Response to Action Item 5-5 PTLR TeR

MCB Issue List Regarding PTLR

Issue #1 (AI 5-5.16)

The staff has identified several inconsistencies between the information provided in the APR1400 Final Safety Analysis Report (FSAR) and what is provided in Technical Report APR1400-Z-M-NR-14008-P, Revision 0, “Pressure Temperature Limits Methodology for RCS Heatup and Cooldown” (PTLR). These include the following:

- Surveillance capsule lead factors (1.5 in FSAR, 1.4 in PTLR)
- RT_{NDT} for beltline forging (68.4 °F in FSAR, 68 °F in PTLR)
- RT_{PTS} for limiting beltline material (123 °F in FSAR, 122.6 °F in PTLR)
- Uncertainty in expected neutron fluence at specimen locations (20% in FSAR, 16.28% in PTLR)
- Surveillance capsule neutron fluence values (end of life neutron fluence only provided in FSAR, more provided in PTLR)
- In APR1400 FSAR Section 5.3.1.6.7, the applicant states that when data from the surveillance capsules becomes available, it will be used to adjust the pressure and temperature limit curves. However, this statement is not provided in the PTLR.

Such inconsistencies interfere with the staff’s from making and documenting a clear determination regarding compliance of the APR1400 design with NRC regulatory requirements. Revise the APR1400 FSAR and the PTLR as necessary to correct all inconsistencies in the information reported.

Response

The following information will be corrected:

- Surveillance capsule lead factors in FSAR will be corrected to 1.4.
- RT_{NDT} in FSAR 5.3.2.1.1 and in PTLR will be corrected as indicated on the Attachment, pages 2 and 3.
- RT_{PTS} for limiting beltline material in FSAR will be corrected to 122.6^{o}F
- The uncertainty of 20% in FSAR is for the design maximum value of fluence calculation while 16.28% is an uncertainty for the best-estimate value of fluence calculation. Therefore, the uncertainty value in FSAR will be changed to reflect the best-estimate value, 16.28%.
- Surveillance capsule neutron fluence values in FSAR Table 5.3-7 will be supplemented.
- Statement in FSAR Section 5.3.1.6.7 (“When actual post-irradiation surveillance data become available for each reactor vessel, the data are used to adjust plant operation limit curves.”) will be added in PTLR.

Non-Proprietary

Response to Action Item 5-5 PTLR TeR

Impact on DCD
DCD 5.3.1.6.5, 5.3.1.6.6, 5.3.2.1.1, 5.3.2.3 and Table 5.3-7 will be revised as indicated on the attached markup.

Impact on PRA
There is no impact on the PRA.

Impact on Technical Specifications
There is no impact on the Technical Specification.

Impact on Technical/Topical/Environmental Reports
Technical Report APR1400-Z-M-NR-14008-P will be revised as indicated on the attached markup.

Table 5.3-4. The test specimens contained in the capsule assemblies are used for monitoring the neutron-induced property changes of the reactor vessel materials. These capsules, therefore, are positioned near the inside wall of the reactor vessel so that the irradiation conditions (fluence, flux spectrum, temperature) of the test specimens resemble as closely as possible the irradiation conditions of the reactor vessel. The neutron fluence of the test specimens is expected to be approximately ~~1.5~~ times higher than that seen by the adjacent vessel wall, and the measured changes in properties of the surveillance materials are therefore able to predict the radiation induced changes in the reactor vessel beltline materials. The capsule assemblies are placed in capsule holders positioned circumferentially about the core at locations that include the regions of maximum flux. Figure 5.3-5 presents the typical exposure locations for the capsule assemblies in the plan view.

1.4

All capsule assemblies are inserted into their respective capsule holders during the final reactor assembly operation. The design also permits the remote installation of replacement capsule assemblies. The capsule holders are welded to the vessel cladding on the inside surface, and the welds are subject to inspection according to the requirements for permanent structural attachments as given in ASME Sections III and XI.

5.3.1.6.6 Withdrawal Schedule

The capsule assemblies remain within their holders until the specimens in the assemblies have been exposed to predetermined removal schedule based on effective full power years (EFPYs). At that time, the capsule assembly is removed, and the surveillance materials are evaluated. The target fluence levels for the surveillance capsules are determined at the azimuthal locations for the recommended withdrawal schedule of ASTM E185, extended to a design life of 60 years. The fluence values in Table 5.3-7 are accurate within +~~20~~ percent, ~~-20~~ percent. The uncertainty is composed of errors in the calculational method and errors in the combined radial and axial power distribution.

−16.28 +16.28

Withdrawal schedules may be modified to coincide with the refueling outages or plant shutdowns most closely approaching the withdrawal schedule. The two standby capsules are provided in the event they are needed to monitor the effect of a major core change or annealing of the vessel or to provide supplemental toughness data for evaluating a flaw in the beltline.

RT_{NDT} for the reactor vessel beltline forging is -23.3 °C (-10 °F), and an initial RT_{NDT} for the remaining material including weld materials of the reactor coolant system (RCS) is -12.2 °C (10 °F). RT_{NDT} is determined in accordance with Article NB-2300 of ASME Section III.

As a result of fast neutron irradiation in the region of the core, the RT_{NDT} of irradiated material increases with operation. The maximum integrated neutron fluence on the reactor vessel wall beltline region is estimated to be 9.5×10^{19} n/cm^2. There are no longitudinal seam welds, and two of the circumferential seam welds are located near the fringes of core beltline region. It is conservatively assumed that weld materials are subjected to the maximum neutron fluence of 9.5 x 10^{19}n/cm^2 for evaluating RT_{NDT}. The techniques used to analytically and experimentally predict the integrated fast neutron ($E \geq 1$ MeV) fluxes of the reactor vessel are described in Subsection 5.3.1.6.

The shift in RT_{NDT} of reactor vessel beltline materials can be analytically predicted based on the procedures described in NRC RG 1.99 because the RCS operating temperature (cold leg) is 290.6 °C (555 °F), which is above 274 °C (525 °F) as shown in Table 5.1.1-1. The surveillance program is prepared to obtain the reliable irradiation data for the adjustment or qualification of operating parameters. Reactor vessel shell materials are designed to limit RT_{NDT} values at 1/4T location within 93.3 °C (200 °F) at the end-of-life. The RT_{NDT} values at 1/4T location at the end-of-life are expected to be 20.2 °C (68.4 °F) for beltline forging and ~~47.2 °C (117 °F)~~ 46.9 °C (116.5 °F) for weld material per NRC RG 1.99 based on the weight percent of residual elements in Subsection 5.2.3.1.

The measured shift in RT_{NDT} for a specimen is applied to the adjacent section of the reactor vessel for later stages in plant life because the measured neutron spectra and flux at the specimen and reactor vessel inside radius are close. The measured shift in RT_{NDT} is adjusted for the difference in calculated flux magnitudes between the surveillance specimens and the point of interest in the reactor vessel wall.

The maximum fluence of the reactor vessel is obtained from the measured exposure by application of the calculated azimuthal neutron flux variation. The neutron fluence and the actual shift in RT_{NDT} are established periodically during plant operation by testing the reactor vessel surveillance material specimens that are irradiated in capsules secured to the inside wall of the reactor vessel, as described in Subsection 5.3.1.6 and shown in Figures

Table 4-2 Initial RT_{NDT} and ART for the APR1400 Reactor Vessel Materials through 60 Years

Material	Initial RT_{NDT} (℃(℉))	ART (℃(℉)) at 60 years			Fluence (n/㎠) at 60 years		
		surface	1/4t	3/4t	surface	1/4t 3)	3/4t 3)
Lower Shell Course (beltline material)	-23.3(-10)	~~21.1(70)~~ 21.2(70.2)	~~20.0(68)~~ 20.2(68.4)	~~17.2(63)~~ 17.4(63.4)	9.5 E+19	5.52E+19	1.86E+19
G-2 Weld (top of lower shell)	-12.2(10)	~~38.3(101)~~ 38.2(100.8)	~~32.2(90)~~ 32.4(90.3)	~~20.0(68)~~ 19.9(67.8)	2.2 E+19	1.28E+19	0.43E+19
G-3 Weld (bottom of lower shell)	-12.2(10)	~~3.3(38)~~ 3.2(37.8)	~~-0.56(31)~~ −0.56(31.0)	~~-6.11(21)~~ −5.9(21.3)	7.9 E+17	4.59E+17	1.55E+17
Limiting Case (weld in beltline) 1)	**-12.2(10)**	~~**50.6(123)**~~ 50.3(122.6)	~~**47.2(117)**~~ 46.9(116.5)	~~**36.7(98)**~~ 36.6(97.8)	**9.5 E+19**	**5.52E+19**	**1.86E+19**
Head Flange	-12.2(10)	NA 2)	NA 2)	NA 2)	< E+12	< E+12	< E+12
Vessel Flange	-12.2(10)	NA 2)	NA 2)	NA 2)	< E+12	< E+12	< E+12
Inlet Nozzles	-12.2(10)	NA 2)	NA 2)	NA 2)	~ E+15	~ E+15	~ E+15
Outlet Nozzles	-12.2(10)	NA 2)	NA 2)	NA 2)	~ E+15	~ E+15	~ E+15
DVI Nozzles	-12.2(10)	NA 2)	NA 2)	NA 2)	~ E+12	~ E+12	~ E+12
RCS Piping	-12.2(10)	NA 2)	NA 2)	NA 2)	negligible	negligible	negligible

Notes to Table 4-2:

1) This case is the basis for the evaluation of P-T limits and PTS assuming conservatively that weld is in the beltline region and subjected to maximum neutron fluence.

2) Not applicable because fast neutron fluence is not significant.

3) Fluences at 1/4t and 3/4t locations are calculated using the equation, $f = f_{surf}(e^{-0.24x})$, in Section 4.3.

5.3.2.3 Pressurized Thermal Shock

The reactor vessel meets the requirements of 10 CFR 50.61 (Reference 30), and NRC SRP BTP 5-3 (i.e., the PTS screening criteria are not projected to be exceeded by expiration of the operations).

RT_{PTS} is evaluated using the procedure described in 10 CFR 50.61, which is provided below:

RT_{PTS} = (Initial) RT_{NDT} + ΔRT_{PTS} + Margin

The calculated maximum RT_{PTS} satisfies the screening criteria in 10 CFR 50.61(b)(2).

The PTS screening criteria are:

a. 132.2 °C (270 °F) for plates, forgings, and axial weld materials

b. 148.9 °C (300 °F) for circumferential weld materials

The following assumptions are applied in the calculation of RT_{PTS} for limiting beltline material:

a. The limiting case is the weld material subjected to the maximum integrated fast neutron fluence of 9.5 × 10^{19} n/cm^2.

b. For the weld material, maximum copper content is 0.05 wt% and maximum nickel content is 0.10 wt%, and the maximum initial RT_{NDT} is -12.2 °C (10 °F).

c. The adjustment in the reference temperature caused by irradiation (ΔRT_{PTS}) is calculated to be 31.4 °C (56.6 °F). The margin required by 10 CFR 50.61 is 31.1 °C (56 °F) for the weld materials.

The calculated RT_{PTS} is ~~50.6 °C (123 °F)~~ 50.3 °C (122.6 °F), which satisfies the above PTS screening criteria for circumferential weld materials as shown in Table 5.3-10.

The calculated RT_{PTS} for lower shell course in Table 5.3-10 is 10.2 °C (50.4 °F), which also satisfies the above PTS screening criteria for forging.

Table 5.3-7

Capsule Assembly Removal Schedule

Capsule	Azimuthal Location	Removal Time(1)	Target Fluence(2) (n/cm^2)	Design Maximum Fluence(3) (n/cm^2)
A	217 °	6 EFPY	0.82×10^{19}	1.0×10^{19}
B	37 °	15 EFPY	1.83×10^{19}	2.6×10^{19}
C	224 °	32 EFPY	3.75×10^{19}	5.4×10^{19}
D	323 °	EOL	6.44×10^{19} ~~(2)~~ Delete	9.5×10^{19}
E	44 °	Standby	-	
F	143 °	Standby	-	

(1) Schedule may be modified to coincide with the refueling outages or scheduled shutdowns most closely approximating the withdrawal schedule.

(2) ~~Expected best estimated fluence level at the end of the plant design life (interface between reactor wall and cladding)~~

Best estimate fluences (best estimate values of expected neutron fluence) at specimen locations in each capsule with ±16.28% uncertainty

(3) Design maximum or conservative fluence values at the reactor vessel inside surface with ±20% uncertainty, which are used to determine the withdrawal schedule

7.3 Withdrawal Schedule

The capsule assemblies remain within their holders until the specimens contained therein have been exposed according to the predetermined removal schedule based on EFPYs. When the scheduled time comes, the capsule assembly is removed and the surveillance materials are evaluated. The capsule assembly removal schedule is presented in Table 7-1.

When actual post-irradiation surveillance data become available for each reactor vessel, the data are used to adjust plant operating limit curves.

Table 7-1 Capsule Assembly Removal Schedule

Capsule	Azimuthal Location	Removal Time 1)	Target Fluence (n/cm^2) 2)	Design Maximum Fluence(3) (n/cm^2)
A	217°	6 EFPY	0.82×10^{19}	1.0×10^{19}
B	37°	15 EFPY	1.83×10^{19}	2.6×10^{19}
C	224°	32 EFPY	3.75×10^{19}	5.4×10^{19}
D	323°	End of Life	6.44×10^{19}	9.5×10^{19}
E	44°	Standby		
F	143°	Standby		

Notes to Table 7-1:

1) Schedule may be modified to coincide with those refueling outages or scheduled shutdowns most closely approximating the withdrawal schedule.

2) Best estimate fluences (best estimate values of expected neutron fluence) ~~Best estimated average value of expected neutron fluence level~~ at specimen locations in each capsule with ~~+16.28%~~ ±16.28% uncertainty.

3) Design maximum or conservative fluence values at the reactor vessel inside surface with ±20% uncertainty, which are used to determine the withdrawal schedule.

Response to Action Item 5-5 PTLR TeR

MCB Issue List Regarding PTLR

Issue #2 (AI 5-5.17)

To allow the staff to independently verify that the methodology used to develop the pressure-temperature (P-T) limits produces limits which meet the requirements of Title 10 of the Code of Federal Regulations, Part 50, Appendix G, the data points (pressure and temperature) corresponding to the all of the P-T limit curves provided in Technical Report APR1400-Z-M-NR-14008-P, Revision 0 are needed. However, the applicant has not provided this information. Therefore, the staff requests that the applicant revise Technical Report APR1400-Z-M-NR-14008-P, Revision 0, to provide the data points corresponding to the all of the P-T limit curves provided in Figure 6-1.

Response

Technical Report APR1400-Z-M-NR-14008-P will be revised to provide the data points corresponding to the all of the P-T limit curves provided in Figure 6-1.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specification.

Impact on Technical/Topical/Environmental Reports

Technical Report APR1400-Z-M-NR-14008-P will be revised as indicated on the attached markup.

6 CALCULATION OF P-T LIMITS FOR THE APR1400 OPERATION

P-T Limits for APR1400 is calculated according to the requirements and methods in Sections 2 through 5. Limiting Condition for Operation (LCO) 3.4.3 (RCS Pressure and Temperature (P-T) Limits), and LCO 3.4.11 (LTOP System) shall be maintained within the limits specified in this PTLR.

6.1 Normal Heatup and Cooldown

For P-T limits calculation, geometrical data and material properties including RT_{NDT} of the limiting materials should be determined. Table 4-1 and Table 4-2 provide the material properties used in the analysis. Heatup or cooldown rates can be between steady-state condition (0 ℃/hr) and 55.6 ℃/hr (100 °F/hr). With this input, allowable pressures are calculated for selected time points of heatup or cooldown transient.

For the reactor vessel beltline region, remote from discontinuities, the 1/4t flaws are postulated at both inside and outside surfaces (1/4t and 3/4t flaws, respectively). For heatup analysis, both 1/4t and 3/4t locations are examined. Thermal stresses are compressive at the inside surface and tensile at the outside surface. The membrane stresses due to pressure are always tensile. The total stresses are always greater at the outside than the inside. However, the allowable stresses decrease more at the inside than the outside because the effects of irradiations are more pronounced inside. For the cooldown analysis, only 1/4t location needs to be examined, because during cooldown the total stresses and irradiation effects are always greater inside.

The nozzle corners are also assessed for allowable pressure at selected time points. For the nozzle, the flaw is postulated to be located on the inside surface.

Finally, the allowable pressure is determined for the vessel flange region. The vessel flange limits are developed considering the postulated outside surface flaw, as described in Subsection 5.1.2 in accordance with G-2000 of ASME Section XI Appendix G. The conditions of stress intensity factor equation (5-1) is maintained in calculating an allowable pressure corresponding to a selected temperature.

Among these pressure values, the lowest one is determined and considered to be the maximum allowable pressure value at each selected time point. In this manner the results at various selected time points are determined to produce a single lower bound P-T limit curve for normal heatup or cooldown. The P-T limit curves associated with the three regions of the reactor vessel for normal plant heatup, cooldown, and inservice leak and hydro pressure test are presented in Figure 6-1.

and data points are listed in Table 6-2

The allowable pressure from plant startup is maintained at a constant value of 43.9 kg/cm^{2}A (625 psia) (20 percent of the preservice hydrostatic test pressure) from the bolt preload temperature condition until the coolant temperature reaches the temperature where a calculated crack-tip metal temperature exceeds the minimum temperature requirement of Table 2-1.

As shown in Figure 6-1, the beltline P-T limit of 55.6 ℃/hr (100 °F/hr) heatup rate is controlling. At 43.9 kg/cm^{2}A (625 psia), which corresponds to 20 percent of the preservice hydrostatic test pressure, the allowable temperature corresponds to the minimum temperature requirement of RT_{NDT} + 66.7 ℃ (120 °F) (54.4 ℃ (130 °F)) per Operating Condition 2.b of Table 2-1. The lowest service temperature of 43.3 ℃ (110 °F), as shown in Section 6.3, is lower than the minimum temperature required, 54.4 ℃ (130 °F). The P-T limits for nozzle corners or flange are not controlling limits at any time during normal plant heatup or cooldown. The allowable pressures for the nozzles are very high and especially those of DVI nozzles are beyond the pressure range shown in Figure 6-1.

The P-T limits presented in Figure 6-1 are corrected P-T limits, meaning that measurement uncertainty, due to instrument errors or sensor location adjustments, is included in the curves. Correction factors for

Table 6-2 Data points for P-T Limit Curve in Figure 6-1

TS