Pressure-Temperature Limits for 19 EFPY
for
Oconee Unit 2
Nuclear Plant
Pressurized Thermal Shock Evaluation
for
Oconee Units 1, 2 & 3

May 1993

Prepared for
Duke Power Company

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1. INTRODUCTION

This report provides the pressure-temperature (P/T) limits for normal operation, both heatup and cooldown, inservice leak and hydrostatic tests, and reactor criticality limits for Oconee Unit 2 at 19 effective full power years (EFPY).

The P/T limits at 21 EFPY for each of the three Oconee Units were provided in Reference 6.

Section 5 contains pressurized thermal shock evaluations for all three Oconee Units.

2. DEVELOPMENT OF TECHNICAL SPECIFICATION PRESSURE-TEMPERATURE LIMITS

The pressure-temperature limits of the reactor coolant pressure boundary (RCPB) of Oconee Unit 2 were established in accordance with the requirements of 10CFR50, Appendix G.[2] The methods and criteria employed to establish operating pressure and temperature limits are described in topical report BAW-10046A.[3] The objective of these limits is to prevent nonductile failure during any normal operating condition, including anticipated operation occurrences and system hydrostatic tests. The loading conditions of interest include the following:

1. Normal operations, including heatup and cooldown.
2. Inservice leak and hydrostatic tests.
3. Reactor core operation.

The major components of the RCPB have been analyzed in accordance with 10CFR50, Appendix G. The closure head region, the reactor vessel (RV) outlet nozzle, and the beltline region have been identified as the only regions of the reactor vessel (and consequently of RCPB) that require the pressure-temperature limits. Since the closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt preload), this region largely controls the pressure-temperature limits for the first several effective full power years. The reactor vessel outlet nozzle also affects the pressure-temperature limit curves for the first several EFPYs. This is due to the high local stresses at the inside corner of the nozzle. After several years of neutron radiation exposure, the vessel beltline region materials will control the pressure-temperature limits of the RCPB due to increased RT_{NDT} value.

The limit curves for Oconee Unit 2 were based on the predicted values of the adjusted reference temperature of the limiting beltline region material at the end of 19 EFPY. The adjusted reference temperatures were calculated by adding the predicted radiation-induced ΔRT_{NDT} to the initial RT_{NDT} plus the margin term. The predicted ΔRT_{NDT} was calculated using the respective vessel's neutron fluence and chemistry. Table 1 summarizes the predicted reactor vessel inside surface peak fluence value at 19 EFPY for Oconee Unit 2. Regulatory Guide 1.99, Rev. 2,[1] was used to predict the radiation-induced RT_{NDT} values as a function of the material's copper and nickel content and neutron fluence. Using these fluence values and the vessel's chemistry, the adjusted RT_{NDT} values of the beltline region at the end of the 19 EFPYs are provided in Table 2. The adjusted RT_{NDT} values are given for the 1/4t and 3/4t vessel wall locations (t = wall thickness). The assumed RT_{NDT} of the closure head region and the outlet nozzle steel forgings was 60°F, in accordance with BAW-10046A.[3]

Using the methodology documented in BAW-10046A, pressure-temperature limits for the closure head region, the outlet nozzle, and the beltline region were determined for the heatup and cooldown rates summarized in Tables 3 and 4. Differential pressure corrections were then applied to the unadjusted P/T limits to account for the pressure differential between the analyzed regions of the reactor vessel and the system pressure sensor on the reactor coolant system. These differential pressure corrections were based on the reactor coolant pump (RCP) operational constraints, also summarized in Tables 3 and 4. The RCP operational restriction, of no more than one pump during normal cooldown at temperature below 200°F, can be modified to allow running a 1-1 RCP combination briefly during cooldown, provided that any one of the following procedures is observed[4]:

a. After terminating the cooldown, begin a soak period. A 1/2 hour soak time should be maintained prior to operating the 1-1 combination. The soak condition should be as near constant temperature as practical, with as near a zero heatup or cooldown rate as possible. Specifically, the 1/2 hour soak time is based on maintaining less than a 10°F/hr rate of temperature change during the soak, i.e. the final temperature should be within approximately 5°F of the initial temperature. Reactor coolant system (RCS) pressure should be maintained below the <u>cooldown</u> P/T limit. This applies to each of the Oconee Units between approximately 160 and 180°F during cooldown.

b. Alternately, maintaining RCS pressure at least 20 psi below the <u>cooldown</u> limit while running the 1-1 combination is an option, requiring no soak time.

At the request of Duke Power Company, instrumentation errors for pressure and temperature were not applied. The maximum allowable pressure as a function of fluid temperature was obtained through a point-by-point comparison of the three limiting regions, corrected for sensor location. The maximum allowable pressure was taken to be the lowest of the three calculated pressures. The resulting corrected data points determine the bounding P/T Technical Specification curves.

Table 1. Fluence at 19 EFPY on Inside Surface (Max Location)[6]

Unit	Fluence (n/cm^2)
Oconee 2	0.598×10^{19}

Table 2. 19 EFPY RT_{NDT}(s)

Unit/Location	1/4T (°F)	3/4T (°F)
Oconee 2	200	147

Table 3. Operational Constraints for Plant Heatup

I. RCS Temperature Constraints	
RC Temperature	Maximum Heatup Rate
T ≤ 280°F	50°F/hr
T > 280°F	100°F/hr
II. RCS Pump Constraints	
RC Temperature	Allowed Pump Combination
T > 250°F	Any
T ≤ 250°F	No more than 1 pump per loop

Table 4. Operational Constraints for Plant Cooldown

I. RCS Temperature Constraints	Maximum Cooldown Rate
T > 280°F	≤ 50°F in any 1/2 hour period
150°F < T ≤ 280°F	≤ 25°F in any 1/2 hour period
T ≤ 150°F	≤ 10°F in any 1 hour period
II. RCS Pump Constraints	
RC Temperature	Allowed Pump Combinations
T > 270°F	Any
200°F < T ≤ 270°F	No more than 1 pump per loop
T ≤ 200°F	No more than 1 pump 1-1 with restriction*

*Brief 1/1 RCP combination operation is permitted if 1/2 hr "soak" time is maintained during cooldown, between approximately 160°F and 180°F.

3. RESULTS

The P/T Technical Specification limits for Oconee Unit 2 through 19 EFPY for normal heatup, normal cooldown, inservice leak and hydrotests, and reactor core operation, as required by 10CFR50, Appendix G are provided in Figures 1 through 3. The limits have invoked additional operational constraints, identified in Tables 3 and 4, that must be maintained for operation.

Protection against nonductile failure is ensured by maintaining the RCS pressure below the upper limits of the P/T limit curves. The acceptable pressure and temperature combinations for RV operation are below and to the right of the limit curve. The RV is not permitted to be critical until the P/T combinations are to the right of the criticality limit curve.

4. REFERENCES

1. U.S. Nuclear Regulatory Commission, Radiation Damage to Reactor Vessel Material, Regulatory Guide 1.99, Revision 2, May 1988.

2. Code of Federal Regulation, Title 10, Part 50, Fracture Toughness Requirements for Light-Water Nuclear Power Reactor, Appendix G, Fracture Toughness Requirements, Federal Register, Vol. 48, No. 194, May 17, 1983.

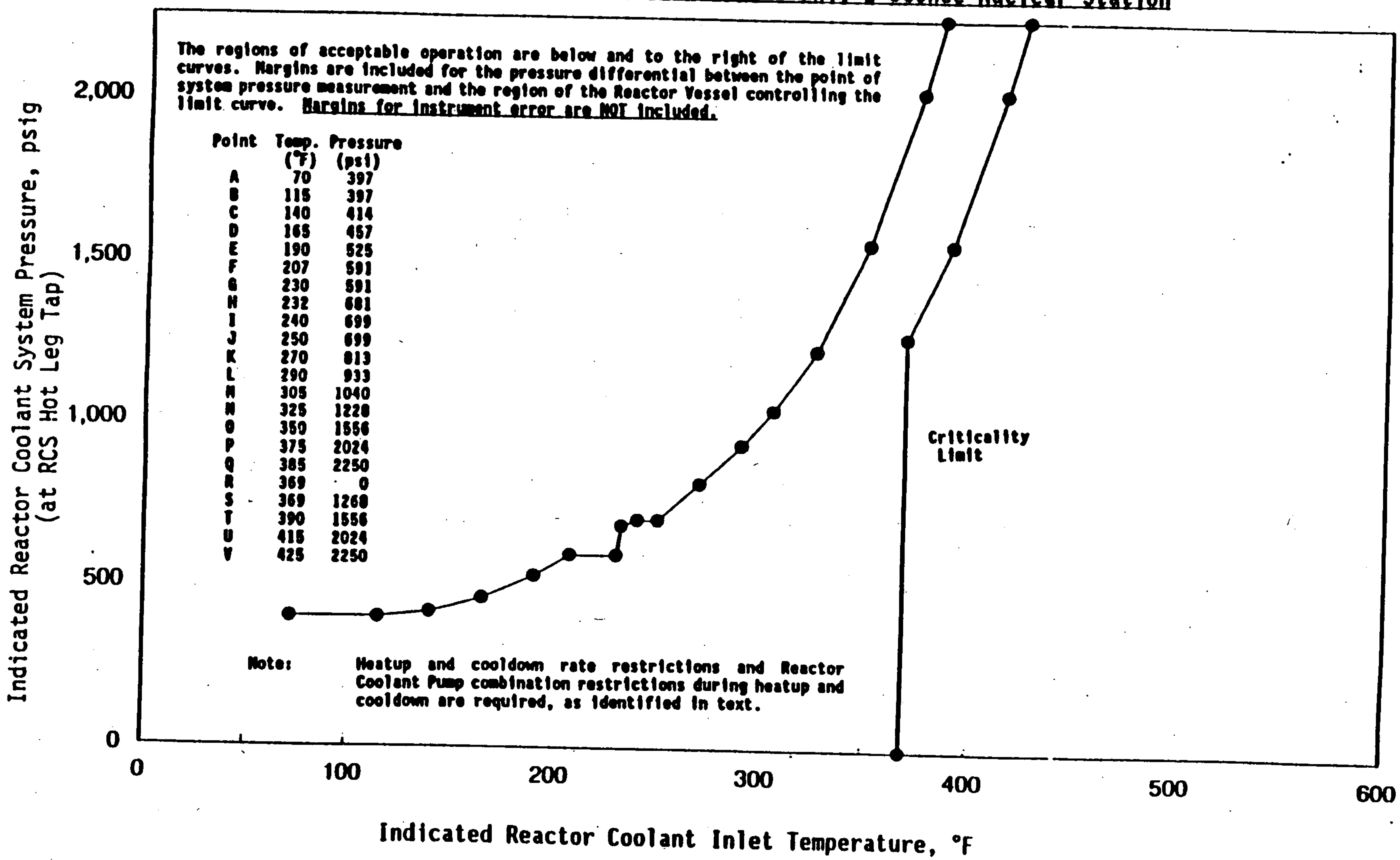
3. B&W Report, BAW-10046A, Rev. 2, Methods of Compliance with Fracture Toughness and Operational Requirements of Appendix G to 10CFR50, Babcock & Wilcox, Lynchburg, Virginia, June 1986.

4. B&W Document 51-1178914-00, "Oconee Cooldown P/T Limits - RCP Options," April 5, 1990.

5. B&W Document 32-1223859-00, "OC-2 P/T Limits at 19 EFPY, April 1993.

6. B&W Document 77-1219637-00, "Pressure-Temperature Limits for 21 EFPY for Oconee Units 1, 2, and 3 Nuclear Plants," February 1993.

Figure 1. Reactor Coolant System Normal Operational Heatup Limitations
Applicable for First 19.0 EFPY - Unit 2 Oconee Nuclear Station



Point	Temp. (°F)	Pressure (psi)
A	70	397
B	115	397
C	140	414
D	165	457
E	190	525
F	207	591
G	230	591
H	232	681
I	240	699
J	250	699
K	270	813
L	290	933
M	305	1040
N	325	1228
O	350	1556
P	375	2024
Q	385	2250
R	369	0
S	369	1268
T	390	1556
U	415	2024
V	425	2250

Figure 2. Reactor Coolant System Normal Operational Cooldown Limitations

Applicable for First 19.0 EFPY - Unit 2 Oconee Nuclear Station

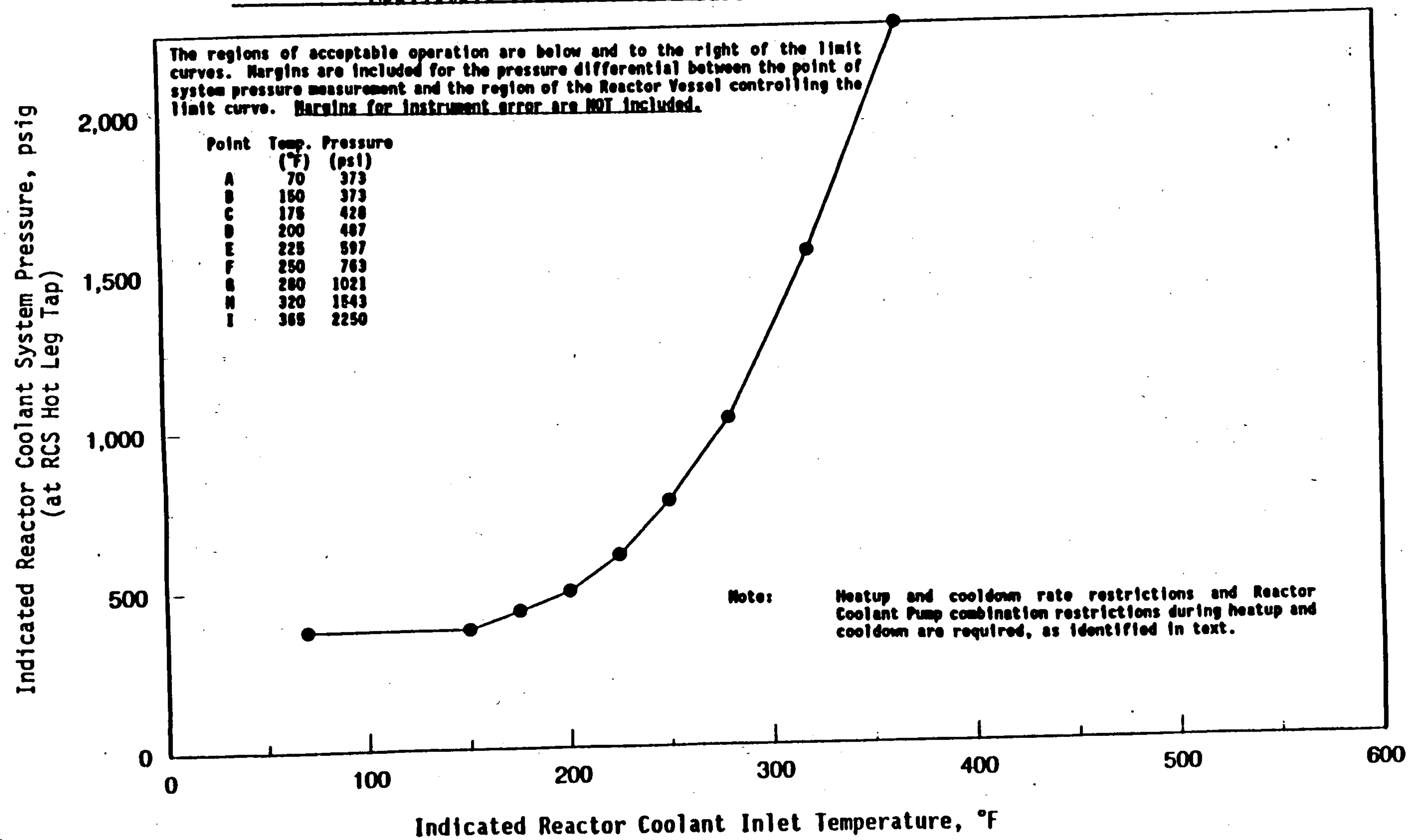
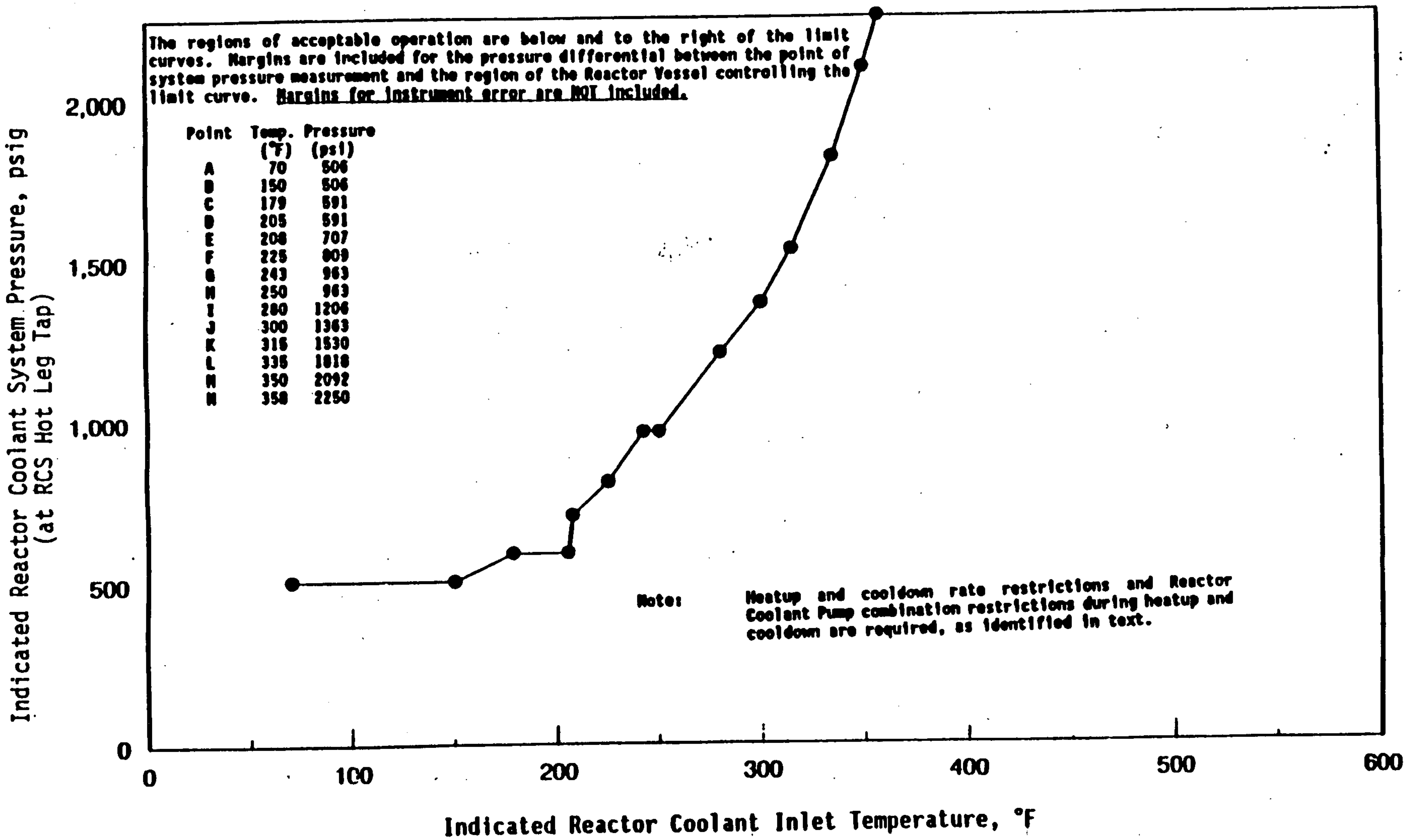


Figure 3. Reactor Coolant System Inservice Leak and Hydrostatic Test Heatup and Cooldown Limitation Applicable for First 19.0 EFPY - Unit 2 Oconee Nuclear Station



5. PRESSURIZED THERMAL SHOCK EVALUATIONS

5.1 Introduction

The Pressurized Thermal Shock (PTS) rule, 10CFR50.61,[1] became effective on July 23, 1985, to (1) establish a screening criterion related to the fracture resistance of pressurized water reactor (PWR) vessels during pressurized thermal shock events; (2) require an analysis and schedule for implementation of flux reduction programs that are reasonably practicable to avoid exceeding the screening criterion; and (3) require detailed safety evaluations to be performed before plant operation beyond the screening criterion will be allowed. The intention of this regulation was to improve safety of the PWR vessels by identifying those corrective actions that may be required to prevent or mitigate potential PTS events. Effective June 14, 1991, the U.S. Nuclear Regulatory Commission (NRC) amended this regulation to change the procedure for calculating the RT_{PTS} value. The amendment updates the calculation procedure and makes it consistent with the calculation procedure for RT_{NDT} described in Regulatory Guide 1.99, Revision 2.[2]

Transients and accidents can be postulated to occur in PWRs resulting in severe overcooling (thermal shock) of the reactor vessel concurrent with high pressure. In these PTS events, rapid cooling of the reactor vessel internal surface causes a temperature distribution across the reactor vessel wall, which produces a thermal stress on the reactor vessel, with a maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress varies with the rate of change of temperature and with time during the transient, and its effect is compounded by coincident pressure stresses.

Severe reactor system overcooling events with pressurization of the reactor vessel (PTS events) are postulated to result from many causes. These include system transients, some of which are initiated by instrumentation and control system malfunctions (including stuck open valves in either the primary or secondary system), and accidents such as small piping breaks, main steam line breaks, and feedwater line breaks resulting in loss-of-coolant accidents. As long as the fracture resistance of the reactor vessel material is relatively high, these events are not expected to cause vessel failure. However, the fracture resistance of the reactor vessel material decreases with integrated exposure to fast neutrons during the life of the plant. The rate of decrease is dependent on the chemical composition of the reactor vessel beltline materials (both base metals and weld metals). If the fracture resistance of the reactor vessel is sufficiently reduced by neutron irradiation, severe PTS events could cause small flaws near the inner surface to propagate into the vessel wall. The assumed initial flaw might be enlarged into a crack through the vessel wall to a sufficient extent which could threaten the vessel integrity and, therefore, compromise core cooling capability.

The designation RT_{PTS} (reference temperature for pressurized thermal shock) is the nil ductility temperature of the material as defined by 10CFR50.61, Paragraph (b)(2), for use as a screening criterion.

On the basis of these studies, the NRC concluded that the PWR reactor pressure vessels with conservatively calculated values of RT_{PTS} less than 270°F for plate, forging, and axial weld materials, and less than 300°F for circumferential weld materials present an acceptably low risk of vessel failure from PTS events.

5.2 Summary

Table 5 provides a summary of the pressurized thermal shock evaluations as required by 10CFR50.61 for the Oconee Units 1, 2, and 3 reactor vessels.

A brief description of the status of each pressure vessel is as follows:

5.2.1 Oconee Unit 1 (Table 6)

Projected values of RT_{PTS} for all materials in the reactor vessel beltline region are below the screening criteria at 32 Effective Full Power Years (End-of-Life).

5.2.2 Oconee Unit 2 (Table 7)

Projected values of RT_{PTS} for all materials in the reactor vessel beltline region are below the screening criteria at 32 Effective Full Power Years (End-of-Life).

5.2.3 Oconee Unit 3 (Table 8)

Projected values of RT_{PTS} for all materials in the reactor vessel beltline region are below the screening criteria at 32 Effective Full Power Years (End-of-Life).

Table 5. Summary of Pressurized Thermal Shock Evaluations for Oconee Units 1, 2, and 3 Reactor Vessels

Plant	RT_{PTS} Value for Controlling Material	Reference
Oconee Unit 1	Base Metal: C2197-2 -- 147 vs. 270	Table 6
	Weld Metal: SA-1073 -- 214 vs. 270	Table 6
Oconee Unit 2	Base Metal: AMX 77 -- 86 vs. 270	Table 7
	Weld Metal: WF-25 -- 284 vs. 300	Table 7
Oconee Unit 3	Base Metal: 4680 -- 204 vs. 270	Table 8
	Weld Metal: WF-200 -- 234 vs. 300	Table 8
	WF-67 -- 234 vs. 300	Table 8

5.3 Basis of Input Data

The pressurized thermal shock regulations require that the data used to perform the specified calculations must be traceable by including the source of all values included in the assessment. The relationship of the material on which any measurements are made to the actual material in the reactor pressure vessel must be described. For the fluence values, the assessment must specify the basis for all projections including the assumptions regarding core loading patterns such as standard versus low-leakage cores.

The following describes the sources for all data used to evaluate the Oconee Units 1, 2, and 3 reactor vessels.

5.3.1 Fluence Estimates

The basis of the fluence estimates for the reactor vessel beltline materials at the inside surface is BAW-2108, Revision 1.[3]

5.3.2 Chemical Compositions

The bases of the chemical composition of the reactor vessel beltline materials are BAW-1820[4] and BAW-2121P.[5]

5.3.3 Material Properties

The basis of the material properties which represent actual measured properties of the reactor vessel beltline region materials is BAW-1820. In cases where the NRC regulations did not provide a generic initial value of RT_{NDT} for either SA-533, Grade B plate, or SA-508, Class 2 forging material, the statistical average value of these materials was calculated using the data base presented in BAW-10046P.[6] These values are as follows:

Plate Material: SA-533, Grade B $RT_{NDT} = +1°F$
Forging Material: SA-508, Class 2 $RT_{NDT} = +3°F$

5.4 Reactor Vessel Pressurized Thermal Shock Calculations

For the purpose of comparison with the PTS criterion, the value of RT_{PTS} for each of the reactor vessel materials must be calculated as described in the following paragraphs. The calculation must be made for each weld and plate, or forging, in the reactor vessel beltline region.

Equation 1: $RT_{PTS} = I + M + \Delta RT_{PTS}$

a. "I" represents the initial reference temperature (RT_{NDT}) of the unirradiated material measured as defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Paragraph NB-2331.[7] Measured values must be used if credible values are available; if not, the following generic mean values must be used: 0°F for welds made with Linde 80 flux, and 56°F for welds made with Linde 0091, 1092, and 124 and ARCOS B-5 weld fluxes.

b. "M" represents the margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and the calculational procedures. In Equation 1, M is 66°F for welds and 48°F for base metal if generic values of "I" are used, and M is 56°F for welds and 34°F for base metal if measured values of "I" are used.

c. ΔRT_{PTS} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

Equation 2: $\Delta RT_{PTS} = (CF) f^{(0.28-0.10 \log f)}$

"CF" (°F) represents the chemistry factor, a function of copper and nickel content, as illustrated in Table 1 and 2 in 10CFR50.61. Linear interpolation is permitted. "Copper-Wt%" and "Nickel-Wt%" are the best estimate values for the material, which will normally be the mean of the measured values for a plate or forging or weld sample made with the weld wire heat number that matches the critical vessel weld.

"f" represents the best estimate neutron fluence, in units of 10^{19} n/cm^2 (E > 1 MeV), at the inside surface of the vessel where the material in question receives the highest fluence for the period of service in question.

Table 6. Evaluation of Oconee Unit 1 Reactor Vessel in Accordance with Pressurized Thermal Shock Criterion

Material Description			Chemical Composition w/o[a,b]						
Reactor Vessel Beltline Region Location	Heat Number	Type	Copper	Nickel	Initial RT_{NDT}	Margin	IS Fluence[e] n/cm² 32 EFPY	Calculated RT_{PTS}	PTS Screening Criteria
Lower Nozzle Belt	AHR 54	SA-508, Cl.2	0.16	0.65	+3[c]	48	1.18E+18	105	270
Intermediate Shell	C2197-2	SA-302B, Mod.	0.15	0.50	+1[c]	48	7.96E+18	147	270
Upper Shell	C3278-1	SA-302B, Mod.	0.12	0.60	+1[c]	48	9.04E+18	130	270
Upper Shell	C3265-1	SA-302B, Mod.	0.10	0.50	+1[c]	48	9.04E+18	112	270
Lower Shell	C2800-1	SA-302B, Mod.	0.11	0.63	+1[c]	48	8.68E+18	120	270
Lower Shell	C2800-2	SA-302B, Mod.	0.11	0.63	+1[c]	48	8.68E+18	120	270
Interm. Circum. Weld (100%)	SA-1135	ASA/Linde 80	0.25	0.54	0[d]	66	1.18E+18	142	300
Upper Circum. Weld (I.D. 61%)	SA-1229	ASA/Linde 80	0.26	0.61	0[d]	66	7.96E+18	236	300
Upper Circum. Weld (O.D. 39%)	WF-25	ASA/Linde 80	0.35	0.68	0[d]	66	N/A	N/A	N/A
Middle Circum. Weld (100%)	SA-1585	ASA/Linde 80	0.21	0.59	0[d]	66	8.68E+18	222	300
Lower Circum. Weld (100%)	WF-9	ASA/Linde 80	0.21	0.59	0[d]	66	5.06E+16	77	300
Interm. Longit. Weld (100%)	SA-1073	ASA/Linde 80	0.21	0.64	0[d]	66	6.28E+18	214	270
Upper Longit. Weld (100%)	SA-1493	ASA/Linde 80	0.20	0.55	0[d]	66	7.23E+18	204	270
Lower Longit. Weld (100%)	SA-1430	ASA/Linde 80	0.20	0.55	0[d]	66	7.29E+18	205	270
Lower Longit. Weld (100%)	SA-1426	ASA/Linde 80	0.20	0.55	0[d]	66	7.29E+18	205	270

[a] BAW-1820.[4]

[b] BAW-2121P.[5]

[c] BAW-10046P.[6]

[d] 10CFR50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock.[1]

[e] BAW-2108, Revision 1.[3]

Table 7. Evaluation of Oconee Unit 2 Reactor Vessel in Accordance with Pressurized Thermal Shock Criterion

Material Description			Chemical Composition w/o[a,b]				IS Fluence[e] n/cm²		
Reactor Vessel Beltline Region Location	Heat Number	Type	Copper	Nickel	Initial RT_{NDT}	Margin	32 EFPY	Calculated RT_{PTS}	PTS Screening Criteria
Lower Nozzle Belt	AMX 77	SA-508, Cl.2	0.06	0.76	+3[c]	48	8.42E+18	86	270
Upper Shell	AAW-163	SA-508, Cl.2	0.04	0.75	+20[a]	34	9.57E+18	80	270
Lower Shell	AWG-164	SA-508, Cl.2	0.02	0.80	+20[a]	34	9.19E+18	74	270
Upper Circum. Weld (100%)	WF-154	ASA/Linde 80	0.31	0.59	0[d]	66	8.42E+18	253	300
Middle Circum. Weld (100%)	WF-25	ASA/Linde 80	0.35	0.68	0[d]	66	9.19E+18	284	300
Lower Circum. Weld (100%)	WF-112	ASA/Linde 80	0.31	0.59	0[d]	66	5.36E+16	80	300

[a] BAW-1820.[4]

[b] BAW-2121P.[5]

[c] BAW-10046P.[6]

[d] 10CFR50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock.[1]

[e] BAW-2108, Revision 1.[3]

Table 8. Evaluation of Oconee Unit 3 Reactor Vessel in Accordance with Pressurized Thermal Shock Criterion

Material Description			Chemical Composition w/o[a,b]						
Reactor Vessel Beltline Region Location	Heat Number	Type	Copper	Nickel	Initial RT_{NDT}	Margin	IS Fluence[f] n/cm^2 32 EFPY	PTS Calculated RT_{PTS}	Screening Criteria
Lower Nozzle Belt	4680	SA-508, Cl.2	0.20[c]	0.91	+3[d]	48	8.26E+18	204	270
Upper Shell	AWS-192	SA-508, Cl.2	0.01	0.73	+20[e]	34	9.39E+18	74	270
Lower Shell	ANK-191	SA-508, Cl.2	0.02	0.76	+20[e]	34	9.01E+18	73	270
Upper Circum. Weld (100%)	WF-200	ASA/Linde 80	0.24	0.63	0[e]	66	8.26E+18	234	300
Middle Circum. Weld (I.D. 75%)	WF-67	ASA/Linde 80	0.24	0.60	0[e]	66	9.01E+18	234	300
Middle Circum. Weld (O.D. 25%)	WF-70	ASA/Linde 80	0.35	0.59	0[e]	66	N/A	N/A	300
Lower Circum. Weld (100%)	WF-169-1	ASA/Linde 80	0.18	0.63	0[e]	66	5.26E+16	77	300

[a] BAW-1820.[4]

[b] BAW-2121P.[5]

[c] Conservative estimated value based on generic SA-508, Class 2 initial RT_{NDT} data.

[d] BAW-10046P.[6]

[e] 10CFR50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock.[1]

[f] BAW-2108, Revision 1.[3]

5.5 References

1. Code of Federal Regulations, Title 10, Part 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.

2. U.S. Nuclear Regulatory Commission, Radiation Damage to Reactor Vessel Material, Regulatory Guide 1.99, Revision 2, May 1988.

3. L. Petrusha, Fluence Tracking System, BAW-2108, Revision 1, B&W Nuclear Service Company, Lynchburg, Virginia, May 1992.

4. J. D. Aadland, Babcock & Wilcox Owners' Group 177-Fuel Assembly Reactor Vessel and Surveillance Program Materials Information, BAW-1820, Babcock & Wilcox, Lynchburg, Virginia, December 1984.

5. L. B. Gross, Chemical Composition of B&W Fabricated Reactor Vessel Beltline Welds, BAW-2121P, B&W Nuclear Service Company, Lynchburg, Virginia, April 1992.

6. H. S. Palme, H. W. Behnke, and W. J. Keyworth, Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G, BAW-10046P, Babcock & Wilcox, Lynchburg, Virginia, March 1976.

7. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III, Nuclear Power Plant Components, Division 1, Subsection NB, Class 1 Components, (updated frequently).