Kim Maza
Site Vice President
Harris Nuclear Plant
5413 Shearon Harris Road
New Hill, NC 27562

10 CFR 50.90

May 12, 2020
RA-19-0359

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

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400/Renewed License No. NPF-63

Subject: License Amendment Request to Correct Non-Conservative Technical Specification 3/4.4.9, "Pressure/Temperature Limits – Reactor Coolant System"

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy) hereby requests an amendment to the Shearon Harris Nuclear Power Plant, Unit 1 (HNP) Renewed Facility Operating License. The proposed change revises HNP Technical Specification (TS) 3/4.4.9, "Pressure/Temperature Limits – Reactor Coolant System," to reflect an update to the pressure and temperature limit curves in Figures 3.4-2 (Reactor Coolant System Cooldown Limitations) and 3.4-3 (Reactor Coolant System Heatup Limitations). The proposed change also reflects that the revised HNP pressure and temperature limit curves in TS Figures 3.4-2 and 3.4-3 will be applicable until 55 effective full power years (EFPY). Lastly, the proposed change revises TS Figure 3.4-4 (Maximum Allowed PORV Setpoint for the Low Temperature Overpressure Protection System) to reflect that the setpoint values are based on 55 EFPY reactor vessel data.

The proposed change is necessary because Duke Energy identified after the removal and examination of reactor pressure vessel surveillance capsule Z at the end of cycle 21 that the existing HNP TS Figures 3.4-2 and 3.4-3 are non-conservative. Current plant operations are administratively controlled consistent with Nuclear Regulatory Commission (NRC) Administrative Letter (AL) 98-10, "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety."

The Enclosure provides a description and assessment of the proposed change. Attachment 1 provides the existing HNP TS pages marked to show the proposed change. Attachment 2 provides existing TS Bases pages marked to show the proposed change for information only.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that the proposed change involves no significant hazards consideration. The basis for this determination is included in the Enclosure.

Duke Energy requests approval of the proposed amendment within one year of the date this submittal is accepted by the NRC staff for review. Once approved, Duke Energy will implement the license amendment within 120 days.

There are no regulatory commitments contained in this submittal.

In accordance with 10 CFR 50.91, Duke Energy is notifying the State of North Carolina of this license amendment request by transmitting a copy of this letter and enclosure to the designated State Official.

If there are any questions or if additional information is needed, please contact Mr. Art Zaremba, Manager – Nuclear Fleet Licensing at 980-373-2062 or Arthur.Zaremba@duke-energy.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 12, 2020.

Sincerely,

Kim E. Maza

Kim Maza
Site Vice President
Harris Nuclear Plant

Enclosure: Description and Assessment of the Proposed Change

Attachments:

1. Technical Specifications Markup
2. Technical Specifications Bases Markup

cc (with Enclosure/Attachments):

L. Dudes, NRC Regional Administrator, Region II
J. Zeiler, NRC Senior Resident Inspector, HNP
T. Hood, NRC Project Manager, HNP
W. L. Cox, III, Section Chief N.C. DHSR

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ENCLOSURE

Description and Assessment of the Proposed Change

Subject: License Amendment Request to Correct Non-Conservative Technical Specification 3/4.4.9, “Pressure/Temperature Limits – Reactor Coolant System”

ATTACHMENTS:

1. Technical Specifications Markup
2. Technical Specifications Bases Markup

1. SUMMARY DESCRIPTION

Duke Energy Progress, LLC (Duke Energy) hereby requests an amendment to the Shearon Harris Nuclear Power Plant, Unit 1 (HNP) Renewed Facility Operating License. The proposed change revises HNP Technical Specification (TS) 3/4.4.9, "Pressure/Temperature Limits – Reactor Coolant System" to reflect an update to the pressure and temperature (P/T) limit curves in Figures 3.4-2 (Reactor Coolant System Cooldown Limitations) and 3.4-3 (Reactor Coolant System Heatup Limitations). The proposed change also reflects that the revised HNP P/T limit curves in TS Figures 3.4-2 and 3.4-3, as well as the existing power-operated relief valve (PORV) setpoints in Figure 3.4-4 (Maximum Allowed PORV Setpoint for the Low Temperature Overpressure Protection System), will be applicable until 55 effective full power years (EFPY). There are no changes being proposed to the PORV setpoints in Figure 3.4-4.

2. DETAILED DESCRIPTION

2.1 System Design and Operation

All components of the HNP Reactor Coolant System (RCS) are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients and reactor trips. HNP is required to limit the pressure and temperature changes during RCS heatup and cooldown within the design assumptions and the stress limits for cyclic operation.

The HNP TS contain P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing and data for the maximum rate of change of reactor coolant temperature. Each P/T limit curve defines an acceptable region for normal operation. The typical use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

Operating limits are established that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB).

Both the HNP Updated Final Safety Analysis Report Section 5.3.2 and the TS 3/4.4.9 Bases provide additional details regarding the methodology that was used to develop the existing P/T limit curves that are contained in the HNP TS.

2.2 Current Technical Specifications Requirements

HNP Limiting Condition for Operation (LCO) 3.4.9.1 (Applicability: Modes 1, 2 and 3) requires that the reactor coolant temperature and pressure (except for the pressurizer) be maintained in accordance with Figures 3.4-2 and 3.4-3 (P/T curves). LCO 3.4.9.2 (Applicability: Modes 4, 5 and 6 with reactor vessel head on) also requires that the reactor coolant temperature and pressure (except for the pressurizer) be maintained in accordance with Figures 3.4-2 and 3.4-3. The maximum heatup and cooldown rates are specified in TS Table 4.4-6 for LCO 3.4.9.2.

LCO 3.4.9.1 and 3.4.9.2 limits apply to all components of the RCS, except the pressurizer, and define allowable operating regions and permit many operating cycles while providing a wide margin to non-ductile failure. Violating either LCO 3.4.9.1 or 3.4.9.2 limits would result in

placing the reactor vessel outside of the bounds of the stress analyses and could increase stresses in other RCPB components.

The existing HNP TS P/T limits curves in Figures 3.4-2 and 3.4-3 were approved by the issuance of Amendment No. 100 on July 28, 2000 (Reference 1).

TS Figure 3.4-4 provides the PORV setpoints for low temperature overpressure protection (LTOP) and the existing values are based on 36 EFPY reactor vessel data.

2.3 Reason for the Proposed Change

By letter dated October 23, 2019 (Reference 2), Duke Energy submitted the summary technical report for the reactor pressure vessel (RPV) surveillance program (Framatome ANP-3798NP Revision 0) in accordance with 10 CFR 50, Appendix H requirements and communicated that a change to the HNP TS P/T limit curves would be required. The Framatome analysis of the surveillance capsule revealed that existing HNP TS Figures 3.4-2 and 3.4-3 are non-conservative. Specifically, the analysis indicates a shift in the Adjusted Reference Temperature (ART) of 14°F for the 1/4T location and 11°F for the 3/4T location, which has the effect of shifting the TS heatup and cooldown curves to the right.

Although the operational curves used by the main control room to operate the plant are conservative with respect to the P/T limit curves developed as part of the capsule Z testing project and thus remain valid, the TS P/T limit curves (i.e., Figures 3.4-2 and 3.4-3) are non-conservative and require a revision to satisfy regulatory requirements.

2.4 Description of the Proposed Change

TS Figure 3.4-2, “Reactor Coolant System Cooldown Limitations – Applicable to Up to 36 EFPY,” is revised as follows:

- The existing RCS cooldown limitations curves are superseded entirely by new curves applicable up to 55 EFPY.

- The cooldown limitations curve for a rate of “100˚F/HR” is removed.

- The “RT_{NDT} [reference nil-ductility temperature] at 1/4 T” value of “191˚F” is revised to “212˚F” in the “Material Property Bases”.

- The “RT_{NDT} at 3/4 T” value of “179˚F” is revised to “198˚F” in the “Material Property Bases”.

- The title of Figure 3.4-2 is revised to state “Reactor Coolant System Cooldown Limitations – Applicable Up to 55 EFPY”.

TS Figure 3.4-3, “Reactor Coolant System Heatup Limitations – Applicable Up to 36 EFPY,” is revised as follows:

- The existing RCS heatup limitations curves are superseded entirely by new curves applicable up to 55 EFPY.
- The “RT_{NDT} at 1/4 T” value of “191˚F” is revised to “212˚F” in the “Material Property Bases”.
- The “RT_{NDT} at 3/4 T” value of “179˚F” is revised to “198˚F” in the “Material Property Bases”.
- The title of Figure 3.4-3 is revised to state “Reactor Coolant System Heatup Limitations – Applicable Up to 55 EFPY”.

TS Figure 3.4-4, “Maximum Allowed PORV Setpoint for the Low Temperature Overpressure Protection System,” is revised as follows:

- The note “*VALUES BASED ON 36 EFPY REACTOR VESSEL DATA” is revised to state “*VALUES BASED ON 55 EFPY REACTOR VESSEL DATA”.

The TS markup provided in Attachment 1 reflects the proposed changes described above.

The proposed changes are supported by changes to the TS Bases. In addition to reflecting the proposed changes to the TS, the TS 3/4.4.9 Bases are revised for clarity and consistency. The regulation at 10 CFR 50.36 states “A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications.” Changes to the TS Bases will be made in accordance with the Technical Specifications Bases Control Program following approval of the requested amendment. The proposed TS Bases changes in Attachment 2 are consistent with the proposed TS changes and provide the purpose for each requirement in the specification consistent with the Commission’s Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 2, 1993 (58 FR 39132). Therefore, the HNP TS Bases changes are provided for information and approval of the TS Bases is not requested.

3. TECHNICAL EVALUATION

Pressure/Temperature Limit Curves Development

The proposed bounding TS cooldown (Figure 3.4-2) and heatup (Figure 3.4-3) limitation curves presented in Attachment 1 are derived from uncorrected P/T limits calculated for the HNP reactor vessel through 60 years of operation (55 EFPY). For various heatup and cooldown rates (°F/hr), the P/T limits were calculated for the reactor vessel beltline shell, nozzle and closure head locations for normal heatup and normal cooldown conditions as well as ISLH test conditions. P/T limits were calculated in accordance with the requirements of the 2007 Edition with 2008 Addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Appendix G, which permits the use of the K_{Ic} fracture

curve. The requirements of 10 CFR 50, Appendix G ("Fracture Toughness Requirements") were also complied with in calculating the P/T limit curves through 55 EFPY.

Heatup and cooldown limit curves were calculated using the most limiting value of RT_{NDT} corresponding to the limiting material in the beltline region of the RPV. Limiting RT_{NDT} is also referred to as limiting ART. Per HNP TS Figures 3.4-2 and 3.4-3, the "controlling material" (i.e., the limiting material) is Plate B4197-2, which is the Intermediate Shell Plate in the beltline region of the RPV.

The materials outside of the traditional beltline region which are expected to receive fluence values greater than $1x10^{17}$ n/cm^2 were evaluated. The evaluation found 12 reactor vessel locations outside the traditional beltline region with 55 EFPY fluence values greater than $1x10^{17}$ n/cm^2. The locations were:

1. Upper to Intermediate Shell Circumferential Weld AC,
2. Upper Shell,
3. Inlet Nozzle Weld 15-A, 15-B, 15-C,
4. Inlet Nozzle,
5. Outlet Nozzle Weld 16-A, 16-B, 16-C,
6. Outlet Nozzle,
7. Upper Shell Longitudinal Welds BE/BF,
8. Lower to Bottom Head Circumferential Weld,
9. Torus Shell,
10. Torus Meridional Welds, CA, CB, CC, CD, CE, CF,
11. Torus to Bottom Head Dome Weld AF,
12. Bottom Head Dome.

The materials at these twelve locations were analyzed for ART and were found not to be limiting. The controlling material for this analysis remains the intermediate shell plate, heat number B4197-2, which is a location inside the traditional beltline region. Thus, limiting ART values for the 1/4T and 3/4T locations of the reactor vessel beltline wall were obtained for Intermediate Shell Plate B4197-2 with fluence projections up to 55 EFPY and are reported in Table 1 below.

The Regulatory Guide (RG) 1.99, Revision 2 methodology was used along with fast neutron fluence estimates on the reactor vessel to calculate ART values in Table 1. A NRC-approved Framatome calculation-based fluence methodology (BAW-2241P-A) that is in accordance with the requirements of RG 1.190 was used to predict the fluence in the HNP reactor vessel. The cumulative fast (E > 1.0 MeV) fluence for several locations of interest at 55 EFPY at the wetted surface (i.e., inner clad surface) are provided in Table 2. The cumulative fast (E > 1.0 MeV) fast fluence for several locations of interest at 55 EFPY at the inner RPV surface (i.e., clad/vessel interface) are provided in Table 3.

Table 1: Limiting ART Values for HNP Reactor Vessel Components at 55 EFPY

Vessel Component	***Material Identification***	***Wall Location***	***Limiting RT_{NDT}***
Beltline Region (7.75") Base Metal/Axial Flaw	Intermediate Shell Plate Heat B4197-2	1/4T	211.1°F (rounded to 212°F)
		3/4T	196.9°F (rounded to 198°F)
Beltline Region (7.75") Circumferential Weld	Intermediate Shell to Lower Shell Weld AB (Wire Heat 5P6771)	1/4T	134.9°F (bounded by base metal axial flaw)
		3/4T	118.3°F (bounded by base metal axial flaw)
Nozzle Shell Region (9.25") Base Metal/Axial Flaw	Upper Shell Plate Heat C0123-1	1/4T	120.0°F (rounded to 121°F)
		3/4T	93.5°F (rounded to 94°F)

Table 2: Fast Neutron Fluence (E > 1 MeV) for the Reactor Vessel “Wetted” Inside Surface

Description	***55 EFPY Fluence (n/cm^2)***
Forgings / Plates (original 40-year beltline)	
Intermediate Shell Plates (ISP)	6.97E+19
Lower Shell Plates (LSP)	6.79E+19
Wetted Surface Max	**6.97E+19**
Welds (original 40-year beltline)	
Upper Shell Plates (USP) to ISP Circular Welds (AC)	3.49E+18
ISP to LSP Circular Weld (AB)	6.77E+19
ISP Longitudinal Weld (BC/BD)	2.60E+19
LSP Longitudinal Weld (BA/BB)	2.53E+19
Extended Beltline	
USP	3.04E+18
USP Longitudinal Weld (BE/BF)	3.49E+18
Inlet Nozzle Lower Weld (15-A, 15-B, 15-C)	3.87E+17
Outlet Nozzle Lower Weld	1.86E+17
LSP to Bottom Head Circular Weld	1.65E+18

Table 3: Fast Neutron Fluence (E > 1 MeV) for the Reactor Vessel Clad-Base Metal Interface	
Description	***55 EFPY Fluence (n/cm²)***
Forgings / Plates (original 40-year beltline)	
Intermediate Shell Plates (ISP)	6.87E+19
Lower Shell Plates (LSP)	6.70E+19
Inner RPV Surface Max	**6.87E+19**
Welds (original 40-year beltline)	
Upper Shell Plates (USP) to ISP Circular Welds (AC)	3.46E+18
ISP to LSP Circular Weld (AB)	6.68E+19
ISP Longitudinal Weld (BC/BD)	2.57E+19
LSP Longitudinal Weld (BA/BB)	2.51E+19
Extended Beltline	
USP	3.00E+18
USP Longitudinal Weld (BE/BF)	3.46E+18
Inlet Nozzle Lower Weld (15-A, 15-B, 15-C)	3.83E+17
Outlet Nozzle Lower Weld	1.84E+17
LSP to Bottom Head Circular Weld	1.65E+18
1/4T at Maximum Peak Location	4.04E+19
3/4T at Maximum Peak Location	9.99E+18

The proposed TS P/T limit curves for HNP provided in Attachment 1 (i.e., the proposed change) were drawn using data points from a Framatome technical report prepared for HNP. The data points for the 55 EFPY P/T limit curves are provided in Tables 4, 5 and 6 below. The 55 EFPY P/T limit curves are based on the limiting beltline material ART values, which are affected by both the fluence and the initial material properties of that material. The P/T limits are not adjusted for instrument error.

Table 4: Uncorrected Heatup P/T Limits 55 EFPY (pressure in psig)									
	Heatup Rates, °F/hr								
Temp, °F	***0***	***5***	***10***	***15***	***20***	***30***	***40***	***50***	***100***
75	621	621	621	621	621	621	621	621	554
80	621	621	621	621	621	621	621	621	554
85	621	621	621	621	621	621	621	621	554
90	621	621	621	621	621	621	621	621	554
95	621	621	621	621	621	621	621	621	554
100	621	621	621	621	621	621	621	621	554
105	621	621	621	621	621	621	621	621	554
110	621	621	621	621	621	621	621	621	554
115	621	621	621	621	621	621	621	621	554
120	621	621	621	621	621	621	621	621	554
125	704	702	700	699	693	666	643	621	554
130	711	709	707	706	701	673	648	621	554
135	719	717	715	713	710	680	654	631	554
140	728	726	723	721	719	689	661	636	554

Table 4: Uncorrected Heatup P/T Limits 55 EFPY (pressure in psig)									
	Heatup Rates, °F/hr								
Temp, °F	***0***	***5***	***10***	***15***	***20***	***30***	***40***	***50***	***100***
145	738	735	733	730	728	699	669	643	554
150	749	746	743	740	738	709	678	650	554
155	761	758	755	752	749	721	689	659	555
160	774	771	767	764	761	734	700	669	557
165	789	785	781	778	774	748	712	681	560
170	805	801	797	793	789	765	727	692	565
175	823	818	814	809	805	783	743	706	570
180	843	838	833	828	823	803	761	722	577
185	865	859	853	848	843	825	781	740	583
190	889	883	877	870	864	849	803	760	593
195	916	909	902	895	889	876	827	781	604
200	946	938	930	923	915	901	854	805	617
205	979	970	961	953	945	929	883	832	632
210	1015	1005	996	986	977	960	916	861	648
215	1055	1044	1033	1023	1013	994	953	894	667
220	1099	1087	1075	1064	1053	1032	993	931	688
225	1148	1135	1122	1109	1097	1074	1037	970	711
230	1202	1187	1173	1159	1146	1120	1085	1014	736
235	1262	1245	1229	1214	1199	1171	1139	1063	765
240	1328	1310	1292	1275	1258	1227	1197	1116	798
245	1401	1380	1361	1342	1324	1289	1256	1175	833
250	1481	1459	1437	1416	1396	1357	1321	1240	871
255	1570	1545	1521	1498	1476	1433	1392	1312	914
260	1669	1641	1614	1589	1564	1516	1472	1391	961
265	1777	1747	1717	1689	1661	1608	1559	1478	1013
270	1898	1864	1831	1799	1769	1710	1655	1575	1071
275	2031	1993	1957	1922	1888	1823	1762	1681	1134
280	2177	2136	2096	2057	2019	1947	1880	1798	1204
285	2340	2294	2249	2206	2164	2084	2010	1926	1281
290	2519	2468	2419	2371	2324	2236	2153	2069	1367
295	2717	2661	2606	2553	2502	2404	2312	2226	1461
300	2858	2858	2813	2754	2697	2589	2487	2392	1564
305	2858	2858	2858	2858	2858	2793	2680	2575	1677
310	2858	2858	2858	2858	2858	2858	2858	2777	1802
315	2858	2858	2858	2858	2858	2858	2858	2858	1940
320	2858	2858	2858	2858	2858	2858	2858	2858	2092
325	2858	2858	2858	2858	2858	2858	2858	2858	2260
330	2858	2858	2858	2858	2858	2858	2858	2858	2445
335	2858	2858	2858	2858	2858	2858	2858	2858	2649
340	2858	2858	2858	2858	2858	2858	2858	2858	2858
345	2858	2858	2858	2858	2858	2858	2858	2858	2858
350	2858	2858	2858	2858	2858	2858	2858	2858	2858

Table 5: Uncorrected Cooldown P/T Limits 55 EFPY (pressure in psig)									
	Cooldown Rates, °F/hr								
Temp, °F	***0***	***5***	***10***	***15***	***20***	***30***	***40***	***50***	***100***
75	621	621	621	621	609	584	558	532	399
80	621	621	621	621	612	587	561	535	402
85	621	621	621	621	615	590	564	538	406
90	621	621	621	621	619	593	568	542	411
95	621	621	621	621	621	597	572	546	416
100	621	621	621	621	621	602	576	551	422
105	621	621	621	621	621	606	581	556	428
110	621	621	621	621	621	612	587	562	436
115	621	621	621	621	621	618	593	569	440
120	697	685	673	661	649	625	601	576	450
125	704	692	680	668	656	632	608	584	460
130	711	699	688	676	664	641	617	593	472
135	719	708	696	684	673	650	627	602	485
140	728	717	705	694	683	659	636	613	500
145	738	727	716	704	693	670	648	625	517
150	749	738	727	716	705	683	661	639	535
155	761	750	739	729	718	697	676	655	555
160	774	764	753	743	733	712	692	672	577
165	789	779	769	759	749	729	710	691	605
170	805	795	786	776	767	748	730	712	632
175	823	814	805	796	787	769	752	735	662
180	843	834	826	817	809	792	776	761	696
185	865	857	849	841	833	818	805	791	733
190	889	882	874	867	860	847	834	822	775
195	916	909	903	897	890	878	867	857	821
200	946	940	934	928	923	913	904	895	872
205	979	973	969	964	959	951	944	938	928
210	1015	1011	1007	1003	1000	994	989	985	992
215	1055	1052	1049	1046	1044	1040	1038	1037	1055
220	1099	1097	1095	1094	1093	1092	1093	1095	1099
225	1148	1147	1147	1147	1147	1148	1148	1148	1148
230	1202	1202	1202	1202	1202	1202	1202	1202	1202
235	1262	1262	1262	1262	1262	1262	1262	1262	1262
240	1328	1328	1328	1328	1328	1328	1328	1328	1328
245	1401	1401	1401	1401	1401	1401	1401	1401	1401
250	1481	1481	1481	1481	1481	1481	1481	1481	1481
255	1570	1570	1570	1570	1570	1570	1570	1570	1570
260	1669	1669	1669	1669	1669	1669	1669	1669	1669
265	1777	1777	1777	1777	1777	1777	1777	1777	1777
270	1898	1898	1898	1898	1898	1898	1898	1898	1898
275	2031	2031	2031	2031	2031	2031	2031	2031	2031
280	2177	2177	2177	2177	2177	2177	2177	2177	2177
285	2340	2340	2340	2340	2340	2340	2340	2340	2340

Table 5: Uncorrected Cooldown P/T Limits 55 EFPY (pressure in psig)									
	Cooldown Rates, °F/hr								
Temp, °F	***0***	***5***	***10***	***15***	***20***	***30***	***40***	***50***	***100***
290	2519	2519	2519	2519	2519	2519	2519	2519	2519
295	2717	2717	2717	2717	2717	2717	2717	2711	2551
300	2858	2844	2830	2815	2801	2771	2741	2711	2551
305	2858	2844	2830	2815	2800	2771	2741	2710	2551
310	2858	2844	2830	2815	2800	2771	2741	2710	2550
315	2858	2844	2829	2815	2800	2770	2740	2709	2550
320	2858	2844	2829	2815	2800	2770	2740	2709	2550
325	2858	2844	2829	2815	2800	2770	2739	2706	2550
330	2858	2844	2829	2815	2800	2770	2737	2705	2551
335	2858	2844	2829	2815	2800	2768	2737	2705	2551
340	2858	2844	2829	2815	2799	2768	2736	2705	2555
345	2858	2844	2829	2814	2798	2768	2736	2704	2555
350	2858	2844	2829	2814	2798	2767	2736	2704	2555

Table 6: Uncorrected ISLH 10 °F/HR for 55 EFPY (pressures in psig)		
Temp, °F	***Heatup***	***Cooldown***
75	621	621
80	621	621
85	621	621
90	621	621
95	621	621
100	621	621
105	621	621
110	621	621
115	621	621
120	621	621
125	934	907
130	943	917
135	953	928
140	965	940
145	977	954
150	991	969
155	1006	986
160	1023	1004
165	1042	1025
170	1062	1048
175	1085	1073
180	1110	1101
185	1138	1132
190	1169	1166
195	1203	1203
200	1240	1245
205	1282	1291

Table 6: Uncorrected ISLH 10°F/HR for 55 EFPY (pressures in psig)		
Temp, °F	***Heatup***	***Cooldown***
210	1327	1342
215	1378	1398
220	1434	1461
225	1496	1529
230	1564	1603
235	1639	1682
240	1722	1770
245	1815	1867
250	1916	1975
255	2029	2094
260	2153	2225
265	2290	2370
270	2441	2530
275	2609	2707
280	2794	2903
285	2999	3120
290	3225	3359
295	3475	3623
300	3751	3773
305	3811	3773
310	3811	3773
315	3811	3773
320	3811	3773
325	3811	3772
330	3811	3772
335	3811	3772
340	3811	3772
345	3811	3772
350	3811	3772

Note that the existing TS Figure 3.4-2 (Cooldown Limitations) contains a P/T limit curve for a cooldown rate of 100°F/HR. The proposed change (i.e., revised Figure 3.4-2) does not contain a curve for the cooldown rate of 100°F/HR because when the RCS fluid temperature is 290°F or greater, the allowable pressure is above the reactor vessel design pressure of 2485 psig.

Regulatory Issue Summary (RIS) 2014-11 Considerations

10 CFR 50, Appendix G requires that P/T limits be developed to bound all ferritic materials in the RPV. RIS 2014-11, “Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components” (Reference 3), clarifies that P/T limit calculations for ferritic RPV materials other than those materials with the highest reference temperature may define P/T curves that are more limiting because the consideration of stress levels from structural discontinuities (such as RPV inlet and outlet nozzles) may produce a lower allowable pressure.

For HNP, thickness transitions (i.e., discontinuities) exist at the lower shell to Torus shell weld (lower transition) and in the upper shell plate to intermediate shell weld (upper transition). P/T limits, which include the impact of the structural discontinuities in the HNP reactor vessel shell upper and lower transition regions at 55 EFPY, were developed in accordance with the guidelines of the ASME Code, Section XI, Appendix G, 2007 Edition including Addenda through 2008. These P/T limits for the thickness transition sections of the reactor vessel were then compared against the P/T limits calculated for the reactor vessel beltline shell, nozzle and closure head locations to verify that they were not more limiting than the traditional beltline P/T limits. Duke Energy determined that the P/T limit curves for the transition regions are not more limiting than the traditional beltline P/T limits for any heatup and cooldown rates. Therefore, the transition region P/T limits are bounded by the P/T limits in Tables 4, 5 and 6 above.

Low Temperature Overpressure (LTOP) Setpoint Considerations

Duke Energy has performed calculations to demonstrate that for the PORV setpoints in TS Figure 3.4-4, the predicted system pressure overshoots resulting from mass or heat input transients will not exceed the proposed HNP P/T limits presented in Tables 4 and 5 above and reflected in the Attachment 1 figures.

The LTOP analysis calculated a peak pressure of 559 psig (including location adjustment and instrument uncertainty). The calculation evaluated a mass input transient and a heat input transient with a single PORV opening in response to each transient. The calculation assumed the existing PORV setpoints in TS Figure 3.4-4. The LTOP analysis utilizes the new limiting pressures determined for the heatup and cooldown curves at 55 EFPY for the reactor vessel beltline region and vessel flange region to conclude that the vessel peak pressure will not be exceeded during low temperature operations. Therefore, the current PORV low-pressure setpoints in TS Figure 3.4-4 remain acceptable for plant operation to 55 EFPY.

The LTOP analysis also evaluated the enable temperature utilizing the 55 EFPY highest adjusted reference temperature. The current 325°F LTOP enable temperature was evaluated using the ASME Boiler and Pressure Vessel Code 2007 Edition including Addenda through 2008 Section XI Appendix G Article G-2215, and determined to remain acceptable.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The following regulatory requirements and guidance documents are applicable to the proposed change.

10 CFR 50.36

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications," establishes the requirements related to the content of the TSs. Pursuant to 10 CFR 50.36(c) TSs will include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) LCOs, (3) Surveillance Requirements (SRs), (4) design features; and (5) administrative controls.

HNP LCOs 3.4.9.1 (Applicability: Modes 1, 2 and 3) and 3.4.9.2 (Applicability: Modes 4, 5 and 6 with reactor vessel head on) limit the pressure and temperature changes during RCS heatup

and cooldown (i.e., to the right and below the P/T curves in Figures 3.4-2 and 3.4-3), to prevent non-ductile RPV failure. The proposed change revises the HNP P/T limit curves in TS and reflects that the curves are applicable until 55 EFPY. Based on the determination that the proposed TS Figures 3.4-2 and 3.4-3 are acceptable up to 55 EFPY (see Section 3 above), Duke Energy concludes that HNP LCOs 3.4.9.1 and 3.4.9.2 will continue to meet the requirements of 10 CFR 50.36(c)(2)(i) with the proposed change.

<u>10 CFR 50.60</u>

Section 50.60 of 10 CFR, “Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation,” imposes fracture toughness and material surveillance program requirements, which are set forth in 10 CFR 50, Appendices G, “Fracture Toughness Requirements,” and H, “Reactor Vessel Material Surveillance Program Requirements.” With the proposed change, HNP meets the requirements set forth in 10 CFR 50, Appendices G and H. Therefore, HNP also satisfies the requirements of 10 CFR 50.60 for the proposed change.

<u>10 CFR 50, Appendix G</u>

Appendix G to 10 CFR 50 requires that the P/T limits for the facility’s RPV be at least as conservative as those obtained by following the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME Code. Using the ART values, P/T limits curves were determined in accordance with the requirements of 10 CFR 50, Appendix G. Therefore, Duke Energy concludes for the proposed change that the HNP RPV will continue to meet RPV integrity regulatory requirements through 55 EFPY.

<u>10 CFR 50, Appendix H</u>

Appendix H to 10 CFR 50 establishes requirements for each facility related to its RPV material surveillance. These regulatory requirements will continue to be met for the proposed change with the surveillance capsule removal schedule prescribed in Section 9.0 of Framatome report ANP-3798NP that was provided to the NRC staff in the Appendix H submittal dated October 23, 2019 (Reference 2).

<u>RG 1.99, Revision 2</u>

RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” contains guidance on methodologies the NRC considers acceptable for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. This RG was used along with fluence to calculate ART values for the HNP reactor vessel materials at 55 EFPY. Therefore, the proposed change has no effect on how Duke Energy applies RG 1.99, Revision 2 for HNP.

<u>RG 1.190</u>

RG. 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” dated March 2001, describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence. Framatome neutron transport evaluation methodologies (proprietary) utilized for the HNP neutron fluence evaluation followed the guidance of RG 1.190.

Regulatory Issue Summary (RIS) 2014-11

RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components" clarifies that P/T limit calculations for ferritic RPV materials other than those materials with the highest reference temperature may result in more limiting P/T curves because of higher stresses due to structural discontinuities, such as those in RPV inlet and outlet nozzles. Duke Energy appropriately considered RIS 2014-11 by performing analyses to include the impact of structural discontinuities in the HNP RPV shell upper and lower transition regions at 55 EFPY.

The proposed change does not affect plant compliance with any of the above regulations or guidance and will ensure that the lowest functional capabilities or performance levels of equipment required for safe operation are met.

4.2 Precedent

The NRC has previously approved changes similar to the proposed change in this License Amendment Request for other nuclear power plants including:

H.B. Robinson Steam Electric Plant, Unit No. 2: Application dated November 2, 2015 (ADAMS Accession No. ML15307A069); NRC Safety Evaluation dated November 22, 2016 (ADAMS Accession No. ML16285A404).

Browns Ferry Nuclear Plant, Unit 2: Application dated June 19, 2014 (ADAMS Accession No. ML14175A307); NRC Safety Evaluation dated June 2, 2015 (ADAMS Accession No. ML15065A049).

4.3 No Significant Hazards Consideration Determination Analysis

Duke Energy Progress, LLC (Duke Energy) requests an amendment to the Shearon Harris Nuclear Power Plant, Unit 1 (HNP) Renewed Facility Operating License. The proposed change revises HNP Technical Specification (TS) 3/4.4.9, "Pressure/Temperature Limits – Reactor Coolant System" to reflect an update to the pressure and temperature limit curves in Figures 3.4-2 (Reactor Coolant System Cooldown Limitations) and 3.4-3 (Reactor Coolant System Heatup Limitations). The proposed change also reflects that the revised HNP pressure and temperature limit curves in TS Figures 3.4-2 and 3.4-3, as well as the existing power-operated relief valve (PORV) setpoints in Figure 3.4-4, will be applicable until 55 effective full power years (EFPY).

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

Response: No.

The proposed change revises HNP TS to reflect updated pressure and temperature (P/T) limit curves in Figures 3.4-2 and 3.4-3 that are applicable until 55 EFPY. The proposed change also reflects that the existing PORV setpoints are applicable until 55 EFPY. The proposed change does not involve physical changes to the plant or alter the reactor coolant system (RCS) pressure boundary (i.e., there are no changes in operating pressure, materials or seismic loading). The proposed TS Figures 3.4-2 and 3.4-3, with an applicability term of 55 EFPY, provide continued assurance that the fracture toughness of the reactor pressure vessel (RPV) is consistent with analysis assumptions and NRC regulations. The methodology used to develop the proposed P/T limit curves provides assurance that the probability of a rapidly propagating failure will be minimized. The proposed P/T limit curves, with the applicability term of 55 EFPY, will continue to prohibit operation in regions where it is possible for brittle fracture of reactor vessel materials to occur, thereby assuring that the integrity of the RCS pressure boundary is maintained.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change revises HNP TS to reflect updated P/T limit curves in Figures 3.4-2 and 3.4-3 that are applicable until 55 EFPY. The proposed change also reflects that the existing PORV setpoints are applicable until 55 EFPY. The proposed change does not affect the design or assumed accident performance of any structure, system or component or introduce any new modes of system operation or failure modes. Compliance with the proposed P/T limit curves will provide sufficient protection against brittle fracture of reactor vessel materials to assure that the RCS pressure boundary performs as previously evaluated.

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

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

Response: No.

The proposed change revises HNP TS to reflect updated P/T limit curves in Figures 3.4-2 and 3.4-3 that are applicable until 55 EFPY. The proposed change also reflects that the existing PORV setpoints are applicable until 55 EFPY. HNP complies with applicable regulations (i.e., 10 CFR 50, Appendices G and H) and adheres to Nuclear Regulatory Commission (NRC)-approved methodologies (i.e., Regulatory Guides 1.99 and 1.190) with respect to the proposed P/T limit curves in TS Figures 3.4-2 and 3.4-3 in

order to provide an adequate margin of safety to the conditions at which brittle fracture may occur. The proposed P/T limit curves for HNP, with an applicability term of 55 EFPY, will continue to provide assurance that the P/T limits are not exceeded.

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

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

4.4 Conclusions

In conclusion, based on the considerations discussed above: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

1. NRC letter, *Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Regarding Pressure/Temperature Limits (TAC No. MA8642)*, July 28, 2000 (ADAMS Accession No. ML003736272).

2. Duke Energy letter, *Submittal of the Summary Technical Report for the Reactor Pressure Vessel Surveillance Program Capsule Z*, October 23, 2019 (ADAMS Accession No. ML19296C841).

3. Regulatory Issue Summary 2014-11, *Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components*, October 14, 2014.

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400 / Renewed License No. NPF-63

License Amendment Request to Correct Non-Conservative Technical Specification 3/4.4.9, “Pressure/Temperature Limits – Reactor Coolant System”

Attachment 1

Technical Specifications Markup

Replace with Insert #1

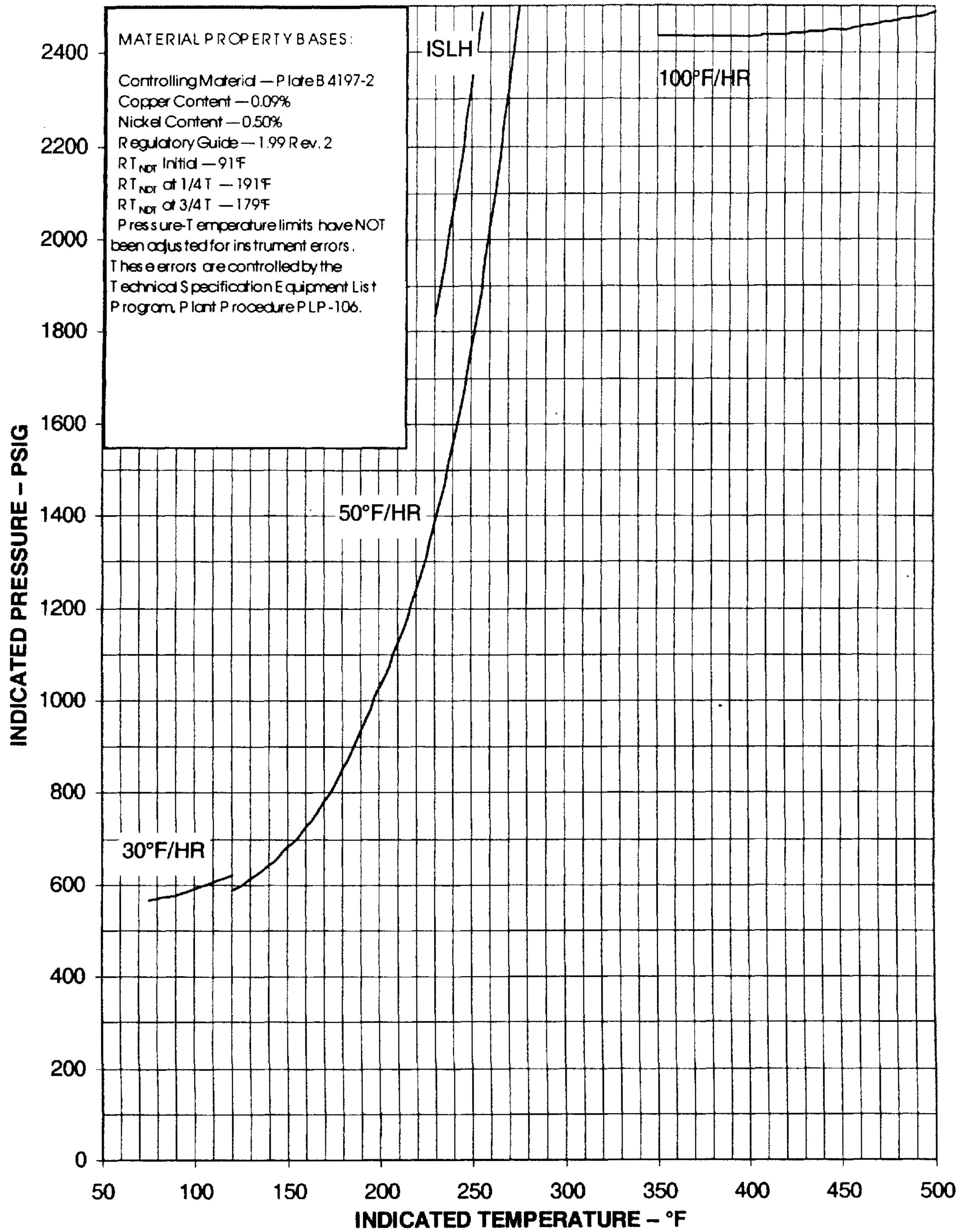


FIGURE 3.4-2
REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS – APPLICABLE TO UP TO 36 EFPY

Insert #1

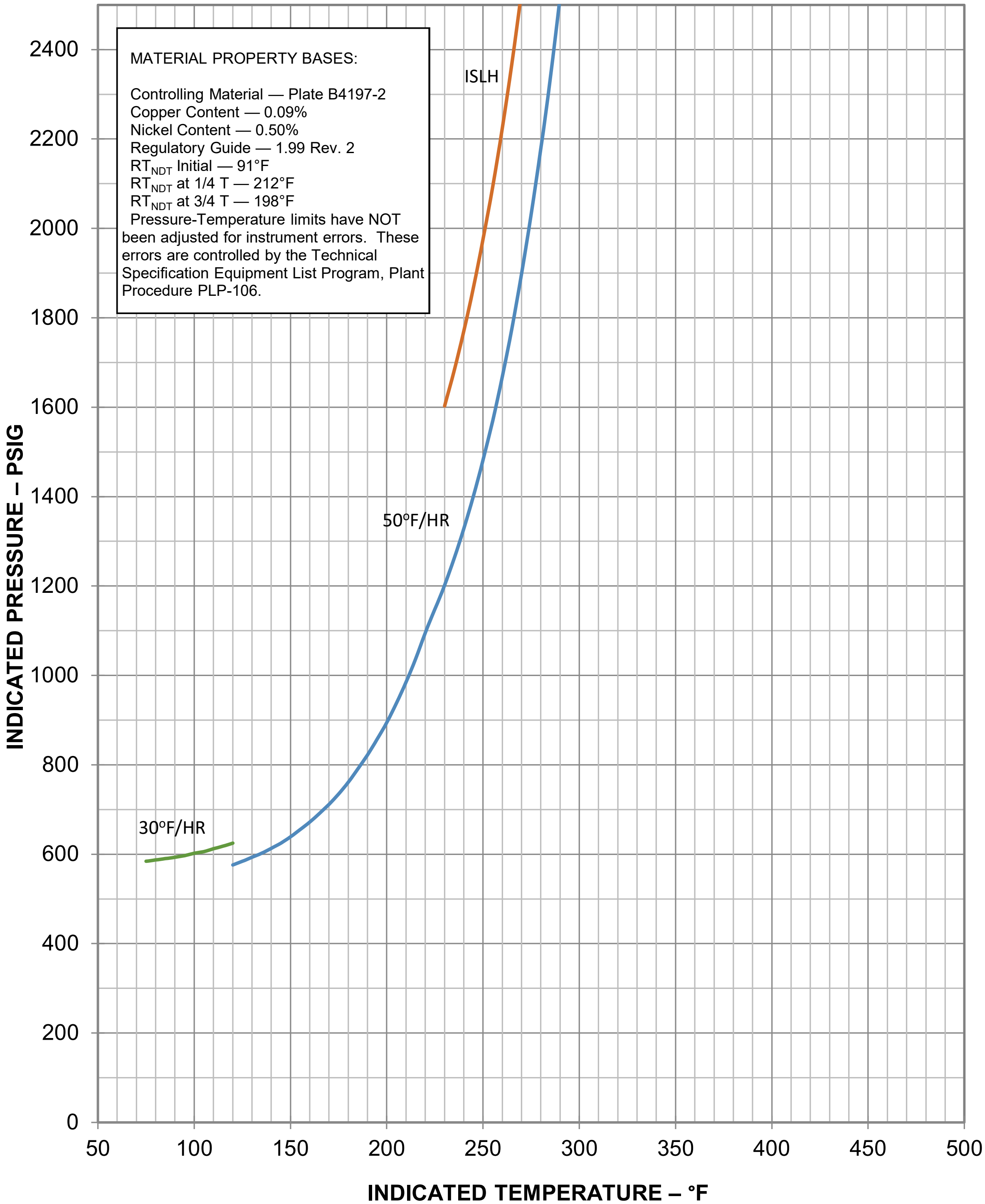


FIGURE 3.4–2
REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS—APPLICABLE UP TO 55 EFPY

Replace with
Insert #2

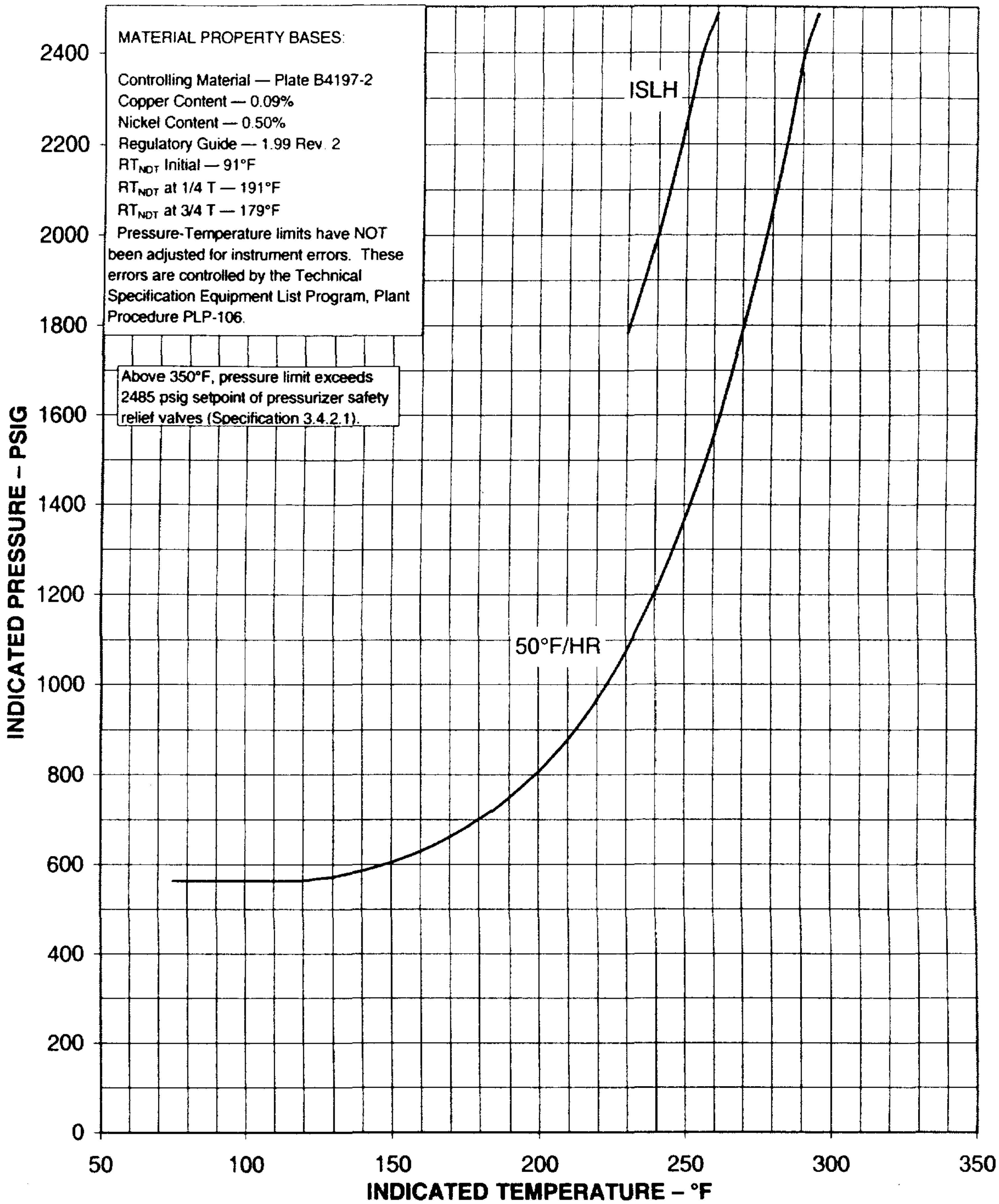


FIGURE 3.4–3
REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS—APPLICABLE UP TO 36 EFPY

Insert #2

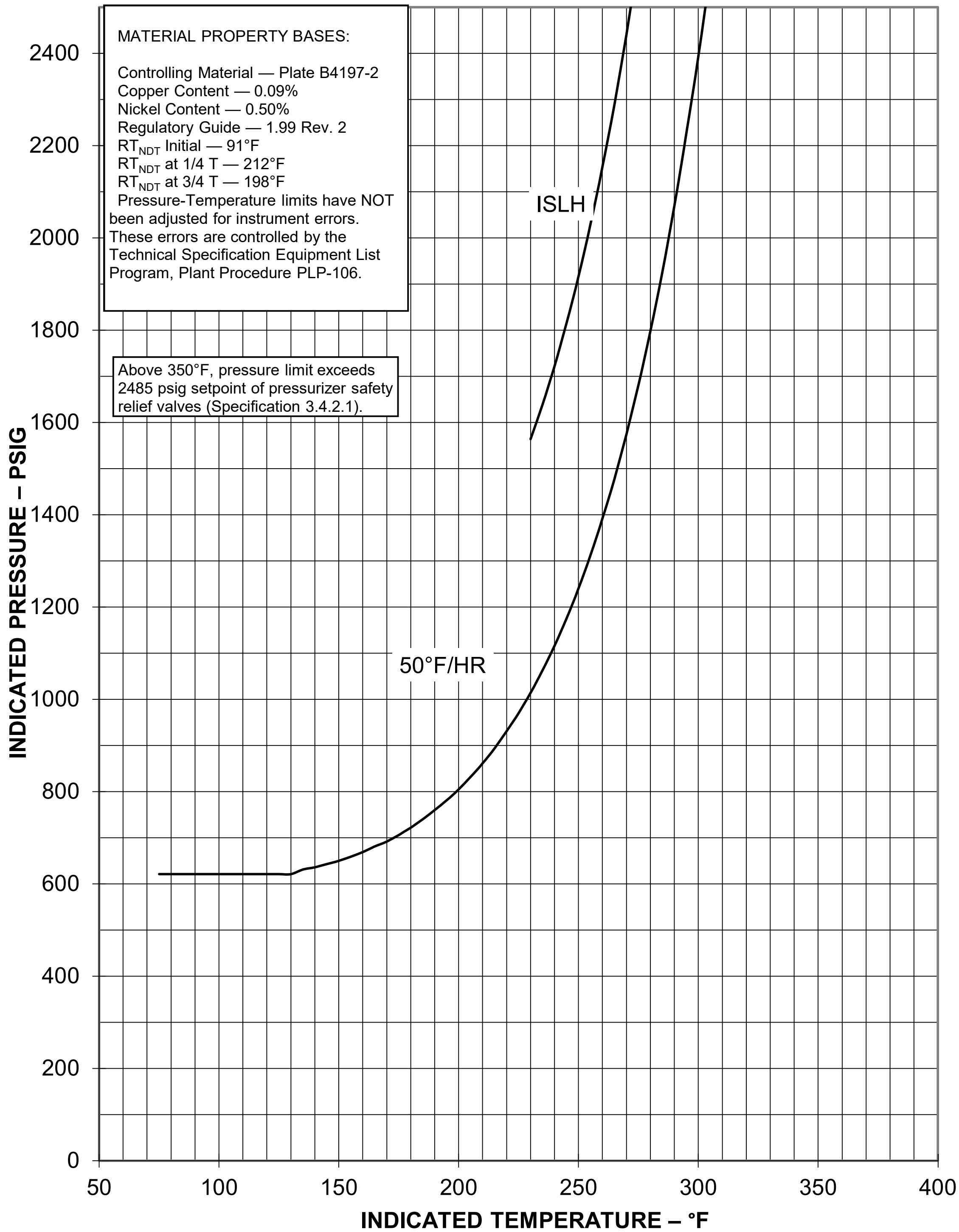
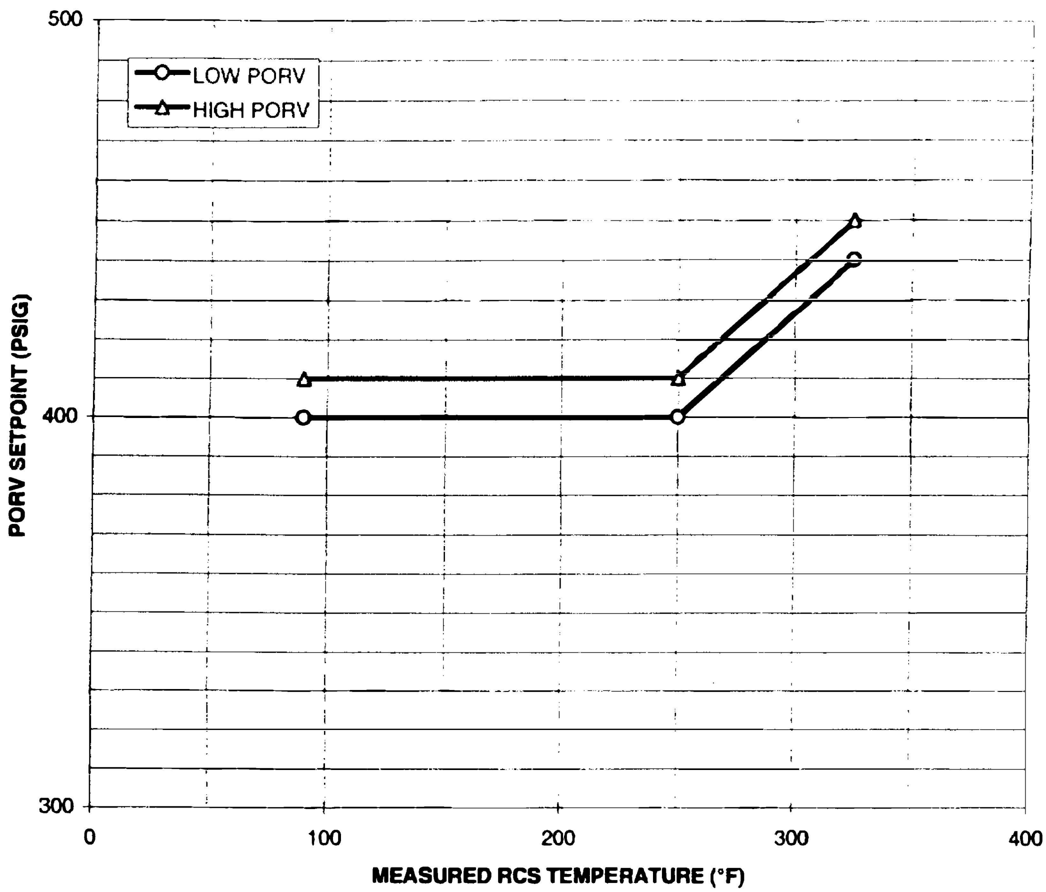


FIGURE 3.4–3
REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS—APPLICABLE UP TO 55 EFPY



RCS TEMP (°F)	LOW PORV* (psig)	HIGH PORV* (psig)
90	400	410
250	400	410
325	440	450

Change from 36 to 55

* VALUES BASED ON ~~36~~ 55 EFPY REACTOR VESSEL DATA

INSTRUMENT ERRORS ARE CONTROLLED BY THE TECHNICAL SPECIFICATION EQUIPMENT LIST PROGRAM, PLANT PROCEDURE PLP-106.

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400 / Renewed License No. NPF-63

License Amendment Request to Correct Non-Conservative Technical Specification 3/4.4.9, “Pressure/Temperature Limits – Reactor Coolant System”

Attachment 2

Technical Specifications Bases Markup

(Provided for Information Only)

SPECIFIC ACTIVITY (Continued)

distinction between the radionuclides above and below a half-life of 15 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture occur, since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, ASME Code Case N-640, and 10 CFR 50 Appendix G and H. 10 CFR 50, Appendix G also addresses the metal temperature of the closure head flange and vessel flange regions. The minimum metal temperature of the closure flange region should be at least 120°F higher than the limiting RT NDT for these regions when the pressure exceeds 20% (621 psig for Westinghouse plants) of the preservice hydrostatic test pressure. For Shearon Harris Unit 1, the minimum temperature of the closure flange and vessel flange regions is 120°F because the limiting RT NDT is 0°F (see Table B 3/4 4-1).

Delete

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1. The reactor coolant temperature and pressure and system cooldown and heatup rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 and Table 4.4-6 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

Replace with:
considered to be 0°F for conservatism based on the maximum closure head design specification requirement.

PRESSURE/TEMPERATURE LIMITS (Continued)

b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 625°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness testing of the ferritic materials in the reactor vessel was performed in accordance with the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code. The fracture toughness testing of the ferritic materials associated with the replacement reactor vessel head (RRVH) was performed in accordance with the 2001 Edition, with Addenda up to and including 2003, of Section III of the ASME Boiler and Pressure Vessel Code. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Replace with : 55

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of $\boxed{36}$ effective full power years (EFPY) of service life.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the value of ΔRT_{NDT}, including margin, computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

Insert: as

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TABLE B 3/4.4-1
REACTOR VESSEL TOUGHNESS

COMPONENT	GRADE	HEAT NO	Cu (wt.%)	Ni (wt.%)	T_{NDT} (°F)	INITIAL RT_{NDT} (°F)	CHARPY UPPER SHELF ENERGY TRANSVERSE FT LB
Closure Hd.	A508,Gr3,CL1	16W84	0.04	0.83	-40	-40	210
Vessel Flange	A508, CL2	5302-V1	-	-	-10	-8	110
Inlet Nozzle	"	438B-4	-	0.77	-20	-20	120.9
" "	"	438B-5	-	0.835	-10	-10	106.6
" "	"	438B-6	-	0.79	-40	-40	121.6
Outlet Nozzle	"	439B-4	-	0.835	-10	-10	103.7
" "	"	439B-5	-	0.815	-10	-10	96.2
" "	"	439B-6	-	0.81	-10	-10	100.1
Nozzle Shell	A533B,CL1	C0224-1	.12	0.57	-20	-1	90
" "	"	C0123-1	.12	0.60	0	42	84
Inter. Shell*	"	A9153-1	.09	.46	-10	60	83
" "*	"	B4197-2	.09	.50	-10	91	71
Lower Shell*	"	C9924-1	.08	.47	-10	54	98
" "*	"	C9924-2	.08	.47	-20	57	88
Bottom Hd. Torus	"	A9249-2	-	0.53	-40	- -**	94
" " Dome		A9213-2	-	0.515	-40	-8	125
Weld (Inter & Lower Shell Vertical Weld Seams)*		4P4784	.05	.91	-20	-20	> 94
Weld (Inter. to Lower Shell Girth Seam)*		5P6771	.03	.94	-80	-20	80

Revise

Revise

* For Beltline Materials, copper and nickel values are "best estimates".

**The initial RT_{NDT} for the Torus Plate material is a generic value based on the class of material calculated in accordance with Regulatory Guide 1.99 Rev. 2 and is considered proprietary.

Add

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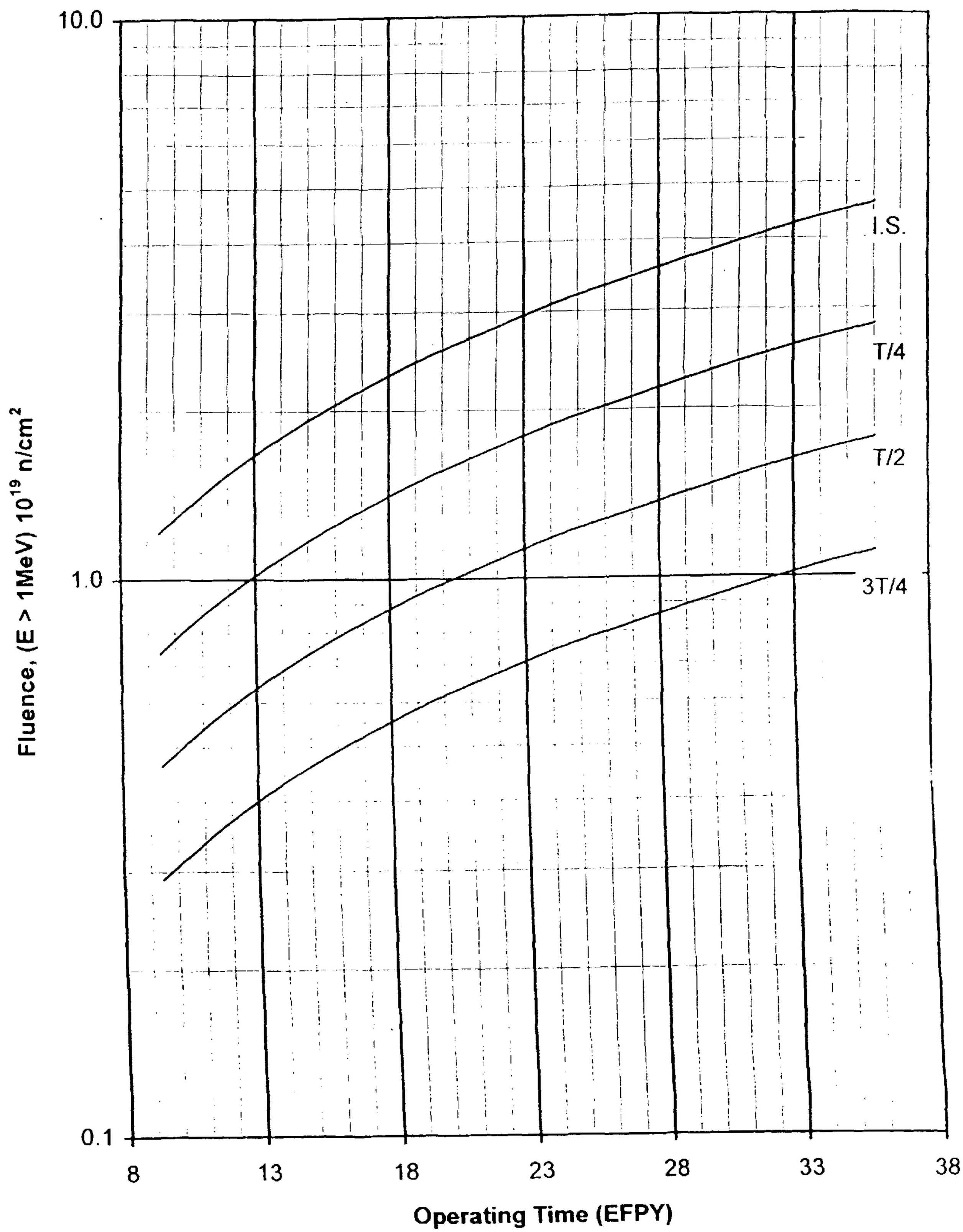

FIGURE B 3/4.4-1
FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

PRESSURE/TEMPERATURE LIMITS (Continued)

The cooldown and heatup limits of Figures 3.4-2 and 3.4-3 are based upon an adjusted RT_{NDT} (initial RT_{NDT} plus predicted adjustments for this shift in RT_{NDT} plus margin).

In accordance with Regulatory Guide 1.99, Revision 2, the results from the material surveillance program, evaluated according to ASTM E185, may be used to determine ΔRT_{NDT} when two or more sets of credible surveillance data are available. Capsules will be removed and evaluated in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The cooldown and heatup curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various cooldown and heatup rates are calculated using methods derived from Appendix G in Section XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and ASME Code Case N-640 for the reactor vessel controlling material.

Delete

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The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures for the beltline shell region a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. A semielliptical inside corner flaw is assumed for the nozzle regions with a depth of one-quarter of the nozzle belt wall thickness. The inlet nozzle is used in the calculation procdures since the inner radius of this tapered nozzle is larger at the corner than the inner radius of the more tapered outlet nozzle. The dimensions of these postulated cracks, referred to in Appendix G of ASME Section XI as reference flaws, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which cooldown and heatup curves are generated.

Insert: surface

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Replace with: upper shell

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The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the

PRESSURE/TEMPERATURE LIMITS (Continued)

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metal temperature at that time. K_{IR} is obtained from reference fracture toughness curves defined in the ASME Code. Pressure-temperature limits are developed for the vessel using the K_{IR} curve defined in Appendix A to the ASME Code, as permitted by ASME Code Case N-640. For the Replacement Reactor Vessel Head (RRVH), pressure-temperature limits are developed, using $K_{IR} = K_{Ic}$ curve from ASME Section XI, Appendix G, 2007 Edition through 2008 Addenda. For the remaining components of the primary pressure boundary, pressure-temperature limits are based on the K_{IR} curve defined in Appendix G to the ASME Code. The K_{IR} curves are given by the equations:

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reactor vessel and the

Replace with:

Vessel regions: $K_{IR} = K_{IC} = 33.2 + 20.734 \exp[0.02(T\text{-}RT_{NDT})]$

$$K_{IR} = K_{Ic} = 33.2 + 2.806 \exp[0.02(T\text{-}RT_{NDT} + 100°F)] \quad (1a)$$

Remaining regions:

$$K_{IR} = K_{Ia} = 26.8 + 1.233 \exp[0.0145(T\text{-}RT_{NDT} + 160°F)] \quad (1b)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\,K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice leak and hydrostatic (ISLH) test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. The pressure stress intensity factors are obtained and allowable pressures are calculated from equation 2.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall and the inlet nozzle corner. During cooldown, the controlling location of the flaw is always at the inside surface because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest. The composite limit curves are developed considering the controlling reactor vessel component, either the beltline shell, RV head flange limit, or the inlet nozzle.

PRESSURE/TEMPERATURE LIMITS (Continued)

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The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T inside surface location is at a higher temperature than the fluid adjacent to the inside surface. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside surface. The thermal gradients during heatup produce compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of