)[b]	Fluence (n/cm^2, E > 1.0 MeV)[a]
S	280°	1.90	1.28	4.79×10^{18} (c)
V	290°	0.91	3.18	5.30×10^{18} (c)
T	270°	2.80	7.27	3.87×10^{19} (c)
X	50°	1.63	20.39	4.49×10^{19} (c)
U[f]	30°	1.41 (2.02)	29.8	6.00×10^{19} (d)
Y	150°	0.92 (1.04)	Standby	(e)
W	40°	0.59 (0.61)	Standby	(e)
Z[g]	230°	0.59 (0.61)	Standby	(e)

Notes:

(a) Updated in Capsule X dosimetry analysis. Lead Factor in Parentheses are for Future Cycles.

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) Capsule U will reach a fluence of approximately 6.00×10^{19} (50 EFPY Peak Fluence) at approximately 29.8 EFPY. Thus, it should be pulled at the closest outage to 29.8 EFPY.

(e) If further material data is desired, then it is recommended that these capsules be moved to a higher lead factor location and then removed once their accumulated neutron fluence equals the license renewal (50 EFPY) fluence on the vessel inner surface.

(f) Moved to Capsule "S" Location (280°) at Cycle 8.

(g) Capsule Z was inadvertently removed from the H.B. Robinson 2 Reactor Vessel. At this time it is unconfirmed that Capsule Z was re-installed into the vessel or placed in the spent fuel pool.

Table 4-1 Chemical Composition (wt %) and Heat Treatment of Material for the H.B. Robinson Unit 2 Reactor Vessel Surveillance Material[a]

Chemical Composition					
Element	**Plate W10201-4**	**Plate W10201-5**	**Plate W10201-6**	**Weld Metal**	**Correlation Monitor Material**
C	0.19	0.20	0.19	0.16	0.24
Mn	1.35	1.29	1.32	0.98	1.34
P	0007	0.010	0.010	0.021	0.011
S	0.019	0.021	0.015	0.014	0.023
Si	0.23	0.22	0.19	0.34	0.23
Mo	0.48	0.46	0.49	0.46	0.51
Cu	0.12	0.10	0.09	0.34	0.20
V	---	---	---	0.001	---
Ni	---	---	---	0.66	0.18
Cr	---	---	---	0.024	0.11
Co	---	---	---	---	---

Heat Treatment	
Plate W10201-4,	1550°F to 1600°F, 4 hours, Water Quench
Plate W10201-5, &	1200°F to 1250°F, 4 hours, Air Cooled
Plate W10201-6	1125°F to 1175°F, 15 1/2 hours, Furnace cooled to 600°F
Weld Metal	1125°F to 1175°F, 30 hours, Furnace cooled to 600°F
Correlation Monitor	1650°F, 4 hours, Water Quenched 1200°F – 6 hours, Air Cooled

Notes:

a) The data given in this column (originally) is from WCAP-7373 & WCAP-10304.

Table 5-2 Charpy V-notch Data for the H.B. Robinson Unit 2 Surveillance Weld Metal Irradiated to a Fluence of 4.49×10^{19} n/cm^2 (E> 1.0 MeV)							
Sample	**Temperature**		**Impact Energy**		**Lateral Expansion**		**Shear**
Number	**F**	**C**	**ft-lbs**	**Joules**	**mils**	**mm**	**%**
W3	0	-18	4	5	0	0.00	0
W2	100	38	14	19	4	0.10	15
W6	175	79	28	38	16	0.41	35
W4	200	93	38	52	22	0.56	40
W8	250	121	74	100	49	1.24	100
W7	350	177	78	106	51	1.30	100
W5	375	191	85	115	56	1.42	100
W1	425	218	82	111	54	1.37	100

Table 5-10 Comparison of the H.B. Robinson Unit 2 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

Material	Capsule	Fluence ($x\ 10^{19}\ n/cm^2$)	30 ft-lb Transition Temperature Shift Predicted (°F) [(a)]	30 ft-lb Transition Temperature Shift Measured (°F) [(b)]	Upper Shelf Energy Decrease Predicted (%) [(a)]	Upper Shelf Energy Decrease Measured (%)[(c)]
Inter. Shell Plate W10201-4 (Longitudinal)	S	0.479	45.39	32.51	18	10
	X	4.49	78.86	104.73	30	1
Surveillance Program Weld Metal	V	0.530	179.17	209.32	39	38
	T	3.87	293.68	288.15	52	46
	X	4.49	300.64	265.93	54	29
Heat Affected Zone Material	V	0.530	- -	59.21	- - -	26
	T	3.87	- -	(d)	- - -	24
	X	4.49	- -	210.13	- - -	19
Correlation Monitor Material	S	0.479	- -	72.79	- - -	3
	V	0.530	- -	69.39	- - -	5
	T	3.87	- -	156.83	- - -	5
	X	4.49	- -	125.21	- - -	0

Notes:

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1 (See Appendix B)

(c) Values are based on the definition of upper shelf energy given in ASTM E185-82.

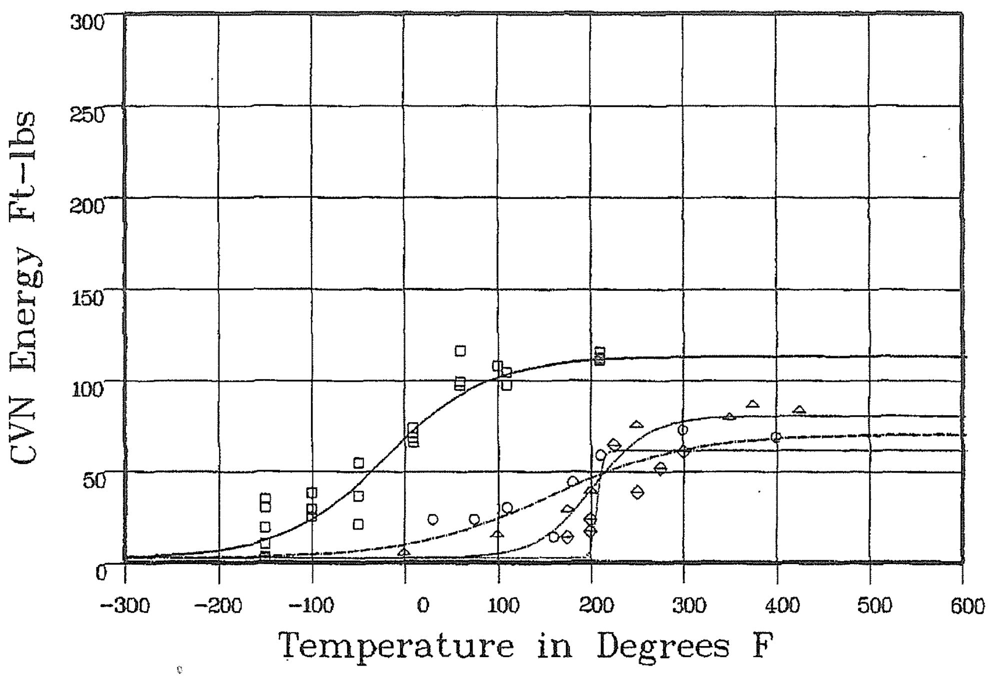
(d) Only 2 specimens were tested from capsule T to confirm the upper shelf energy, thus, there is insufficient data to determine the measured 30 ft-lb shift.

SURVELLIANCE PROGRAM WELD MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:54:57 on 10-24-2001

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	113	0	-86.29	0	-40.35	0
2	0	2.19	0	70	-43	123.02	209.32	214.59	254.94
3	0	2.2	0	61	-52	201.86	288.15	205.94	246.3
4	0	2.19	0	80	-33	179.64	265.93	211.38	251.74



Curve Legend

1 □———— 2 ○---------- 3 ◇———— 4 △————

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	HB2	UNIRR	WELD N/A		W5214
2	HB2	V	WELD N/A		W5214
3	HB2	T	WELD N/A		W5214
4	HB2	X	WELD N/A		W5214

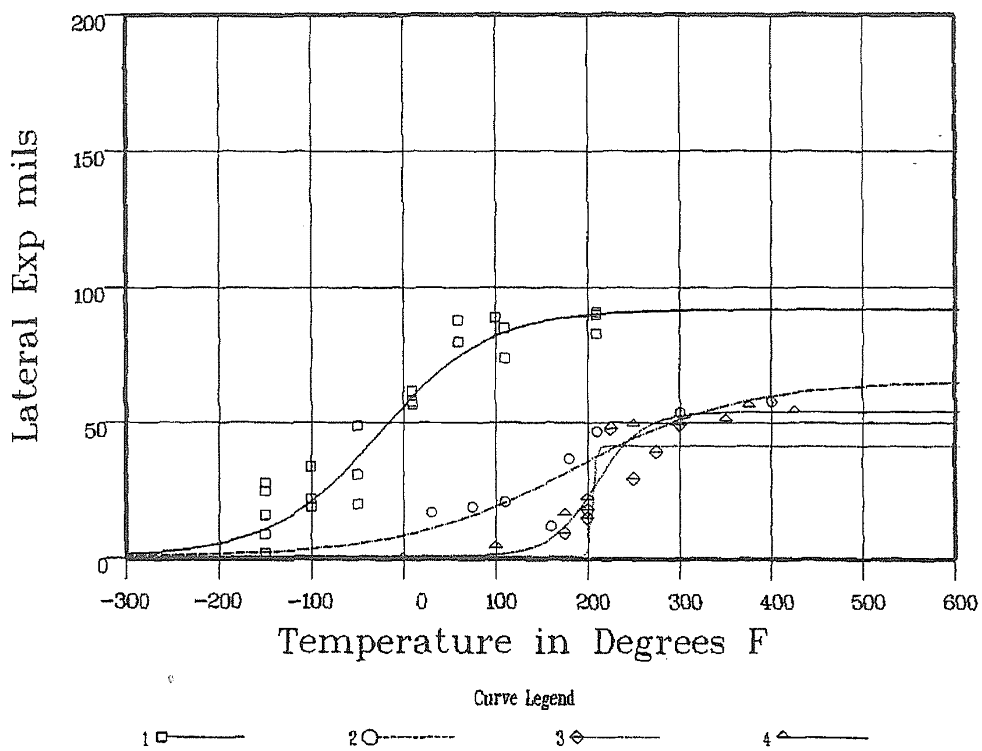
Figure 5-4 Charpy V-Notch Impact Energy vs. Temperature for H.B. Robinson Unit 2 Reactor Vessel Surveillance Weld Metal

SURVEILLANCE PROGRAM WELD MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:59:52 on 10-24-2001

Results

Curve	Fluence	USE	d-USE	T @ LE35	d-T @ LE35
1	0	91.98	0	-60.64	0
2	0	65.7	-26.27	190.26	250.9
3	0	41.57	-50.4	204.2	264.85
4	0	54.2	-37.77	219.24	279.89



Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	HB2	UNIRR	WELD N/A		W5214
2	HB2	V	WELD N/A		W5214
3	HB2	T	WELD N/A		W5214
4	HB2	X	WELD N/A		W5214

Figure 5-5 **Charpy V-Notch Lateral Expansion vs. Temperature for H.B. Robinson Unit 2 Reactor Vessel Surveillance Weld Metal**

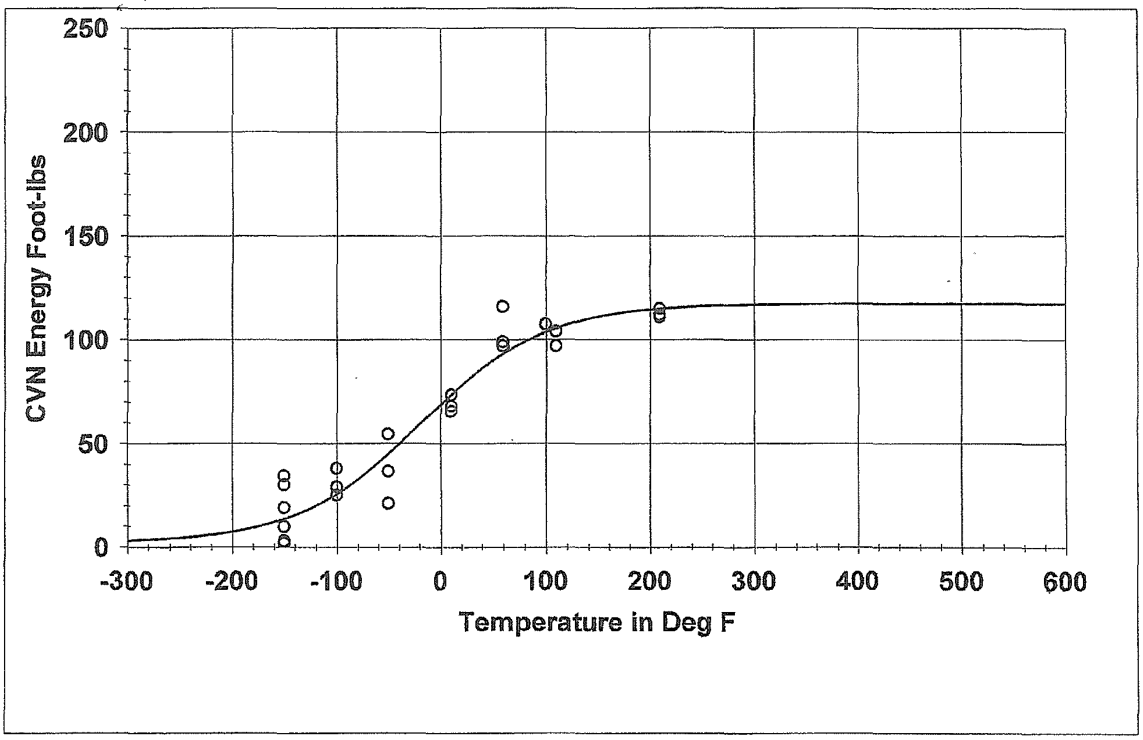
H B Robinson Weld Data

PREP-4 Hyperbolic Tangent Curve Printed on 10/19/2009 10:38:40 PM
Page 1
Coefficients of Curve 1

A = 59.8 B = 57.6 C = 118.8 T0 = -18.8

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

Upper Shelf Energy=117.4 Lower Shelf Energy=2.2
Temp.@30 ft-lbs=-86.8 Deg F Temp.@50 ft-lbs=-39.2 Deg F
Plant: H B Robinson 2 Material: SAW Heat: W5214 (S)
Orientation: TL Capsule: Unirr. Fluence: 0.



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-150.00	30.00	13.60	16.40
-150.00	2.00	13.60	-11.60
-150.00	3.00	13.60	-10.60
-150.00	10.00	13.60	-3.60
-150.00	19.00	13.60	5.40
-150.00	34.50	13.60	20.90
-100.00	29.00	25.60	3.40

H B Robinson Weld Data

Plant: H B Robinson 2   Material: SAW   Heat: W5214 (S)
Orientation: TL   Capsule: Unirr.   Fluence: 0.

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-100.00	38.00	25.60	12.40
-100.00	25.00	25.60	-.60
-50.00	54.50	45.01	9.49
-50.00	21.00	45.01	-24.01
-50.00	36.50	45.01	-8.51
10.00	73.50	73.50	.00
10.00	65.50	73.50	-8.00
10.00	68.00	73.50	-5.50
60.00	97.00	93.24	3.76
60.00	116.00	93.24	22.76
60.00	99.00	93.24	5.76
100.00	107.50	103.67	3.83
110.00	104.00	105.58	-1.58
110.00	97.00	105.58	-8.58
210.00	112.00	115.00	-3.00
210.00	111.00	115.00	-4.00
210.00	115.00	115.00	.00

Correlation Coefficient = .000

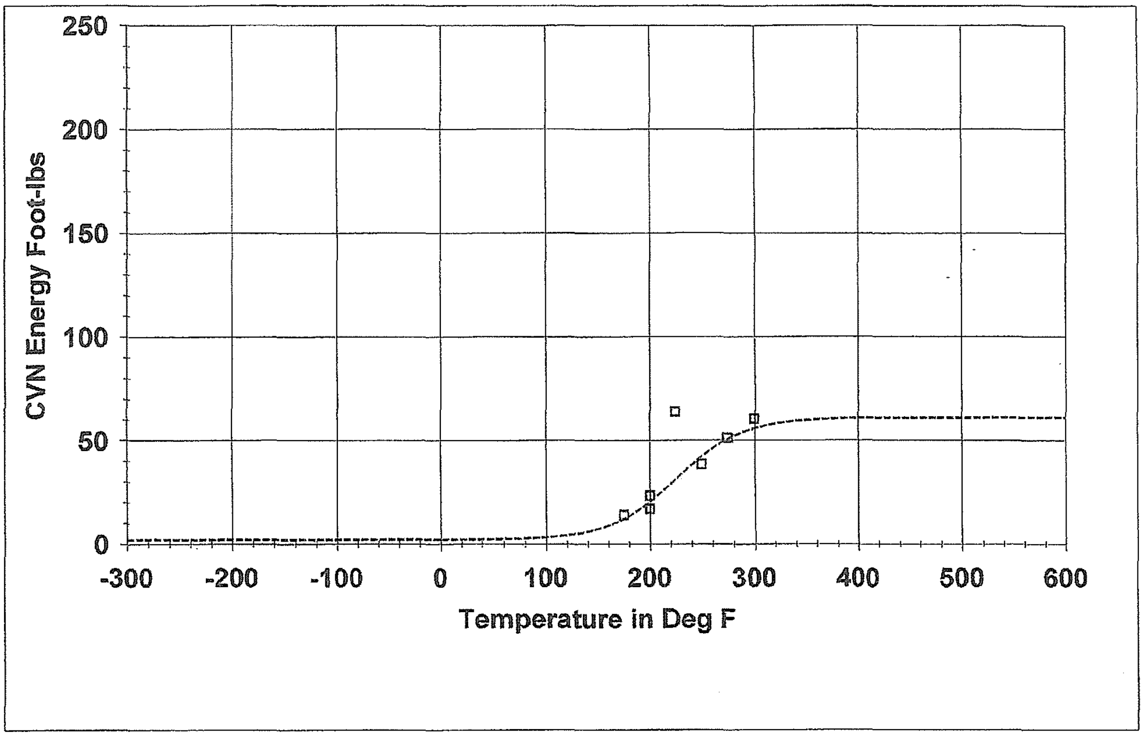
H B Robinson Weld Data

PREP-4 Hyperbolic Tangent Curve Printed on 10/19/2009 10:40:29 PM
Page 1
Coefficients of Curve 2

A = 31.6 B = 29.4 C = 63.57 T0 = 225.94

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

Upper Shelf Energy=61.0 Lower Shelf Energy=2.2
Temp.@30 ft-lbs=222.5 Deg F Temp.@50 ft-lbs=272.6 Deg F
Plant: H B Robinson 2 Material: SAW Heat: W5214 (S)
Orientation: TL Capsule: T Fluence: 4.42E+19



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
175.00	14.00	12.06	1.94
200.00	23.50	20.23	3.27
200.00	17.00	20.23	-3.23
225.00	64.00	31.17	32.83
250.00	38.50	42.22	-3.72
275.00	51.50	50.65	.85
300.00	60.50	55.79	4.71

Correlation Coefficient = .000

H B Robinson Weld Data

PREP-4 Hyperbolic Tangent Curve Printed on 10/19/2009 10:41:23 PM
Page 1
Coefficients of Curve 3

A = 34.35 B = 32.15 C = 142.03 T0 = 141.21

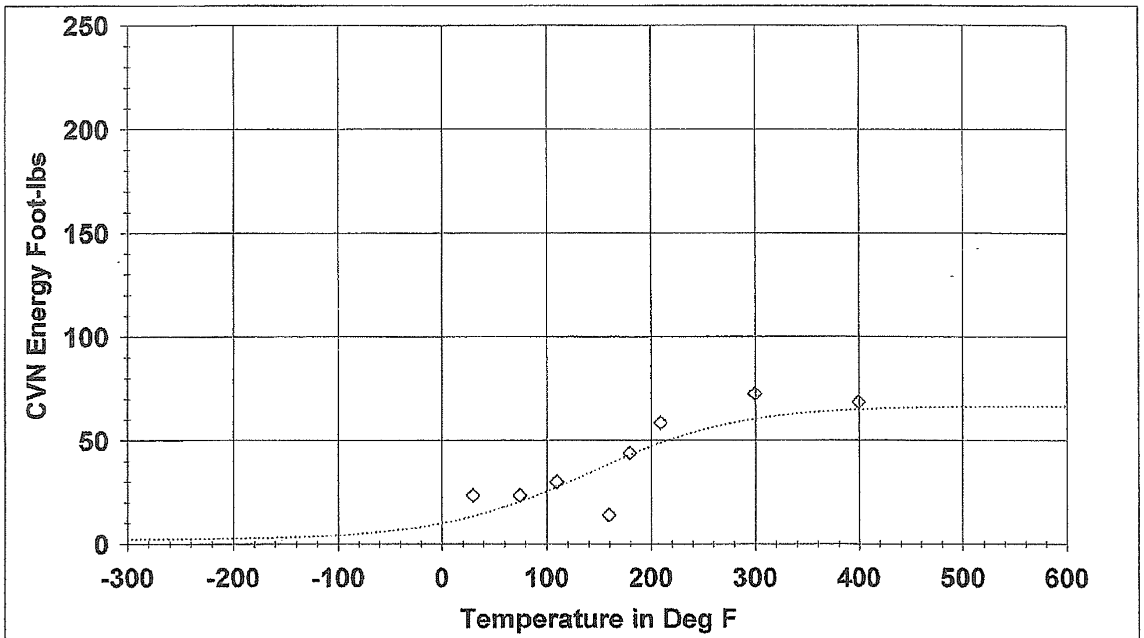
Equation is A + B * [Tanh((T-To)/C)]

Upper Shelf Energy=66.5 Lower Shelf Energy=2.2

Temp.@30 ft-lbs=121.9 Deg F Temp.@50 ft-lbs=216.7 Deg F

Plant: H B Robinson 2 Material: SAW Heat: W5214 (S)

Orientation: TL Capsule: V Fluence: 6.01E+18



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
30.00	23.50	13.31	10.19
75.00	23.50	20.36	3.14
110.00	30.00	27.40	2.60
160.00	14.00	38.58	-24.58
180.00	44.00	42.92	1.08
210.00	58.50	48.81	9.69
300.00	72.50	60.29	12.21

H B Robinson Weld Data

Plant: H B Robinson 2 Material: SAW Heat: W5214 (S)
Orientation: TL Capsule: V Fluence: 6.01E+18

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
400.00	68.50	64.86	3.64

Correlation Coefficient = .000

APPENDIX G

CVGRAPH TANH CURVE-FITS FOR W5214 SURVEILLANCE WELD DATA
(from Reference 32)

Table G-1. Fitted Results for CVGRAPH Hyperbolic Tangent Curve-Fits [32]

Plant	Capsule	A	B	C	T0	T30
Palisades	Unirradiated	53.14	50.94	100.7	-10.76	-60.1
Palisades	SA-60-1	28.35	26.15	158.11	188.85	198.9
Palisades	SA-240-1	27.35	25.15	111.62	208.13	220
H. B. Robinson 2	Unirradiated	56.05	53.85	107.57	-29.1	-85.8
H. B. Robinson 2	T	31.35	29.15	9.09	203.64	203.3
H. B. Robinson 2	V	36.35	34.15	150.19	151.23	123
H. B. Robinson 2	X	40.97	38.78	59.96	197.18	179.8
Indian Point 2	Unirradiated	59.32	57.12	86.23	-16.54	-65.4
Indian Point 2	V	39.1	36.9	123.56	163.14	132.1
Indian Point 2	Y	34.35	32.15	91.6	140.88	128.5
Indian Point 3	Unirradiated	60.38	58.19	45.25	-37.64	-63.8
Indian Point 3	T	46.35	44.15	98.27	124.16	86
Indian Point 3	Y	35.6	33.4	90.16	122.54	107.3
Indian Point 3	Z	39.1	36.9	97.52	188.96	164.5
Indian Point 3	X	38.6	36.4	121.83	157.96	128.7

Palisades Unirradiated Capsule Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 04/09/2010 06:36 PM

Page 1

Coefficients of Curve 1

A = 53.14 B = 50.94 C = 100.7 T0 = -10.76 D = 0.00E+00

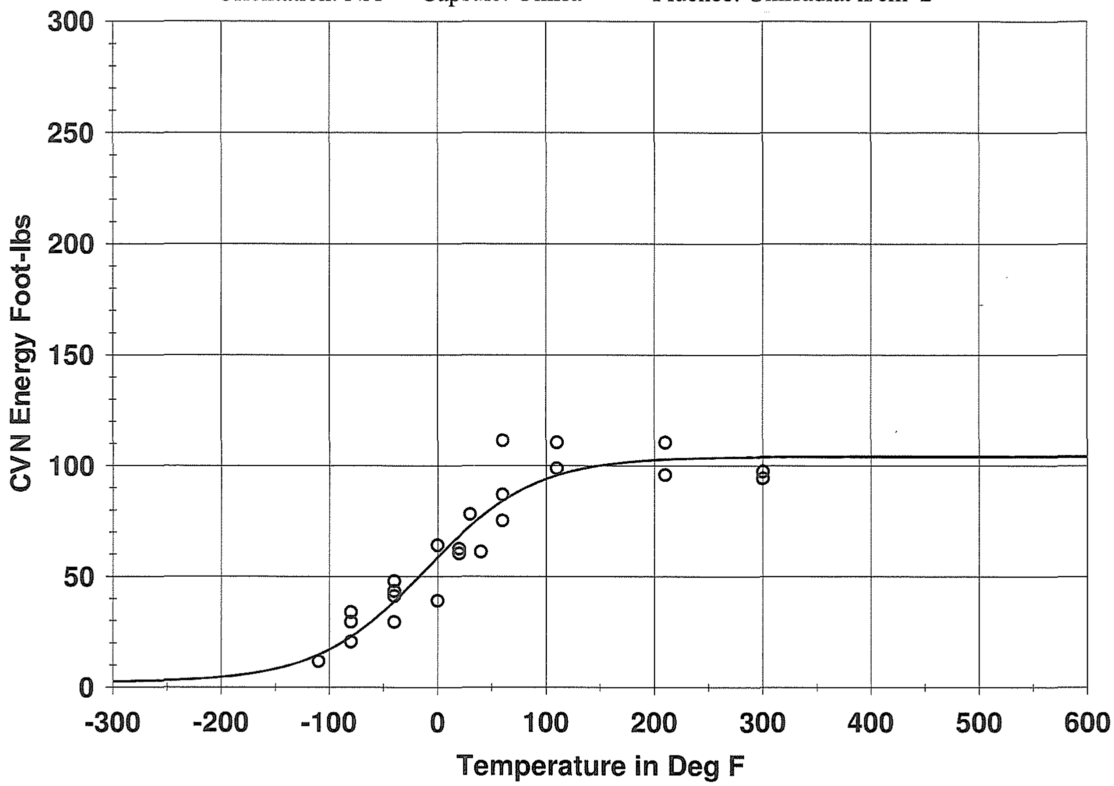
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=104.1 Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=-60.1 Deg F Temp@50 ft-lbs=-16.9 Deg F

Plant: PALISADES Material: SAW Heat: W5214

Orientation: NA Capsule: Unirra Fluence: Unirradiat n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-110.00	11.80	14.66	-2.86
-110.00	11.80	14.66	-2.86
-80.00	33.90	22.76	11.14
-80.00	20.60	22.76	-2.16
-80.00	29.50	22.76	6.74
-40.00	47.90	38.75	9.15
-40.00	43.50	38.75	4.75
-40.00	29.50	38.75	-9.25
-40.00	41.29	38.75	2.54

Palisades Unirradiated Capsule Report

Page 2
Plant: PALISADES Material: SAW Heat: W5214
Orientation: NA Capsule: Unirra Fluence: Unirradiat n/cm^2

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
.00	64.19	58.56	5.63
.00	39.09	58.56	-19.47
20.00	62.70	68.24	-5.54
20.00	60.50	68.24	-7.74
30.00	78.19	72.70	5.49
40.00	61.20	76.85	-15.65
60.00	87.00	84.02	2.98
60.00	75.19	84.02	-8.83
60.00	111.40	84.02	27.38
110.00	110.59	95.60	14.99
110.00	98.80	95.60	3.20
210.00	110.59	102.83	7.76
210.00	95.90	102.83	-6.93
300.00	97.40	103.87	-6.47
300.00	94.40	103.87	-9.47

Correlation Coefficient = .947

Palisades SA-60-1 Capsule Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 10:39 AM

Page 1

Coefficients of Curve 1

A = 28.35 B = 26.15 C = 158.11 T0 = 188.85 D = 0.00E+00

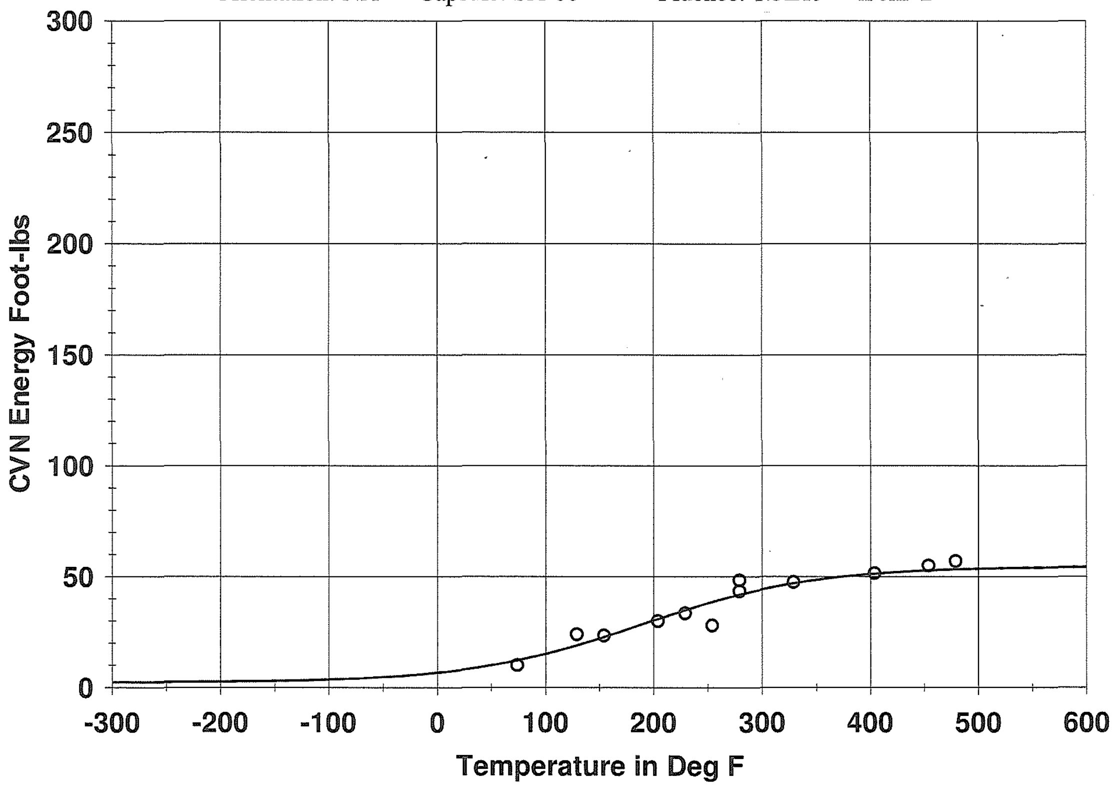
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=54.5(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=198.9 Deg F Temp@50 ft-lbs=375.7 Deg F

Plant: PALISADES Material: SAW Heat: W5214

Orientation: NA Capsule: SA-60- Fluence: 1.5E19 n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
74.00	10.00	12.11	-2.11
129.00	24.00	18.90	5.10
154.00	23.50	22.68	.82
204.00	30.00	30.85	-.85
229.00	33.50	34.85	-1.35
254.00	28.00	38.55	-10.55
279.00	43.50	41.83	1.67
279.00	48.50	41.83	6.67
329.00	47.50	46.91	.59

Palisades SA-60-1 Capsule Report

Plant: PALISADES Material: SAW Heat: W5214
Orientation: NA Capsule: SA-60- Fluence: 1.5E19 n/cm^2

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
404.00	51.50	51.27	.23
454.00	55.00	52.73	2.27
479.00	57.00	53.20	3.80

Correlation Coefficient = .957

Palisades SA-240-1 Capsule Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 10:34 AM
Page 1
Coefficients of Curve 1

A = 27.35 B = 25.15 C = 111.62 T0 = 208.13 D = 0.00E+00

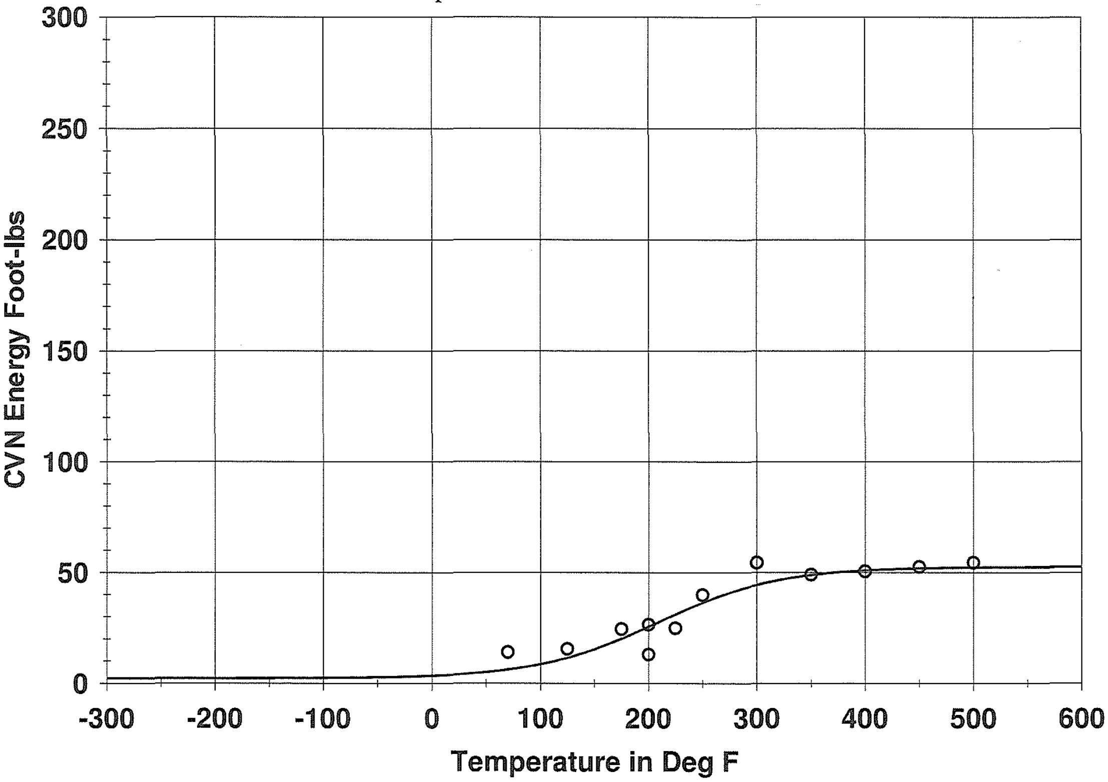
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=52.5(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=220.0 Deg F Temp@50 ft-lbs=372.9 Deg F

Plant: PALISADES Material: SAW Heat: W5214

Orientation: NA Capsule: SA-240 Fluence: 2.38E19 n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
70.00	14.00	6.10	7.90
125.00	15.50	11.45	4.05
175.00	24.50	20.10	4.40
200.00	13.00	25.52	-12.52
200.00	26.50	25.52	.98
225.00	25.00	31.12	-6.12
250.00	40.00	36.36	3.64
300.00	54.50	44.37	10.13
350.00	49.00	48.83	.17

Palisades SA-240-1 Capsule Report

Plant: PALISADES Material: SAW Heat: W5214
Orientation: NA Capsule: SA-240 Fluence: 2.38E19 n/cm^2

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
400.00	50.50	50.93	-.43
450.00	52.50	51.85	.65
500.00	54.50	52.23	2.27

Correlation Coefficient = .935

H. B. Robinson 2 Unirradiated Capsule Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 11:43 AM

Page 1

Coefficients of Curve 1

A = 56.05 B = 53.85 C = 107.57 T0 = -29.1 D = 0.00E+00

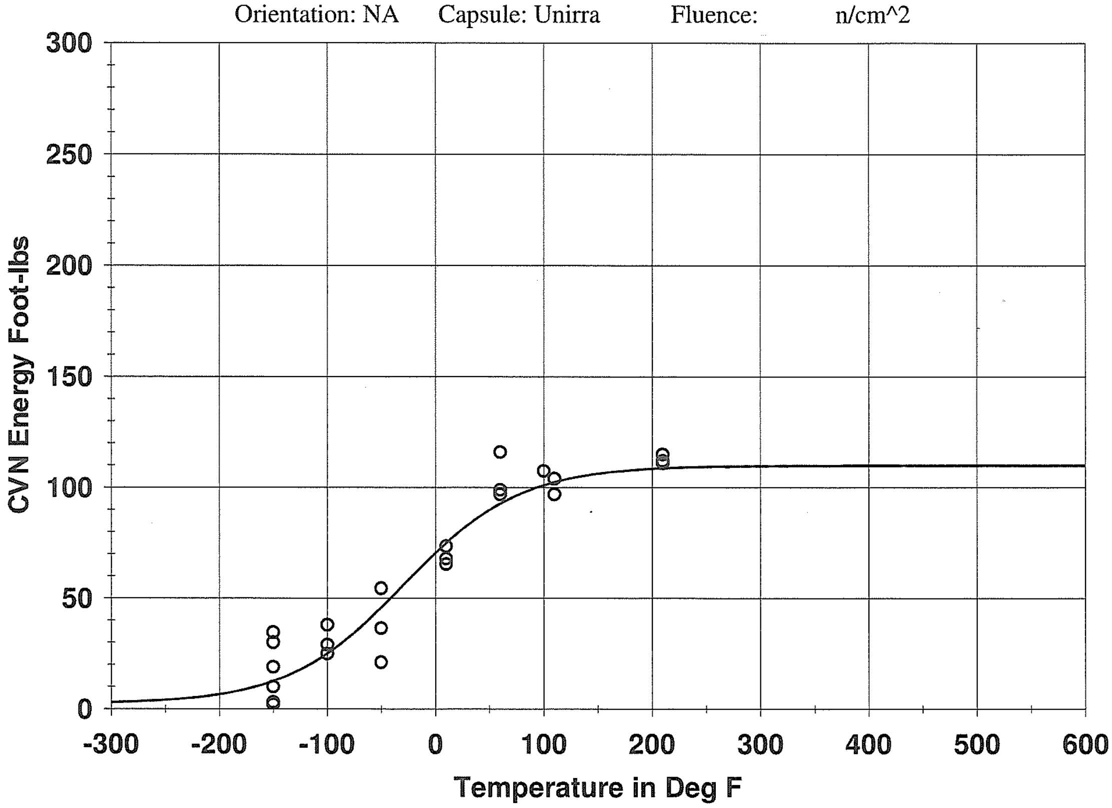
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=109.9(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=-85.8 Deg F Temp@50 ft-lbs=-41.2 Deg F

Plant: H B Robinson 2 Material: SAW Heat: W5214

Orientation: NA Capsule: Unirra Fluence: n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-150.00	19.00	12.49	6.51
-150.00	10.00	12.49	-2.49
-150.00	30.00	12.49	17.51
-150.00	34.50	12.49	22.01
-150.00	3.00	12.49	-9.49
-150.00	2.00	12.49	-10.49
-100.00	25.00	24.94	.06
-100.00	38.00	24.94	13.06
-100.00	29.00	24.94	4.06

H. B. Robinson 2 Unirradiated Capsule Report

Page 2
Plant: H B Robinson 2 Material: SAW Heat: W5214
Orientation: NA Capsule: Unirra Fluence: n/cm^2

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-50.00	36.50	45.72	-9.22
-50.00	21.00	45.72	-24.72
-50.00	54.50	45.72	8.78
10.00	65.50	74.80	-9.30
10.00	73.50	74.80	-1.30
10.00	68.00	74.80	-6.80
60.00	97.00	92.64	4.36
60.00	99.00	92.64	6.36
60.00	116.00	92.64	23.36
100.00	107.50	100.94	6.56
110.00	97.00	102.36	-5.36
110.00	104.00	102.36	1.64
210.00	115.00	108.65	6.35
210.00	111.00	108.65	2.35
210.00	112.00	108.65	3.35

Correlation Coefficient = .962

H. B. Robinson 2 Capsule T Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 04/14/2010 09:40 PM
Page 1
Coefficients of Curve 1

A = 31.35 B = 29.15 C = 9.09 T0 = 203.64 D = 0.00E+00

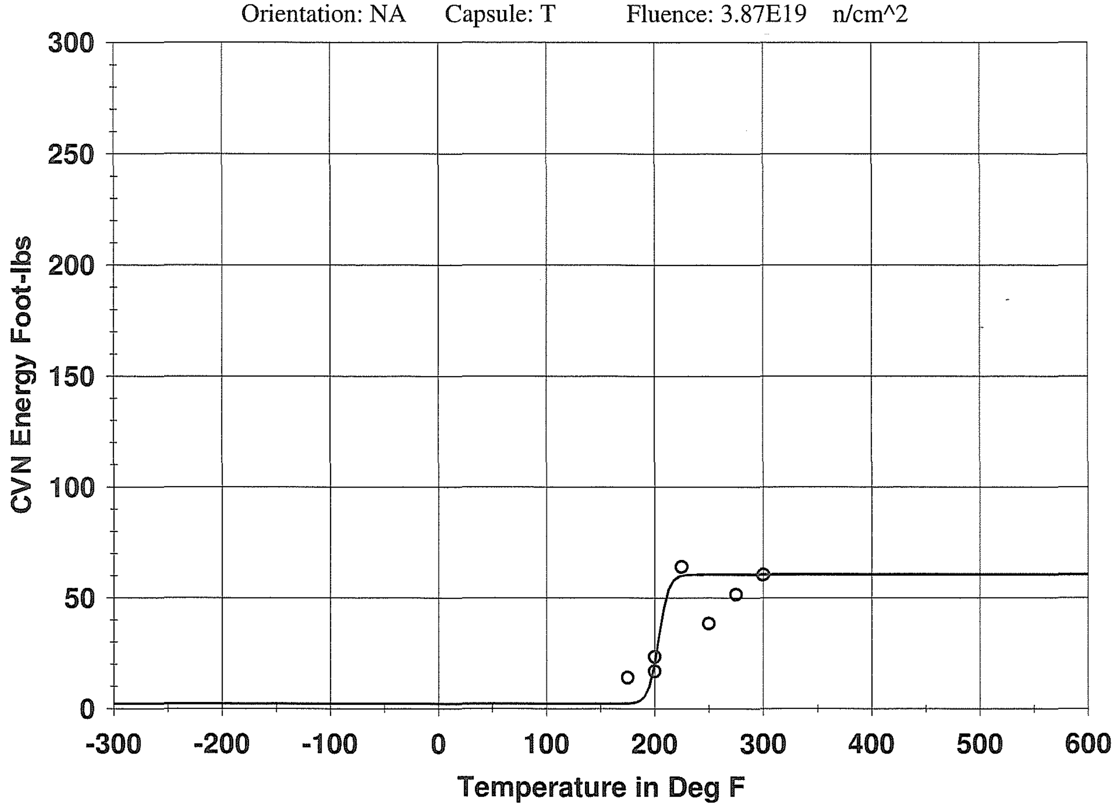
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=60.5(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=203.3 Deg F Temp@50 ft-lbs=210.6 Deg F

Plant: H B ROBINSON 2 Material: SAW Heat: W5214

Orientation: NA Capsule: T Fluence: 3.87E19 n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
175.00	14.00	2.31	11.69
200.00	17.00	20.27	-3.27
200.00	23.50	20.27	3.23
225.00	64.00	59.97	4.03
250.00	38.50	60.50	-22.00
275.00	51.50	60.50	-9.00
300.00	60.50	60.50	.00

Correlation Coefficient = .908

H. B. Robinson 2 Capsule V Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 11:47 AM

Page 1

Coefficients of Curve 1

A = 36.35 B = 34.15 C = 150.19 T0 = 151.23 D = 0.00E+00

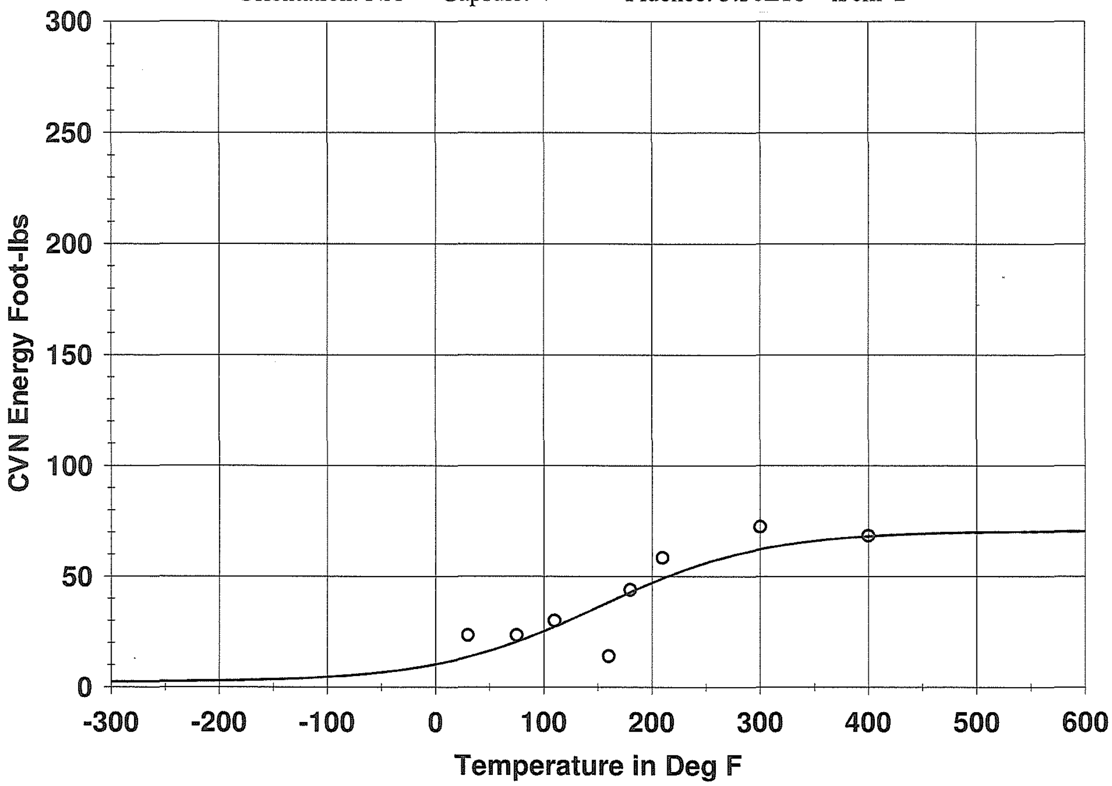
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=70.5(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=123.0 Deg F Temp@50 ft-lbs=214.9 Deg F

Plant: H B ROBINSON 2 Material: SAW Heat: W5214

Orientation: NA Capsule: V Fluence: 5.30E18 n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
30.00	23.50	13.54	9.96
75.00	23.50	20.37	3.13
110.00	30.00	27.20	2.80
160.00	14.00	38.34	-24.34
180.00	44.00	42.81	1.19
210.00	58.50	49.07	9.43
300.00	72.50	62.22	10.28
400.00	68.50	68.10	.40

Correlation Coefficient = .865

H. B. Robinson 2 Capsule X Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 11:45 AM

Page 1

Coefficients of Curve 1

A = 40.97 B = 38.78 C = 59.96 T0 = 197.18 D = 0.00E+00

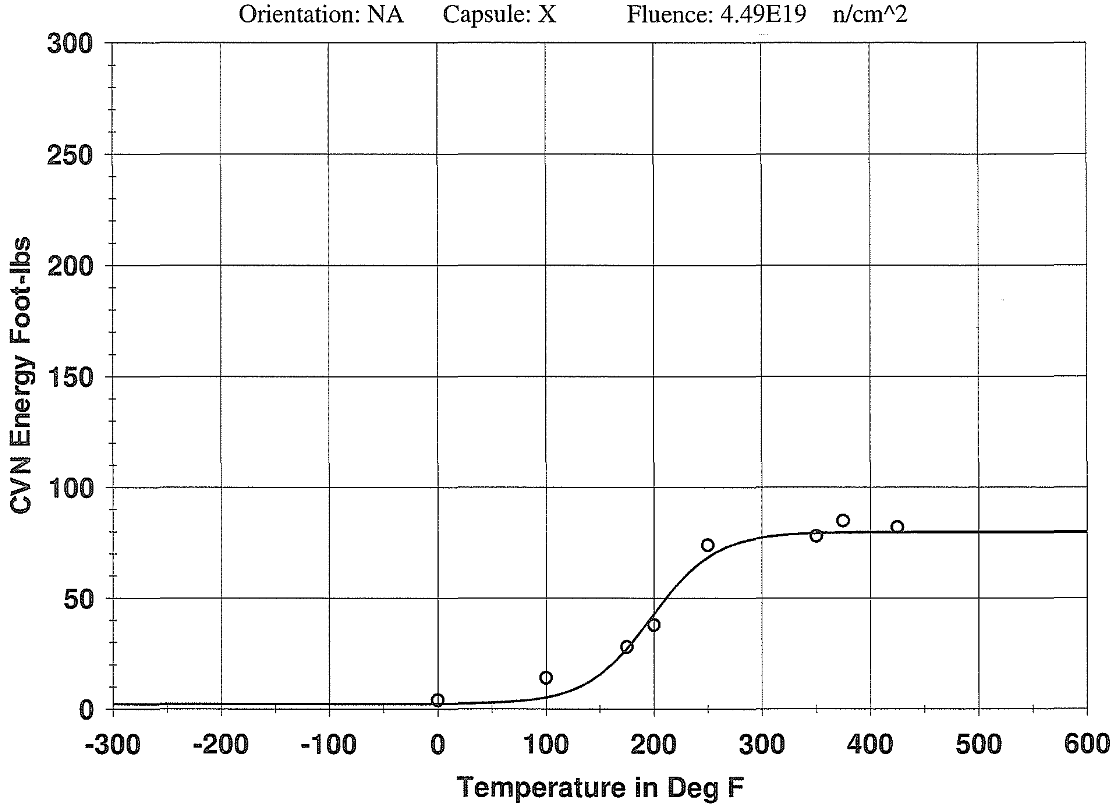
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=79.8(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=179.8 Deg F Temp@50 ft-lbs=211.4 Deg F

Plant: H B ROBINSON 2 Material: SAW Heat: W-5214

Orientation: NA Capsule: X Fluence: 4.49E19 n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
.00	4.00	2.31	1.69
100.00	14.00	5.12	8.88
175.00	28.00	27.25	.75
200.00	38.00	42.80	-4.80
250.00	74.00	68.39	5.61
350.00	78.00	79.28	-1.28
375.00	85.00	79.54	5.46
425.00	82.00	79.71	2.29

Correlation Coefficient = .992

Indian Point 2 Unirradiated Capsule Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 10:55 AM

Page 1

Coefficients of Curve 1

A = 59.32 B = 57.12 C = 86.23 T0 = -16.54 D = 0.00E+00

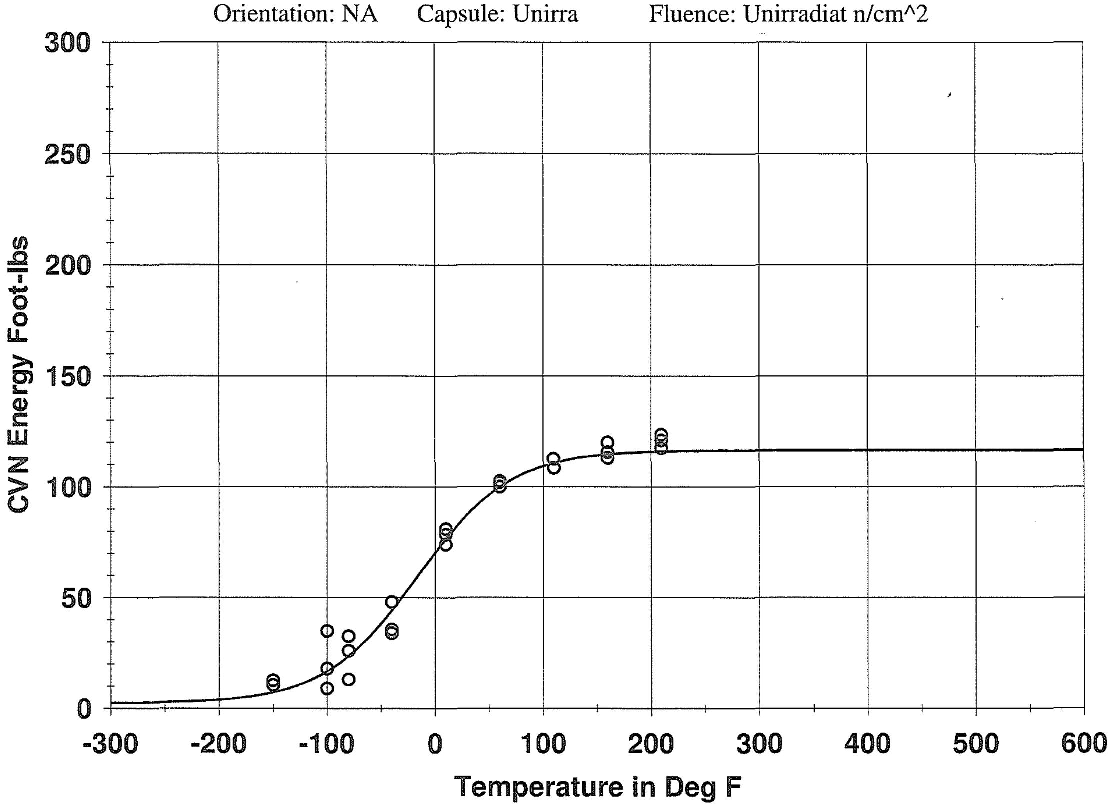
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=116.4(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=-65.4 Deg F Temp@50 ft-lbs=-30.7 Deg F

Plant: INDIAN POINT 2 Material: SAW Heat: W5214

Orientation: NA Capsule: Unirra Fluence: Unirradiat n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-150.00	12.50	7.15	5.35
-150.00	10.50	7.15	3.35
-100.00	35.00	16.61	18.39
-100.00	18.00	16.61	1.39
-100.00	9.00	16.61	-7.61
-80.00	13.00	23.52	-10.52
-80.00	26.00	23.52	2.48
-80.00	32.50	23.52	8.98
-40.00	35.50	44.15	-8.65

Indian Point 2 Unirradiated Capsule Report

Page 2
Plant: INDIAN POINT 2 Material: SAW Heat: W5214
Orientation: NA Capsule: Unirra Fluence: Unirradiat n/cm^2

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-40.00	48.00	44.15	3.85
-40.00	34.00	44.15	-10.15
10.00	74.00	76.37	-2.37
10.00	81.00	76.37	4.63
10.00	78.50	76.37	2.13
60.00	102.50	99.89	2.61
60.00	102.00	99.89	2.11
60.00	100.00	99.89	.11
110.00	112.50	110.68	1.82
110.00	108.50	110.68	-2.18
110.00	108.50	110.68	-2.18
160.00	120.00	114.57	5.43
160.00	115.50	114.57	.93
160.00	113.00	114.57	-1.57
210.00	123.50	115.85	7.65
210.00	121.00	115.85	5.15
210.00	117.50	115.85	1.65

Correlation Coefficient = .990

Indian Point 2 Capsule V Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 02:35 PM

Page 1

Coefficients of Curve 1

A = 39.1 B = 36.9 C = 123.56 T0 = 163.14 D = 0.00E+00

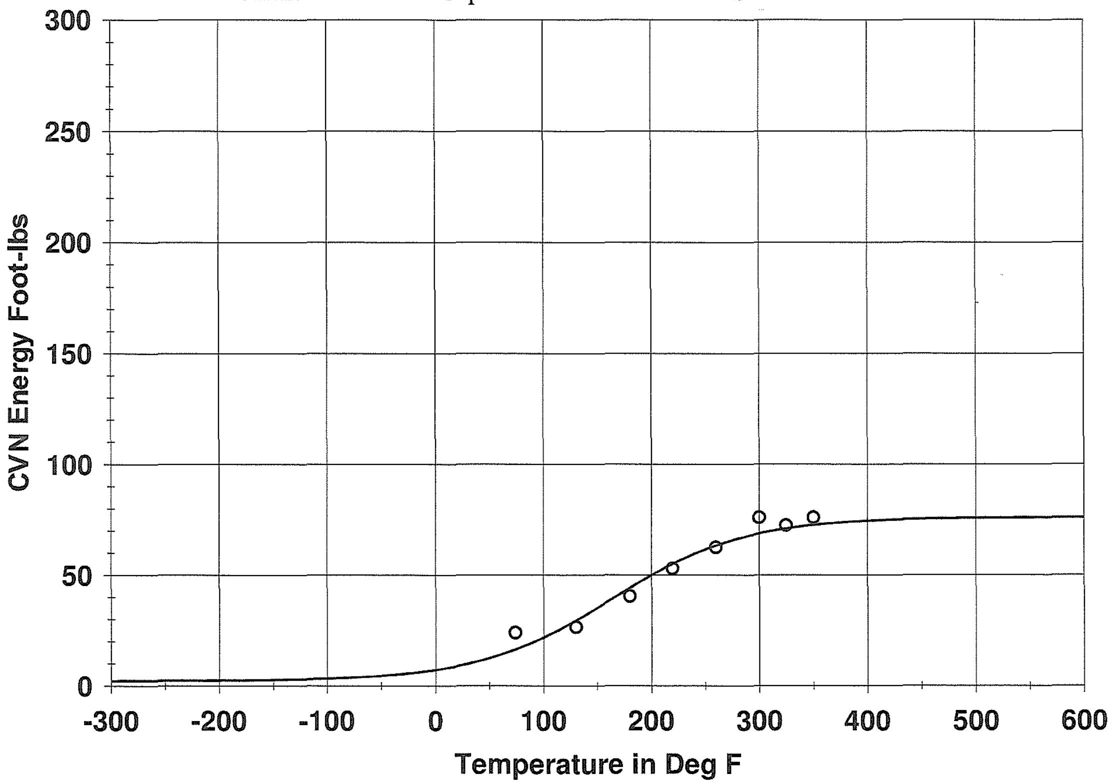
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=76.0(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=132.1 Deg F Temp@50 ft-lbs=200.8 Deg F

Plant: INDIAN POINT 2 Material: SAW Heat: W5214

Orientation: NA Capsule: V Fluence: 4.92E18 n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
74.00	24.00	16.30	7.70
130.00	26.50	29.44	-2.94
180.00	40.50	44.11	-3.61
220.00	53.00	54.98	-1.98
260.00	62.50	63.27	-.77
300.00	76.00	68.74	7.26
325.00	72.50	70.99	1.51
350.00	76.00	72.58	3.42

Correlation Coefficient = .978

Indian Point 2 Capsule Y Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 10:59 AM

Page 1

Coefficients of Curve 1

A = 34.35 B = 32.15 C = 91.6 T0 = 140.88 D = 0.00E+00

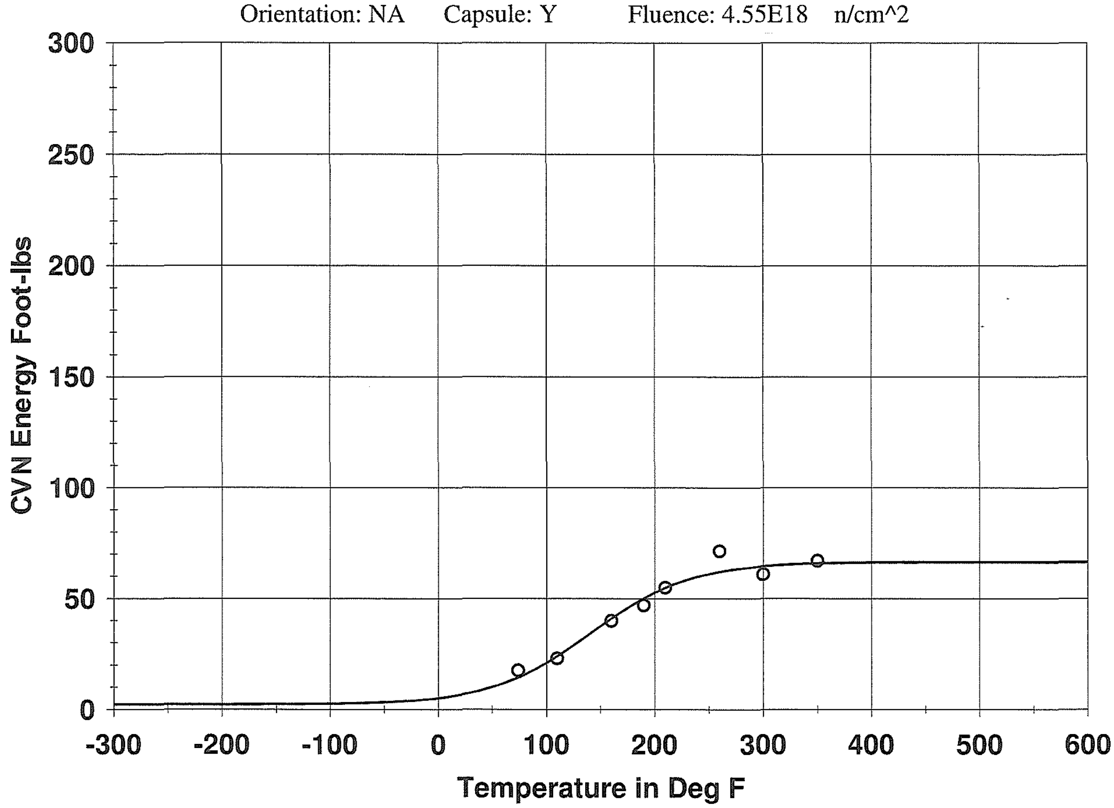
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=66.5(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=128.5 Deg F Temp@50 ft-lbs=189.6 Deg F

Plant: INDIAN POINT 2 Material: SAW Heat: W5214

Orientation: NA Capsule: Y Fluence: 4.55E18 n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
74.00	17.50	14.32	3.18
110.00	23.00	23.90	-.90
160.00	40.00	40.96	-.96
190.00	47.00	50.11	-3.11
210.00	55.00	54.86	.14
260.00	71.50	62.06	9.44
300.00	61.00	64.57	-3.57
350.00	67.00	65.84	1.16

Correlation Coefficient = .978

Indian Point 3 Unirradiated Capsule Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 11:16 AM

Page 1

Coefficients of Curve 1

A = 60.38 B = 58.19 C = 45.25 T0 = -37.64 D = 0.00E+00

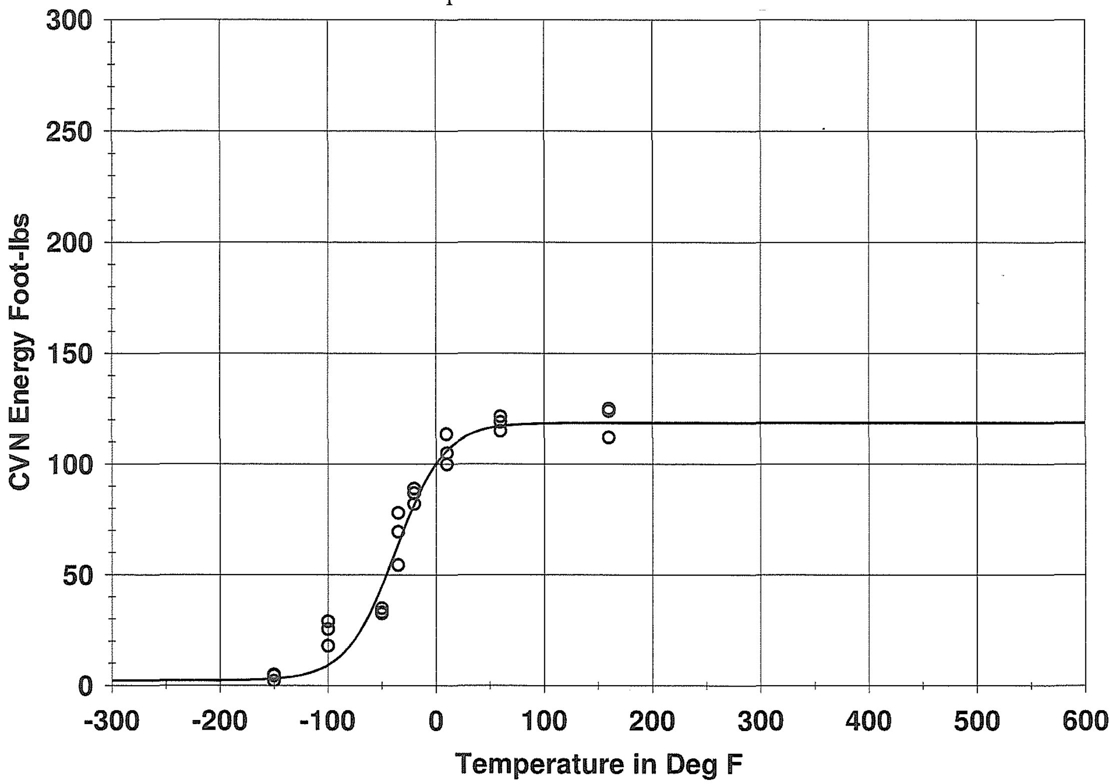
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=118.6(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=-63.8 Deg F Temp@50 ft-lbs=-45.8 Deg F

Plant: INDIAN POINT 3 Material: SAW Heat: W5214

Orientation: NA Capsule: Unirra Fluence: Unirradiat n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-150.00	5.00	3.01	1.99
-150.00	2.00	3.01	-1.01
-150.00	4.50	3.01	1.49
-100.00	29.00	9.15	19.85
-100.00	18.00	9.15	8.85
-100.00	25.50	9.15	16.35
-50.00	35.00	44.88	-9.88
-50.00	33.00	44.88	-11.88
-50.00	32.50	44.88	-12.38

Indian Point 3 Unirradiated Capsule Report

Page 2
Plant: INDIAN POINT 3 Material: SAW Heat: W5214
Orientation: NA Capsule: Unirra Fluence: Unirradiat n/cm^2

Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
-35.00	78.00	63.78	14.22
-35.00	69.50	63.78	5.72
-35.00	54.50	63.78	-9.28
-20.00	87.00	81.98	5.02
-20.00	82.00	81.98	.02
-20.00	89.00	81.98	7.02
10.00	100.00	105.94	-5.94
10.00	105.00	105.94	-.94
10.00	113.50	105.94	7.56
60.00	115.00	117.04	-2.04
60.00	119.00	117.04	1.96
60.00	121.50	117.04	4.46
160.00	124.00	118.55	5.45
160.00	125.00	118.55	6.45
160.00	112.00	118.55	-6.55

Correlation Coefficient = .981

Indian Point 3 Capsule T Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 11:19 AM

Page 1

Coefficients of Curve 1

A = 46.35 B = 44.15 C = 98.27 T0 = 124.16 D = 0.00E+00

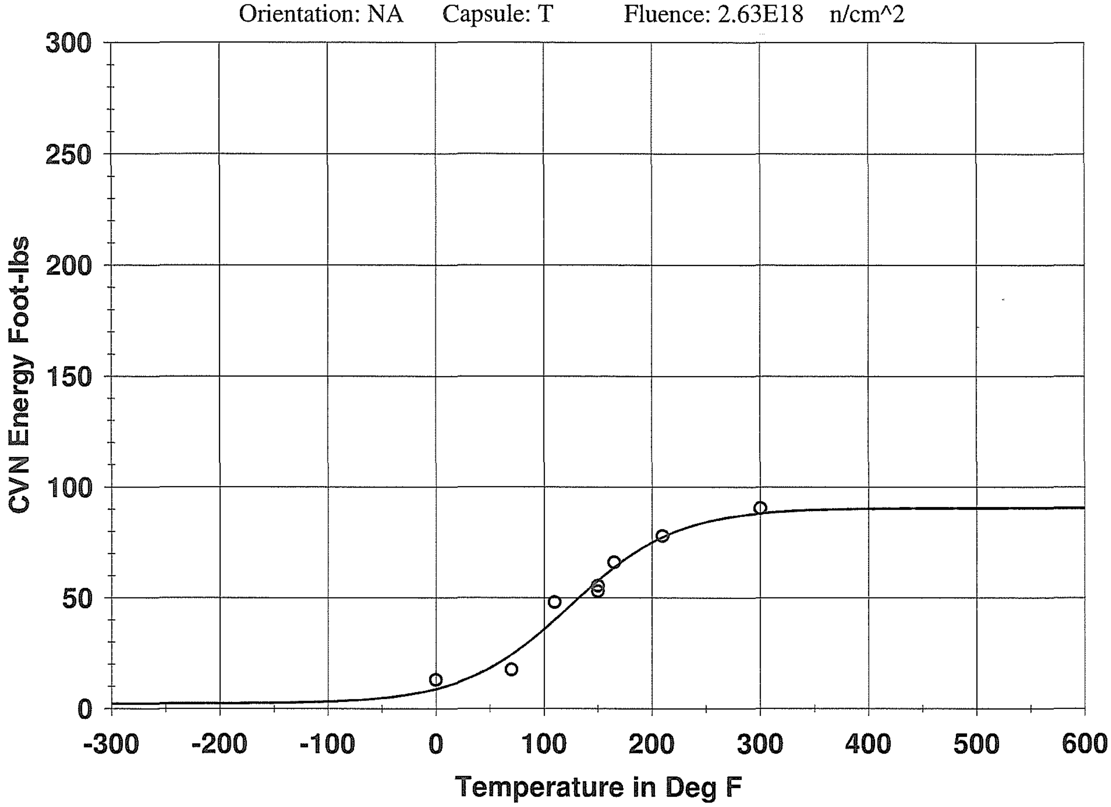
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=90.5(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=86.0 Deg F Temp@50 ft-lbs=132.4 Deg F

Plant: INDIAN POINT 3 Material: SAW Heat: W5214

Orientation: NA Capsule: T Fluence: 2.63E18 n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
.00	13.00	8.73	4.27
70.00	17.50	24.21	-6.71
110.00	48.00	40.03	7.97
150.00	55.50	57.70	-2.20
150.00	53.00	57.70	-4.70
165.00	66.00	63.71	2.29
210.00	78.00	77.39	.61
300.00	90.50	88.10	2.40

Correlation Coefficient = .984

Indian Point 3 Capsule Y Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 11:18 AM

Page 1

Coefficients of Curve 1

A = 35.6 B = 33.4 C = 90.16 T0 = 122.54 D = 0.00E+00

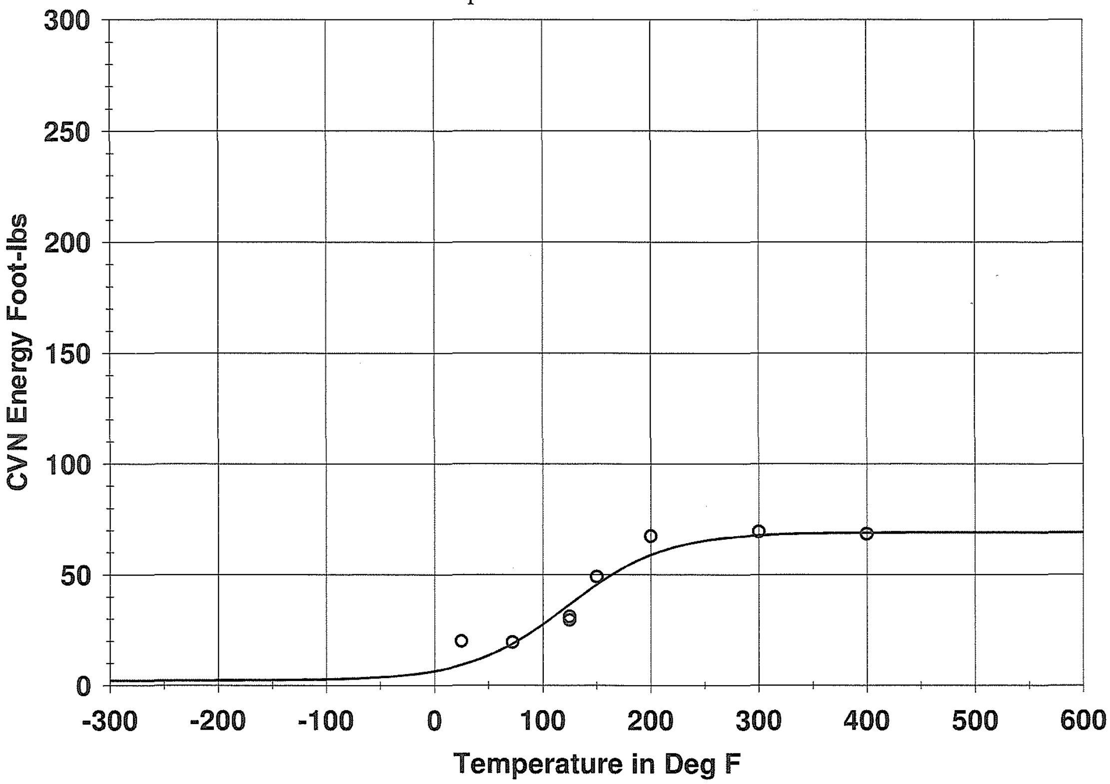
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=69.0(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=107.3 Deg F Temp@50 ft-lbs=164.2 Deg F

Plant: INDIAN POINT 3 Material: SAW Heat: W5214

Orientation: NA Capsule: Y Fluence: 6.92E18 n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
25.00	20.00	9.08	10.92
72.00	19.50	18.62	.88
125.00	31.00	36.51	-5.51
125.00	29.50	36.51	-7.01
150.00	49.00	45.47	3.53
200.00	67.50	58.84	8.66
300.00	69.50	67.72	1.78
400.00	68.50	68.86	-.36

Correlation Coefficient = .960

Indian Point 3 Capsule Z Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 11:21 AM

Page 1

Coefficients of Curve 1

A = 39.1 B = 36.9 C = 97.52 T0 = 188.96 D = 0.00E+00

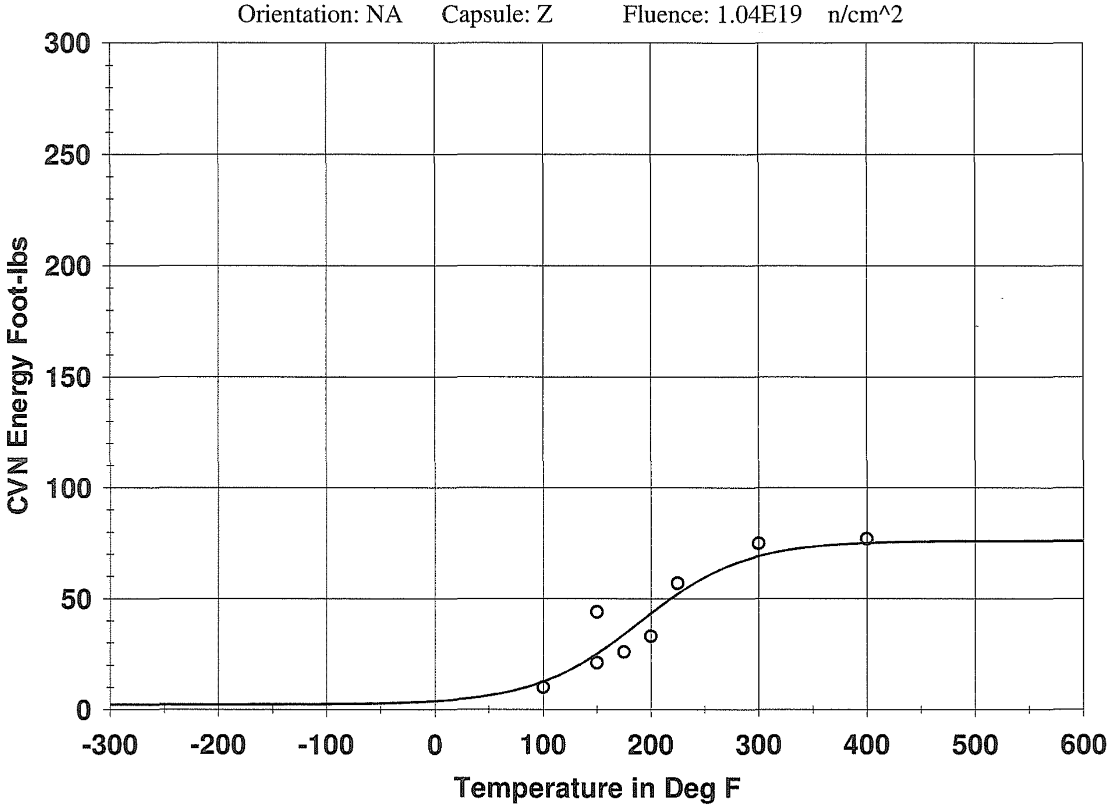
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=76.0(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=164.5 Deg F Temp@50 ft-lbs=218.7 Deg F

Plant: Indian Point 3 Material: SAW Heat: W5214

Orientation: NA Capsule: Z Fluence: 1.04E19 n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
100.00	10.00	12.45	-2.45
150.00	21.00	25.10	-4.10
150.00	44.00	25.10	18.90
175.00	26.00	33.85	-7.85
200.00	33.00	43.26	-10.26
225.00	57.00	52.15	4.85
300.00	75.00	69.14	5.86
400.00	77.00	75.04	1.96

Correlation Coefficient = .929

Indian Point 3 Capsule X Report

CVGRAPH 5.0.2 Hyperbolic Tangent Curve Printed on 03/25/2010 02:34 PM

Page 1

Coefficients of Curve 1

A = 38.6 B = 36.4 C = 121.83 T0 = 157.96 D = 0.00E+00

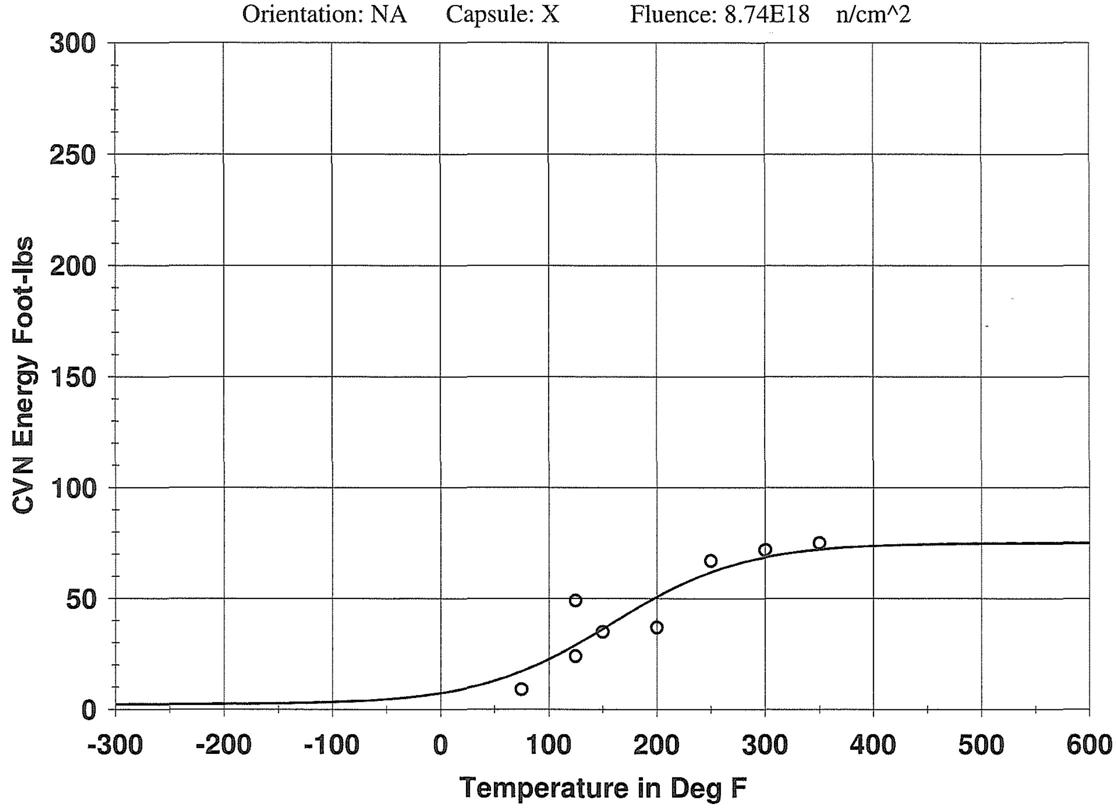
Equation is A + B * [Tanh((T-To)/(C+DT))]

Upper Shelf Energy=75.0(Fixed) Lower Shelf Energy=2.2(Fixed)

Temp@30 ft-lbs=128.7 Deg F Temp@50 ft-lbs=197.5 Deg F

Plant: INDIAN POINT 3 Material: SAW Heat: W5214

Orientation: NA Capsule: X Fluence: 8.74E18 n/cm^2



Charpy V-Notch Data

Temperature	Input CVN	Computed CVN	Differential
75.00	9.00	17.05	-8.05
125.00	49.00	28.99	20.01
125.00	24.00	28.99	-4.99
150.00	35.00	36.22	-1.22
200.00	37.00	50.68	-13.68
250.00	67.00	61.84	5.16
300.00	72.00	68.56	3.44
350.00	75.00	72.02	2.98

Correlation Coefficient = .907

APPENDIX H

CALCULATION OF TIME-WEIGHTED AVERAGE TEMPERATURES FOR SURVEILLANCE CAPSULES CONTAINING WELD HEAT NO. W5214

(from Reference 33)

Calculation of Time-weighted Average Temperatures for Surveillance Capsules Containing Weld Heat No. W5214

The irradiation temperatures for the various surveillance capsules have been reviewed and verified in support of the credibility analysis for all sources of W5214 weld metal [33].

The Palisades time-weighted average capsule and vessel temperatures were determined from plant operating history data [23]. The weighted average temperatures considered that the capsules were inserted for only two and three cycles. The resulting time weighted temperature for capsule SA-60-1 and SA-240-1 is 535.0°F and 535.7°, respectively, as determined from the values shown in Table 8. The vessel time-weighted average temperature is determined to be 535.2°F, as shown in Table H-7.

Discussions with staff at H. B. Robinson 2 indicate that the values for irradiation temperature of HB2 T, HB2 V, and HB2 X are 547°F, which is consistent with the temperature values, listed in Table D-5 of WCAP-15085 [12].

Cycle by cycle T_{cold} temperatures were reviewed for Indian Point Unit 2 and Indian Point Unit 3 in order to generate time weighted temperatures corresponding to each of the surveillance capsules which contain W5214 weld metal. For Indian Point Unit 2 the data was reviewed from Calculation Number FCX-00538 [29]. For Indian Point Unit 3 the data was reviewed from Calculation Number IP3-CALC-RV-03720 [30].

The radiation exposure and temperature history of Indian Point Unit 2 relative to Capsule Y and Capsules V is shown below.

IP2 History			
Cycle	EFPY	T cold	Capsule
1a	1.041	540.6	
1b	1.419	528.6	
2a	1.666	529.2	
2b	2.337	524.2	Capsule Y Removed
3	3.266	522.9	
4	4.112	524.0	
5	5.173	522.7	
6	6.323	523.2	
7	7.414	522.7	
8	8.622	523.1	Capsule V Removed

Time weighted temperature values for these (2) surveillance capsules is derived in Table H-1 and Table H-2.

TABLE H-1. IP2 CAPSULE Y			
Cycle	EFPY	Cold Leg Temp	EFPY x Cold Leg Temp
1a	1.041	540.6	562.7646
1b	1.419	528.6	750.0834
2a	1.666	529.2	881.6472
2b	2.337	524.2	1225.0554
SUM	6.463		3419.5506
Time Weighted Average Temperature = 3419.5506/6.463=529.0965			

TABLE H-2. IP2 CAPSULE V			
Cycle	EFPY	Cold Leg Temp	EFPY x Cold Leg Temp
1a	1.041	540.6	562.7646
1b	1.419	528.6	750.0834
2a	1.666	529.2	881.6472
2b	2.337	524.2	1225.0554
3	3.266	522.9	1707.7914
4	4.112	524	2154.688
5	5.173	522.7	2703.9271
6	6.323	523.2	3308.1936
7	7.414	522.7	3875.2978
8	8.622	523.1	4510.1682
SUM	41.373		21679.6167
Time Weighted Average Temperature = 21679.6167/41.373=524.0039			

The radiation exposure and temperature history of Indian Point Unit 3 relative to Capsule T, Y, Z, and X is shown below.

IP3 HISTORY			
Cycle	EFPY	T cold	Capsule
1A	1.342	539.4	Capsule T Removed
1B	1.382	540	
2	2.280	539.4	
3	3.30	539.4	Capsule Y Removed
4	4.424	539.4	
5	5.566	537.5	Capsule Z Removed
6	6.579	537.5	
7	7.839	539.8	
8	8.972	539.8	
9	10.518	540.1	
10	12.310	540	
11	13.791	540.4	
12A	15.361	540.1	
12B	15.601	540	Capsule X Removed
13	17.433	539.5	
14	19.297	538.7	

Time weighted temperature values for each of the four- (4) surveillance capsules with W5214 weld metal from Indian Point Unit 3 is derived in Tables H-3, H-4, H-5, and H-6.

TABLE H-3. IP3 CAPSULE T			
Cycle	EFPY	Cold Leg Temp	EFPY x Cold Leg Temp
1A	1.342	539.4	723.8748
Average Temperature = 723.9/1.342 = 539.4			

TABLE H-4. IP3 CAPSULE Y			
Cycle	EFPY	Cold Leg Temp	EFPY x Cold Leg Temp
1A	1.342	539.4	723.8748
1B	1.382	540	746.28
2	2.28	539.4	1229.832
3	3.3	539.4	1780.02
SUM	8.304		4480.0068
Time Weighted Average Temperature = 4480.0068/8.304=539.507			

TABLE H-5. IP3 CAPSULE Z			
Cycle	EFPY	Cold Leg Temp	EFPY x Cold Leg Temp
1A	1.342	539.4	723.8748
1B	1.382	540	746.28
2	2.28	539.4	1229.832
3	3.3	539.4	1780.02
4	4.424	539.4	2386.3056
5	5.566	537.5	2991.725
SUM	18.294		9858.0374
Time Weighted Average Temperature =9858.0374/18294=538.867			

TABLE H-6. IP3 CAPSULE X			
Cycle	EFPY	Cold Leg Temp	EFPY x Cold Leg Temp
1A	1.342	539.4	723.8748
1B	1.382	540	746.28
2	2.28	539.4	1229.832
3	3.3	539.4	1780.02
4	4.424	539.4	2386.3056
5	5.566	537.5	2991.725
6	6.579	537.5	3536.2125
7	7.839	539.8	4231.4922
8	8.972	539.8	4843.0856
9	10.518	540.1	5680.7718
10	12.31	540	6647.4
11	13.791	540.4	7452.6564
12A	15.361	540.1	8296.4761
12B	15.601	540	8424.54
SUM	109.265		58970.672
Time Weighted Average Temperature =58970.672/109.265=539.703			

Table H-7. PALISADES VESSEL TIME-WEIGHTED AVERAGE TEMPERATURE						
Operating Cycle	**Cycle Length (EFPD)**	**Factored Days**	**Cycle Average Temperature (Ti)**	**Factored Days x Time Weighted Cycle Avg. Ti**	**Cycle Length (EFPD) from CNS-04-02-01, Revision 1**	**Cycle Length Adjusted by 0.98xEFPD**
1	371.7(1)	0.041	523	21.5	379.3	371.7
2	440.1(1)	0.049	529	25.8	449.1	440.1
3	342.5(1)	0.038	534	20.2	349.5	342.5
4	321.0(1)	0.036	536	19.0	327.6	321.0
5	386.7(1)	0.043	536	22.9	394.6	386.7
6	326.7(1)	0.036	536	19.4	333.4	326.7
7	362.5(1)	0.040	536	21.5	369.9	362.5
8	366.1(1)	0.041	537	21.8	373.6	366.1
9	292.5(1)	0.032	534	17.3	298.5	292.5
10	349.7(1)	0.039	534	20.7	356.8	349.7
11	421.9(1)	0.047	533	24.9	430.5	421.9
12	399.3(1)	0.044	534	23.6	407.4	399.3
13	419.6	0.046	536	24.9		
14	449.3	0.050	537	26.7		
15	401.3	0.044	537	23.8		
16	444.3	0.049	537	26.4		
17	493.1	0.055	537	29.3		
18	472	0.052	537	28.0		
19	459.2	0.051	537	27.3		
20	499.8	0.055	537	29.7		
21	519.2(2)	0.057	537	30.9		
22	498.8(2)	0.055	537	29.6		
	SUM	**1**				
SUM	**9037.4**					

(1) EFPD for Cycles 1 - 12 have been reduced by 2% per Ref. 23.
(2) EFPD for Cycles 21 and 22 are projected values.

SUM	**535.2°F**
Time Weighted Average Temperature	**535.2°F**

APPENDIX I

LISTING OF DESIGN INPUTS FOR WELD HEAT NO. W5214 SURVEILLANCE DATA RE-EVALUATION

Listing of Design Inputs for Weld Heat No. W5214 Surveillance Data Re-evaluation

A roadmap to the references used in this report is provided in this Appendix. This roadmap provides a snapshot of the various references used to obtain the design inputs for chemistry factor, unirradiated and irradiated Charpy V-notch (CVN) data, the TANH methodology used to fit the CVN data, capsule fluences, the regulatory guidance documents and codes used, and the inputs used for the estimation of the time weighted average temperatures for the various surveillance capsules. It is to be noted that several of the inputs used in this report have been superseded by more recent reports based on the availability of new data (such as re-evaluation of capsule reports and updated fluence calculations).

The measured chemistries for the surveillance capsule materials were obtained from the respective surveillance capsule reports that contain the weld heat No. W5214. Fluence data that were historically used in previous reports and submittals were also obtained from the respective capsule reports. More recent fluence values for the surveillance capsules were obtained from an updated fluence calculation by Westinghouse [18]. It is to be noted that even though the design inputs used in this capsule re-evaluation may be available in more than one source, the roadmap points to specific references in order to use the most current and valid inputs in the analysis.

Table I-1. Listing of Design Inputs and References Used

Reference	Report Reference No.	Description	Chemistry Factor	CVN Data	TANH Methodology	Fluence	Regulatory Documents/Codes	Time Weighted Average Temperature	SI Calculation Package
BAW-2398	14	Palisades capsule SA-240-1	X	X					
BAW-2341	15	Palisades capsule SA-60-1	X	X					
J. Kneeland Letter Dated Feb 2, 1999	16	Palisades Unirradiated		X					
WCAP-7373	27	H. B. Robinson Unirradiated		X					
WCAP-15805	12	HBR2 Capsule X	X	X		X			
SwRI Project No. 17-2108	6	IP 2 Capsule V	X	X		X			
WCAP-7323	8	IP 2 Unirradiated		X					
SwRI Project No 02-5212	7	IP 2 Capsule Y		X					
WCAP-16251-NP	13	IP 3 Capsule X	X	X		X			
WCAP-8475	9	IP 3 Unirradiated		X					
WCAP-9491	10	IP 3 Capsule T		X					
WCAP-10300	11	IP 3 Capsule Y		X					
N. Haskell (Palisades) Letter to NRC	26	Palisades RAI	X						
WCAP-15353, Rev 0	3, 34	Palisades Fluence Evaluation				X			
Email from S. Anderson (Westinghouse) to Tim Griesbach (SIA): 04/15/2010	18	Revised Fluence Values for Design Inputs to PTS Evaluation				X			
D. S. Hood (NRC) Letter to N. Haskall (Palisades)	1	Fluence evaluation and schedule for reaching PTS Screening Criteria				X			

Table I-1. Listing of Design Inputs and References Used (cont.)

Reference	Report Reference No.	Description	Chemistry Factor	CVN Data	TANH Methodology	Fluence	Regulatory Documents/Codes	Time Weighted Average Temperature	SI Calculation Package
CAPL-01-009	17	Neutron Fluence Analysis for Palisades Surveillance Capsule SA-240-1				X			
CE NPSD-1119	20	CEOG Cu, Ni Chemistries	X						
ASTM E-185-66	22	Surveillance Testing			X		X		
M. A. EricksonKirk et.al, Journal of Pressure Vessel Technology, v. 131, 2009	25	CVN Data Fitting Methodology			X				
CVGRAPH 5	4	Software program to fit CVN Data			X				
10CFR50.61	2	PTS Rule					X		
Reg Guide 1.190	5	Fluence Calculation Methodology					X		
Generic Letter 92-01	19	RPV Integrity Assessment Workshop					X		
ASME Section III	21	ASME Boiler and Pressure Vessel Code					X		
NUREG/CR-6413, ORNL/TM-13133	24	Irradiation Data for A302B and A533B Correlation Monitor Materials			X		X		
NRC RVID2	31	NRC Reactor Vessel Integrity Database		X			X		
FCX-00538	29	IP2 Vessel Head Temperature						X	
IP3-CALC-RV-03720	30	IP3 Vessel Head Temperature						X	
CNS-04-02-01, Rev. 1	28	Evaluations of Palisades RPV Through Period of Extended Operation						X	
LAR of 2-21-2000 EA-DOR-09-01 Rev. 0	23	Cycle 1 - 12 EFPD for Time Weighted Average Temperature Calculation						X	
0901132.301, Rev. 0	32	Determination of ΔT_{30} Values for the Heat No. W5214							X
0901132.302, Rev. 0	33	Verification of the Time-Weighted Average Temperatures for IP2 and IP3 Capsules							X

ATTACHMENT 4

Westinghouse Review of PTS Screening Criteria Assessment

Westinghouse Letter, "Westinghouse Review of Reactor Vessel Weld Heat W5214 Surveillance Data for Palisades," Stephen T. Byrne (W) to Keith Smith (PNP), LTR-RIDA-10-107-NP, Rev. 0, May 13, 2010.

3 pages follow

Westinghouse Non-Proprietary Class 3

Westinghouse

Westinghouse Electric Company
Nuclear Services
Engineering Services
20 International Drive
Windsor, Connecticut 06095
USA

To: Keith Smith, Palisades Plant
cc: Nathan A. Palm

Date: May 13, 2010

From: Stephen T. Byrne
Ext: 860-731-6703
Fax: 860-731-6709

Our ref: LTR-RIDA-10-107-NP, Rev. 0

Subject: Review of Reactor Vessel Weld Heat W5214 Surveillance Data for Palisades

References:

1. U.S. Nuclear Regulatory Commission, 10 CFR 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," 60 FR 65468, December 19, 1995, as amended at 61 FR 39300, July 29, 1996; 72 FR 49500, August 28, 2007; 73 FR 5722, January 31, 2008.
2. Structural Integrity Associates Report, 0901132.401, Rev. 0, "Evaluation of Surveillance Data for Weld Heat No. W5214 for Application to Palisades PTS Analysis," April 20, 2010.

Executive Summary

The purpose of this letter is to document a review of the SIA report [2] that updated the pressurized thermal shock (PTS) screening criteria assessment for the Palisades reactor pressure vessel.

A chemistry factor (CF) value of 227.4°F was determined in [2] based on post-irradiation surveillance test results specific to the Palisades vessel weld 2-112 A/C, that contains weld wire heat number W5214 with Ni-200 wire addition. The CF evaluation incorporates data from four different surveillance programs, and applies a conservative methodology to normalize the data for differences in chemistry and irradiation temperature. The calculated 227.4°F CF value is more precise than a CF derived only from Cu and Ni content and more conservative than the results from the plant-specific surveillance capsules. RT_{PTS} projections for Palisades weld 2-112 A/C were made using the 227.4°F CF. The 270°F PTS screening criterion [1] for this axial weld is projected to be reached at a neutron fluence of 1.685E19 n/cm^2 (E > 1 MeV), which will be attained during the first quarter of 2017.

The assessment performed in [2] is reasonably and conservatively based. Therefore, it is conservative to project that the PTS screening criterion will not be reached in the Palisades reactor vessel until the first quarter of the year 2017. This projection is based on the assumption that the neutron fluence

accumulation rate will continue as expected such that the 60° weld location fluence will reach 1.685E19 n/cm^2 (E > 1 MeV) in the March 2017 – April 2017 time-frame.

Review and Discussion

The integrity of the Palisades reactor pressure vessel is determined in part through an evaluation of the limiting RT_{PTS} for the beltline materials. The RT_{PTS} is determined in accordance with 10 CFR 50.61 [1] and compared to screening criteria specific to the orientation of an assumed flaw. The Palisades reactor pressure vessel beltline consists of the lower and intermediate shell courses that are fabricated from six low-alloy steel plates and the adjoining welds. The focus of the evaluation is the limiting weld fabricated using wire heat number W5214 with Ni-200 wire addition. The chemistry factor determined for this weld from Table 1 of [1] using best estimate copper (Cu) and nickel (Ni) content is stated in [2] as 231.08°F.

In [2], an update is provided to an earlier evaluation of data from reactor vessel surveillance capsules irradiated in Palisades and from three other reactor vessels. The data are specific to submerged arc weldments produced using wire heat number W5214 with Ni-200 wire addition. The data specific to Palisades are from supplemental capsules SA-60-1 and SA-240-1. The weld material was obtained from the Palisades retired steam generator and irradiated in the Palisades vessel. The three other sources of weld data applicable to wire heat number W5214 plus Ni-200 wire are H.B. Robinson Unit 2, Indian Point Unit 2, and Indian Point Unit 3. These other sources represent three unique surveillance program welds. The transition temperature shift measurements were determined using a consistent curve-fitting methodology. The neutron fluence calculated for each evaluated surveillance capsule is obtained from recent sources.

The surveillance data credibility analysis followed the criteria in [1], (c)(2)(i). Each of the five criteria were addressed subject to the extent of the information available from each of the four sources.

The evaluation of surveillance data follows the approach described in [1], (c)(2) and (c)(3) with the addition of other considerations. Those considerations include differences in best estimate weld chemistry and irradiation temperature. Determination of chemistry factor (CF) given differences in copper and nickel content between the surveillance program weld and the Palisades reactor vessel weld is addressed in accordance with [1], (c)(2)(ii). Accounting for the effect of differences in the irradiation temperature between the individual plant-specific surveillance capsules and the Palisades reactor vessel is performed using informal guidance provided by the NRC Staff and included in the referenced report. (For each 1°F that the capsule temperature exceeds the mean vessel temperature, the transition temperature shift is decreased by 1°F. Likewise, a positive adjustment was applied to the shift for capsule temperatures lower than that of the vessel.) The two measured transition temperature shifts from Palisades (supplemental capsules SA-60-1 and SA-240-1) and the combined set of twelve measured shifts from the four sources, including adjustments for differences in copper and nickel content and for differences in irradiation temperature, were then analyzed using Equation 5 from [1]. The resultant CF

value was then assessed for predictability relative to each of the twelve adjusted transition temperature shifts.

The evaluation of the two measured transition temperature shifts from Palisades resulted in a CF of 198.8°F. When assessed for predictability, the SA-60-1 and SA-240-1 measured shifts are within approximately 3°F of the prediction (within the one-sigma scatter of 28°F). An additional evaluation of the correlation monitor material from those Palisades supplemental capsules demonstrated the correlation monitor shift measurements to also be consistent with predictions. Therefore, the plant-specific measurements are fully consistent with predictions.

The evaluation of the combined twelve measured transition temperature shifts for four separate weldments produced using wire heat number W5214 with Ni-200 wire addition resulted in a CF of 227.4°F. When assessed for predictability, the twelve measured shifts are within approximately 46°F of the prediction (within the two-sigma scatter of 56°F).

The 227.4°F CF value was used to project the RT_{PTS} for the Palisades reactor vessel weld 2-112 A/C. The 270°F PTS screening criterion [1] is projected to be reached at a neutron fluence of 1.685E19 n/cm^2 (E > 1 MeV). That fluence will be attained during the first quarter of 2017. The RT_{PTS} projection is based on the assumption that the neutron fluence accumulation rate will continue as expected such that the 60° weld location fluence will reach 1.685E19 n/cm^2 (E > 1 MeV) in the March 2017 – April 2017 time-frame.

In conclusion, this review finds that the assessment performed in [2], including the CF evaluation and the RT_{PTS} projection, is reasonably and conservatively based.

If you have any questions or desire further information, please contact the undersigned.

ELECTRONICALLY APPROVED[1]
Stephen T. Byrne, Author
Reactor Internals Design & Analysis II

Verified by: ELECTRONICALLY APPROVED[1]
Paul R. Sotherland, Reviewer
Reactor Internals Design & Analysis II

Approved by: ELECTRONICALLY APPROVED[1]
John F. Kielb, Acting Manager
Reactor Internals Design & Analysis II

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

ATTACHMENT 5

List of ENO Representatives Planning to Attend Meeting

Brian Kemp, Design Engineering Manager, PNP
Paula Anderson, Licensing Manager, PNP
Jeff Erickson, Licensing, PNP
Keith Smith, Project Manager, PNP
Bill Server, President, ATI Consulting
Tim Griesbach, Senior Associate, Structural Integrity Associates
Steve Byrne, Fellow Engineer, Westinghouse

ATTACHMENT 6

Planned Meeting Agenda

- Introduction of Meeting Attendees
- Purpose of Meeting
- Historical Status of PTS at Palisades
- Current Activities to Manage PTS at Palisades
- Integration of New Data
- Update of WCAP-15353
- PTS Assessment for Limiting Weld Metal
- Future Plans
- Proposed Schedule
- Questions
- Identification of Action Items