ArevaEPRDCPEm Resource

From: Eudy, Michael
Sent: Tuesday, April 08, 2014 4:53 PM
To: ArevaEPRDCPEm Resource
Cc: McLellan, Judith
Subject: FW: Rev 6 AIA Editorial Comments
Attachments: NRC-14-010.pdf

From: Eudy, Michael
Sent: Wednesday, March 26, 2014 2:41 PM
To: HOTTLE Nathan (AREVA) (Nathan.Hottle@areva.com)
Cc: Segala, John; Wunder, George; Xu, Jim; Istar, Ata; ArevaEPRDCDocsPE Resource
Subject: Rev 6 AIA Editorial Comments

Nathan,

With respect to your proposed FSAR markups for AIA, the staff has found some inconsistencies in AREVA letter NRC:14:010 dated March 24, 2014. We are providing these editorial comments as a courtesy and are not planning to issue an RAI nor make them items to be confirmed by staff in a future revision. Please let me know if you have any questions.

Michael A. Eudy - Project Manager
U.S. Nuclear Regulatory Commission
NRO/DNRL/LB3
301-415-3104

EPR FSAR, Revision 6-Interim:
See the following clerical comments:

1- Page 1.6-2, Table 1.6-1, fifth cell of “Title” column: should read “…..the U.S. EPR Topical Report, December 2012”

2- Pages 7.1-15, 7.1-41, 7.1-55 and 7.1-57, for consistency, “ANP-10266, Addendum A.” should read “ANP-10266, Addendum A, (Reference 42).”

3- Page 9.5-38, Section 9.5.1.6.5 Quality Assurance, “ANP-10266 (Reference 41).” should read “ANP-10266, (Reference 42).”

4- Page 17.5-2, Section 17.5.4 References, for consistency – first reference should read, “…..the U.S. EPR Topical Report,” AREVA NP Inc., December 2012.”

5- Throughout the EPR FSAR: in “References” sections, some of the referenced items were deleted, and were denoted as “Deleted.” If the reference item is “non-applicable,” it shall be denoted as “This reference item intentionally left blank” or, simply, “Intentionally left blank”

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AREVA

March 24, 2014
NRC:14:010

U.S. Nuclear Regulatory Commission
Document Control Desk
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U.S. EPR Design Certification Application - Closure of FSAR Group A Chapters 2, 13, 17 and AIA

Ref. 1: Letter, Pedro Salas (AREVA NP Inc.) to Document Control Desk (U.S. Nuclear Regulatory Commission), "Closure Plan for U.S. EPR Design Certification Application – FSAR Group A Chapters," NRC:13:092, December 20, 2013.

The purpose of this letter and enclosure is to communicate the closure by AREVA Inc. (AREVA) of FSAR Group A Chapters 2, 13, and 17, and Section 19.2.7, "Beyond Design Basis Large Commercial Aircraft Impact Assessment (AIA)," of the U.S. EPR Design Certification application. Reference 1 provided the closure plan for the U.S. EPR FSAR Group A chapters. AREVA selected chapters 2, 5, 8, 10, 13, and 17 along with the AIA as the Group A FSAR chapters for near-term closure. AREVA committed to support the completion of the Group A chapters in calendar year 2014, which includes supporting the review by the ACRS. FSAR Chapters 2, 13, 17, and AIA have been closed by a chapter "freeze" procedure, which provides assurance that the FSAR chapters are complete and accurate, and no further changes are anticipated.

The final FSAR mark-ups for Chapters 2, 13, 17, AIA and associated conforming changes to other chapters are enclosed. This FSAR information will be available for confirmation in FSAR revision (Revision 6) that will be provided to the NRC by April 3, 2014.

AREVA is in the process of finalizing the remaining Group A FSAR chapters (5, 8, and 10) and will provide details on the final changes to those chapters in a subsequent communication.

If you have any questions related to this information, please contact Len Gucwa by telephone at (434) 832-3466, or by email at Len.Gucwa.ext@areva.com.

Sincerely,

JT Guerra/for

Pedro Salas, Director
Regulatory Affairs
AREVA Inc.

cc: G. F. Wunder
J. P. Segala

Enclosure 1

FINAL MARKUPS FOR GROUP A CHAPTERS FOR FSAR REVISION 6

FSAR Chapter / Section	
2	Site Characteristics
13	Conduct of Operations
17*	Quality Assurance and Reliability Assurance
19.2.7	Beyond Design Basis Large Commercial Aircraft Impact Assessment

* - excludes FSAR Section 17.4, “Reliability Assurance Program”

U.S. EPR Final Safety Analysis Report Markups

For Chapter 2 Freeze

Table 1.8-2—U.S. EPR Combined License Information Items
Sheet 6 of 40

Item No.	Description	Section
2.5-3	A COL applicant that references the U.S. EPR design certification will compare the final strain-dependent soil profile with the U.S. EPR design soil parameters and verify that the site-specific seismic response is enveloped by the CSDRS and the soil profiles discussed in Sections 2.5.2, 2.5.4.7 and 3.7.1 and summarized in Table 3.7.1-6, Table 3.7.1-8 and Table 3.7.1-9.	2.5.2.6 2.5.4.7
2.5-4	A COL applicant that references the U.S. EPR design certification will verify that site-specific foundation soils beneath the foundation basemats of Seismic Category I structures have the capacity to support the bearing pressure with a factor of safety of 3.0 under static conditions or 2.0 under dynamic conditions, whichever is greater.	2.5.4.10.1
2.5-5	A COL applicant that references the U.S. EPR design certification will investigate site-specific surface and subsurface geologic, seismic, geophysical, and geotechnical aspects within 25 miles around the site and evaluate any impact to the design. The COL applicant will evaluate the potential for surface deformation at the site~~demonstrate that no capable faults exist at the site~~ in accordance with the requirements of 10 CFR 100.23 and of 10 CFR 50, Appendix S. If the potential for surface deformation is present at the site, the COL applicant will evaluate the effects of potential surface deformation on the design and operation of the U.S. EPR.~~If non-capable surface faulting is present under foundations for safety-related structures, the COL applicant will demonstrate that the faults have no significant impact on the structural integrity of safety-related structures, systems, or components~~.	2.5.3
2.5-6	A COL applicant that references the U.S. EPR design certification will present site-specific information about the properties and stability of soils and rocks that may affect the nuclear power plant facilities under both static and dynamic conditions, including the vibratory ground motions associated with the CSDRS and the site-specific SSE.	2.5.4
2.5-7	A COL applicant that references the U.S. EPR design certification will verify that the predicted tilt settlement value of ½ in per 50 ft in any direction across the foundation basemat of a Seismic Category I structure is not exceeded. Settlement values larger than this may be demonstrated acceptable by performing additional site-specific evaluations.	2.5.4.10.2

Table 2.1-1—U.S. EPR Site Design Envelope
Sheet 3 of 7

U.S. EPR Site Design Envelope	
Maximum Ground Water	3.3 ft below grade
Minimum Coefficient of Static Friction for Category I Structures (representative of all interfaces between basemat and soil)	0.5
NAB Coefficients of Friction	$0.5 \le \mu \le 0.7$
EPGB Coefficient of Side Wall Friction	$\mu \ge 0.36$
Inventory of Radionuclides Which Could Potentially Seep Into the Groundwater (Refer to Section 11.2)	
~~See Table 2.1-2—Bounding Values for Component Radionuclide Inventory~~	
Flood Level (Refer to Section 2.4)	
Maximum Flood (or Tsunami)	1 ft below grade
Wind (Refer to Section 3.3)	
Maximum Speed (Other than Tornado and Hurricane)	145 mph (Based on 3-second gust at 33 ft above ground level and factored for 50-yr mean recurrence interval)
Importance Factor	1.15 (Safety-related structures for 100-year mean recurrence interval.)
Tornado (Refer to Sections 3.3 and 3.5)	
Maximum Pressure and Rate of Drop	1.2 psi at 0.5 psi/s
Maximum Rotational Speed	184 mph
Maximum Translational Speed	46 mph

Table 2.1-1—U.S. EPR Site Design Envelope
Sheet 6 of 7

U.S. EPR Site Design Envelope					
Main Control Room/Technical Support Center Intake Atmospheric Dispersion Factors for Onsite Accident Dose Analysis (χ/Q)					
Time Period	Vent Stack Base	Releases via Safeguard Building Canopy	Equipment Hatch Releases via Material Lock	~~Depressurization Shaft Releases~~	Main Steam Relief Train Silencer
0–2 hours (s/m^3)	1.93E-03	6.52E-03	(See Note 6)	~~(See Note 6)~~	4.30E-03
2–8 hours (s/m^3)	1.73E-03	5.68E-03	(See Note 6)	~~(See Note 6)~~	3.71E-03
8–24 hours (s/m^3)	6.74E-04	2.34E-03	(See Note 6)	~~(See Note 6)~~	1.46E-03
1–4 days (s/m^3)	5.12E-04	1.63E-03	(See Note 6)	~~(See Note 6)~~	1.12E-03
4–30 days (s/m^3)	4.72E-04	1.50E-03	(See Note 6)	~~(See Note 6)~~	1.03E-03

Table 2.1-1—U.S. EPR Site Design Envelope
Sheet 7 of 7

U.S. EPR Site Design Envelope					
Main Control Room/Technical Support Center Unfiltered Inleakage Atmospheric Dispersion Factors for Onsite Accident Dose Analysis (χ/Q)					
Time Period	Vent Stack Base	Releases via Safeguard Building Canopy	Equipment Hatch Releases via Material Lock	~~Depressurization Shaft Releases~~	Main Steam Relief Train Silencer
0–2 hours (s/m^3)	4.30E-03	1.67E-02	(See Note 6)	~~(See Note 6)~~	1.76E-02
2–8 hours (s/m^3)	3.71E-03	1.47E-02	(See Note 6)	~~(See Note 6)~~	1.48E-02
8–24 hours (s/m^3)	1.46E-03	5.96E-03	(See Note 6)	~~(See Note 6)~~	5.88E-03
1–4 days (s/m^3)	1.12E-03	4.28E-03	(See Note 6)	~~(See Note 6)~~	4.55E-03
4–30 days (s/m^3)	1.03E-03	3.89E-03	(See Note 6)	~~(See Note 6)~~	4.16E-03

Note:

1. The effect of the extreme liquid winter precipitation event on roof loads is negligible due to the lack of parapets.
2. Deleted.
3. By definition, zero percent exceedance temperature values exclude peaks of temperatures less than two hours in duration. The zero percent exceedance temperature values are based on conservative estimates of 100-year return period values and historic extreme values, whichever is bounding.
4. For maximum values, data from the summer months or June, July, and August are used. For minimum values, data from the winter months of December, January, and February are used.
5. The shear wave velocities (strain compatible best estimate average values directly beneath the foundation basemat) of soft, medium, and hard soils are 1000 ft/sec, 1640 ft/sec, and greater than or equal to 6601 ft/sec, respectively.

6. The atmospheric dispersion parameters for the equipment hatch ~~and depressurization shaft releases~~ are bounded by the parameters for the releases via the Safeguard Building canopy.

7. Soil densities provided in this table are associated with the structural design of basemats and walls below grade. Soil densities used in dynamic soil structure interaction analyses are defined in terms of the value of the shear wave velocity in Table 3.7.2-9. The shear wave velocity layers and depths are defined in Table 3.7.1-6, Table 3.7.1-8 and Table 3.7.1-9.

8. Minimum angle of internal friction is associated with the soil's ability to develop the minimum coefficient of static friction.

9. Maximum angle of internal friction is associated with the passive soil pressure coefficient so that soil pressure acting on walls below grade does not exceed capacity. If the maximum angle of internal friction is higher than 30 degrees, a site-specific analysis will be performed using the site-specific soil parameters and site-specific SSE to demonstrate that the capacity of the below grade walls is not exceeded.

~~Table 2.1-2—Bounding Values for Component Radionuclide Inventory~~

Table 2.1-2---Deleted

~~Nuclide~~	~~Activity (μCi/g)~~	~~Nuclide~~	~~Activity (μCi/g)~~
~~Br-84~~	~~1.7E-02~~	~~Y-91~~	~~8.1E-05~~
~~I-129~~	~~4.6E-08~~	~~Y-92~~	~~1.4E-04~~
~~I-131~~	~~7.4E-01~~	~~Y-93~~	~~6.5E-05~~
~~I-132~~	~~3.7E-01~~	~~Zr-95~~	~~9.3E-05~~
~~I-133~~	~~1.3E+00~~	~~Nb-95~~	~~9.4E-05~~
~~I-134~~	~~2.4E-01~~	~~Mo-99~~	~~1.1E-01~~
~~I-135~~	~~7.9E-01~~	~~Tc-99~~	~~1.1E-09~~
~~Cs-134~~	~~1.7E-01~~	~~Tc-99m~~	~~4.6E-02~~
~~Cs-136~~	~~5.3E-02~~	~~Ru-103~~	~~7.8E-05~~
~~Cs-137~~	~~1.1E-01~~	~~Ru-106~~	~~2.7E-05~~
~~Cr-51~~	~~2.0E-03~~	~~Ag-110m~~	~~2.0E-07~~
~~Mn-54~~	~~1.0E-03~~	~~Te-129m~~	~~1.5E-03~~
~~Fe-55~~	~~7.6E-04~~	~~Te-129~~	~~2.4E-03~~
~~Fe-59~~	~~1.9E-04~~	~~Te-131m~~	~~3.7E-03~~
~~Co-58~~	~~2.9E-03~~	~~Te-131~~	~~2.6E-03~~
~~Co-60~~	~~3.4E-04~~	~~Te-132~~	~~4.1E-02~~
~~Zn-65~~	~~3.2E-04~~	~~Ba-140~~	~~6.2E-04~~
~~W-187~~	~~1.8E-03~~	~~La-140~~	~~1.6E-04~~
~~Rb-88~~	~~1.0E+00~~	~~Ce-141~~	~~8.9E-05~~
~~Sr-89~~	~~6.4E-04~~	~~Ce-143~~	~~7.6E-05~~
~~Sr-90~~	~~3.3E-05~~	~~Ce-144~~	~~6.9E-05~~
~~Sr-91~~	~~1.0E-03~~	~~Np-239~~	~~8.7E-04~~
~~Y-91m~~	~~5.2E-04~~	~~H-3~~	~~1.0E+00~~

Table 2.1-3—Deleted

Table 2.1-4—Deleted

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whichever is greater. Snow pack and snowfall are adjusted for density differences and ground level values are adjusted to represent appropriate weights on roofs.

A COL applicant that references the U.S. EPR design certification will provide site-specific characteristics for regional climatology.

2.3.1.2 Meteorological Data for Evaluating the Ultimate Heat Sink

As described in Section 9.2.5, the ultimate heat sink (UHS) is designed to operate for a nominal 30 days following a LOCA without requiring any makeup water to the source, or it must be demonstrated that replenishment or use of an alternate or additional water supply can provide continuous capability of the heat sink to perform its safety-related functions. The tower basin contains a minimum 72-hour supply of water.

Meteorological conditions resulting in the maximum evaporative and drift loss of water for the UHS over a 72 hour period are presented in Table 9.2.5-3. The UHS cooling tower basin is designed considering the wet bulb temperature in Table 9.2.5-2 and maintains its cooling function for the Table 9.2.5-3 meteorological conditions.

Water makeup to the UHS cooling tower basin beyond 72 hours is site-specific. As described in Section 9.2.5.3, the COL applicant will describe the means for providing UHS makeup sufficient to meet the maximum evaporative and drift water loss after 72 hours through the remainder of the 30 day period consistent with RG 1.27.

Meteorological conditions resulting in minimum water cooling are presented in Table 9.2.5-4. These conditions reflect a 1 day period where evaporative cooling is at a minimum. The UHS heat loads peak and decline within the first day, such that extending the 1 day meteorological profile for 5 consecutive days does not cause the UHS cooling tower basin water temperature to exceed the maximum temperature of 95°F listed in Table 9.2.5-2. The potential for water freezing in the UHS basin and site-specific makeup water source~~water storage facility~~ is addressed in Section 2.4.

2.3.2 Local Meteorology

A COL applicant that references the U.S. EPR design certification will provide site-specific characteristics for local meteorology.

2.3.3 Onsite Meteorological Measurement Program

A COL applicant that references the U.S. EPR design certification will provide the site-specific, onsite meteorological measurement program.

2.3.4 Short-Term Atmospheric Dispersion Estimates for Accident Releases

Atmospheric dispersion factors (χ/Q values) considered to be representative of potential future nuclear plant sites in the U.S. were used to calculate the consequences from postulated accidental releases of radioactive and hazardous materials.

χ/Q values for ground-level releases were calculated at the exclusion area boundary (EAB) and at the low population zone (LPZ) for appropriate time periods up to 30 days after an accident. The accident χ/Q values were either extracted from Reference 1 or were calculated following the methodology in NRC RG 1.145. The ground-level χ/Q values used for short-term atmospheric dispersion dose analyses at the EAB and LPZ receptor locations are provided in Table 2.1-1.

In addition to the offsite accident consequences evaluated at the EAB and LPZ, onsite accident dose consequences at the Main Control Room (MCR) and Technical Support Center (TSC) were evaluated. MCR and TSC χ/Q values, provided in Table 2.1-1 for the main air supply and the unfiltered inleakage, are used for these analyses from potential post-accident release points. These multiple potential release points affecting the MCR and the TSC include:

- The vent stack.
- Main steam relief train (MSRT) releases for steam generator overpressure protection.
- Safeguard Building roofs via the Safeguard Building canopies.
- An open equipment hatch.
- ~~Safeguard Building depressurization shaft.~~

The information in these tables conforms to the guidance in RG 1.23, RG 1.145, and RG 1.194. Conformance with RG 1.78 is addressed in Sections 2.2, 6.4, 9.4, and 9.5.

The input variables used in calculating the accident χ/Q values are shown in Table 2.3-1—ARCON96 Input Parameters for Control Room Air Intake χ/Q Values and Table 2.3-2—ARCON96 Input Parameters for Unfiltered Inleakage Control Room χ/Q Values.

Figure 2.3-1—U.S. EPR Release Points, Control Room Air Intakes, and Unfiltered Inleakage Locations provides the relative locations of the release points and the control room air intakes. Section 15.0.3 addresses the dose calculation methodology for accident analyses.

A COL applicant that references the U.S. EPR design certification will confirm that site-specific χ/Q values, based on site-specific meteorological data, are bounded by those specified in Table 2.1-1 at the EAB, LPZ, and the control room.

For site-specific χ/Q values that exceed the bounding χ/Q values, a COL applicant that references the U.S. EPR design certification will demonstrate that the radiological consequences associated with the controlling design basis accident continue to meet the dose reference values given in 10 CFR 50.34 and the control room operator dose limits given in GDC 19 using site-specific χ/Q values.

A COL applicant that references the U.S. EPR design certification will provide a description of the atmospheric dispersion modeling used in evaluating potential design basis events to calculate concentrations of hazardous materials (e.g., flammable or toxic clouds) outside building structures resulting from the onsite and/or offsite airborne releases of such materials.

2.3.5 Long-Term Atmospheric Dispersion Estimates for Routine Releases

A COL applicant that references the U.S. EPR design certification will provide the site-specific, long-term diffusion estimates for routine releases. In developing this information, the COL applicant should consider the guidance provided in RG 1.23, RG 1.109, RG 1.111, and RG 1.112. The maximum annual average χ/Q value at the site boundary, provided in Table 2.1-1, is used to calculate radionuclide concentrations associated with routine gaseous effluent releases, addressed in Section 11.3~~11.3~~, for comparison with environmental release limits and dose limits given in 10 CFR 20. If a reactor site has an annual average χ/Q value that exceeds the reference value, then a site-specific evaluation will be performed.

A COL applicant that references the U.S EPR design certification will also provide estimates of annual average atmospheric dispersion (χ/Q values) and deposition (D/Q values) for 16 radial sectors to a distance of 50 miles from the plant as part of its environmental assessment.

2.3.6 References

1. EPRI ALWR Utility Requirements Document, “Electric Power Research Institute Advanced Light Water Reactor Utility Requirements Document,” Volume II-Revision 8, March 1999.

Table 2.3-1—ARCON96 Input Parameters for Control Room Air Intake χ/Q Values
Sheet 1 of 2

Parameter	Value(s)
Wind instrument heights	Site specific
Wind speed units of measure	Site specific
Release mode	Ground level (used for each pathway)
Building area	Assumed to be zero for each pathway
Vertical velocity	Assumed to be zero for each pathway
Stack flow	Assumed to be zero for each pathway
Stack radius	Assumed to be zero for each pathway
Terrain elevation difference	Assumed to be zero for each pathway
Direction to source	Site specific; EPR FSAR used the direction that produced the highest χ/Q values
Initial diffusion coefficients	Assumed to be zero for each pathway
Minimum wind speed value for ARCON96	0.5 m/sec
Surface roughness for ARCON96	0.2
Sector averaging constant for ARCON96	4.3
Wind direction window for ARCON96	90 degrees
Control Room air intake location employed in analysis	Intake closest to stack
Control Room air intake elevation	32.1 meters (Mid-point of intake)
Control Room air intake horizontal distance to stack base	69.0 meters
Control Room air intake horizontal distance to Main Steam Relief Train, via Silencer: SG-4 Silencer to MCR Div. 3 Air Intake (AI) SG-3 Silencer to MCR Div. 3 AI SG-1 Silencer to MCR Div. 3 AI SG-2 Silencer to MCR Div. 3 AI	 53.0 meters 46.0 meters 78.0 meters 71.0 meters
Control Room air intake horizontal distances to Canopy exhausts (referred to as the Canopy release point in the present application) ~~1) Near depressurization shaft (Safeguard Building Div. 4)~~ ~~2)~~ Southeast side of SAB Div. 4	 ~~30.1 meters~~ 65.3 meters

Table 2.3-1—ARCON96 Input Parameters for Control Room Air Intake χ/Q Values
Sheet 2 of 2

Parameter	Value(s)
Control Room air intake horizontal distance to Material Lock (for the Equipment Hatch release)	97.5 meters
~~Control Room air intake horizontal distance to the depressurization shaft of Safeguard Building Div. 4~~	~~31.4 meters~~
Release heights	Silencer – 33.9 meters Stack – 32.1 meters(1) Canopy Pt. 1 – 15.5 meters Canopy Pt. 2 – 11.5 meters elevation Material Lock (for Equipment Hatch release) – 32.1 meters ~~Depressurization Shaft – 7 meters~~

Note:

1. Stack release height assumed to be the same as the mid-point of the control room air intake.

Table 2.3-2—ARCON96 Input Parameters for Unfiltered Inleakage Control Room χ/Q Values
Sheet 1 of 2

Parameter	Value(s)
Wind instrument heights	Site specific
Wind speed units of measure	Site specific
Release mode	Ground level (used for each pathway)
Building area	Assumed to be zero for each pathway
Vertical velocity	Assumed to be zero for each pathway
Stack flow	Assumed to be zero for each pathway
Stack radius	Assumed to be zero for each pathway
Terrain elevation difference	Assumed to be zero for each pathway
Direction to source	Site specific; EPR FSAR used the direction that produced the highest χ/Q values
Initial diffusion coefficients	Assumed to be zero for each pathway
Minimum wind speed value for ARCON96	0.5 m/sec
Surface roughness for ARCON96	0.2
Sector averaging constant for ARCON96	4.3
Wind direction window for ARCON96	90 °F
Unfiltered inleakage ~~air intake~~ elevation	32.1 meters
Unfiltered inleakage ~~air intake~~ horizontal distance to stack base	46.0 meters (same distance as SG-3 Silencer to MCR Div. 3 Air Intake)
Unfiltered inleakage ~~air intake~~ horizontal distance to Main Steam Relief Train, via Silencer: SG-1 Silencer SG-2 Silencer SG-3 Silencer SG-4 Silencer	 70.0 meters 62.0 meters 22.0 meters 32.0 meters
Unfiltered inleakage ~~air intake~~ horizontal distances to Canopy exhausts (referred to as the Canopy release point in the present application) ~~1) Near depressurization shaft (Safeguard Building Div. 4)~~ ~~2)~~ Southeast side of SAB Div. 4	 ~~12.7 meters~~ 45.3 meters

Table 2.3-2—ARCON96 Input Parameters for Unfiltered Inleakage Control Room χ/Q Values
Sheet 2 of 2

Parameter	Value(s)
Unfiltered inleakage ~~air intake~~ horizontal distance to Material Lock (for the Equipment Hatch release)	75.2 meters
~~Unfiltered inleakage air intake horizontal distance to the depressurization shaft of Safeguard Building Div. 4~~	~~17.3 meters~~
Release heights	Silencer – 33.9 meters Stack – 33.9 meters(1) Canopy Pt. 1 – 15.5 meters Canopy Pt. 2 – 11.5 meters elevation Material Lock (for Equipment Hatch release) – 32.1 meters ~~Depressurization Shaft – 7.0 meters~~

Note:

1. The slant distance from the stack to the ingress point is approximately the same as the slant distance from the SG-3 silencer to the control room air intake; therefore, the SG-3 run, with a release height of 33.9 meters is also used for the stack scenario.

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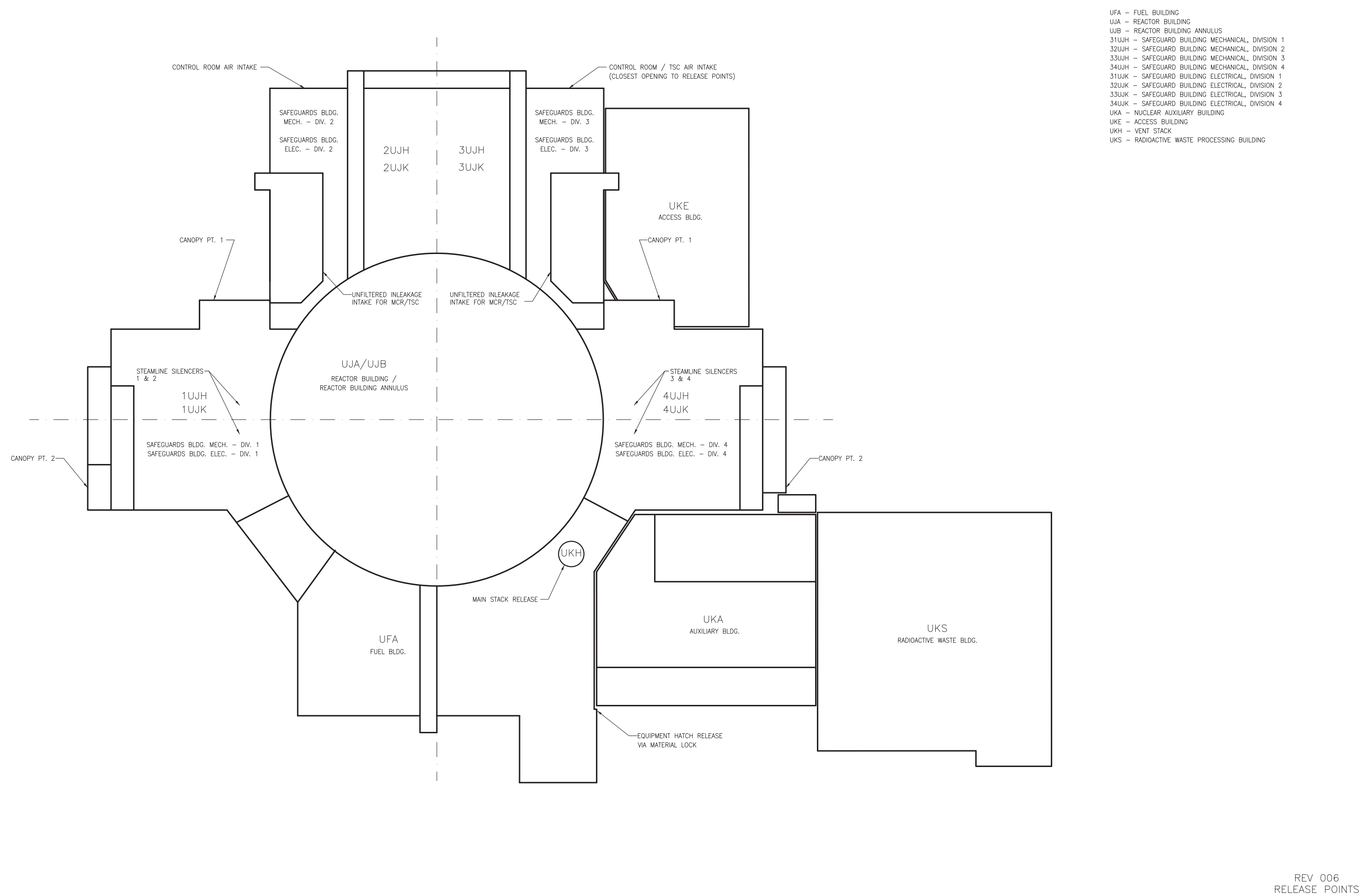
Figure 2.3-1—U.S. EPR Release Points, ~~and~~ Control Room Air Intakes, and Unfiltered Inleakage Locations



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- Emergency Power Generator Building (EPGB)—center of basemat (Figures 3.7.2-101, 3.7.2-102, and 3.7.2-103) and +51 ft, 6 in. (Figures 3.7.2-148, 3.7.2-149, and 3.7.2-150.
- Essential Service Water Building (ESWB)—Pump Slab on elevation +14 ft, 0 in (Figures 3.7.2-107, 3.7.2-108, and 3.7.2-109) and Fan Deck on elevation +63 ft, 0 in (Figures 3.7.2-104, 3.7.2-105, and 3.7.2-106).

9. Exceedances will require additional evaluation to determine if safety-related structures, systems, and components of the U.S. EPR at the location(s) in question will be affected.

As a result of the reconciliation process described above, the applicant may redesign selected features of the U.S. EPR, as required. Redesigned features will be identified as exceptions to the FSAR and addressed by the COL applicant.

2.5.3 Surface ~~Faulting~~Deformation

~~No surface faulting~~The potential for surface deformation is considered to be ~~present under foundations for Seismic Category I structures in the U.S. EPR~~absent from the site (GDC 2).

A COL applicant that references the U.S. EPR design certification will investigate site-specific surface and subsurface geologic, seismic, geophysical, and geotechnical aspects within 25 miles around the site and evaluate any impact to the design. The COL applicant will ~~demonstrate that no capable faults exist at the site~~evaluate the potential for surface deformation at the site in accordance with the requirements of 10 CFR 100.23 and of 10 CFR 50, Appendix S. If the potential for surface deformation is present at the site, the COL applicant will evaluate the effects of potential surface deformation on the design and operation of the U.S. EPR.~~If non-capable surface faulting is present under foundations for safety-related structures, the COL applicant will demonstrate that the faults have no significant impact on the structural integrity of safety-related structures, systems, or components.~~

2.5.4 Stability of Subsurface Materials and Foundations

The stability of subsurface materials under the foundations for Seismic Category I structures is demonstrated in Section 3.8.5 for the U.S. EPR soil profiles described in Section 3.7.1 and Section 3.7.2. As described in Section 3.8.5, lateral soil pressure loads under saturated conditions are considered for the design of below-grade walls. Soil loads are based on the parameters described in Section 2.5.4.2.

A COL applicant that references the U.S. EPR design certification will present site-specific information about the properties and stability of soils and rocks that may affect the nuclear power plant facilities under both static and dynamic conditions, including the vibratory ground motions associated with the CSDRS and the site-specific SSE.

Earthquake induced soil pressures for the design of the U.S. EPR are developed in accordance with Section 3.5.3 of ASCE 4-98 (Reference 2). Maximum ground water and maximum flood elevations used for determining lateral soil loads for the U.S. EPR are as specified in Table 2.1-1.

A COL applicant that references the U.S. EPR design certification will reconcile the site-specific soil and backfill properties with those used for design of U.S. EPR Seismic Category I structures and foundations described in Section 3.8.

2.5.4.3 Foundation Interfaces

Foundation interfaces with underlying materials are site specific and will be addressed by the COL applicant. The COL applicant will confirm that the site soils and backfill material have (1) minimum sliding coefficient of friction of 0.5, (2) adequate shear strength to provide adequate static and dynamic bearing capacity, (3) adequate elastic and consolidation properties to satisfy the limits on settlement described in Section 2.5.4.10.2, (4) adequate dynamic properties (i.e., shear wave velocity and strain-dependent modulus-reduction and hysteretic damping properties), and (5) properties so that the earthquake design loading on the below grade walls is not exceeded (i.e., the site-specific angle of internal friction, unit soil weight, and seismic wall movements do not cause design limits of the walls to be exceeded because of the passive lateral earth pressure on the walls to support the Seismic Category I structures of the U.S. EPR under earthquake loading).

2.5.4.4 Geophysical Surveys

Geophysical surveys are site specific and will be addressed by the COL applicant.

2.5.4.5 Excavations and Backfill

Excavations and backfill are site-specific and will be addressed by the COL applicant. Additional backfill requirements are identified in Section 3.8.5.4. Mud mats may be provided under foundations for ease of construction. Mud mats may be designed as structural plain concrete elements on a site-specific basis in accordance with ACI 318 (Reference 3).

2.5.4.6 Ground Water Conditions

Ground water conditions are described in Section 2.4 and provided in Table 2.1-1 for the U.S. EPR. Ground water conditions are considered in the structural design of the U.S. EPR, as described in Section 3.8. However, groundwater conditions are not explicitly considered in the SSI analyses described in Section 3.7.1 and Section 3.7.2.

The COL applicant will address site-specific ground water conditions.

~~The 40-foot extension for the grid of borings is established from a Boussinesq analysis of the zone of influence of the foundation basemat which shows that the net change in the effective vertical overburden stress is less than 7 percent at a distance of 40 feet from the edge of the foundation basemat.~~ The grid need not be of equal spacing in the two orthogonal directions, but it should be oriented in accordance with the true dip and strike of the rock. If geologic conditions are such that true dip and strike are not obvious, or if the dip is practically flat, then the orientation of the grid can be consistent with the major orthogonal lines of the NI Common Basemat Structures.

The depth of borings should be determined on the basis of the geologic conditions. Borings should be extended to a depth sufficient to define the site geology and to sample materials that may swell during excavation, may consolidate subsequent to construction, may be unstable under earthquake loading, or whose physical properties would affect foundation behavior or stability. At least one-fourth of the primary borings should penetrate sound rock or, for a deep soil site, to a maximum depth of 250 feet below the foundation basemat. At this depth of 250 feet, the change in the vertical stress during or after construction for the combined foundation loading is less than 10 percent of the in-situ effective overburden stress. Other primary borings may terminate at a depth of 160 feet below the foundation (i.e., equal to the equivalent radius of the structure). It is recommended that the shear wave velocity should be measured to a depth of 350 ft to 500 ft beneath the foundation basemat of the NI Common Basemat Structures. Thus, a limited number of borings should penetrate significantly deeper than the 250 ft criterion cited above.

2.5.4.10.5 Site Investigation for Non-uniform Sites

At sites that are judged to be non-uniform, potentially non-uniform, highly variable or potentially highly variable based on not meeting the criteria stated in Section 2.5.4.10.3, the investigation effort may have to be extended to determine if the site is acceptable for the U.S. EPR.

The U.S. EPR foundation/structural system for the NI Common Basemat Structures has significant margin. Therefore, it is expected that all but the most variable of sites will meet the criteria stated in Section 2.5.4.10.3. As stated in RG 1.132, where variable conditions are found, the spacing of boreholes should be closer to adequately define the media properties and their variability. Where cavities or other discontinuities of engineering significance may occur, the normal exploratory work should be supplemented by secondary borings or soundings at a spacing close enough to detect such features.

~~The depth of the secondary borings is 160 feet below the foundation basemat of the NI Common Basemat Structures. At this depth, the maximum change in vertical stress during or after construction is about 11 percent of the in-situ effective overburden stress. The depth of borings should be extended beyond 160 feet if the geologic~~

~~investigation indicates the possible presence of karst conditions, under-consolidated clays, loose sands, intrusive dikes, or other forms of geologic impacts at depth greater than 160 feet.~~

2.5.4.11 Design Criteria

Section 3.8.5 provides design criteria and design methods used in analysis and design of foundations, including a description of computer programs used in the analyses and a description of soil loads on embedded walls.

2.5.4.12 Techniques to Improve Subsurface Conditions

Techniques used for improving subsurface conditions are site specific and will be addressed by the COL applicant.

2.5.5 Stability of Slopes

No slope failure potential is considered in the design of safety-related SSC in the U.S. EPR.

A COL applicant that references the U.S. EPR design certification will evaluate site-specific information concerning the stability of earth and rock slopes, both natural and manmade (e.g., cuts, fill, embankments, dams, etc.), of which failure could adversely affect the safety of the plant. As noted in Section 3.7.1, the evaluation of slope stability is performed for the seismic level of the site-specific GMRS.

2.5.6 References

1. NUREG/CR-6372, “Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts,” U.S. Nuclear Regulatory Commission, November 1997.

2. ASCE 4-98 Standard, “Seismic Analysis of Safety-Related Nuclear Structures and Commentary,” American Society of Civil Engineers, 1999.

3. ACI 318-2005, “Building Code Requirements for Structural Concrete and Commentary,” ACI Committee 318, American Concrete Institute, 2005.

4. NUREG/CR-0693, “Seismic Input and Soil–Structure Interaction,” Final Report, U.S. Nuclear Regulatory Commission, January 1979.

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13.3 Emergency Planning

A COL applicant that references the U.S. EPR design certification will provide a site-specific emergency plan in accordance with 10 CFR 50.47 and 10 CFR 50 Appendix E. Emergency planning is, in part, within the scope of a COL applicant. Design features, facilities, functions and equipment that are technically relevant to the design and are not site-specific, and which affect some aspect of emergency planning or the capability of a licensee to cope with plant emergencies are described in this section.

A COL applicant that references the U.S. EPR design certification will address the Requested Information in Fukushima Recommendation 9.3 regarding Emergency Preparedness Communications and Staffing as outlined in Enclosure 5 of the request for additional information pursuant to the 10 CFR 50.54(f) letter dated March 12, 2012 (ML12053A340).

A space of at least 1875 ft^2 suitable for a technical support center (TSC), which demonstrates compliance with the design requirements of NUREG-0696, Section 2.4 (Reference 1) for staffing levels of 25 persons (20 utility and 5 NRC) at 75 ft^2 per person, and Revision 1 of NUREG-0654/FEMA REP-1 (Reference 2), is provided within the integrated operations area adjacent to the main control room (MCR). This space is within the Safeguard Building. It is also within the control room envelope (CRE) which maintains habitability during normal, off-normal and emergency conditions; refer to Figure 6.4-1—Control Room Envelope Plan View 1 and Figure 6.4-2—Control Room Envelope Plan View 2. A detailed description of CRE habitability, including radiological protective provisions, is provided in Section 6.4. The control room air conditioning system is described in Section 9.4.1.

Voice communications between the TSC and the plant, local and offsite emergency response facilities, local and state governments and the NRC are provided by the plant telephone, paging and radio systems. These are described in Section 9.5.2.2.1 through Section 9.5.2.2.4.

Data communications within the TSC is provided through the process information and control system (PICS), which is described in Section 7.1.1.3.2. This non-safety related digital I&C system provides a screen-based interface capable of monitoring plant parameters during: normal, off-normal and emergency conditions. It electronically provides MCR safety parameter information to the TSC and to the NRC through the emergency response data system (ERDS). Safety-related information systems are described in detail in Section 7.5, with accident monitoring systems described in Section 7.5.1.2 and information systems provided in the emergency response facilities described in Section 7.5.1.3.

Space suitable for an operational support center (OSC), which demonstrates conformance with the design requirements for staffing levels consistent with current operating practices of NUREG-0654/FEMA REP-1 Revision 1 (Reference 2), is provided within the Access Building. This building also contains a personnel decontamination area. Adequate voice communications in these facilities is provided by the plant telephone, paging and radio systems as described in Section 9.5.2.2.1 through Section 9.5.2.2.4. The Access Building is described in Section 1.2 and Section 12.3.1.6.

Next File

13.8 References

1. NUREG-0696, "Functional Criteria for Emergency Response Facilities," U.S. Nuclear Regulatory Commission, February 1981.

2. NUREG-0654/FEMA REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, U.S. Nuclear Regulatory Commission, November 1980.

3. Deleted.~~Letter from Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), "U.S. EPR Vital Equipment List (Safeguards Information)," dated November 30, 2007.~~

4. Letter OG-1789, Tony Stallard, Chairman, B&WOG Operator Support Committee, to Chief, Reactor Systems Branch (NRC), "Transmittal of B&W Owners Group Emergency Operating Procedures Technical Bases Document, Revision 9," dated April 26, 2000 and attachments (ML003711891).

5. Letter from Richards, Stuart A. (NRC) to Kelly, Michael, Chairman, B&W Owners Group Operator Support Committee, "Completion of Review of the Babcock and Wilcox Emergency Operating Procedures Guidelines (TAC No. M54946)," with attachment, dated November 5, 1999.

6. NUREG-0711, "Human Factors Engineering Program Review Model," Revision 2, U.S. Nuclear Regulatory Commission, February 2004.

7. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980.

8. NUREG-0737, "Clarification of TMI Action Plan Requirements: Supplement 1," U.S. Nuclear Regulatory Commission, January 1983.

9. ANS 3.2-1994, "Administrative Controls and Quality Assurance for the Operational Phase of NPPs," American Nuclear Society, 1994.

10. NUREG-0800, "Standard Review Plan", Section 13.5.2.1, "Operating and Emergency Operating Procedures, Appendix A, Review Procedures for the Evaluation of Procedures Generation Packages," Revision 2, U.S. Nuclear Regulatory Commission, March 2007.

11. NUREG-1358, "Lessons Learned from the Special Inspection Program for Emergency Operating Procedures," Supplement 1, U.S. Nuclear Regulatory Commission, 1992.

12. NUREG-1358, "Lessons Learned From the Special Inspection Program for Emergency Operating Procedures," U.S. Nuclear Regulatory Commission, April 1989.

13. NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," U.S. Nuclear Regulatory Commission, August 1982.

14. ANP-10295P, Revision ~~3~~4, "U.S. EPR Security Design Features Technical Report," AREVA NP Inc., ~~February 2012~~ April 2013.

15. ANP-10296, Revision ~~0~~2, "U.S. EPR Design Features that Enhance Security," AREVA NP Inc., ~~December 2008~~June 2013.

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Table 1.6-1—Reports Referenced
Sheet 1 of 5

Report No. (See Notes 1, 2, and 3)	Title	Date Submitted to NRC	FSAR Section Number(s)
ANF-89-060P-A ANF-89-060NP-A Supplement 1	Generic Mechanical Design Report High Thermal Performance Spacer and Intermediate Flow Mixer	3/28/91	4.2
ANP-10263P-A ANP-10263NP-A	Codes and Methods Applicability Report for the U.S. EPR	11/06/07	4, ~~5.1~~ 5.2, 15, and 16
~~ANP-10264NP-A~~	~~U.S. EPR Piping Analysis and Pipe Support Design Topical Report~~	~~11/07/08~~	~~3.6, 3.7, 3.8, 3.9, 3.10, 3.12, App. 3A, and App. 3C~~
~~[*ANP-10264NP Revision 1*~~	~~*U.S. EPR Piping Analysis and Pipe Support Design Topical Report*]*~~	~~5/10~~	~~3.9 and 3.12~~
~~[~~ANP-10266~~A~~ Revision 4~~1~~	AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR Topical Report~~]*~~	6/18/07	7.1, 9.5, ~~14.3,~~ 15.0, 17.0, 17.4, 17.5, 18.1, 18.7, and 18.11
ANP-10268P-A ANP-10268NP-A	U.S. EPR Severe Accident Evaluation Topical Report	2/26/08	6.2.5, ~~15.4,~~ 19.1, and 19.2
ANP-10269P-A ANP-10269NP-A	The ACH-2 CHF Correlation for the U.S. EPR Topical Report	3/10/08	4.4, ~~5, 7,~~ 15, and 16~~-19~~
~~[~~ANP-10272 -A Revision 3	Software Program Manual TELEPERM XSTM Safety Systems Topical Report~~]*~~	7/11	7.1 and 7.6
~~[~~ANP-10275P-A ANP-10275NP-A	U.S. EPR Instrument Setpoint Methodology Topical Report~~]*~~	2/26/08	1.9, 7.1, 7.2, 7.3 and 16
ANP-10278P-A ANP-10278NP-A Revision 1	U.S. EPR Realistic Large Break Loss of Coolant Accident Topical Report	3/08/12	15 and 16
ANP-10282P ANP-10282NP	POWERTRAX/E Online Core Monitoring Software for the U.S. EPR Technical Report	11/27/07	4.4
ANP-10283P-A Revision 2 ANP-10283NP-A Revision 2	U.S. EPR Pressure-Temperature Limits Methodology for RCS Heat-Up and Cool-Down ~~Technical~~Topical Report	~~4~~8/12	5.3 and 16
ANP-10285P Rev 1 ANP-10285NP Rev 1	U.S. EPR Fuel Assembly Mechanical Design Topical Report	5/31/2013	4.1,4.2, 4.3, ~~15.6,~~ and 16
ANP-10286P-A ANP-10286NP-A	U.S. EPR Rod Ejection Accident Methodology Topical Report	3/08/12	4.3, ~~and~~ 15 and 16

Table 1.8-2—U.S. EPR Combined License Information Items
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Item No.	Description	Section
17.4-2	A COL applicant that references the U.S. EPR design certification will provide the information requested in Regulatory Guide 1.206, Section C.I.17.4.4.	17.4.4
17.6-1	A COL applicant that references the U.S. EPR design certification will describe the process for determining which plant structures, systems, and components (SSCs) will be included in the scope of the Maintenance Rule Program in accordance with 10 CFR 50.65(b). The program description will identify that additional SSCs functions may be added to or subtracted from the Maintenance Rule scope prior to fuel load, when additional information is developed (e.g., emergency operating procedures, or EOP), and after the license is issued.	17.6.1
17.6-2	A COL applicant that references the U.S. EPR design certification will provide the process for determining which SSCs within the scope of the Maintenance Rule program will be tracked to demonstrate effective control of their performance or condition in accordance with 10 CFR 50.65(a)(2).	17.6.2
17.6-3	A COL applicant that references the U.S. EPR design certification will provide a program description for monitoring SSCs in accordance with 10 CFR 50.65(a)(1).	17.6.2
17.6-4	A COL applicant that references the U.S. EPR design certification will identify and describe the program for periodic evaluation of the Maintenance Rule program in accordance with 10 CFR 50.65(a)(3).	17.6.3
17.6-5	A COL applicant that references the U.S. EPR design certification will describe the program for maintenance risk assessment and management in accordance with 10 CFR 50.65(a)(4). Since the removal of multiple SSCs from service can lead to a loss of Maintenance Rule functions, the program description will address how removing SSCs from service will be evaluated. For qualitative risk assessments, the program description will explain how the risk assessment and management program will preserve plant-specific key safety functions.	17.6.4
17.6-6	A COL applicant that references the U.S. EPR design certification will describe the program for selection, training, and qualification of personnel with Maintenance-Rule-related responsibilities consistent with the provisions of Section 13.2 as applicable. Training will be commensurate with maintenance rule responsibilities, including Maintenance Rule Program administration, the expert panel process, operations, engineering, maintenance, licensing, and plant management.	17.6.5

Table 1.8-2—U.S. EPR Combined License Information Items
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Item No.	Description	Section
17.6-7	A COL applicant that references the U.S. EPR design certification will describe the relationship and interface between the Maintenance Rule Program and the Reliability Assurance Program.	17.6.6
17.6-8	A COL applicant that references the U.S. EPR design certification will describe the plan or process for implementing the Maintenance Rule Program ~~as described~~ in the COL application, which includes establishing program elements through sequence and milestones and monitoring or tracking the performance and/or condition of SSCs as they become operational.	17.6.8
17.6-9	A COL applicant that references the U.S. EPR design certification will describe the program for Maintenance Rule implementation.	17.6
18.1-1	A COL applicant that references the U.S. EPR design certification will execute the NRC approved HFE program as described in this section.	18.1
18.1-2	A COL applicant that references the U.S. EPR design certification will be responsible for HFE design implementation for a new Emergency Operations Facility (EOF) or changes resulting from the addition of the U.S. EPR to an existing EOF.	18.1.1.3
18.5-1	A COL applicant that references the U.S. EPR design will confirm that actual staffing levels and qualifications of plant personnel specified in Section 13.1 of the COL application remain bounded by regulatory requirements and results of the staffing and qualifications analysis.	18.5
18.8-1	A COL applicant that references the U.S. EPR design certification will describe how HFE principles and criteria are incorporated into the development program for site procedures.	18.8
18.9-1	A COL applicant that references the U.S. EPR design certification will describe how HFE principles and criteria are incorporated into the development of training program scope, structure, and methodology.	18.9
19.0-1	A COL applicant that references the U.S. EPR design certification will either confirm that the PRA in the design certification bounds the site-specific design information and any design changes or departures, or update the PRA to reflect the site-specific design information and any design changes or departures.	19.0
19.1-1	A COL applicant that references the U.S. EPR design certification will describe the uses of PRA in support of licensee programs and identify and describe risk-informed applications being implemented during the combined license application phase.	19.1.1.2

Equipment

The SICS is implemented with various types of I&C technology to support its functions. Manual controls are implemented with buttons and switches. Indications are provided via dedicated indicators. A limited number of indications are provided on the QDS for situational awareness. The QDS consists of a display, computer, and input devices such as a touch screen or trackball.

The SICS is implemented with the TXS I&C platform, the QDS platform, and hardwired I&C equipment.

Qualification Requirements

The safety-related equipment used in SICS is qualified for environmental, seismic, electromagnetic interference and radio frequency interference (EMI/RFI) conditions in accordance with the environmental qualification program described in Section 3.11.

Quality Requirements

Safety-related hardwired I&C will meet the general quality requirements outlined in ANP-10266~~A~~. The non-safety-related portions of the SICS are designed, fabricated, erected, and tested under the quality assurance program described in ANP-10266~~A~~, Addendum A. This quality assurance program is consistent with the guidance of Generic Letter 85-06.

Diversity Requirements

There are no diversity requirements for SICS. See the U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report (ANP-10304) (Reference 8) for further information on defense-in-depth and diversity.

Data Communications

Data communications implemented in the SICS include:

- PS-SICS (QDS) – uni-directional (PS to SICS), point-to-point data connections implemented with the TXS Ethernet protocol.

Power Supply

The safety-related portion of the SICS is powered from the Class 1E uninterruptible power supply (EUPS). The EUPS provides backup power with two-hour batteries and the EDGs in the case of a loss of offsite power (LOOP). In the event of a station blackout (SBO), the EUPS has the capability of receiving power from the station blackout diesel generators (SBODGs).

Qualification Requirements

*[The PICS is intended to be used during normal, accident, and severe accident conditions as long as it is available. The PICS equipment is located in Safeguard Buildings that provide a mild environment during and following design basis events (DBEs). Equipment selected for use in the PICS will be rated by the manufacturer to operate under the mild environmental conditions expected to exist at its location during the events that the equipment is expected to be used.]**

EMI/RFI Requirements

The equipment used in the PICS is evaluated for EMI/RFI performance using the principles described in RG 1.180, IEC 61000-3, IEC 61000-4, and IEC 61000-5 for limits for electromagnetic compatibility. Strict compliance with these requirements is not required; however, the following shall be demonstrated:

- The electromagnetic emissions from this equipment are sufficiently low so that safety-related equipment in proximity is not adversely affected.
- The electromagnetic susceptibility of this equipment is adequate so that emissions from other equipment do not cause adverse effects within the system. Examples of adverse effects include: spurious actuation of plant components that results in an undesirable plant transient, large electrical surges that can damage equipment and other adjacent equipment, or corruption of data that can result in confusing indications to the operator.

Quality Requirements

*[In its role as the primary operator interface, the PICS is required to be of supplemented quality to perform its functions in a reliable manner. The PICS is designed using a robust engineering process with appropriate reviews, verifications, tests, and approvals. Supplemented quality is achieved in the design of the PICS through the following measures:]**

- The PICS is designed, fabricated, erected, and tested under the quality assurance program described in ANP-10266~~A~~, Addendum A (Reference 42). *[This quality assurance program is consistent with the guidance of Generic Letter 85-06 (Reference 43).]**
- The design of the PICS is accomplished through a phased approach as described in the software development lifecycle.
- A criticality analysis is performed for the PICS software in accordance with accepted industrial practice.
- V&V of the PICS software is performed according to a V&V plan that is consistent with accepted industrial practice.

Redundant DUs are provided in both divisions 1 and 4. This configuration is chosen so that the control rods remain operable given a failure of a single CU. Hardwired outputs from the DUs are sent to the Control Rod Drive Control System (CRDCS).

The MSIs provide a communication path between the RCSL and the PICS via redundant GWs for both display of information and transfer of manual commands. The MSIs also provide a path to the SU for testing and maintenance of the various functional units of the RCSL.

Equipment

The RCSL is implemented with the TXS I&C platform.

The AUs, CUs, DUs and MSIs generally consist of subracks, I/O modules, function processors, and communication modules, and optical link modules. SUs and GWs are non-safety-related and consist of industrial grade computers. Fiber optic and copper cable is used for the various data and hardwired connections.

Qualification Requirements

The RCSL equipment is located in Safeguard Buildings that provide a mild environment during and following DBEs. Equipment used in the RCSL will be rated by the manufacturer to operate under the mild environmental conditions expected to exist at its location during the events that the equipment is expected to be used.

Quality Requirements

For the RCSL equipment, the quality requirements will be consistent with the Quality Assurance Plan for non-safety-related equipment as described in ANP-10266~~A~~, Addendum A.

Diversity Requirements

There are no diversity requirements for the RCSL equipment.

Data Communications

Non-safety-related data communications implemented in the RCSL are:

- AU-CU – bi-directional, networked data connections implemented with the TXS Profibus protocol.
- CU-DU – bi-directional, networked data connections implemented with the TXS Profibus protocol.
- AU-MSI - bi-directional, networked data connections implemented with the TXS Profibus protocol.

The DAUs interface with the SICS via hardwired connections to receive manual system level commands and to display information.

Equipment

The DAS generally consists of various modules, such as threshold comparators, voting, and alarm modules. Copper cable is used for the hardwired connections. Specialized components may be used.

Qualification Requirements

The DAS equipment must function properly under conditions during and following AOOs or PAs concurrent with a SWCCF of the PS. The DAS equipment is located in Safeguard Buildings that provide a mild environment during and following AOOs or PAs. Equipment selected for use in the DAS shall be rated by the manufacturer to operate under the mild environmental conditions expected to exist at its location during the events for which the equipment is expected to respond.

Quality Requirements

As a system relied on to mitigate AOOs and PAs concurrent with a SWCCF of the PS, the DAS is required to be of sufficient quality to perform its functions in a reliable manner. The DAS is therefore designed using a robust engineering process with appropriate reviews, verification, tests, and approvals. Sufficient quality is achieved in the design of the DAS through the following measures:

- The DAS is designed, fabricated, erected, and tested under the quality assurance program described in ANP-10266~~A~~, Addendum A (Reference 42). This quality assurance program is consistent with the guidance of Generic Letter 85-06 (Reference 43).
- The design of the DAS is accomplished through a phased approach including the following (or equivalent) phases:
  - System requirements phase.
  - System design phase.
  - Software/hardware requirements phase.
  - Software/hardware design phase.
  - Software/hardware implementation phase.
  - Software/hardware validation phase.
  - System integration phase.

The SCDS is organized into four independent divisions located in the following buildings:

- Safeguard Buildings.
- Emergency Power Generating Buildings.
- Essential Service Water Pump Buildings.

In each division, there are safety-related and non-safety-related SCDS equipment to interface with safety-related and non-safety-related sensors, respectively. The safety-related SCDS and non-safety-related SCDS equipment is located in separate cabinets.

The SCDS is composed of non-computerized signal conditioning modules and signal distribution modules that are part of the TXS platform. Multiple signal conditioning modules or signal distribution modules may be used for a particular signal, depending on the required conditioning and the number of DCS systems to which the output signal is required to go.

The SCDS receives hardwired signal inputs from sensors or black boxes. The SCDS sends hardwired signal outputs to the SICS, DAS, PS, SAS, RCSL, and PAS, as needed. Outputs from safety-related SCDS equipment to non-safety-related DCS systems are electrically isolated by the signal distribution modules.

Equipment

The SCDS is implemented with TXS signal conditioning and distribution equipment.

The SCDS is implemented primarily with subracks, signal conditioning modules, and signal distribution modules.

Qualification Requirements

The equipment used in the SCDS is qualified for environmental, seismic, electromagnetic interference, and radio frequency interference (EMI/RFI) conditions in accordance with the environmental qualification program described in Section 3.11.

Quality Requirements

The SCDS is designed under the TXS quality program described in Section 7.1.1.2.1. The non-safety-related portions of the SCDS are designed, fabricated, erected, and tested under the quality assurance program described in ANP-10266~~A~~, Addendum A. This quality assurance program is consistent with the guidance of Generic Letter 85-06.

A feedback signal is sent from the rod control unit to the RCSL. This feedback signal is used by the RCSL to generate a digital position indication of the RCCA and is based on the number of rod movement steps sent from the CRDCS to the operating coils of the CRDM. A description of the CRDM and its associated operating coils is provided in Section 3.9.4.

The rod position measurement system (RPMS), described in Section 7.1.1.5.14, uses analog rod position measurement coils located within the CRDM to provide an indication of RCCA position that is separate from the position signal developed by the rod control unit of the CRDCS.

The CRDCS receives DC power from the NUPS to move and hold the CRDMs. The reactor trip breakers are upstream of the CRDCS. Refer to Section 8.3 for more information on the NUPS and the reactor trip breakers.

Within the CRDCS, the safety-related trip contactor modules interrupt power to the CRDMs when a trip signal is received from the PS. The trip contactors get a signal from each division of the PS and are arranged to implement two-out-of-four logic. The contactor modules are environmentally qualified, including seismic, EMI, and RFI effects.

The DAS provides a reactor trip signal to the CRDCS in case of an AOO or PA concurrent with a CCF of the PS. The reactor trip signal is sent to the rod control unit to drop the rods in a diverse manner from the trip contactors.

Drop orders are issued for a partial or full reactor trip in support of the reactor limitation functions. Refer to Section 7.7.1 for a description of the reactor control and limitation functions.

The non-safety-related components of the CRDCS are designed such that a seismic event does not result in damage that disables the safety function of the trip contactors.

The non-safety-related portion of the CRDCS will be designed, procured, installed, and tested in accordance with the Quality Assurance Plan for non-safety-related equipment as described in ANP-10266~~A~~, Addendum A.

Refer to Section 4.6.2 for more information on the reactivity control systems.

7.1.1.5.2 Incore Instrumentation System

Classification

The incore instrumentation system (ICIS) is classified as safety-related.

34. BTP 7-17, "Guidance on Self-Test and Surveillance Test Provisions," U.S. Nuclear Regulatory Commission, Standard Review Plan, Branch Technical Position, Rev. 5, March 2007.

35. BTP 7-18, "Guidance on the Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems," U.S. Nuclear Regulatory Commission, Standard Review Plan, Branch Technical Position, Rev. 5, March 2007.

36. BTP 7-19, "Guidance for Evaluation of Diversity and Defense-In-Depth in Digital Computer-Based Instrumentation and Control Systems," U.S. Nuclear Regulatory Commission, Standard Review Plan, Branch Technical Position, Rev. 5, March 2007.

37. BTP 7-21, "Guidance on Digital Computer Real-Time Performance," U.S. Nuclear Regulatory Commission, Standard Review Plan, Branch Technical Position, Rev. 5, March 2007.

38. BTP 5-2, "Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures," U.S. Nuclear Regulatory Commission, Standard Review Plan, Branch Technical Position, Rev. 3, March 2007.

39. Deleted.

40. EPRI TR-106439, "Guidance on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," Electric Power Research Institute, October 1996.

41. Deleted.

42. ~~[~~ANP-10266~~A~~, Revision ~~1~~4, "AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR Topical Report," AREVA NP Inc., ~~April 2007~~December 2012.~~]*~~

43. Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment that is Not Safety-Related," U.S. Nuclear Regulatory Commission, April 16, 1985.

44. [*ANP-10310P, Revision 2, "Methodology for 100% Combinatorial Testing of the U.S. EPR Priority Module Technical Report," AREVA NP Inc., May 2013.*]*

45. Letter, Sandra Sloan (AREVA NP Inc.) to Document Control Desk (NRC), "Request for Alternatives to IEEE Std 603-1991 to Satisfy 10 CFR 50.55a(h)(3) Requirements - U.S. EPR Design Certification," May 24, 2011.

46. [*ANP-10315P, Revision 2, "U.S. EPR Protection System Surveillance Testing and Teleperm XS Self-Monitoring Technical Report," May 2013.*]*

47. NEI 12-06, Revision 0. "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Nuclear Energy Institute, August 2012.

standards and is stored in accordance with manufacturer's recommendations. An adequate inventory of firefighting equipment is maintained to outfit a full complement of brigade members with consideration of the possibility of sustained fire response operations (i.e., multiple crews).

9.5.1.6.5 Quality Assurance

The overall plant quality assurance plan (QAP) includes the QA program for fire protection. The QAP provides reasonable assurance that the fire protection systems are designed, fabricated, erected, tested, maintained and operated so that they will function as intended. As stated in Section 17.5, the QAP for the design of the U.S. EPR is addressed in AREVA NP Topical Report ANP-10266~~-A~~ (Reference 41). The AREVA QAP implements quality requirements for the fire protection system in accordance with RG 1.189, Regulatory Position 1.7, directly by reference.

As stated in Section 17.2, a COL applicant that references the U.S. EPR design certification will provide the Quality Assurance Programs associated with the construction and operations phase. The program description to be provided by the applicant also includes a description of the fire protection system quality assurance program to be applied during fabrication, erection, installation and operations.

9.5.1.7 References

1. NFPA 10, "Standard for Portable Fire Extinguishers," National Fire Protection Association Standards, 2007.

2. NFPA 13, "Standard for Installation of Sprinkler Systems," National Fire Protection Association Standards, 2007.

3. NFPA 14, "Standard for the Installation of Standpipe and Hose Systems," National Fire Protection Association Standards, 2007.

4. NFPA 15, "Standard for Water Spray Fixed Systems for Fire Protection," National Fire Protection Association Standards, 2007.

5. NFPA 20, "Standard for the Installation of Stationary Pumps for Fire Protection," National Fire Protection Association Standards, 2007.

6. NFPA 22, "Standard for Water Tanks for Private Fire Protection," National Fire Protection Association Standards, 2003.

7. NFPA 24, "Standard for Installation of Private Fire Service Mains and Their Appurtenances," National Fire Protection Association Standards, 2007.

8. NFPA 25, "Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems," National Fire Protection Association Standards, 2002.

41. ANP-10266, Revision 4 ~~A~~, “AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR Topical Report,” AREVA NP Inc, December 2012~~June 2007~~.

42. NFPA 804, “Standard for Fire Protection for Advanced Light Water Reactor Electric-Generating Plants,” National Fire Protection Association Standards, 2006.

43. NFPA 105, “Installation of Smoke Door Assemblies and Other Protective Openings,” National Fire Protection Association Standards, 2007.

44. ASCE/SEI Std. 43-05, “Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities,” American Society of Civil Engineers/Structural Engineering Institute, 2005.

45. ANSI/AWWA D100-2005, “Welded Steel Tanks for Water Storage,” American National Standards Institute/American Water Works Association, 2005.

46. ASCE 7-10, “Minimum Design Loads for Buildings and Other Structures,” American Society of Civil Engineers, 2010.

17.0 Quality Assurance and Reliability Assurance

This FSAR chapter contains the following information:

- Section 17.1, Section 17.2, Section 17.3, and Section 17.5 address the Quality Assurance Plan (QAP) for the U.S. EPR. The basis for these sections is ~~AREVA NP Topical Report~~ ANP-10266~~A~~, Revision ~~1~~4, ~~“~~AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR Topical Report~~,” which has been approved by the NRC~~ (Reference 1 of Section 17.5).
- Section 17.4 addresses the Design Reliability Assurance Program (D-RAP) for the U.S. EPR.
- Section 17.6 addresses the U.S. EPR Maintenance Rule Program.

17.1 Quality Assurance During Design

This information is provided in Section 17.5.

Next File

17.4.2 Reliability Assurance Program Implementation

The RAP for the design stage is implemented in several phases. The first phase is the design certification phase, which defines the overall structure of the RAP, including guidance for procedures and other activities which will be implemented in future phases. A design-specific PRA model is used to develop a list of SSC and insights. The risk-significant SSC are identified in this phase for inclusion in the program using the probabilistic, deterministic, or other methods previously indicated.

The second phase is the site-specific phase, which introduces the plant site-specific design information to the RAP process. A COL applicant that references the U.S. EPR design certification will identify the site-specific SSC within the scope of the RAP. Also in this phase, the RAP is modified or appended based on consideration specific to the site.

Risk-significant SSC are subject to the appropriate quality requirements through the implementation of the RAP. Safety-related SSC that are also determined to be risk-significant in the RAP have a full 10 CFR 50 Appendix B quality assurance program applied along with the applicable GDC.

For non-safety-related SSC that have been determined to be "risk-significant" under the RAP in Section 17.4, the U.S. EPR design applies additional quality assurance measures and design requirements consistent with the guidance in SRP 17.5, Part V, "Non-Safety Related SSC Quality Controls." These additional quality assurance measures are described in the approved topical report ANP-10266~~A~~, Revision ~~1~~4, "AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR Topical Report," Addendum A, and are applied to all risk-significant SSC during the design certification phase.

All risk-significant SSC will be included in the scope of the COL applicant's Maintenance Rule program in accordance with 10 CFR 50.65(b) in the high safety significance category. This is done so that the risk-significant SSC are subject to performance monitoring criteria which are established consistent with the reliability and availability assumptions used in the PRA.

Tier 1 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) provide confirmation that as the SSC design progresses, the procurement and construction information for risk-significant SSC is consistent with the RAP related key assumptions and insights. This confirmation occurs by verifying that appropriate quality requirements are specified in the documents approved for the procurement and construction of risk-significant SSC.

Beyond the writing of design specifications, consistency with RAP related key assumptions and insights during the construction and initial testing phases are verified by confirming that the systems are as built in accordance with the system level ITAAC

- Detection of component failures.
- The effect of component failure on the other systems.

As a result of the expert panel review, a list of non-site-specific systems and structures within the RAP scope, and an indication of whether they are PRA based input versus added by the expert panel, is provided in Table 17.4-3.

17.4.3 Organization, Design Control, Procedures and Instructions, Corrective Actions, and Audit Plans

AREVA ~~NP~~ is an integrated design and engineering organization that is responsible for formulating and implementing Phase 1 of the RAP.

The AREVA ~~NP~~ RAP implementation plan includes RAP scope, objectives, design consideration, the identification and prioritization of SSC, RAP organization, and expert panel. This RAP implementation plan is described in the following paragraphs.

The AREVA ~~NP~~ engineering organization is responsible for the safety analyses, risk and reliability analyses, and the PRA necessary to support the development of the RAP. PRA and design engineering personnel report to the manager of nuclear island engineering. Therefore, risk and reliability personnel are directly involved with the design organization and are responsible for keeping the design staff cognizant of the risk-significant items of the RAP, program needs, and project status. Risk and reliability personnel participate in the design change control process to incorporate RAP-related inputs into the design process. Additionally, a cognizant representative of risk and reliability is present at design reviews to identify interfaces between the performance of risk-significant SSC and the reliability assumptions in the PRA. Meetings between risk and reliability personnel and the designer are held to manage interface issues.

AREVA ~~NP~~ engineering design procedural controls are applied to the RAP. Specific procedures provide guidance for the design control process, control of design changes, and storage and retrieval controls.

The design control procedure defines the process for performing, documenting, and verifying design activities. This includes the development or modification of system designs, evaluations, analyses, calculations and design document preparation (e.g., specifications, drawings, reports).

The procedure for design change control defines the process for evaluating design changes in engineering controlled documents so that the total effect is considered before a change is approved, and the affected documents are identified and changed accordingly. The procedure identifies the information and organizations responsible for these interfaces, including PRA review. If a proposed change could affect the safety, availability, or capacity factor of the U.S. EPR, system reliability is analyzed.

There are several AREVA ~~NP~~ corporate quality assurance and design control procedures which provide guidance for the development of a high-quality process for the reliability assurance program and for maintaining the appropriate documentation of it. The documentation development and maintenance procedure establishes the requirements and responsibilities for the preparation, approval, and issue of documents controlled by the engineering design organizations. The QA records procedure provides requirements for QA record retention. The self-assessment, corrective action, and audit procedures specify the responsibilities associated with respective audits of the engineering organization. This self-assessment is also used to promptly identify, document, and determine corrective actions for conditions that are adverse to quality.

The above AREVA ~~NP~~ corporate processes provide configuration control of the list of SSC within the scope of RAP thereby demonstrating that the U.S. EPR reliability assurance implementation program will maintain the scope of RAP SSC throughout the design process.

17.4.4 Reliability Assurance Program Information Needed in a COL Application

A COL applicant that references the U.S. EPR design certification will provide the information requested in Regulatory Guide 1.206, Section C.I.17.4.4.

17.4.5 References

1. NUMARC 93-01, Nuclear Utilities Management and Resources Council, “Industry Guideline for Monitoring Effectiveness of Maintenance at Nuclear Power Plants,” April 1996.

17.5 Quality Assurance Program Description

The Quality Assurance Plan (QAP) for the U.S. EPR is addressed in ~~AREVA NP Topical Report~~ ANP-10266~~A~~, Revision ~~1~~4, "AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR Topical Report"~~, which has been approved by the NRC~~ (Reference 1). Changes to the approved U.S. EPR QAP have not reduced the commitments in the program description as accepted by the NRC in Reference 2. As noted in the ~~referenced~~QAP topical report, the QAP is applicable to the design certification of the U.S. EPR. The QAP is based on the eighteen-point criteria of 10 CFR 50, Appendix B, and ANSI/ASME NQA-1-1994. Consistent with Section I of SRP Section 17.5, design certification does not include fabrication, erection, installation, or operations.

17.5.1 QA Program Responsibilities

The scope and responsibilities for the U.S. EPR QAP are addressed in the referenced topical report.

17.5.2 SRP Section 17.5 and the QA Program Description

As noted in Section 17.5.2 of Regulatory Guide 1.206, "The NRC staff revised the SRP to add the new Section 17.5, 'Quality Assurance Program Description-Design Certification, Early Site Permit and New License Applicants.' This new SRP section addresses QAPD provisions for COL applicants. The NRC staff reviews and evaluates QAPDs in accordance with the applicable sections of the SRP. Section 17.5 of the SRP is the principal guidance for NRC reviews of a QAPD submitted by a COL applicant. A COL applicant may submit its QAPD in two phases. The first phase could apply to design, fabrication, construction, and testing QA activities, and the second phase could apply to operational QA activities. The requirements for the two phases are fully defined in SRP 17.5. Regardless of the approach, the NRC would review and evaluate QAPDs before issuing the COL. Chapter 17 of the FSAR should incorporate the QAPD (or QAPDs) by reference."

While the purpose of Regulatory Guide 1.206 is to provide guidance regarding the information to be submitted in a combined license application, as noted in Section 1.1,~~.~~ AREVA ~~NP~~ has structured the FSAR for the U.S. EPR to be consistent to the extent practical with the format and content that would be expected for a COL applicant.

The QAP for the U.S. EPR is described in the QAP topical report. The QAP provides the specific applicability and application of the criteria of 10 CFR 50, Appendix B and the Basic, Supplemental and applicable Subpart requirements of ANSI/ASME NQA-1-1994 to the U.S. EPR Design Certification Project. Each section of the QAP delineates the applicability of the criteria to the U.S. EPR design certification.

17.5.3 Evaluation of the QAPD Against the SRP and QAPD Submittal Guidance

The U.S. EPR QAP has been approved by the NRC and conforms to the guidance provided in NUREG-0800.

Per Section 1.8 of the QAP, AREVA ~~NP~~ Inc. does not delegate any of the activities associated with planning, establishing, or implementing the overall QA program to others and retains the responsibility for the program.

17.5.4 References

1. ~~AREVA NP Inc. Topical Report~~ ANP-10266~~A~~, Revision ~~1~~4, "AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR Topical Report," ~~April 2007 (ML071790218)~~December 2012 ~~(ML12354A475)~~.

2. ANP-10266A, Revision 1, "AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR Topical Report," April 2007 ~~(ML071790218)~~.

Next File

17.6 Description of Applicant's Program for Implementation of 10 CFR 50.65, the Maintenance Rule

A COL applicant that references the U.S. EPR design certification will describe the program for Maintenance Rule implementation.

17.6.1 Scoping per 10 CFR 50.65(b)

A COL applicant that references the U.S. EPR design certification will describe the process for determining which plant structures, systems, and components (SSCs) will be included in the scope of the Maintenance Rule Program in accordance with 10 CFR 50.65(b). The program description will identify that additional SSCs functions may be added to or subtracted from the Maintenance Rule scope prior to fuel load, when additional information is developed (e.g., emergency operating procedures, or EOP), and after the license is issued.

17.6.2 Monitoring per 10 CFR 50.65(a)

A COL applicant that references the U.S. EPR design certification will provide a program description for monitoring SSCs in accordance with 10 CFR 50.65(a)(1).

A COL applicant that references the U.S. EPR design certification will provide the process for determining which SSCs within the scope of the Maintenance Rule Program will be tracked to demonstrate effective control of their performance or condition in accordance with paragraph 50.65(a)(2).

17.6.3 Periodic Evaluation per 10 CFR 50.65(a)(3)

A COL applicant that references the U.S. EPR design certification will identify and describe the program for periodic evaluation of the Maintenance Rule Program in accordance with 10 CFR 50.65(a)(3).

17.6.4 Risk Assessment and Management per 10 CFR 50.65(a)(4)

A COL applicant that references the U.S. EPR design certification will describe the program for maintenance risk assessment and management in accordance with 10 CFR 50.65(a)(4). Since the removal of multiple SSCs from service can lead to a loss of Maintenance Rule functions, the program description will address how removing SSCs from service will be evaluated. For qualitative risk assessments, the program description will explain how the risk assessment and management program will preserve plant-specific key safety functions.

17.6.5 Maintenance Rule Training and Qualification

A COL applicant that references the U.S. EPR design certification will describe the program for selection, training, and qualification of personnel with Maintenance-

Rule-related responsibilities consistent with the provisions of Section 13.2 as applicable. Training will be commensurate with maintenance rule responsibilities, including Maintenance Rule Program administration, the expert panel process, operations, engineering, maintenance, licensing, and plant management.

17.6.6 Maintenance Rule Program Role in Implementation of Reliability Assurance Program (RAP) in the Operations Phase

A COL applicant referencing the U.S. EPR Design Certification will describe the relationship and interface between the Maintenance Rule Program and the Reliability Assurance Program (refer to Section 17.4).

17.6.7 Maintenance Rule Program Relationship with Industry Operating Experience Activities

Industry operating experience (IOE) comprises information from a variety of sources that is applicable and available to the nuclear industry with the intent of minimizing, through shared experiences, adverse plant conditions or situations. Sources of IOE include information programs organized by the reactor vendor, safety-related equipment suppliers, the NRC, the Institute of Nuclear Power Operations (INPO), and the Electric Power Research Institute (EPRI).

IOE is reviewed for plant-specific applicability and, where appropriate, is applied in various elements of the Maintenance Rule Program and procedures, including scoping, performance/condition criteria development, monitoring, goal-setting, corrective action, training, program assessment, and maintenance and procurement activities. The specific steps for employing IOE in the various Maintenance Rule Program areas will be contained in the plan or process for maintenance rule implementation described in Section 17.6.8.

17.6.8 Maintenance Rule Program Implementation

A COL applicant referencing the U.S. EPR Design Certification will describe the plan or process for implementing the Maintenance Rule Program ~~as described~~ in the COL application, which includes establishing program elements through sequence and milestones and monitoring or tracking the performance and/or condition of SSCs as they become operational.

The detailed design phase involves performing design support and configuration measures. Support measures such as calculations, selection and suitability reviews, and design reviews (as described in Section 4.5.1 of the U.S. EPR HFE Program Management Plan (Reference 2)) are used to validate the design and maintain or manage the design configuration. HFE design evaluation activities are conducted throughout basic and detailed design. Verification and validation (V&V) activities are performed after the iterative design/evaluation process in order to develop a design that meets requirements.

The construction and operation phase involves acceptance testing before and after installation, verifying configuration management for design documentation (see Section 18.11), and monitoring system and operator performance throughout the life of the plant (see Section 18.12).

18.1.5.2 Relationship Between HFE and Other Engineering Disciplines

Reference 3 requires that the HFE and Control Room Design Team follow the same design processes as other engineering disciplines. Section 4.0 of the U.S. EPR HFE Program Management Plan (Reference 2) describes the relationship between HFE program design documentation and general design documentation.

18.1.5.3 HFE Program Element Documentation

The U.S. EPR HFE program is described in Section 18.1. Section 2.0 of the U.S. EPR HFE Program Management Plan (Reference 2) describes the general HFE requirements, standards, and specifications utilized in the design of the U.S. EPR. Section 18.10 of this FSAR and Section 6.3 of the U.S. EPR HFE Program Management Plan (Reference 2) describe the uses of HFE facilities such as mockups and simulators as well as methods and tools employed for the various testing and validation techniques.

Sections 18.2 through 18.12 provide information on the types of documents generated as part of the U.S. EPR HFE program.

18.1.6 References

1. [*NUREG-0711, "Human Factors Engineering Program Review Model," Revision 2, U.S. Nuclear Regulatory Commission, 2004.*

2. *ANP-10327P, Revision 0, "U.S. EPR HFE Program Management Plan Technical Report," AREVA NP Inc., April 2013.]**

3. ANP-10266~~-A~~, Revision ~~1~~4, "AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR," AREVA NP Inc., ~~June 2007~~December 2012.

4. U.S. EPR Human Performance Monitoring Implementation Plan, AREVA NP Inc., 2010.

Activities such as concept testing, mock-up activities, trade-off evaluations, and performance-based tests are utilized at various stages of the design. The criteria used to decide which type of testing or evaluation technique is applicable are described in the U.S. EPR Human Factors Verification and Validation Implementation Plan (Reference 17).

18.7.8 HSI Design Results and Documentation

As described in Section 4.5 of EPR HFE Program Management Plan (Reference 2), the HSI designs are documented using specific design control process requirements. The various configuration management, design change controls, design verification, and design quality control tools are also described in Reference 1.

18.7.9 References

1. ANP-10266~~NPA~~, Revision ~~0~~4, "AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR," AREVA NP Inc., December ~~2008~~2012.

2. [*ANP-10327P, Revision 0, "U.S. EPR HFE Program Management Plan Technical Report," AREVA NP Inc., April 2013.*]*

3. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980.

4. NUREG-0711, "Human Factors Engineering Program Review Model," Rev. 2, U.S. Nuclear Regulatory Commission, February 2004.

5. ANP-10304, Revision 6, "U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report," AREVA NP Inc., May 2013.

6. NUREG-0700, "Human-System Interface Design Review Guidelines," Revision 2, U.S. Nuclear Regulatory Commission, May 2002.

7. NUREG/CR-6633, "Advanced Information Systems: Technical Basis and Human Factors Review Guidance," U.S. Nuclear Regulatory Commission, March 2000.

8. NUREG/CR-6634, "Computer-Based Procedure Systems: Technical Basis and Human Factors Review Guidance," U.S. Nuclear Regulatory Commission, March 2000.

9. NUREG/CR-6635, "Soft Controls: Technical Basis and Human Factors Review Guidance," U.S. Nuclear Regulatory Commission, March 2000.

10. NUREG/CR-6636, "Maintainability of Digital Systems: Technical Basis and Human Factors Review Guidance," U.S. Nuclear Regulatory Commission, March 2000.

11. NUREG-0696, "Functional Criteria for Emergency Response Facilities," U.S. Nuclear Regulatory Commission, February 1981.

18.11.4 Results Summary

Throughout the design implementation, the HFE Issues Tracking Database is updated as new HEDs are discovered during the process. Resolution for these HEDs is also updated in the HFE Issues Tracking Database. A results summary report is generated detailing the status of HEDs tracked including any that remain unresolved and concludes HFE issues have been adequately addressed. The results summary report concludes the design implementation was performed in accordance with the prescribed process for validating that the as built design conforms to the standard design resulting from the HFE V&V process. Also included are the methods and criteria used during the design implementation process and the results of the verification. This report becomes part of the final design documentation owned by the U.S. EPR operator.

18.11.5 References

1. NUREG-0711, "Human Factors Engineering Program Review Model," U.S. Nuclear Regulatory Commission, 1994.

2. [*ANP-10327P, Revision 0, "U.S. EPR HFE Program Management Plan Technical Report," AREVA NP Inc., April 2013.*]*

3. ANP-10266~~A~~, Revision ~~1~~4, "AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR," AREVA NP Inc., ~~April 2007~~December 2012.

4. NUREG-0700, "Human-System Interface Design Review Guidelines," Revision 2, U.S. Nuclear Regulatory Commission, May 2002.

5. [*U.S. EPR Human Factors Engineering (HFE) Design Implementation Plan, AREVA NP Inc., 2010.*]*

Next File

U.S. EPR Final Safety Analysis Report Markups

For AIA Freeze

19. Letter from D. Matthews, NRC to R. Ford, AREVA NP, "Approval of AREVA NP Inc. Safeguards Protection Program and Reviewing Official, and Transmittal of Beyond Design Basis, Large Commercial Aircraft Characteristics Specified by the Commission", December 21, 2007.

20. Deleted.

21. Achenbach, J.A., Miller, R.B., Srinivas, V., "Large-Scale Hydrogen Burn Equipment Experiments," EPRI NP-4354, Electric Power Research Institute, 1985.

22. NUREG/CR-5334, "Severe Accident Testing of Electrical Penetration Assemblies," SAND89-0327, November 1989.

23. ANP-10314, Revision 0, "The Operating Strategies for Severe Accidents Methodology for the U.S. EPR Technical Report," AREVA NP Inc., July 2010.

24. ANP-10317, Revision ~~1~~3, "Design Requirements for the U.S. EPR Aircraft Hazard Protection Structures," AREVA NP Inc., ~~April~~ July 2013.

25. ANP-10295P, Revision 4, "U.S. EPR Security Design Features Technical Report," AREVA NP Inc., April 2013.

26. ANP-10296, Revision ~~1~~2, "U.S. EPR Design Features that Enhance Security," AREVA NP Inc., ~~April~~June 2013.

27. ANP-10329, Revision 0, "U.S. EPR Mitigation Strategies for an Extended Loss of AC Power Event Technical Report," AREVA NP Inc., May 2013.

28. Regulatory Guide 1.126, "Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design Basis Pressure," August 2010.

29. Regulatory Guide 1.155, "Station Blackout," August 1988.

All Boxed changes are related to AIA in Section 19.2.7