Response to Action Item 5-5 Section 5.3.1

MCB Issue List Regarding APR-1400, FSAR Section 5.3.1

Issue #1 (AI 5-5.11)

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 However, the MIL-E-18193B electrode specification was cancelled in 1989 and superseded by a different military specification, MIL-E-23765.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 specifications of 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.

Impact on DCD

Table 5.2-2 will be revised as indicated on the attached markup (See sheet 1 of the attachment).

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

There is no impact on any Technical, Topical or Environmental Reports.

Response to Action Item 5-5 Section 5.3.1

MCB Issue List Regarding APR-1400, FSAR Section 5.3.1

Issue #2 (AI 5-5.12)
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. (Please see the attached markup.)

Impact on DCD
FSAR Table 5.2-2 will be revised as indicated on the attached markup (See sheet 2 of the attachment).

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
There is no impact on any Technical, Topical or Environmental Reports.

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 Delete c. d.	a. SFA 5.1 E-7018, E-7016 b. SFA 5.5 E-8018-C3, E-8018-G, E-8016-G ~~c. MIL-E-18193 B-4~~ d. SFA 5.23 EA3 EA3(N) e. SFA 5.18 ER70S-6	Primary piping straight to the RV primary nozzle
P-1	P-8	a. NiCrFe filler metal 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	Buttering (NiCrFe filler metal) of J-grooves in hot leg pipe
P-3	P-3 Delete b.	a. SFA 5.5 (2) E-8016-C3, E-8018-G, E-8016-G ~~b. MIL-E-18193 B-4 (2)~~ c. SFA 5.23 EA3 (2)	RV upper shell to RV flange
P-3	P-8 EA3(N)(2)	a. NiCrFe filler metal 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	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

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

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

Response to Action Item 5-5 Section 5.3.1

MCB Issue List Regarding APR-1400, FSAR Section 5.3.1

Issue #3 (AI 5-5.13)

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 and Table 7-1 of the technical report (APR1400-Z-M-NR-14008-P) 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 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. Additionally, design maximum fluence values for withdrawal time determination are added in FSAR Table 5.3-7 and Table 7-1 of the technical report (APR1400-Z-M-NR-14008-P). On this definition, FSAR Table 5.3-7 and Table 7-1 of the technical report (APR1400-Z-M-NR-14008-P) will be revised for the consistency between two tables.

Response to Action Item 5-5 Section 5.3.1

Impact on DCD
Table 5.3-7 will be revised as indicated on the attached markup (See sheet 1 of the attachment).

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
Table 7-1 of the technical report (APR1400-Z-M-NR-14008-P) will be revised as indicated on the attached markup (Attachment). There is no impact on other Topical or Environmental Reports.

Response to Action Item 5-5 Section 5.3.1

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 x 10^{20} n/cm^2. This value is calculated using the end of life design maximum fluence at vessel inside surface (9.5 x 10^{19} n/cm^2 (E > 1.0 MeV)) of FSAR Table 4.3-7 which is added in FSAR Table 5.3-7 and lead factor (1.4). The lead factor (1.5→1.4) in Section 5.3.1.6.5 and the uncertainty of the best estimate value (±20%→±16.28%) in Section 5.3.1.6.6 will be revised.

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 56.5°F which is calculated in accordance with RG 1.99 Rev. 2 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.

Response to Action Item 5-5 Section 5.3.1

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

Case \ Withdrawal Sequence	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	15.3 [f)]	-	-	-
Determined withdrawal time for APR1400	6 [g)]	15	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 x 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 RT_{NDT} 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 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 and Table 7-1 of the technical report (APR1400-Z-M-NR-14008-P) will be revised as follows:

Response to Action Item 5-5 Section 5.3.1

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}	9.5×10^{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 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.

Impact on DCD

Table 5.3-7 will be revised as indicated on the attached markup (See sheet 1 of the attachment). Additionally, Section 5.3.1.6.5, and Section 5.3.1.6.6 will be revised as indicated on the attached markup (See sheet 2 of the attachment).

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

Table 7-1 of the technical report (APR1400-Z-M-NR-14008-P) will be revised as indicated on the attached markup (See sheet 3 of the attachment). There is no impact on other Topical or Environmental Reports.

Response to Action Item 5-5 Section 5.3.1

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, design maximum fluences, and the notes. See response to Issue #3, question 2 for the explanation on meeting ASTM E 185-82, Table 1.

Impact on DCD

Table 5.3-7 will be revised as indicated on the attached markup (See sheet 1 of the attachment).

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

Table 7-1 of the technical report (APR1400-Z-M-NR-14008-P) will be revised as indicated on the attached markup (See sheet 3 of the attachment). There is no impact on other Topical or Environmental Reports.

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)	9.5 × 10^{19}
E	44 °	Standby	- Delete	
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

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~~ 1.4 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.

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

design maximum fluence values

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 target fluence

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.

The design maximum fluence values in Table 5.3-7 are accurate within + 20 percent, -20 percent. The target fluence identified in Table 5.3-7 is an actual fluence which a capsule may experience until it is withdrawn.

AI 5-5.13_5.3.1_#3 Attachment (3/3)

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: Best estimate fluences (best estimate values of expected neutron fluence)

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

2) Best-estimated average value of expected neutron fluence level at specimen locations in each capsule with +16.28% uncertainty. ±16.28%

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