MCB Issue List Regarding APR1400 FSAR Section 5.3.1

Issue #1

Title 10 of the Code of Federal Regulations (10 CFR) Section 50.55a states that systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the American Society of Mechanical Engineers (ASME) Code. The materials used to fabricate the reactor vessel (RV) are acceptable if they are in accordance with ASME Code, Section III, NB-2000. APR1400 Final Safety Analysis Report (FSAR), Tier 2, Table 5.2-2 states that a weld electrode with the specification of MIL-E-18193 B-4 will be used for welding primary piping to the RV nozzle and for welding the RV upper shell to the RV flange. However, the MIL-E-18193B electrode specification was cancelled in 1989 and superseded by a different military specification, MIL-E-23765.

Revise FSAR Section 5.3.1 to replace the MIL-E-18193 B-4 weld electrode with a weld electrode that meets the requirements of the ASME Code, Section III, NB-2000. If an ASME specification is not used, then the applicant must provide a justification describing how the weld electrode meets the requirements of the ASME Code.

Response

FSAR Table 5.2-2 states that MIL-E-18193 B-4 will be used for P-1 to P-3 or P-3 to P-3 welding. However, MIL-E-18193 B-4 is not used for the APR1400. Therefore, it is reasonable to delete MIL-E-18193 B-4 from FSAR Table 5.2-2. Other welding materials for P-1 to P-3 or P-3 to P-3 welding in FSAR Table 5.2-2 satisfy ASME Code, Section III, NB-2000. FSAR Table 5.2-2 will be revised as shown in Appendix #1 (Page 1/3) attached.

Issue #2

FSAR, Tier 2, Table 5.2-2 states that NiCrFe filler metal will be used for buttering of j-groove welds in the RV closure head and in the RV control element drive mechanism (CEDM) nozzles. However, in order for the staff to make a finding on the acceptability of the filler metal, the material specification must be provided. The material specification designated must meet the requirements of the ASME Code, Section III, NB-2000.

Provide a specification for the NiCrFe filler metal used or provide a combined license (COL) applicant action item requiring any applicant that references the APR1400 design to select the specific material
used.

Response

The specifications for the NiCrFe filler metals (SFA 5.11 ENiCrFe-7 and SFA 5.14 ERNiCrFe-7(A)) will be added to FSAR Table 5.2-2 as shown in Appendix #1 (Pages 1/3 and 2/3) attached.

Issue #3

Per 10 CFR Part 50, Appendix H, the reactor vessel surveillance program (RVSP) is acceptable if it complies with ASTM Standard E 185-82. ASTM E 185-82 provides a recommended surveillance capsule withdrawal schedule that defines the time and/or projected neutron fluence at which a surveillance capsule should be removed. The applicant has provided the surveillance capsule withdrawal schedule in APR1400 FSAR, Table 5.3-7, and in Technical Report APR1400-Z-M-NR-14008-P, Revision 0, "Pressure- Temperature Limits Methodology for RCS Heatup and Cooldown," Table 7-1.

While reviewing the information provided in both documents, the staff has identified the following issues:

1. Although the projected values are the same, the APR1400 FSAR and Technical Report APR1400-Z-M-NR-14008-P, Revision 0, are inconsistent in their definition of the "target fluence." FSAR Table 5.3-7, Note 2 defines the "target fluence" as the expected fluence level at the "interface between the reactor wall and cladding," while Technical Report APR1400-Z-M-NR-14008-P, Revision 0, Table 7-1 defines the "target fluence" as the expected fluence level at "the specimen locations in each capsule." While the staff expects that there will be a relationship between the surveillance capsule fluence and the RV clad-to-base metal interface peak fluence (as defined by the ASTM E 185-82 requirements), the ambiguity introduced by the different definitions of "target fluence" make it difficult for the staff to interpret specifically what the applicant means when the term is used. Typically, the staff would expect "target fluence" to be defined in terms of the RV clad-to-base interface metal peak fluence that a specific surveillance capsule is intended to simulate.

   Revise both documents to provide a consistent definition of "target fluence" and revise the numerical fluence values given in the FSAR, Technical Report, or both, as necessary, to be consistent with the definition.

   Response

   FSAR Table 5.3-7 will be revised to provide a consistent definition of "target fluence". Since the target fluence is defined as an actual fluence which a capsule may experience until it is withdrawn, the target fluences of FSAR Table 5.3-7 and the Table 7-1 of the technical report (APR1400-Z-M-NR-14008-P) should be the best estimate fluences at the surveillance specimen location. Best estimate fluence is the fluence which is calculated using realistic axial and radial core power distributions. On this definition, FSAR Table 5.3-7 will be revised for the consistency between two tables as shown in Appendix #1 (Page 3/3) attached.

2. Based on the review of the projected neutron fluence values and the inconsistency described above, the staff cannot determine whether the recommended withdrawal schedule is in accordance with ASTM E-185-82.

   Explain how the RVSP given in FSAR Table 5.3-7 specifically meets the requirements of ASTM E 185-82. Specifically identify which of the withdrawal schedule programs of ASTM E 185-82, Table 1 the applicant proposes to comply with for the APR1400 design. Revise FSAR Table 5.3-7 and Technical Report APR1400-Z-M-NR-14008-P, Revision 0, Table 7-1 to clearly document, based on surveillance capsule and target RV clad-to-base metal interface fluences, how the proposed RVSP meets the ASTM E 185-82 requirements.

Response

Definitions:

1) A design maximum fluence is the fluence calculated using conservative axial and radial core power distributions.
2) A best estimate fluence is the fluence calculated using realistic axial and radial core power distributions.

The target fluence identified in FSAR Table 5.3-7 is an actual fluence which a capsule may experience until it is withdrawn. That is, it is best estimate fluence at the surveillance capsule location. However, please note that instead of the best estimate fluence in FSAR Table 5.3-7, the design maximum fluence at the surveillance capsule location is used to determine the withdrawal schedule of surveillance capsules for the conservatism of the schedule. The design maximum fluence at the surveillance specimen location at the end of the plant design life is 1.3×10^{20} n/cm^2. This value is calculated using the design maximum fluence at vessel inside surface (9.5×10^{19} n/cm^2 (E > 1.0 MeV)) and lead factor (1.4). The lead factor in Section 5.3.1.6.5 will be revised from 1.5 to 1.4.

The withdrawal schedule of the APR1400 surveillance capsules is prepared primarily in accordance with the withdrawal schedule program for the predicted transition temperature shift at vessel inside surface of ≤ 56°C (≤ 100°F) in Table 1 of ASTM E 185-82. The basis of the withdrawal schedule is as follows:

1) APR1400 plants have the maximum predicted transition temperature shift of 43°F which is calculated in accordance with RG 1.99 Rev. 1 using the design maximum fluence at the reactor vessel inside surface. According to Table 1 of ASTM E 185-82, the minimum number of capsules is 3 because the shift of RT_{NDT} is less than 100°F. However, the number of capsules selected for APR1400 plants is 4 instead of 3 because it has a longer design life (60 years, 55.8 EFPY) than the design life (40 years, 32 EFPY) defined in ASTM E 185-82.

2) APR1400 surveillance capsule withdrawal schedule and its technical basis are summarized in the table and its notes below.

Withdrawal sequence / Case	First	Second	Third	Fourth
Case 1	6[a)]	15[a)]	32[b)]	EOL[a)]
Case 2	2.14[c)]	40.7[d)]	41.85[e)]	-
Case 3	55.8 [f)]	-	-	-
Determined withdrawal time for 1400	6[g)]	155	32[h)]	EOL

Notes:

a) This is withdrawal time identified in Table 1 of ASTM E 185-82
b) At the end of life (EOL) time for the plants with the design life of 40 years
c) At the time when the accumulated neutron fluence of the capsule exceeds 5.0×10^{18} n/cm^2
d) At the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location.
e) At the time that corresponds to the three quarters of EOL, 55.8 EFPY. (Three quarters is determined considering the number of capsules, 4.)
f) At the time when the highest predicted delta RTNDT of all encapsulated materials is approximated 50°F
g) For the first withdrawal time, 6 EFPY is selected because the first withdrawal of 2.14 EFPY is considered to be too early in the plant life to use the surveillance capsule test data effectively and it would result in a long time span before the second capsule withdrawal at 15 EFPY.
h) Earlier time, 32 EFPY is selected as a third withdrawal time considering the end of life requirement for 40 year design life defined on Table 1 in ASTM E185-82.

Therefore, FSAR Table 5.3-7 will be revised as follows:

Capsule Assembly Removal Schedule

Capsule	Azimuthal Location	Removal Time[(1)]	Target Fluence[(2)] (n/cm2)
A	217°	6 EFPY	0.82×10^{19}
B	37°	15 EFPY	1.83×10^{19}
C	224°	32 EFPY	3.75×10^{19}
D	323°	EOL	6.44×10^{19}
E	44°	Standby	-
F	143°	Standby	-

Note

(1) Schedule may be modified to coincide with the refueling outages or scheduled shutdowns most closely approximating the withdrawal schedule.
(2) Best estimate fluence (best estimate value of expected neutron fluence) level at specimen locations in each capsule with +16.28% uncertainty.

3. Per ASTM E 185-82, the end of life (EOL) surveillance capsule should be withdrawn at a neutron fluence level that is not less than once or greater than twice the peak EOL RV clad-to-base metal interface fluence. The APR1400 withdrawal schedule is based on a design life of 60 years and calls for a capsule to be withdrawn at an EOL fluence of 6.44×10^{19} n/cm^2 (E > 1.0 MeV). However, both the APR1400 FSAR and Technical Report APR1400-Z-M-NR-14008-P, Revision 0, state that the expected APR1400 RV clad-to-base metal interface peak fluence at EOL is estimated to be 9.5×10^{19} n/cm^2 (E > 1.0 MeV). Therefore, the current withdrawal schedule is not in accordance with ASTM E 185-82.

   Revise the capsule withdrawal schedule and target fluences, as necessary, to ensure that the RVSP is in accordance with ASTM E 185-82.

   Response
   FSAR Table 5.3-7 will be revised to incorporate the modified information on target fluences and the notes. See response to Issue #3-2 for the explanation on meeting ASTM E 185-82, Table 1.

Sec 5.3.1 Issue #1,2
Appendix #1

Table 5.2-2 (4 of 5)

Base Material Type (5)	Base Material Type (5)	Type of Weld Material	Example of Use
Weld Materials for Reactor Coolant Pressure Boundary Components			
P-1	P-1	a. SFA 5.1 E-7018, E-7016 b. SFA 5.18 ER70S-6 c. SFA 5.23, EA-3(N)	Primary piping straight to primary piping elbows
P-1	P-3	a. SFA 5.1 E-7018, E-7016 b. SFA 5.5 E-8018-C3, E-8018-G, E-8016-G Delete ~~c. MIL-E-18193 B-4~~ c. ~~d.~~ SFA 5.23 ~~EA3~~ EA3(N) d. ~~e.~~ SFA 5.18 ER70S-6	Primary piping straight to the RV primary nozzle
P-1	P-8	a. NiCrFe filler metal SFA 5.11 ENiCrFe-7, SFA 5.14 ERNiCrFe-7(A) b. SFA 5.4 E309L-16 c. SFA 5.9 ER309L	Primary piping surge nozzle to surge nozzle safe end
P-1	P-43	NiCrFe filler metal SFA 5.11 ENiCrFe-7, SFA 5.14 ERNiCrFe-7(A)	Buttering (NiCrFe filler metal) of J-grooves in hot leg pipe
P-3	P-3	a. SFA 5.5 (2) E-8016-C3, E-8018-G, E-8016-G Delete ~~b. MIL-E-18193 B-4 (2)~~ b. ~~c.~~ SFA 5.23 ~~EA3 (2)~~ EA3(N)(2)	RV upper shell to RV flange
P-3	P-8	a. NiCrFe filler metal SFA 5.11 ENiCrFe-7, SFA 5.14 ERNiCrFe-7(A) b. SFA 5.4 E309L-16 c. SFA 5.9 ER309L	POSRV nozzle to POSRV safe end
P-3	P-43	a. NiCrFe filler metal SFA 5.11 ENiCrFe-7, SFA 5.14 ERNiCrFe-7(A)	Buttering (NiCrFe filler metal) of J-grooves in RV closure head
P-8	P-8	a. SFA 5.4 E308, E308L, E308L-16, E309, E309L-16, E316, E347 b. SFA 5.9 ER308, ER308L, ER309, ER309L, ER316, ER347	Surge line piping to surge line elbows

Sec 5.3.1 Issue #2
Appendix #1

Page (2/3)

Table 5.2-2 (5 of 5)

Base Material Type (5)	Base Material Type (5)	Type of Weld Material	Example of Use
Weld Materials for Reactor Coolant Pressure Boundary Components (cont.)			
P-8	P-43	a. NiCrFe filler metal	Pressurizer instrument nozzles to pressurizer instrument nozzle safe ends
P-43	P-43	a. NiCrFe filler metal	RV CEDM nozzles to J-groove buttering (NiCrFe filler metal)
Stainless steel cladding (1)		a. SFA 5.4 E308, E308L, E308L-16, E309, E309L, E309L-16 b. SFA 5.9 ER308, ER308L, ER 309, ER309L c. SFA 5.22 E308LT1-1, E309LT1-1	-
Nickel alloy cladding (1)		a. NiCrFe filler metal	-

SFA 5.11 ENiCrFe-7
SFA 5.14 ERNiCrFe-7(A)

(1) Materials exposed to reactor coolant

(2) Special weld wire with low residual elements of copper, nickel and phosphorous as specified when used in the RV core beltline region

(3) Material to be provided in the thermally treated condition

(4) SG secondary side pressure boundary materials including weld materials contain no greater than 0.010 wt% of sulfur (S)

(5) P-number designations are per the ASME Section IX, Table QW-422

Sec 5.3.1 Issue #3
Appendix #1

Page (3/3)

Table 5.3-7

Capsule Assembly Removal Schedule

Capsule	Azimuthal Location	Removal Time(1)	Target Fluence (n/cm^2) (2)
A	217 °	6 EFPY	- 0.82×10^{19}
B	37 °	15 EFPY	- 1.83×10^{19}
C	224 °	32 EFPY	- 3.75×10^{19}
D	323 °	EOL	6.44×10^{19} (2) Delete
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 fluence (Best estimate value of expected neutron fluence) level at specimen locations in each capsule with +16.28% uncertainty.

MCB Issue List Regarding APR1400 FSAR Section 5.3.2

Issue #1

Based on the information provided in the design certification application and Technical Report ANP APR1400-Z-M-NR-14008-P, Revision 0, the staff understands that the applicant has performed its reactor vessel integrity analyses (pressure-temperature limits, pressurized thermal shock, charpy USE) using a "hypothetical weld" placed in the center of the RV beltline region as the limiting material. The staff believes that this weld is not intended to exist in the construction of an actual APR1400 RV, but has been introduced as a means of providing additional conservatism in the APR1400 RV integrity analyses. Confirm that the aforementioned staff's understanding of is correct. If not, explain why the applicant's RV analyses include a circumferential weld at the center of the active core region that is not a part of the actual RV design.

Response
Yes, the staff's understanding is correct.

Issue #2

In Final Safety Analysis Report (FSAR) Section 5.3.2.3, the applicant provided pressurized thermal shock reference temperature (RT_{PTS}) value for the limiting reactor vessel (RV) beltline material, which is a weld conservatively assumed to be in the center of the RV beltline and subjected to a neutron fluence of 9.5 x 10^{19} n/cm^2 (E > 1.0 MeV). However, Title 10 of the Code of Federal Regulations (10 CFR) Section 50.61 requires that projected values of RT_{PTS} be provided for each RV beltline material. For the APR1400 design, the beltline materials to be considered should include, at a minimum, the lower shell section of the RV adjacent to the active height of the core and the two circumferential welds that connect the RV beltline ring forging (lower shell) to the bottom head and to the upper shell section (nozzles section).

Revise the FSAR to provide all values (i.e., initial RT_{NDT}, chemistry factors, fluence values, margins, ΔRT_{NDT}, etc.) used to calculate the RT_{PTS} for all RV materials which meet the definition in 10 CFR 50.61(a)(3) of being in the APR1400 RV beltine. The staff notes that the information requested is currently provided in Table 8-1 of APR1400 Technical Report APR1400-Z-M-NR-14008-P, Revision 0, "Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown."

Response
The Table 8-1 of the APR1400 Technical Report will be added to the Table 5.3-10, as shown below and the descriptions about the lower shell section will be added in FSAR Section 5.3.2.3.

Table 5.3-10 RT_{PTS} for the APR1400 Reactor Vessel Materials at EOL (60 Years)

Material	Initial RT_{NDT} (℃(°F))	Fluence Factor[1)]	Chemistry Factor[2)] (℃(°F))	ΔRT_{PTS} (℃(°F))	Margin[3)] (℃(°F))	RT_{PTS} (℃(°F))	Screening Criterion (℃(°F))
Lower Shell Course (beltline material)	-23.3 (-10)	1.51	11.1 (20)	16.8 (30.2)	16.8 (30.2)	10.2 (50.4)	132.2 (270)
Weld Material (G-2 & G-3)	-12.2 (10)	1.51	20.8 (37.5)	31.4 (56.6)	31.1 (56)	50.3 (122.6)	148.9 (300)
Head Flange	-12.2 (10)	negligible	24.4 (44.0)	NA[4)]	NA[4)]	NA[4)]	132.2 (270)
Vessel Flange	-12.2 (10)	negligible	24.4 (44.0)	NA[4)]	NA[4)]	NA[4)]	132.2 (270)
Inlet Nozzles	-12.2 (10)	negligible	24.4 (44.0)	NA[4)]	NA[4)]	NA[4)]	132.2 (270)
Outlet Nozzles	-12.2 (10)	negligible	24.4 (44.0)	NA[4)]	NA[4)]	NA[4)]	132.2 (270)
DVI Nozzles	-12.2 (10)	negligible	24.4 (44.0)	NA[4)]	NA[4)]	NA[4)]	132.2 (270)

Notes to Table 5.3-10:

1) Fluence Factor = $f^{(0.28-0.10\log f)}$, f = Fluence at Limiting Case, 9.5×10^{19} n/cm^2.

2) Chemistry Factor with Chemical Composition of Cu 0.03 wt% and Ni 1.00 wt% for the lower shell course; Cu 0.05 wt% and Ni 0.10 wt% for the weld material; and Cu 0.07 wt% and Ni 1.00 wt% for the flanges and nozzles.

3) Margin = $2\sqrt{\sigma_u^2 + \sigma_\Delta^2}$

where, $\sigma_u = 0$

σ_Δ = lesser of 17°F or 0.5xΔRT_{PTS} for the forging and lesser of 28°F or 0.5 x ΔRT_{PTS} for the weld material.

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

MCB Issue List Regarding APR1400 Technical Report APR1400-Z-M-NR-14008-P, Revision 0, "Pressure Temperature Limits Methodology for RCS Heatup and Cooldown"

Issue #1

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} for beltline forging in PTLR will be corrected to 68.4 °F.
- RT_{PTS} for limiting beltline material in FSAR will be corrected to 122.6°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.

Issue #2

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.