Exelon Generation.

Oyster Creek
741 Route 9 South
Forked River, NJ 08731

10 CFR Part 50 Appendix G

RA-18-072

June 29, 2018

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

Oyster Creek Nuclear Generating Station
Renewed Facility Operating License No. DPR-16
NRC Docket No. 50-219

Subject: Oyster Creek Nuclear Generating Station Pressure and Temperature Limit Report Revision 1

The purpose of this letter is to transmit the Pressure and Temperature Limits Report (PTLR) Revision 1 for the Oyster Creek Nuclear Generating Station (OCNGS) in accordance with Technical Specification (TS) 6.23.c, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)."

The OCNGS PTLR was revised to update the Effective Full-Power Years (EFPY) to operate the station to the end of plant life.

Please direct any questions you may have regarding this matter to Mr. Gary Flesher, Regulatory Assurance Manager, at (609) 971-4232.

Respectfully,

Timothy A. Moore
Site Vice President
Oyster Creek Nuclear Generating Station

Attachment: Oyster Creek Generating Station Pressure and Temperature Limit Report (PTLR) Revision 1

CC
Regional Administrator – NRC Region I
NRC Senior Resident Inspector – Oyster Creek Nuclear Generating Station
NRC Project Manager, NRR – Oyster Creek Nuclear Generating Station
Manager, Bureau of Nuclear Engineering – New Jersey Department of Environmental Protection
Mayor of Lacey Township, Forked River, NJ

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Exelon Nuclear Corporation

Oyster Creek Generating Station

Pressure and Temperature Limits Report (PTLR) for 40 Effective Full-Power Years (EFPY)

Prepared by: Sara Dennis Date: 6/18/18

Reviewed by: Jess A. N... Date: 6/18/18
ISI Program Manager

Approved by: William Saraceno Date: 6/20/18
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Concurred by: Gerald J. Jarosinski Date: 6/20/18
Corporate Asset Management

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PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) FOR 40 EFFECTIVE FULL- POWER YEARS

1.0 PURPOSE

The purpose of the Oyster Creek Generating Station (OCGS) Pressure and Temperature Limits Report (PTLR) is to present operating limits relating to:

- Reactor Coolant System (RCS) Pressure versus Temperature limits during Heatup, Cooldown and Hydrostatic/Class 1 Leak Testing;
- RCS Heatup and Cooldown rates;
- Reactor Pressure Vessel (RPV) head flange bolt-up temperature limits.

This report has been prepared in accordance with the requirements of Licensing Topical Report SIR-05-044-A, Revision 1-A, contained within BWROG-TP-11-022-A, Revision 1 **(CM-1)** (Reference 6.1).

2.0 APPLICABILITY

This report is applicable to the OCGS RPV for 40 Effective Full-Power Years (EFPY). The following OCGS Technical Specification (TS) is affected by the information contained in this report:

- TS Limiting Conditions for Operation 3.3 ("Reactor Coolant")
- TS Surveillance Requirement 4.3 ("Reactor Coolant")

The Oyster Creek Reactor Vessel Pressure and Temperature Limits for 32 to 50 EFPY have been developed per Reference 6.2. Only the 40 EFPY limits are incorporated in this revision of the PTLR. Future revisions of the PTLR must be revised per the 10CFR50.59 Review process as applicable.

3.0 METHODOLOGY

The limits in this report were derived as follows:

1) The methodology used is in accordance with Reference 6.1, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," Revision 1-A, incorporating the NRC Safety Evaluation in Reference 6.3.

2) The neutron fluence is calculated in accordance with NRC Regulatory Guide 1.190 (RG 1.190), Reference 6.4 using the RAMA computer code, as documented in Reference 6.5.

3) The adjusted reference temperature (ART) values for the limiting beltline materials are calculated in accordance with NRC Regulatory Guide 1.99, Revision 2 (RG 1.99), Reference 6.6, as documented in References 6.7 and 6.8.

4) The pressure and temperature (P-T) limits were calculated in accordance with the BWROG P-T limits topical report, Revision 0-A (Reference 6.9), as documented in References 6.2 and 6.8. However, the P-T limits in References 6.2 and 6.8 also meet the requirements of the most recent NRC-approved revision of the BWROG topical report, Revision 1-A (Reference 6.1).

5) This revision of the pressure and temperature limits is to incorporate the following changes:
   - Revision 0: Initial issue of PTLR.
   - Revision 1: Incorporated P-T limits for 40 EFPY and removed curves for 32 and 36 EFPY.

Changes to the curves, limits, or parameters within this PTLR, based upon new irradiation fluence data of the RPV, or other plant design assumptions in the Updated Final Safety Analysis Report (UFSAR), can be made pursuant to 10 CFR 50.59, provided the above methodologies are utilized. The revised PTLR shall be submitted to the NRC upon issuance.

Changes to the curves, limits, or parameters within this PTLR, based upon revised RPV fluence calculation methodology, cannot be made without prior NRC approval. Such analysis and revisions shall be submitted to the NRC for review prior to incorporation into the PTLR.

4.0 OPERATING LIMITS

The P-T curves included in this report represent steam dome pressure versus minimum vessel coolant temperature and incorporate the appropriate non-beltline limits and irradiation embrittlement effects in the beltline region.

The operating limits for pressure and temperature are required for three categories of operation: (a) hydrostatic pressure tests and leak tests, referred to as Curve A; (b) core not critical operation, referred to as Curve B; and (c) core critical operation, referred to as Curve C, in accordance with 10 CFR 50 Appendix G (Reference 6.10).

Complete P-T curves were developed for 40 EFPY for OCGS, as documented in Reference 6.2. Composite curves bounding all RPV component curves for Curves A, B, and C for OCGS for 40

EFPY are provided in Figures 1 through 3 of this report. Tabulation of the curves is included in Tables 1 through 3. The ART tables for the OCGS vessel beltline materials are shown in Table 4 for 40 EFPY (Reference 6.8). The P-T Curves A, B, and C presented in this report each represent a bounding curve for the RPV beltline, upper vessel, and bottom head regions. The resulting P-T curves are based on the geometry, design and materials information for the OCGS vessel with the following conditions:

- Heatup and Cooldown rate limit during Hydrostatic and Class 1 Leak Testing (Figure 1: Curve A): ≤ **25°F/hour**[1].
- Normal Operating Heatup and Cooldown rate limit (Figure 2: Curve B - non-nuclear heating, and Figure 3: Curve C - nuclear heating): ≤ **100°F/hour**[2].
- RPV head installation temperature limit (Figure 1: Curve A - Hydrostatic and Class 1 Leak Testing; Figure 2: Curve B - non-nuclear heating): ≥ **60°F**.

Minimum temperature limits are set in accordance with 10 CFR 50, Appendix G (Table 1 in Reference 6.10). Regarding the RPV head installation temperature limit, the minimum bolt-up temperature is selected to address the NRC condition in Section 4.0 of Reference 6.3 regarding lowest service temperature (LST) for all ferritic components of the reactor coolant pressure boundary (RCPB), including piping and other non-RPV components. The minimum temperature is set to 60°F for Curves A and B, which bounds the maximum RT_{NDT} for the closure flange material, 36°F for the upper shell plate (Reference 6.2). The minimum criticality temperature is 96°F for Curve C, which is equal to $RT_{NDT,max}$ + 60°F. However, from Reference 6.2, the non-beltline (feedwater nozzle) P-T limits are more limiting than the closure flange limits for Curves B and C, and the minimum temperature is 76°F for Curve B and 116°F for Curve C, based on the non-beltline P-T limits. These temperatures are consistent with the minimum temperature limits and minimum bolt-up temperatures in the current docketed P-T curves (Reference 6.11, approved by the NRC in Reference 6.12). These temperatures also bound the non-RPV ferritic components of the RCPB, such as piping. Non-ductile fracture was considered in the design of ferritic RCPB piping, in accordance with the requirements of the piping code of construction and specifications identified in the OCGS FSAR Table 5.2-1 (Reference 6.13). Consequently, the P-T limits have considered all

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[1] Interpreted as the temperature change in any 1-hour period is less than or equal to 25°F, based on Reference 6.1.

[2] Interpreted as the temperature change in any 1-hour period is less than or equal to 100°F, the bounding heatup/cooldown rate on which the P-T curves are evaluated, discussed in Section 5.0.

ferritic RCPB components, consistent with the requirements of 10CFR50 Appendix G, as described in Reference 6.13.

5.0 DISCUSSION

The ART of the limiting beltline material is used to adjust the beltline P-T curves to account for irradiation effects. RG 1.99 provides the methods for determining the ART (Reference 6.6). The RG 1.99 methods for determining the limiting material and adjusting the P-T curves using ART are discussed in this section. Appendix A in this report provides additional detail regarding RPV material surveillance programs at Oyster Creek.

The OCGS RPV beltline copper (Cu) and nickel (Ni) values were obtained from the evaluation of the OCGS vessel plate and weld materials (Reference 6.7). The Cu and Ni values were used with Tables 1 and 2 of RG 1.99 to determine a chemistry factor (CF) per Regulatory Position 1.1 of RG 1.99 for welds and plates, respectively.

The peak RPV ID fluence used in the P-T curve evaluation for 40 EFPY is 5.69×10^{18} n/cm^2 for OCGS (Table 4). Fluence values were linearly interpolated for 40 EFPY based on the fluence values for 32 and 50 EFPY provided in Reference 6.5, which were calculated using methods that comply with the guidelines of RG 1.190 (Reference 6.4). This fluence value applies to the limiting beltline lower-intermediate shell plate 564-03C for OCGS. The ID fluence value was adjusted based upon an attenuation factor of 0.630 for a postulated 1/4t flaw. As a result, the 1/4t 40 EFPY fluence for the limiting lower-intermediate plate is 3.58×10^{18} n/cm^2 for OCGS. The limiting 1/4t ART for the OCGS beltline for 40 EFPY is 172.5°F.

The RPV fluence values in Reference 6.5 were used as input to the currently docketed P-T curves. An updated fluence evaluation for the OCGS RPV was subsequently performed in Reference 6.14. Based on review of the updated fluence values for 40 EFPY, the previously calculated limiting 1/4t ART value bounds the limiting 1/4t ART calculated using the updated fluence. Accordingly, the 40 EFPY ART values previously documented in Reference 6.8 are provided in Table 4 and used as input to the P-T curves.

The P-T limits are developed to bound all ferritic materials in the RPV, including the consideration of stress levels from structural discontinuities such as nozzles. The latest NRC-approved revision of the BWROG P-T limits topical report, Revision 1-A (Reference 6.1), was developed to incorporate additional requirements related to the effects of nozzles within the beltline region. Revision 0 of this

PTLR (Reference 6.11) and the supporting P-T curves calculation (Reference 6.2) predate the approval of the Revision 1-A BWROG topical report and were prepared to an earlier revision of the topical report, Revision 0-A (Reference 6.9). However, the OCGS P-T limits developed in Reference 6.2 already appropriately addressed the effects of RPV nozzles. The limiting RPV nozzle configuration, the feedwater (FW) nozzle, is considered in the evaluation of the non-beltline (upper vessel) region P-T limits. The OCGS small bore instrument nozzles are outside the beltline region and are bounded by the non-beltline/FW nozzle P-T limits, as agreed by the NRC in Reference 6.12. OCGS has no nozzles within the RPV beltline. Therefore, the 40 EFPY P-T limits are also consistent with the latest NRC-approved BWROG topical report methodology (Reference 6.1). Accordingly, both topical reports are referenced in this PTLR.

The P-T curves for the core not critical and core critical operating conditions at a given EFPY apply for both the 1/4t and 3/4t locations. When combining pressure and thermal stresses, it is usually necessary to evaluate stresses at the 1/4t location (inside surface flaw) and the 3/4t location (outside surface flaw). This is because the thermal gradient tensile stress of interest is in the inner wall during cooldown and is in the outer wall during heatup. However, as a conservative simplification, the thermal gradient stress at the 1/4t location is assumed to be tensile for both heatup and cooldown. This results in the approach of applying the maximum tensile stress at the 1/4t location. This approach is conservative because irradiation effects cause the allowable toughness at 1/4t to be less than that at 3/4t for a given metal temperature. This approach causes no operational difficulties, since the BWR is at steam saturation conditions during normal operation, which is well below the P-T curve limits.

For the core not critical curve (Curve B) and the core critical curve (Curve C), the P-T curves are developed based on a coolant heatup and cooldown temperature rate of ≤ **100°F/hr** for which the curves are applicable. However, the core not critical and the core critical curves were also developed to bound Service Level A/B RPV thermal transients defined on the RPV thermal cycle diagram and the nozzle thermal cycle diagrams. For the hydrostatic pressure and leak test curve (Curve A), a coolant heatup and cooldown temperature rate of ≤ **25°F/hr** must be maintained. The P-T limits and corresponding limits of either Curve A or B may be applied, if necessary, while achieving or recovering from test conditions. So, although Curve A applies during pressure testing, the limits of Curve B may be conservatively used during pressure testing if the pressure test heatup/cooldown rate limits cannot be maintained.

The initial RT_{NDT}, the chemistry (weight-percent Cu and Ni) and adjusted reference temperature at the 1/4 thickness location for all RPV beltline materials significantly affected by fluence (i.e., fluence

> 10^{17} n/cm^2 for E > 1 MeV) are shown in Table 4 for 40 EFPY. The initial RT_{NDT} values shown in Table 4 (obtained from Reference 6.15) were developed using the procedures of Branch Technical Position MTEB 5-2 in Standard Review Plan 5.3.2 in NUREG-0800, and they have been previously approved for use by the NRC (Reference 6.16).

Per the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP), representative weld and plate surveillance materials data for OCGS were reviewed from BWRVIP-135, Revision 3 (Reference 6.17), and in accordance with Appendix A of Reference 6.1. Use of the BWRVIP ISP for OCGS was approved by the NRC in Reference 6.18. The representative heats of weld and plate material in the ISP are not the same as the target weld and plate heats in the vessel, and no surveillance heats are present in the OCGS beltline. Therefore, the chemistry factors (CFs) from the tables in Regulatory Guide 1.99, Revision 2 (Reference 6.6), were used in the determination of the ART values for all materials for the OCGS vessel.

The only computer code used in the determination of the OCGS P-T curves was the ANSYS (Release 8.1 with Service Pack 1) finite element computer program for the feedwater nozzle (non-beltline) stresses. This analysis was performed to determine through-wall thermal and pressure stress distributions for the OCGS feedwater nozzles due to a step-change thermal transient (Reference 6.19). The ANSYS program was controlled under the vendor's 10 CFR 50 Appendix B Quality Assurance Program for nuclear quality-related work. Benchmarking consistent with NRC GL 83-11, Supplement 1 (Reference 6.20) was performed as a part of the computer program verification by comparing the solutions produced by the computer code to hand calculations for several problems. The following inputs were used as input to the finite element analysis:

- With respect to operating conditions, stress distributions were developed for a thermal shock of 450°F, which represents the maximum thermal shock for the feedwater nozzle during normal operating conditions. The stress results for a 450°F shock are appropriate for use in developing the non-beltline P-T curves based on the limiting feedwater nozzle, as a shock of 450°F is representative of the Turbine Roll transient that occurs in the feedwater nozzle as part of the 100°F/hr startup transient. Therefore, these stresses represent the bounding stresses in the feedwater nozzle associated with 100°F/hr heatup/cooldown limits associated with the P-T curves for the upper vessel feedwater nozzle region. The boundary integral equation/influence function (BIE/IF) methodology as presented in Reference 6.1 was used in Reference 6.2 to calculate the thermal stress intensity factor K_{It} by fitting a third order polynomial equation to the path stress distribution for the thermal load case.

- Heat transfer coefficients were calculated from the governing design basis stress report for the OCGS feedwater nozzle and from a model of the heat transfer coefficient as a function of flow rate. The heat transfer coefficients were evaluated at flow rates that bound the actual operating conditions in the feedwater nozzles at OCGS.

- With respect to pressure stress, a unit pressure of 1,000 psig was applied to the internal surfaces of the finite element model. The pressure stress distribution was taken along the same path as the thermal stress distribution. The BIE/IF methodology presented in Reference 6.1 was used in Reference 6.2 to calculate the pressure stress intensity factor K_{Ip} by fitting a third order polynomial equation to the path stress distribution for the pressure load case. The resulting K_{Ip} can be linearly scaled to determine the K_{Ip} for various RPV internal pressures.

- A two-dimensional, axisymmetric finite element model of the feedwater nozzle was constructed using the same modeling techniques that were employed to evaluate the feedwater nozzle in the governing design basis stress report. To model the feedwater nozzle using a two-dimensional model, the analysis was performed as a penetration in a sphere and not in a cylinder. To account for three-dimensional effects on the pressure stresses at the nozzle blend radius, a conversion factor of 3.2 times the cylinder radius was used to model the sphere (Reference 6.19). Material properties were evaluated at 325°F to conservatively bound the 100°F condition where the maximum stresses occurred.

6.0 REFERENCES

6.1 Licensing Topical Report (LTR) BWROG-TP-11-022-A, Revision 1 (SIR-05-044, Revision 1-A), "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," August 2013, ADAMS Accession No. ML13277A557.

6.2 Structural Integrity Associates, Inc. Calculation No. OC-05Q-313, Revision 3, "Revised P-T Curves Based on New Fluence," November 28, 2007.

6.3 U.S. NRC Letter to BWROG dated May 16, 2013, "Final Safety Evaluation for Boiling Water Reactor Owners' Group Topical Report BWROG-TP-11-022, Revision 1, November 2011, 'Pressure-Temperature Limits Report Methodology for Boiling Water Reactors'" (TAC NO. ME7649, ADAMS Accession No. ML13107A062).

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

6.5 TransWare Enterprises Inc. Report No. EXL-FLU-001-R-002, Revision 0, "Fluence Evaluation for Oyster Creek Reactor Pressure Vessel," SI File No. OC-05Q-257.

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

6.7 Structural Integrity Associates, Inc. Calculation No. OC-05Q-301, Revision 1, "Adjusted Reference Temperature Evaluation," May 11, 2006.

6.8 Letter TJG-08-001 from T.J. Griesbach (SI) to Greg Harttraft (Exelon), Revised Calculation of P-T Limit Curves for the Oyster Creek Generating Station, dated February 26, 2008.

6.9 SI Report No. SIR-05-044-A, Revision 0, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," April 2007, ADAMS Accession No. ML072340283, SI File No. GE-10Q-401.

6.10 Part 50 of Title 10 of the Code of Federal Regulations, Appendix G, "Fracture Toughness Requirements."

6.11 Enclosure 3 to AmerGen Letter No. RA-08-004, "Oyster Creek Generating Station, Pressure and Temperature Limits Report (PTLR) for 32 and 36 Effective Full-Power Years (EFPY)," Exelon Nuclear Corporation, Revision 0c, March 4, 2008 (ADAMS Accession No. ML080740287).

6.12 Oyster Creek Nuclear Generating Station License Amendment No. 269, "Relocation of Pressure and Temperature Curves to the Pressure and Temperature Limits Report," September 30, 2008 (ADAMS Accession No. ML082390685).

6.13 Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 5.2, Revision 17, October 2011. SI File No. 1601200.201.

6.14 TransWare Enterprises Inc. Report No. OYC-FLU-001-R-005, Revision 0, "Non-Proprietary Version of Oyster Creek Generating Station Reactor Pressure Vessel Fluence Evaluation," SI File No. 1401181.211.

6.15 General Electric Report GENE-B13-01769, "Pressure-Temperature Curves per Regulatory Guide 1.99, Revision 2 for the Oyster Creek Nuclear Generating Station," July 1995, SI File No. OC-05Q-210.

6.16 Letter from Pao-Tsin Kuo (U.S. NRC) to Mr. Timothy Rausch (AmerGen Energy Company, LLC), "Safety Evaluation Report Related to the License Renewal of Oyster Creek Nuclear Generating Station," Docket No. 50-219, dated March 30, 2007.

6.17 *BWRVIP-135, Revision 3: BWR Vessel and Internals Project, Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations.* EPRI, Palo Alto, CA: 2014. 3002003144. **EPRI PROPRIETARY INFORMATION.**

6.18 Letter from P. S. Tam (NRC) to C. M. Crane (AmerGen Energy Company, LLC), "Oyster Creek Nuclear Generating Station (OCNGS) – Issuance of Amendment RE: Use of Integrated Surveillance Program for Reactor Vessel Specimen Surveillance (TAC NO. MB7005)", dated April 27, 2004.

6.19 Structural Integrity Associates Calculation No. OC-05Q-307, Revision 0, "Feedwater Nozzle Green's Functions," July 20, 2005.

6.20 U. S. Nuclear Regulatory Commission, Generic Letter 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," June 24, 1999.

6.21 Part 50 of Title 10 of the Code of Federal Regulations, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

6.22 GPU Nuclear Technical Data Report TDR-725, Revision 4, "Testing and Evaluation of Irradiated Reactor Vessel Materials Surveillance Program Specimens," January 3, 1996, SI File No. OC-05Q-219.

6.23 Manahan, M. P., et. al., Examination, Testing, and Evaluation of Specimens from the 210° Irradiated Pressure Vessel Surveillance Capsule for the Oyster Creek Nuclear Generating Station, Battelle Columbus Laboratories Report BCL-382-85-1, Rev. 1, October 1985, SI File No. GPUN-27Q-215.

6.24 *BWRVIP-86, Revision 1-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan.* EPRI, Palo Alto, CA: 2012. 1025144. **EPRI PROPRIETARY INFORMATION.**

Figure 1: Oyster Creek Pressure Test (Curve A) P-T Curve (40 EFPY)

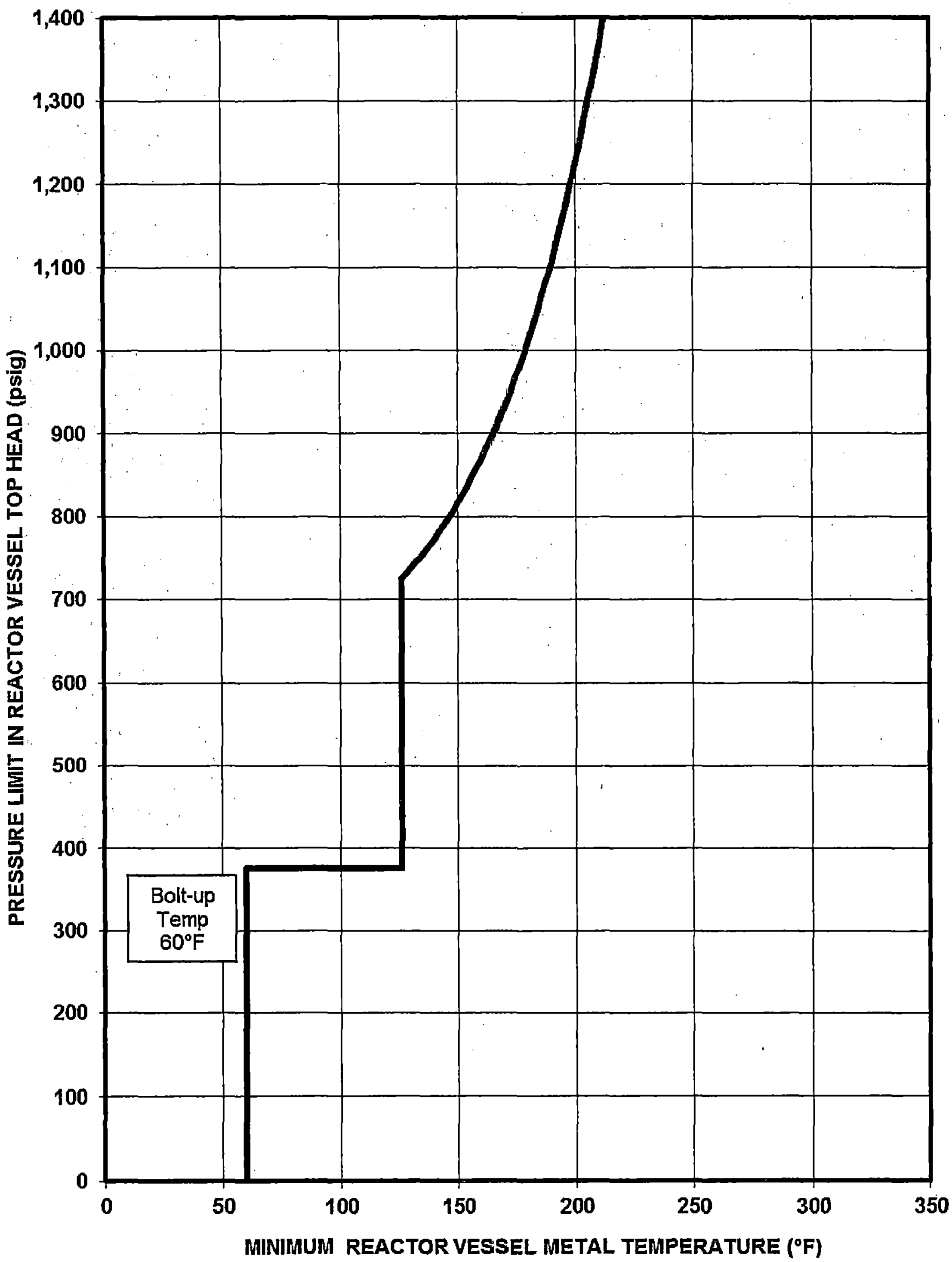

Figure 2: Oyster Creek Core Not Critical (Curve B) P-T Curve (40 EFPY)

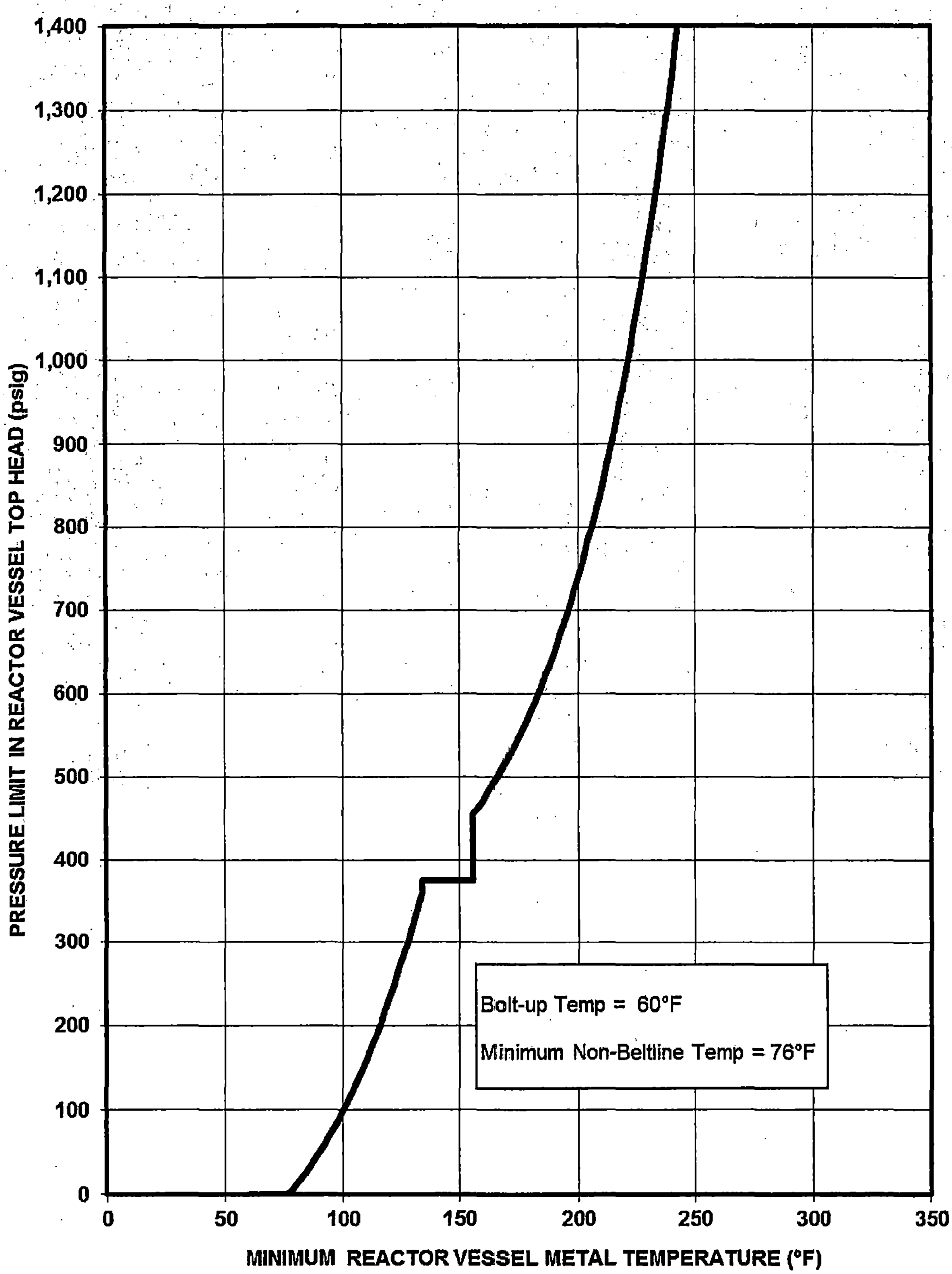

Figure 3: Oyster Creek Core Critical (Curve C) P-T Curve (40 EFPY)

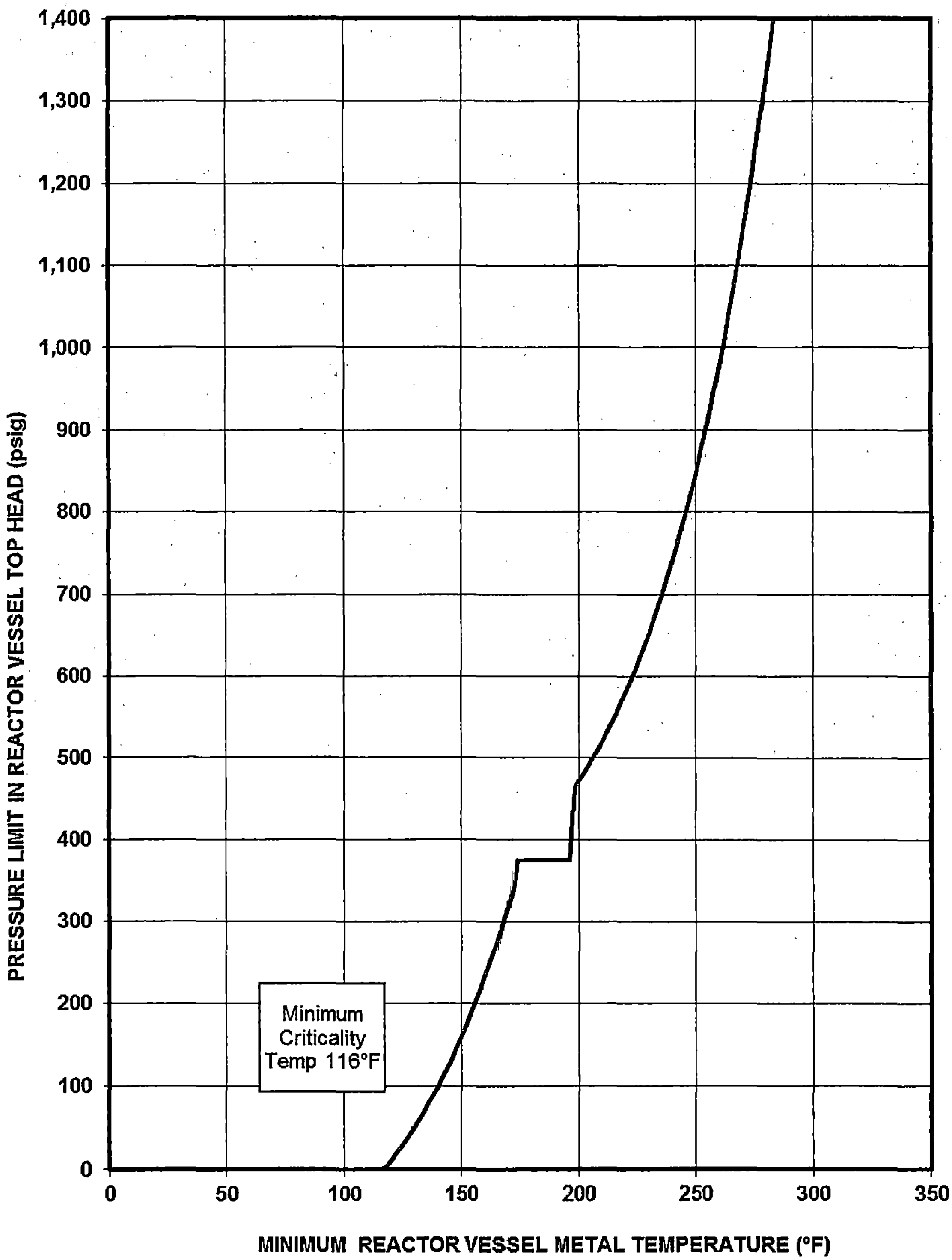

Table 1: Oyster Creek Pressure Test (Curve A) P-T Curve (40 EFPY)

Temperature for P-T Curve (°F)	Pressure for P-T Curve (psig)
60	0.0
60	375.0
62	375.0
64	375.0
66	375.0
68	375.0
70	375.0
72	375.0
74	375.0
76	375.0
78	375.0
80	375.0
82	375.0
84	375.0
86	375.0
88	375.0
90	375.0
92	375.0
94	375.0
96	375.0
98	375.0
100	375.0
102	375.0
104	375.0
106	375.0
108	375.0
110	375.0
112	375.0
114	375.0
116	375.0
118	375.0
120	375.0
122	375.0
124	375.0
126	375.0
126	412.4
126	419.6
126	427.0
126	434.7
126	442.7
126	451.1
126	459.8
126	468.8
126	478.3
126	488.1
126	498.3

Temperature for P-T Curve (°F)	Pressure for P-T Curve (psig)
126	508.9
126	520.0
126	531.5
126	543.5
126	555.9
126	568.9
126	582.4
126	596.5
126	611.1
126	626.3
126	642.2
126	658.7
126	675.9
126	693.7
126	712.3
126	725.2
126	725.2
126	725.2
126	725.2
126	725.2
126	725.2
128	731.2
130	737.5
132	744.0
134	750.8
136	757.8
138	765.2
140	772.8
142	780.8
144	789.0
146	797.7
148	806.6
150	816.0
152	825.7
154	835.8
156	846.3
158	857.3
160	868.7
162	880.5
164	892.9
166	905.8
168	919.1
170	933.1
172	947.6
174	962.7
176	978.4
178	994.7
180	1011.7

Temperature for P-T Curve (°F)	Pressure for P-T Curve (psig)
182	1029.4
184	1047.8
186	1067.0
188	1087.0
190	1107.8
192	1129.4
194	1151.9
196	1175.3
198	1199.7
200	1225.1
202	1251.5
204	1279.0
206	1307.6
208	1337.4
210	1368.4
212	1400.7
214	1434.2
216	1469.2
218	1505.6
220	1543.4
222	1582.8
224	1623.8

Table 2: Oyster Creek Core Not Critical (Curve B) P-T Curve (40 EFPY)

Temperature for P-T Curve (°F)	Pressure for P-T Curve (psig)
60	0.0
60	0.0
62	0.0
64	0.0
66	0.0
68	0.0
70	0.0
72	0.0
74	0.0
76	0.0
78	3.2
80	10.2
82	17.6
84	25.2
86	33.2
88	41.5
90	50.1
92	59.1
94	68.5
96	78.2
98	88.3
100	98.9
102	109.9
104	121.3
106	133.2
108	145.5
110	158.4
112	171.8
114	185.8
116	200.3
118	215.4
120	231.2
122	247.5
124	264.6
126	282.3
128	300.8
130	320.0
132	340.0
134	360.8
134	375.0
136	375.0
138	375.0
140	375.0
142	375.0
144	375.0

Temperature for P-T Curve (°F)	Pressure for P-T Curve (psig)
146	375.0
148	375.0
150	375.0
152	375.0
154	375.0
156	375.0
156	428.5
156	452.9
156	455.3
156	455.3
156	455.3
156	455.3
156	455.3
156	455.3
156	455.3
158	463.5
160	472.0
162	480.9
164	490.2
166	499.9
168	509.9
170	520.3
172	531.2
174	542.5
176	554.3
178	566.6
180	579.3
182	592.6
184	606.4
186	620.8
188	635.8
190	651.4
192	667.6
194	684.5
196	702.0
198	720.3
200	739.4
202	759.2
204	779.8
206	801.3
208	823.6
210	846.8
212	871.0
214	896.2
216	922.4
218	949.7
220	978.1
222	1007.7

Temperature for P-T Curve (°F)	Pressure for P-T Curve (psig)
224	1038.4
226	1070.4
228	1103.8
230	1138.4
232	1174.5
234	1212.1
236	1251.2
238	1291.9
240	1334.3
242	1378.4
244	1424.3
246	1472.0
248	1521.7
250	1573.5
252	1627.3

Table 3: Oyster Creek Core Critical (Curve C) P-T Curve (40 EFPY)

Temperature for P-T Curve (°F)	Pressure for P-T Curve (psig)
100	0
100	0
102	0
104	0
106	0
108	0
110	0
112	0
114	0
116	0
118	3
120	10
122	18
124	25
126	33
128	42
130	50
132	59
134	68
136	78
138	88
140	99
142	110
144	121
146	133
148	146
150	158
152	172
154	186
156	200
158	215
160	231
162	248
164	265
166	282
168	301
170	320
172	340
174	375
176	375
178	375
180	375
182	375

Temperature for P-T Curve (°F)	Pressure for P-T Curve (psig)
184	375
186	375
188	375
190	375
192	375
194	375
196	375
198	463
200	472
202	481
204	490
206	500
208	510
210	520
212	531
214	543
216	554
218	567
220	579
222	593
224	606
226	621
228	636
230	651
232	668
234	684
236	702
238	720
240	739
242	759
244	780
246	801
248	824
250	847
252	871
254	896
256	922
258	950
260	978
262	1008
264	1038
266	1070
268	1104
270	1138
272	1175
274	1212
276	1251
278	1292

Temperature for P-T Curve (°F)	Pressure for P-T Curve (psig)
280	1334
282	1378
284	1424

Table 4: Oyster Creek ART Calculations for 40 EFPY (Reference 6.8)

PLATES

PART NAME	PIECE NO.	CODE NO. (Note 3)	HEAT NO. / FORGING S/N	INITIAL RT_{NDT} °F	CHEMISTRY Cu %	CHEMISTRY Ni %	CF °F	ID Fluence 10^{19} n/cm²	Attenuation	1/4-T Fluence 10^{19} n/cm²	FF	ADJUSTMENTS FOR MARGIN AND BELTLINE IRRADIATION SHIFT ΔRT_{NDT} °F	σ_D °F	σ_I °F	Margin °F	ART °F
Lower Intermediate Shell Plates	564-03A	G-8-7	P-2161-1	17	0.21	0.48	139.4	5.69E+18	0.630	3.58E+18	0.717	99.9	17.0	10.7	40.2	157.1
	564-03B	G-8-8	P-2136-2	8	0.18	0.46	120.7	5.69E+18	0.630	3.58E+18	0.717	86.5	17.0	12.8	42.6	137.1
	564-03C	G-8-6	P-2150-1	31	0.2	0.51	138.2	5.69E+18	0.630	3.58E+18	0.717	99.1	17.0	12.7	42.4	172.5
Lower Shell Plates	564-03D	G-307-1	T-1937-2	30	0.17	0.11	79.45	3.07E+18	0.630	1.93E+18	0.561	44.6	17.0	12.6	42.3	116.9
	564-03E	G-308-1	T-1937-1	21	0.17	0.11	79.45	3.07E+18	0.630	1.93E+18	0.561	44.6	17.0	14.2	44.3	109.9
	564-03F	G-307-5	P-2076-2	3	0.27	0.63	173.9	3.07E+18	0.630	1.93E+18	0.561	97.6	17.0	13.9	43.9	144.5

SHELL AXIAL SEAM WELDS

PART NAME	Weld No.	Weld Metal Type / Heat Number	Flux Lot Number / Flux Type	INITIAL RT_{NDT} °F	CHEMISTRY Cu %	CHEMISTRY Ni %	CF °F	ID Fluence n/cm²	Attenuation	1/4-T Fluence n/cm²	FF	ADJUSTMENTS FOR MARGIN AND BELTLINE IRRADIATION SHIFT ΔRT_{NDT} °F	σ_Δ °F	σ_I °F	Margin °F	ART °F
Lower Intermediate Shell Axial Welds	2-564A, 2-564B, and 2-564C	86054B	ARCOS B-5	-50	0.214	0.05	97.6	5.03E+18	0.630	3.17E+18	0.684	66.8	28.0	0.0	56.0	72.8
Lower Shell Axial Welds	2-564D, 2-564E, 2-564F	86054B	ARCOS B-5	-8	0.214	0.05	97.6	3.03E+18	0.630	1.91E+18	0.559	54.5	27.2	0.0	54.5	101.0

SHELL CIRCUMFERENTIAL SEAM WELD

PART NAME	Weld No.	Weld Metal Type / Heat Number	Flux Lot Number / Flux Type	INITIAL RT_{NDT} °F	CHEMISTRY Cu %	CHEMISTRY Ni %	CF °F	ID Fluence 10^{19} n/cm²	Attenuation	1/4-T Fluence 10^{19} n/cm²	FF	ADJUSTMENTS FOR MARGIN AND BELTLINE IRRADIATION SHIFT ΔRT_{NDT} °F	σ_D °F	σ_I °F	Margin °F	ART °F
Lower Shell to Lower Intermediate Shell Circumferential Weld	3-564	1248	ARCOS B-5	-50	0.206	0.07	98.2	3.07E+18	0.630	1.93E+18	0.561	54.0	27.0	0.0	54.0	58.0

APPENDIX A

Oyster Creek Reactor Vessel Material Surveillance Programs

Oyster Creek:

In accordance with 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements (Reference 6.21), one surveillance capsule has been removed from the OCGS RPV. The first surveillance capsule was removed from the OCGS RPV on February 12, 1983 after 8.38 EFPY (Reference 6.22). The surveillance capsule contained flux wires for neutron fluence measurement, Charpy V-Notch impact test specimens and uniaxial tensile test specimens fabricated using materials from the vessel materials within the core beltline region. The flux wires and test specimens removed from the capsule were tested according to ASTM E185-82. The methods and results of testing are presented in References 6.22 and 6.23, as required by 10 CFR 50, Appendices G and H (References 6.10 and 6.21). There are two remaining OCGS surveillance capsules which will remain in place to serve as backup surveillance material for the BWRVIP program, or as otherwise needed.

Currently, OCGS has made a licensing commitment to replace the existing material surveillance program with the BWRVIP ISP (Reference 6.24) in the license amendment issued by the NRC regarding implementation of the BWRVIP ISP, dated April 27, 2004 (Reference 6.18). The BWRVIP ISP meets the requirements of 10 CFR 50, Appendix H, for Integrated Surveillance Programs, and has been approved by NRC. Under the ISP, there are no further capsules from OCGS to be tested. Representative surveillance capsule materials for the OCGS limiting beltline plate and weld are in the Cooper and Hatch Unit 2 surveillance capsule programs, respectively. The next Cooper surveillance capsule is scheduled to be withdrawn and tested under the ISP in approximately 2029 at 40 EFPY. The next Hatch Unit 2 surveillance capsule is scheduled for withdrawal and testing under the ISP in approximately 2027 at 37 EFPY.