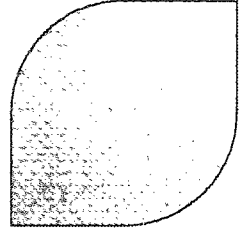


**Presentation to NRC
U.S. EPR
FSAR Chapter 4 Open Item RAIs**

AREVA NP
May 27, 2010

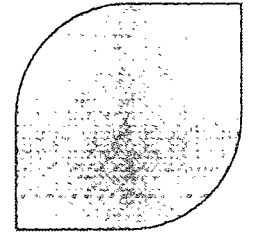


FSAR Chapter 4 Open Item RAIs



- ▶ **Draft responses have been provided to the staff in advance of this meeting**
- ▶ **Today's presentations are intended to facilitate discussion of the draft responses**
- ▶ **Goal of today's discussion is to assure that the responses provide the information requested**

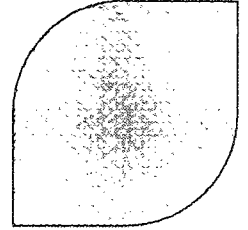
FSAR Chapter 4 Open Item RAIs



- ▶ RAI 339 – 4 questions
- ▶ RAI 343 – 2 questions
- ▶ RAI 344 – 2 questions
- ▶ RAI 366 – 1 question
- ▶ RAI 367 – 1 question

FSAR Chapter 4

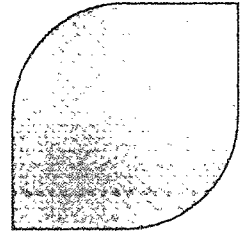
RAI 339



- ▶ **Question 04.02-17 - AREVA NP is requested to acknowledge receipt of this open item which states that ANP-10285P is currently under review by the staff.**
- ▶ **Response to 04.02-17 – AREVA NP acknowledged receipt of this open item.**

FSAR Chapter 4

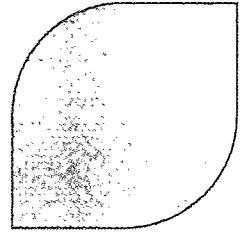
RAI 339



- ▶ **Question 04.05.02-9 – This question is related to the use of Stellite 6 for hardfacing the Radial Key Inserts. The staff requested that the applicable ASME code specifications for Stellite 6 be added to the FSAR.**
- ▶ **Response to 04.05.02-9 – The applicable ASME code specifications were added to the FSAR.**

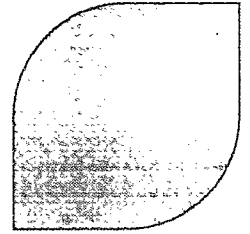
FSAR Chapter 4

RAI 339



- ▶ **Question 04.05.02-10 – AREVA NP was requested to add a COL item related to consideration of neutron fluence in the design of reactor internals.**
- ▶ **Response to 04.05.02-10 – A COL condition was added to the FSAR.**

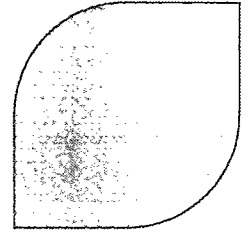
FSAR Chapter 4 RAI 339



- ▶ **Question 04.05.02-11 – AREVA NP was requested to provide a discussion of the prevention of notches on the vertical keys and keyways in the heavy reflector.**
- ▶ **Response to 04.05.02-11 – A discussion of the prevention of notches on the vertical keys and keyways in the heavy reflector was provided.**

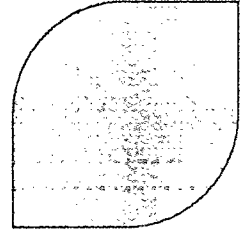
FSAR Chapter 4

RAI 343



- ▶ **Question 04.05.01-6 – AREVA NP was requested to modify the FSAR to reflect the use of F347 material.**
- ▶ **Response to 04.05.01-6 – AREVA NP responded that F347 material is not used in the U.S. EPR and that the use of that term in the referenced RAI response was a typo.**

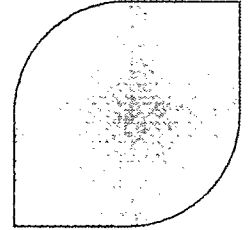
FSAR Chapter 4 RAI 343



- ▶ **Question 04.05.01-7 – AREVA NP was requested to notify the staff when the ASME code case N-785 was approved.**
- ▶ **Response to 04.05.01-7 – The response states that the ASME code case has been approved and describes the basis for the approval. The reference in the FSAR to this code case constitutes a request to use this material as required by 10 CFR 50.55a.**

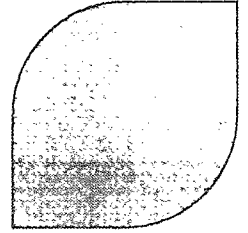
FSAR Chapter 4

RAI 344



- ▶ **Question 04.03-27 – AREVA NP was requested to add a COL item related to the benchmarking of the method in BAW-2241PA.**
- ▶ **Response to 04.03-27 – A COL item related to the benchmarking of the method in BAW-2241PA was added to the FSAR.**

FSAR Chapter 4 RAI 344

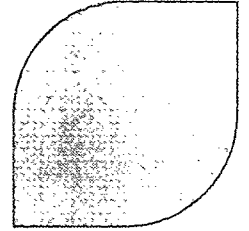


- ▶ **Question 04.03-28 – AREVA NP is requested to acknowledge receipt of this open item which states that ANP-10286P is currently under review by the staff.**

- ▶ **Response to 04.03-28 – AREVA NP acknowledged receipt of this open item.**

FSAR Chapter 4

RAI 366

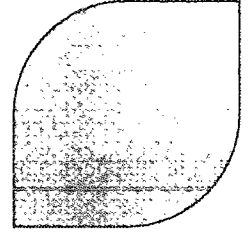


- ▶ **Question 04.06-13 – AREVA NP was requested to clarify an apparent discrepancy in the FSAR regarding the descriptions of the credit taken for reactivity controls systems other than reactor trip.**

- ▶ **Response to 04.06-13 – AREVA NP provided a clarification of the descriptions in the FSAR. No change was necessary to the FSAR.**

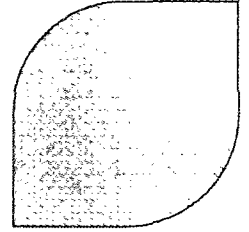
FSAR Chapter 4

RAI 367



- ▶ **Question 04.06-14 – AREVA NP was requested to provide a description of how the limitations in the SE for ANP-10287P would be implemented and verified.**
- ▶ **Response to 04.06-14 – AREVA NP provided a description of how the limitations in the SE for ANP-10287P would be implemented and verified. A COL item was added to the FSAR to address one of the limitations in the SE.**

FSAR Chapter 4 Open Item RAIs



▶ **Future Actions**

▶ **Schedule**

Response to

Request for Additional Information No. 339, Supplement 2

01/08/2010

U.S. EPR Standard Design Certification

AREVA NP Inc.

Docket No. 52-020

SRP Section: 04.02 - Fuel System Design

SRP Section: 04.05.02 - Reactor Internal and Core Support Structure Materials

Application Section: FSAR Chapter 4

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)

QUESTIONS for Component Integrity, Performance, and Testing Branch 1

(AP1000/EPR Projects) (CIB1)

AREVA NP Inc.

Response to Request for Additional Information No. 339, Supplement 2
U.S. EPR Design Certification Application

Page 2 of 5

Question 04.02-17:

OPEN ITEM

Throughout the U.S. EPR Final Safety Analysis Report (FSAR) Tier 2, Section 4.2, AREVA NP refers to licensing topical report ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Topical Report." This document is currently under review by the NRC staff. This RAI is created to track an open item associated with this review. It will be closed upon completion of the review by the NRC staff. AREVA is requested to acknowledge receipt of this open item.

Response to Question 04.02-17:

AREVA NP acknowledges receipt of this open item.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

DRAFT

Question 04.05.02-9:

OPEN ITEM:

AREVA's response to RAI No. 50, Question 04.05.02-1 stated that Stellite 6 is used for hardfacing the Radial Key Inserts, Upper Core Plate Guide Pins and Inserts. Your response also lists the applicable ASME specifications for the Stellite 6 (ASME SFA5.21 Classification ERCCoCr-A, ASME SFA 5.21 Classification ERCoCr-A and ASME SFA5.13 Classification ECoCr-A) that could be used for weld deposition of the Stellite 6 onto the applicable components base material. The staff requests that the applicable ASME code specifications for the hardfacing material, Stellite 6, be included in the U.S. EPR FSAR, Tier 2, Table 4.5.2.

Response to Question 04.05.02-9:

U.S. EPR FSAR Tier 2, Table 4.5-2 will be revised to include the welding filler material specifications ASME SFA-5.21 ERCCoCr-A or ERCoCr-A and ASME SFA-5.13 ECoCr-A for Stellite 6 hardfacing materials.

FSAR Impact:

U.S. EPR FSAR Tier 2, Table 4.5-2 will be revised as described in the response and indicated on the enclosed markup.

DRAFT

Question 04.05.02-10:

OPEN ITEM:

In response to RAI No. 50, Question 04.05.02-4, your response stated that the reentrant corners of the heavy reflector are estimated to experience a peak 60-EFPY neutron fluence of 8.56×10^{22} n/cm² (E>1.0 MeV) which exceeds the threshold for IASCC and void swelling. Therefore, AREVA plans on participating in the industry EPRI/MRP programs to manage IASCC and void swelling to screen the heavy reflector for IASCC and void swelling. To verify that IASCC and void swelling does not impact the safety function of the heavy reflector or create loose parts, an augmented ASME Code, Section XI inspection program will be developed. Therefore, the staff requests that AREVA include in the U.S. EPR DCD, a license condition or a COL Action Item to address this issue.

Response to Question 04.05.02-10:

U.S. EPR FSAR Tier 2, Section 4.5.2.1 will be modified to include consideration of neutron fluence in design of the reactor internals and evaluation of the materials relative to susceptibility to known aging degradation mechanisms such as irradiation-assisted stress corrosion cracking and void swelling.

U.S. EPR FSAR Tier 2, Table 1.8-2 and Section 3.9.3 will be modified to also require the COL applicant to address reactor internals materials with regard to known aging degradation mechanisms such as irradiation assisted stress corrosion cracking or void swelling.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 3.9.3, Section 4.5.2-1, and Table 1.8-2 will be revised as described in the response and indicated on the enclosed markup.

Question 04.05.02-11:

OPEN ITEM:

AREVA's response to RAI No. 50, Question 04.05.02-2b did not address the staff's request for a discussion on the prevention of notches on the vertical keys and keyways that can act as stress concentrations and crack initiation sites, which could lead to the loss of function of the heavy reflector. Therefore, the staff requests a discussion on this topic.

Response to Question 04.05.02-11:

Under normal and upset conditions, the heavy reflector slabs are not subject to significant primary loads. Loading during these conditions is mainly attributable to the loading induced by the preload of [] tie rods which hold the heavy reflector assembly together and attach it to the lower support plate of the core barrel assembly and thermal loading (from both gamma heating and surrounding fluid temperature). The vertical keys within the heavy reflector assembly mainly provide lateral restraint between the heavy reflector slabs during faulted conditions; however, they may also be credited for vertical restraint during faulted conditions in conjunction with the tie rods. Specific configurations are evaluated with consideration of any notch effects (and associated stress concentration factors) to confirm that the structural and fatigue requirements of ASME Section III, Subsection NG are met as specified in U.S. EPR FSAR Tier 2, Section 3.9.5.2

The width of the vertical key is controlled for a distance [] above and below each of the heavy reflector slab interfaces to provide a clearance fit []. The lateral restraint function of the vertical keys is provided only within these portions of the vertical keys. For the remainder of the vertical key length, greater clearances [] between the vertical keys and the heavy reflector slabs are provided. For the keyway within the heavy reflector slabs, the reentrant corners are provided with a small radius for the full length of the keyway to preclude sharp notches.

Each end of the vertical keys is configured with a 'T' connection where the width is increased to engage with the upper and lower heavy reflector slabs for the vertical restraint function. During assembly, a gap is provided between the vertical keys and the lower slab so that vertical keys / heavy reflector slabs are not subjected to tensile loading during normal operation. Within the keyway opening in the upper and lower slabs, a small radius is provided for all reentrant corners to preclude sharp notches.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

U.S. EPR Final Safety Analysis Report Markups

DRAFT



Table 1.8-2—U.S. EPR Combined License Information Items
Sheet 17 of 47

Item No.	Description	Section	Action Required by COL Applicant	Action Required by COL Holder
3.8-12	A COL applicant that references the U.S. EPR design certification will describe the program to examine inaccessible portions of below-grade concrete structures for degradation and monitoring of groundwater chemistry.	3.8.5.7	☑	
3.8-13	A COL applicant that references the U.S. EPR design certification will identify if any site-specific settlement monitoring requirements are required for Seismic Category I foundations based on site-specific soil conditions.	3.8.5.7	☑	
3.8-14	A COL applicant that references the U.S. EPR design certification will describe the design and analysis procedures used for buried conduit and duct banks, and buried pipe and pipe ducts.	3.8.4.4.5	☑	
3.8-15	A COL applicant that references the U.S. EPR design certification will use results from site-specific investigations to determine the routing of buried pipe and pipe ducts.	3.8.4.4.5	☑	
3.8-16	A COL applicant that references the U.S. EPR design certification will perform geotechnical engineering analyses to determine if the surface load will cause lateral and/or vertical displacement of bearing soil for the buried pipe and pipe ducts and consider the effect of wide or extra heavy loads.	3.8.4.4.5	☑	
3.9-1	A COL applicant that references the U.S. EPR design certification will submit the results from the vibration assessment program for the U.S. EPR RPV internals, in accordance with RG 1.20.	3.9.2.4		☑
3.9-2	A COL applicant that references the U.S. EPR design certification will prepare the design specifications and design reports for ASME Class 1, 2, and 3 components, piping, supports and core support structures that comply with and are certified to the requirements of Section III of the ASME Code.	3.9.3		☑



Table 1.8-2—U.S. EPR Combined License Information Items
Sheet 18 of 47

Item No.	Description	Section	Action Required by COL Applicant	Action Required by COL Holder
	<p>04.05.02-10 ↘ <u>The COL applicant will address the results and conclusions from the reactor internals material reliability programs applicable to the U.S. EPR reactor internals with regard to known aging degradation mechanisms such as irradiation-assisted stress corrosion cracking or void swelling.</u></p>			
3.9-3	<p>A COL applicant that references the U.S. EPR design certification will examine the feedwater line welds after hot functional testing prior to fuel loading and at the first refueling outage, in accordance with NRC Bulletin 79-13. A COL applicant that references the U.S. EPR design certification will report the results of inspections to the NRC, in accordance with NRC Bulletin 79-13.</p>	3.9.3.1.1		☒
3.9-4	<p>As noted in ANP-10264NP-A, a COL applicant that references the U.S. EPR design certification will confirm that thermal deflections do not create adverse conditions during hot functional testing.</p>	3.9.3.1.1		☒
3.9-5	<p>As noted in ANP-10264NP-A, should a COL applicant that references the U.S. EPR design certification find it necessary to route Class 1, 2, and 3 piping not included in the U.S. EPR design certification so that it is exposed to wind and tornadoes, the design must withstand the plant design-basis loads for this event.</p>	3.9.3.1.1	☒	
3.9-6	<p>A COL applicant that references the US EPR design certification will identify any additional site-specific valves in Table 3.9.6-2 to be included within the scope of the IST program.</p>	3.9.6.3	☒	
3.9-7	<p>A COL applicant that references the U.S. EPR design certification will submit the preservice testing (PST) program and IST program for pumps, valves, and snubbers as required by 10 CFR 50.55a.</p>	3.9.6		☒



This section refers to U.S. EPR Piping Analysis and Pipe Support Design Topical Report (Reference 2) for information related to the design and analysis of safety-related piping. This topical report presents the U.S. EPR code requirements, acceptance criteria, analysis methods, and modeling techniques for ASME Class 1, 2, and 3 piping and pipe supports. Applicable COL action items in the topical report are identified in the applicable portions of this section. The U.S. EPR design is based on the 2004 ASME Code, Section III, Division 1, with no addenda subject to the limitations and modification identified in 10 CFR 50.55a(b)(1) and the piping analysis criteria and methods, modeling techniques, and pipe support criteria described in Reference 2.

A design specification is required by Section III of the ASME Code for Class 1, 2, and 3 components, piping, supports, and core support structures. In addition, the ASME Code requires design reports for all Class 1, 2, and 3 components, piping, supports and core support structures documenting that the as-designed and as-built configurations adhere to the requirements of the design specification. A COL applicant that references the U.S. EPR design certification will prepare the design specifications and design reports for ASME Class 1, 2, and 3 components, piping, supports and core

04.05.02-10

support structures that comply with and are certified to the requirements of Section III of the ASME Code. The COL applicant will address the results and conclusions from the reactor internals material reliability programs applicable to the U.S. EPR reactor internals with regard to known aging degradation mechanisms such as irradiation-assisted stress corrosion cracking or void swelling addressed in Section 4.5.2.1.

Other sections that relate to this section are described below:

- Section 3.9.6 describes the snubber inspection and test program.
- Section 3.10 describes the methods and criteria for seismic qualification testing of Seismic Category I mechanical equipment and a description of their seismic operability criteria.
- Section 3.12 describes the design of systems and components that interface with the RCS with regard to intersystem LOCAs.
- Section 3.13 describes bolting and threaded fastener adequacy and integrity.
- Section 5.2.2 describes the pressure-relieving capacity of the valves specified for RCPB.
- Section 10.3 describes the pressure-relieving capacity of the valves specified for the steam and feedwater systems.

3.9.3.1 Loading Combinations, System Operating Transients, and Stress Limits

Section 3.9.3.1.1 describes the design and service level loadings used for the design of ASME Class 1, 2, and 3 components, piping, supports, and core support structures,



4.5.2.1 Materials Specifications

The major components for the reactor internals are fabricated from austenitic stainless steel except for the hold-down spring, which is made from martensitic stainless steel and pins and inserts which are coated with Stellite 6 or equivalent which is a cobalt alloy. The materials specifications for the reactor internals and core support materials including weld filler materials are listed in Table 4.5-2—Reactor Vessel Internal Materials, which includes the use of ASME Code Case N-60-5 which is listed as an acceptable code case under RG 1.84. There are no other materials used in the reactor internals or core support structures that are not otherwise allowed under ASME Code, Section III, Subsection NG-2120 (Reference 4). Reactor internals and core support structure weld filler materials are specified in ASME BPV Code, Section II (Reference 2) which is in accordance with GDC 1 and 10 CFR 50.55(a).

04.05.02-10

Design of the reactor internals considers the estimated peak neutron fluence to which the materials may be subjected. The reactor internals materials are evaluated for susceptibility to known aging degradation mechanisms such as irradiation-assisted stress corrosion cracking and void swelling that have been identified in current operating pressurized water reactors and are being addressed in the reactor internals material reliability programs.

4.5.2.2 Controls on Welding

The controls on welding of austenitic stainless steel pressure boundary components provided in Section 5.2.3 apply to the welding of reactor internals and core support components. When Section 5.2.3 is applied to the reactor internals and core support materials, ASME BPV Code, Section III (Reference 4) applies as in accordance with GDC 1 and 10 CFR 50.55(a).

4.5.2.3 Nondestructive Examination

Nondestructive examination (NDE) of base materials is in accordance with ASME Code Section III, Division I, NG-2500 (Reference 4). The NDE methods and acceptance criteria for welds are in accordance with the requirements of the ASME Code Section III, Division 1, NG-5000 (Reference 4) and GDC 1 and 10 CFR 50.55(a).

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel Components

The details provided in Section 5.2.3 concerning the processing, inspections, and tests on unstabilized austenitic stainless steel components to minimize susceptibility to intergranular corrosion caused by sensitization are applicable to the austenitic stainless steel materials used in the reactor internals and core support structures. Section 5.2.3 verifies compliance of reactor internals and core support structures with RG 1.44. The reactor internals and core support structures are fabricated from low carbon austenitic stainless steels which are heat treated in accordance with RG 1.44 to minimize their



Table 4.5-2—Reactor Vessel Internal Materials

Component	Material Specifications
Lower Internals Assembly	ASME SA-182 Grade F304LN (see Notes 1&2) ASME SA-336 Grade F304LN (see Notes 1&2) ASME SA-240 Type 304LN (see Notes 1&2) ASME SA-479 Type 304LN (see Notes 1&2) ASME SA-479 Type 316 Strain Hardened Level 1 (Code Case N-60-5) Carbon content shall be 0.03 wt% or less ASME SB-168 UNS-N06690 ASME SB-637 UNS-N07750, Type 2 Stellite 6 (see Note 3) or equivalent (hard facing)
Upper Internals Assembly	ASME SA-182 Grade F304LN (see Notes 1&2) ASME SA-376 Grade TP304LN (see Notes 1&2) ASME SA-240 Type 304LN (see Notes 1&2) ASME SA-479 Type 304LN (see Notes 1&2) ASME SA-479 Type 316 Strain Hardened Level 1 (Code Case N-60-5) Carbon content shall be 0.03 wt% or less Stellite 6 (see Note 3) or equivalent (hard facing)
Heavy Reflector	ASME SA-336 Grade F304LN (see Notes 1&2) ASME SA-240 Type 304LN (see Notes 1&2) ASME SA-336 Grade F304LN (see Notes 1&2) ASME SA-479 Type 304LN (see Notes 1&2) ASME SA-479 Type 316 Strain Hardened Level 1 (Code Case N-60-5) Carbon content shall be 0.03 wt% or less Stellite 6 (see Note 3) or equivalent (hard facing)
Control Rod Guide Assembly	ASME SA-182 Grade F304LN (see Notes 1&2) ASME SA-240 Type 304LN (see Notes 1&2) ASME SA-479 Type 304LN (see Notes 1&2) ASME SA-376 Grade TP304LN (see Notes 1&2)
Hold Down Spring	ASME SA-182 Grade F6NM
Reactor Vessel Internals Welds	Type 308L/309L/316L austenitic stainless steel per SFA 5.4, 5.9, or 5.22

04.05.02-9

Notes:

1. Solution annealed and rapidly cooled.
2. Carbon content not exceeding 0.03 wt%.

3. ASME SFA-5.21 ERCCoCr-A or ERCoCr-A, ASME SFA-5.13 ECoCr-A.

04.05.02-10 →

Next File

3.9.5.1.2.6 Flow Distribution Device

The flow distribution device is located below, and attached to, the LSP. The flow distribution device is composed of a distribution plate and support columns. The flow distribution device provides a homogeneous flow distribution between the LSP holes.

3.9.5.1.2.7 Heavy Reflector

The heavy reflector is located inside the core barrel between the core and core barrel shells. The heavy reflector increases neutron efficiency due to its neutron reflective properties, protects the RPV from radiation-induced embrittlement, improves the long-term mechanical behavior of the lower internals, and provides lateral support to maintain the geometry of the core. To avoid any welded or bolted connections close to the core, the heavy reflector consists of stacked slabs positioned one above the other (see Figure 3.9.5-3—Reactor Pressure Vessel Heavy Reflector). The heavy reflector rests on the LSP, but does not contact the UCP. The internal contour of the slabs conforms to the core, while the external contour is cylindrical. The top slab is fitted with alignment pins that extend through the UCP to provide proper alignment.

Since the heavy reflector is located between the core and the core barrel, it limits the core bypass flow at the core periphery. It also provides lateral support to the core and contributes to the decrease of neutron fluence on the RPV inner wall.

Additional information on the heavy reflector is provided in 4.3 Nuclear Design.

3.9.5.1.3

Vertical keys are installed into keyways machined into the external contour of the slabs to provide additional lateral and vertical restraint. The reentrant corners of the keyways within the slabs are provided with a small radius.

Internals, and are described in further detail below. The primary functions of the upper internals are:

- Support, locate, restrain, protect, and guide the core components.
- Direct the coolant flow from the core outlet to the RPV outlet nozzles.
- Permit core loading, unloading, and reloading.
- Support, align, and protect the rod cluster control assemblies (RCCAs).
- Guide, support, and protect the incore instrumentation.

The upper internals consist of the:

- Upper support assembly (including the flange, shell, and USP).
- Upper core plate.
- Control rod guide assemblies (CRGAs).

Response to

Request for Additional Information No. 343, Supplement 1

12/16/2009

U.S. EPR Standard Design Certification

AREVA NP Inc.

Docket No. 52-020

SRP Section: 04.05.01 - Control Rod Drive Structural Materials

Application Section: 4.5.1

**QUESTIONS for Component Integrity, Performance, and Testing Branch 1
(AP1000/EPR Projects) (CIB1)**

DRAFT

Question 04.05.01-6:

OPEN ITEM

Figure 05.02.03-12-1 in RAI response 05.02.03-12, dated November 10, 2008, indicates that F347 material will be used to fabricate part of the CRDM pressure housing. However, Table 5.2-2 does not list a forging specification for Grade 347 material. The staff requests that the applicant modify FSAR Table 5.2-2 to list material specifications and grades for all CRDM pressure boundary components.

Response to Question 04.05.01-6:

Three parts were inadvertently marked with the label "F347" in Figure 05.02.03-12-1 (i.e., Parts 1, 3, and 5) of the response to RAI 88, Question 05.02.03-12. The label "F347" should not have contained the letter F. These parts are fabricated from ASME material specification SA-479 Grade 347, which is already listed in U.S. EPR FSAR Tier 2, Table 5.2-2.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

DRAFT

Question 04.05.01-7:**OPEN ITEM**

The staff requested, in RAI 88 Question 05.02.03-1, that the applicant delete SA-479 UNS S41500 from Table 5.2-2 for use in the CRDM pressure housing, provide an alternative material or take the appropriate steps to have SA-479 UNS S41500 included in Table 2A by ASME Code, Section III. The applicant responded, by letter dated November 10, 2008, and stated that it has submitted a request to ASME Code to extend the properties currently provided in Section II Part D for SA-182 Grade F6NM (UNS S41500) to SA-479 (UNS S41500) material. The applicant stated that it expects the Code Case to be issued in the near future. The staff requests that the applicant notify the NRC staff when ASME Code has approved the code case requested by the applicant.

Response to Question 04.05.01-7:

- The request to extend the properties currently provided in Section II Part D for SA-182/182M Grade F6NM to SA-479/479M UNS S41500 was approved October 12, 2009 by ASME as Code Case N-785.
- The basis for approval of this code case is that the chemical composition, material properties, and heat treatment requirements for SA-182/182M Grade F6NM and SA-479/479M UNS S41500 are virtually identical. As this code case has not yet been added to RG 1.84, U.S. EPR FSAR Tier 2, Section 5.2.3.1 will be revised to specifically address acceptability of this code case for use.
- The appropriate control rod drive mechanism (CRDM) materials listed in U.S. EPR FSAR Tier 2, Table 5.2-2 for "Control Rod Drive Mechanism" and in U.S. EPR FSAR Tier 2, Section 4.5.1.1 and Section 4.5.1.3 will be modified to reflect this code case. This code case will also be added to U.S. EPR FSAR Tier 2, Table 5.2-1.
- Code cases listed in U.S. EPR FSAR Tier 2, Table 5.2-1 include those applicable to ASME Section XI and the ASME OM Code. U.S. EPR FSAR Tier 2, Section 5.2.1.2 will be revised to clarify that code cases used in the U.S. EPR design certification are listed in Table 5.2-1, not just code cases applicable to ASME Section III.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 4.5.1.1, Section 4.5.1.3, Section 5.2.1.2, Section 5.2.3.1, Table 5.2-1, and Table 5.2-2 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

DRAFT

4.5 REACTOR MATERIALS

4.5.1 Control Rod Drive System Structural Materials

GDC 1 and 10 CFR 50.55(a) establish the requirements regarding structures, systems, and components (SSC) important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. The specifications and design requirements of the materials selected for the control rod drive mechanism (CRDM) are described in Sections 3.9.4, 4.5, and 5.2.3.

GDC 14 establishes requirements regarding the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The pressure boundary of the CRDM is designed in accordance with ASME Code and the materials are selected based on compatibility with their environment as described in Sections 3.9.4, 4.5, and 5.2.3.

GDC 26 establishes the requirements regarding control rods being capable of reliable control of reactivity changes to prevent exceeding fuel design limits under conditions of normal operation, including anticipated operational occurrences. The CRDM material selection and fabrication support reliable rod movement for reactivity control, which is addressed in Sections 3.9.4, 4.5, and 5.2.3.

4.5.1.1 Materials Specifications

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Parts exposed to reactor coolant are made of corrosion resistant materials. The CRDM pressure boundary materials exposed to reactor coolant include Type 347 stabilized austenitic stainless steel and ASME SA-479 UNS S41500 (Code Case N-785)/SA-182 Grade F6NM (UNS S41500) martensitic stainless steel. The CRDM pressure boundary bolting-studs materials not exposed to reactor coolant include Alloy A-286 austenitic stainless steel bolting studs as well as martensitic stainless steel nuts. These materials are listed in Table 5.2-2.

The CRDM pressure boundary materials and pressure boundary weld filler material, which includes Type 347 austenitic stainless steel and alloy 52/52M/152 nickel base alloys, meet the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB (Reference 1). No Alloy 600 base metal or Alloy 82/182 weld metals are used in the CRDM pressure boundary in accordance with GDC 1 and 10 CFR 50.55(a).

Materials used in the CRDM internals are selected based on a proven AREVA design with 30 years of operating experience. CRDM internals are non-pressure boundary and non-structural components, thus the CRDM internals material specifications are not required to be ASME materials. CRDM internals material specifications are typically per European standards and are listed in Table 4.5-1—Control Rod Drive

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austenitic stainless steel used in these applications is ASME SA-479 UNS S41500 (Code Case N-785)/SA-182 Grade F6NM (UNS S41500). This material is martensitic stainless steel and is delivered in the quenched and tempered condition. The material is tempered between 1050°F and 1120°F as required by the ASME material specifications.

Materials other than austenitic stainless steel used in the non-pressure boundary components of the CRDM include martensitic stainless steel, cobalt-chromium alloy, nickel-base materials, and cobalt base material. The materials not used in pressure boundary applications are selected based on a proven German design with 30 years of operating experience. Materials are selected for their compatibility with the reactor coolant, as described in ASME articles NB-2160 and NB-3120 (Reference 1).

The martensitic stainless steel base metal used in the non-pressure boundary components is delivered in the quenched and tempered condition; tempering is performed at a temperature to between 1256°F and 1436°F.

The cobalt-chromium alloy is delivered in the solution annealed condition.

The nickel-base alloy used is a precipitation hardenable alloy which is extremely resistant to chemical corrosion and oxidation. It is supplied in the solution annealed (followed by quenching) and thermally aged condition for optimum resistance to stress corrosion cracking.

The cobalt alloy is only used in a very small portion of the CRDM where an alternate material will not perform satisfactorily. It has a very low susceptibility to corrosion.

The sliding surfaces of the latch unit are hard chromium plated. This material is only used in a very small portion of the CRDM where an alternate material will not perform satisfactorily.

4.5.1.4 **Cleaning and Cleanliness Control**

Cleanliness of the CRDMs is controlled during manufacture and installation per the requirements of ASME NQA-1-1994 (Reference 3) and RG 1.37 as addressed in Section 5.2.3.

4.5.2 **Reactor Internals and Core Support Materials**

GDC 1 and 10 CFR 50.55(a) establish the requirements regarding SSC important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. The specifications and design requirements of the materials selected for the reactor internals and core support structures are described in Sections 3.9.5, 4.5, and 5.2.3.

5.2 Integrity of the Reactor Coolant Pressure Boundary

This section describes the measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant design lifetime. Consistent with the definition in 10 CFR 50.2, the U.S. EPR RCPB includes all pressure-containing components, such as pressure vessels, piping, pumps, and valves which are part of the reactor coolant system (RCS) or connected to the RCS, up to and including these:

- The outermost containment isolation valve in system piping which penetrates primary reactor containment.
- The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment.
- The RCS safety and relief valves.

Section 3.9 presents the design transients, loading combinations, stress limits, and evaluation methods used in the design analyses of RCPB components and supports to demonstrate that RCPB integrity is maintained.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR 50.55a

The RCPB components are designed and fabricated as Class 1 components in accordance with Section III of the ASME Boiler and Pressure Vessel Code (Reference 1), except for components that meet the exclusion requirements of 10 CFR 50.55a(c) which are designed and fabricated as Class 2 components. The RCPB component classification complies with the requirements of GDC 1 and 10 CFR 50.55a. Table 3.2.2-1—Classification Summary lists the RCPB components, including pressure vessels, piping, pumps, and valves, along with the applicable component codes. Other safety-related plant components are classified in accordance with RG 1.26, as specified in Section 3.2.

The code of record for the design of the U.S. EPR is the 2004 edition of the ASME Boiler and Pressure Vessel Code (no addenda).

The application of Section XI of the 2004 edition of the ASME Boiler and Pressure Vessel Code to the U. S. EPR is described in Section 5.2.4 and Section 6.6 The application of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) (Reference 2) is described in Section 3.9.6.

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5.2.1.2 Compliance with Applicable Code Cases

ASME Section III Code Cases acceptable for use in the U.S. EPR design, subject to the limitations specified in 10 CFR 50.55a, are listed in RG 1.84. Code Cases pertaining to

04.05.01-7 →

ASME Section III, Division 2 are addressed in Section 3.8. Table 5.2-1—ASME Code Cases lists the specific Code Cases used in the U.S. EPR design. A COL applicant that references the U.S. EPR design certification will identify additional ASME Code Cases to be used. Code Cases pertaining to ASME Code Section III, Division 2 are addressed in Section 3.8. ASME Section XI Code Cases acceptable for use for preservice inspection and inservice inspection (ISI), subject to the limitations specified in 10 CFR 50.55a, are listed in RG 1.147 and described in Section 5.2.4 and Section 6.6. ASME OM Code Cases acceptable for use for preservice testing and inservice testing (IST), subject to the limitations specified in 10 CFR 50.55a, are listed in RG 1.192 and described in Section 3.9.6.

Table 5.2-1—ASME Code Cases lists the specific Code Cases used in the U.S. EPR design. A COL applicant that references the U.S. EPR design certification will identify additional ASME Code Cases to be used.

5.2.2 Overpressure Protection

Pressurizer safety relief valves (PSRV) protect the RCPB from overpressure during power operation and during low temperature operation. Auxiliary and emergency systems connected to the RCS are not utilized for RCPB overpressure protection.

Main steam safety valves (MSSV) and main steam relief trains protect the secondary side of the steam generators from overpressure. Secondary side overpressure protection is addressed in Section 10.3.

5.2.2.1 Design Bases

Component design bases for the PSRVs and the secondary side overpressure protection devices are addressed in Section 5.4.13 and Section 10.3, respectively.

The PSRVs are part of the RCPB and are designed to meet the requirements for ASME Section III, Class 1 components (GDC 1, GDC 30, 10 CFR 50.55a). Component classifications are presented in Section 3.2.

The opening set pressures and capacity of the PSRVs are sufficient to limit the RCS pressure to less than 110 percent of the RCPB design pressure during any condition of normal operation, including anticipated operational occurrences (AOO) (GDC 15). The bounding design transient for RCPB overpressure is a turbine trip at full power. This transient bounds all upset, emergency and faulted conditions identified in Section 3.9.1.

The PSRVs maintain the RCS pressure below brittle fracture limits when the RCPB is stressed under operating, maintenance, testing, and postulated accident conditions, including low temperature operation, so that the RCPB behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized (GDC 31).

NiCrFe Alloy 600 base metal or Alloys 82/182 weld metal is not used in RCPB applications. NiCrFe Alloy 690 base metal has controlled chemistry, mechanical properties, and thermo-mechanical processing requirements that produce an optimum microstructure for resistance to intergranular corrosion. Alloy 690 is solution annealed and thermally treated to optimize the resistance to intergranular corrosion.

Alloy 690 and its weld filler metals (Alloy 52/52M/152) in contact with RCS primary coolant have limited cobalt content not exceeding 0.05 wt%. Alloy 690 in contact with RCS primary coolant has limited sulfur content not exceeding 0.02 wt%.

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Code Case N-785 has been applied to CRDM materials as shown in Table 5.2-2. This code case was approved by the ASME Boiler and Pressure Vessel Standards Committee on October 12, 2009. The basis for approval of this case is derived from the fact that the chemical composition, material properties, and heat treatment requirements for SA-182/182M Grade F6NM and SA-479/479M (UNS S41500) are virtually identical.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 Reactor Coolant Chemistry

The RCS water chemistry is controlled to minimize negative impacts of chemistry on materials integrity, fuel rod corrosion, fuel design performance, and radiation fields, and is routinely analyzed for verification. The water chemistry parameters are based on industry knowledge and industry experience as summarized in the EPRI PWR Primary Water Chemistry Guidelines (Reference 3).

The chemical and volume control system (CVCS) provides the primary means for maintaining the required volume of water in the RCS and for the addition of chemicals. The design of the CVCS allows for the addition of chemicals to the RCS to control pH, scavenge oxygen, control radiolysis reactions, and maintain corrosion product particulates within a specified range. Table 5.2-3—Reactor Coolant Water Chemistry - Control Parameters shows the control values for the reactor coolant chemistry parameters and impurity limitations during power operation. These criteria conform to the recommendations of RG 1.44 and the EPRI PWR Primary Water Chemistry Guidelines report.

Enriched boric acid (EBA) is added to the RCS as a soluble neutron poison for core reactivity control. Lithium hydroxide enriched in lithium 7 is used as a pH control agent to maintain a slightly basic pH at operating conditions. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/nickel-base alloy systems. Lithium-7 is also produced in solution from the neutron irradiation of the dissolved boron in the coolant.

In addition to degasification during startup, two chemicals are added to the reactor coolant to control oxygen: (1) hydrazine during startup operations below 250°F; and

Table 5.2-1—ASME Code Cases

Code Case Number	Title
N-60-5	Material for Core Support Structures Section III, Division 1
N-71-18	Additional Materials for Subsection NF, Class 1, 2, 3, and MC Supports Fabricated by Welding, Section III, Division 1
N-284-1 ¹	Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC
N-319-3	Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping, Section III, Division 1
N-729-1	Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining, Partial-Penetration Welds
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N-785 ³	Use of SA-479/SA-479M, UNS S41500 for Class 1, Welded Construction Section III, Division 1
OMN-1, Revision 0 ²	Alternative Rules for Preservice and Inservice Testing of Active Electric Motor-Operated Valve Assemblies in Light-Water Reactor Power Plants
OMN-13, Revision 0 ²	Requirements for Extending Snubber Inservice Visual Examination Interval at LWR Power Plants

NOTES:

1. See Section 3.8 for use.
2. See Section 3.9.6 for use.
3. See Section 5.2.3.1 for use.

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3. See Section 5.2.3.1 for use.

Table 5.2-2—Material Specifications for RCPB Components
Sheet 4 of 5

Component	Material
Latch housing 04.05.01-7 →	ASME SA-479 UNS S41500 (Code Case N-785)/SA-182 Grade F6NM (see Note 1) (UNS S41500)
Seamless tube	ASME SA-312 Grade TP347 (Seamless) (see Note 3)
Bolt	ASME SA-453 Grade 660 (see Note 7)
Nut	ASME SA-437 Grade B4C (see Note 1)
Welding filler material	SFA 5.4 E347 SFA 5.9 ER347 SFA 5.14 ERNiCrFe-7 SFA 5.14 ERNiCrFe-7A
RCPB Valves Pressurizer Safety Relief Valves	
Bodies A vendor for the PSRV has not been chosen for the U.S. EPR	SA-182 Grade F304 (see Notes 3 & 4), Grade F304L (see Note 3), Grade F304LN (see Note 3), Grade F316 (see Notes 3 & 4), Grade F316L (see Note 3), Grade F316LN (see Note 3) SA-351 Grade CF3, Grade CF3A, Grade CF3M, Grade CF8 (see Note 4), Grade CF8A (see Note 4), Grade CF8M (see Note 4 & 10)
Bonnets	SA-182 Grade F304 (see Notes 3 & 4), Grade F304L (see Note 3), Grade F304LN (see Note 3), Grade F316 (see Notes 3 & 4), Grade F316L (see Note 3), Grade F316LN (see Note 3) SA-351 Grade CF3, Grade CF3A, Grade CF3M, Grade CF8 (see Note 4), Grade CF8A (see Note 4), Grade CF8M (see Note 4 & 10) SA-240 Type 304 (see Notes 3 & 4), Type 304L (see Note 3), Type 304LN (see Note 3), Type 316 (see Notes 3 & 4), Type 316L (see Note 3), 316LN (see Note 3)
Discs	SA-182 Grade F304 (see Notes 3 & 4), Grade F304L (see Note 3), Grade F304LN (see Note 3), Grade F316 (see Notes 3 & 4), Grade F316L (see Note 3), Grade F316LN (see Note 3) SA-351 Grade CF3, Grade CF3A, Grade CF3M, Grade CF8 (see Note 4), Grade CF8A (see Note 4), Grade CF8M (see Note 4 & 10) SA-479 Type 304 (see Notes 3 & 4), Type 304L (see Note 3), Type 304LN (see Note 3), Type 316 (see Notes 3 & 4), Type 316L (see Note 3), 316LN (see Note 3), XM-19 (see Note 3) SA-564 Type 630 (Conditions H1075, H1100, H1150) SB-637 UNS N07718 (see Note 8)
Stems	SA-479 Type 304 (see Notes 3 & 4), Type 304L (see Note 3), Type 304LN (see Note 3), Type 316 (see Notes 3 & 4), Type 316L (see Note 3), Type 316LN (see Note 3), Type XM-19 (see Notes 3 & 9) SA-564 Type 630 SB-637 UNS N07718 (see Note 8)

Response to

Request for Additional Information No. 344(4062), Supplement 1

02/23/2010

U.S. EPR Standard Design Certification

AREVA NP Inc.

Docket No. 52-020

SRP Section: 04.03 - Nuclear Design

Application Section: 04.03

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)

DRAFT

Question 04.03-27:

OPEN ITEM

The staff requests that a combined license (COL) information item to be added to Table 1.8-2 of the FSAR in regard to collection of plant specific surveillance capsule data to be used to benchmark BAW-2241PA's applicability to the specific plant.

The capsule withdrawal and reporting requirements will follow 10 CFR Part 50 Appendix H.

Response to Question 04.03-27:

U.S. EPR FSAR Tier 2, Table 1.8-2 will be revised to include a COL Item for U.S. EPR FSAR Tier 2, Section 5.3.1.6.2, Plant Specific Monitoring. U.S. EPR FSAR Tier 2, Section 5.3.1.6.2 will be revised to state that a COL applicant that references the U.S. EPR design certification will provide plant-specific surveillance capsule data to benchmark BAW-2241P-A and demonstrate applicability to the specific plant.

FSAR Impact:

The U.S. EPR FSAR Tier 2, Table 1.8-2 and Section 5.3.1.6.2 will be revised as described in the response and indicated on the enclosed markup.

DRAFT

Question 04.03-28:

OPEN ITEM

Throughout the U.S. EPR Final Safety Analysis Report (FSAR) Tier 2, Section 4.3, AREVA NP refers to licensing topical report ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology Topical Report." This document is currently under review by the NRC staff. This RAI is created to track an open item associated with this review. It will be closed upon completion of the review by the NRC staff. AREVA is requested to acknowledge receipt of this open item.

Response to Question 04.03-28:

AREVA NP acknowledges receipt of this open item.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

DRAFT

U.S. EPR Final Safety Analysis Report Markups

DRAFT



Table 1.8-2—U.S. EPR Combined License Information Items
Sheet 24 of 49

Item No.	Description	Section	Action Required by COL Applicant	Action Required by COL Holder
5.2-2	A COL applicant that references the U.S. EPR design certification will identify additional ASME code cases to be used.	5.2.1.2	☒	
5.2-3	A COL applicant that references the U.S. EPR design certification will identify the implementation milestones for the site-specific ASME Section XI preservice and inservice inspection program for the reactor coolant pressure boundary, consistent with the requirements of 10 CFR 50.55a (g). The program will identify the applicable edition and addenda of the ASME Code Section XI, and will identify additional relief requests and alternatives to Code requirements.	5.2.4	☒	
5.3-1	A COL applicant that references the U.S. EPR design certification will identify the implementation milestones for the material surveillance program.	5.3.1.6	☒	
5.3-2	A COL applicant that references the U.S. EPR design certification will provide a plant-specific pressure and temperature limits report (PTLR), consistent with an approved methodology.	5.3.2.1		☒
5.3-3	A COL applicant that references the U.S. EPR design certification will provide plant-specific RT _{PTS} values in accordance with 10 CFR 50.61 for vessel beltline materials.	5.3.2.3		☒
5.3-4	A COL applicant that references the U.S. EPR design certification will provide plant-specific surveillance data to benchmark BAW-2241P-A and demonstrate applicability to the specific plant.	5.3.1.6.2		
5.4-1	A COL applicant that references the U.S. EPR design certification will identify the edition and addenda of ASME Section XI applicable to the site specific Steam Generator inspection program.	5.4.2.5.2.2	☒	

04.03-27

specimens; i.e. major axis of the specimen is parallel to the surface and normal to the major working direction (the transverse direction). The CT specimens and Charpy V-notch specimens from the weld metal are oriented so that the major axis of the specimen (axis normal to the crack plane for CT specimens) is parallel to the RV inside surface and normal to the weld bead direction. Weld metal tension specimens are oriented in the same direction as the Charpy V-notch specimens with the gage length consisting entirely of weld metal (the transverse direction). The Charpy V-notch specimens from the HAZ are oriented so that the major axis of the specimen is parallel to the RPV inside surface and normal to the weld bead direction. The Charpy V-notch root is in the HAZ about 1/32 inch from the fusion line.

5.3.1.6.1 Fluence Monitoring

The neutron fluence on the vessel material test specimens and the vessel itself is determined based on core-follow calculations of the cycle-by-cycle operation. The fluence and uncertainty methodologies, described in BAW-2241P-A, "Fluence and Uncertainty Methodologies" (Reference 9), explain how the calculations are performed. The calculations conform to RG 1.190 and thus meet the requirements of 10 CFR Part 50, Appendix H.

As noted in RG 1.190, the bases for the bias and random uncertainties in the calculations are:

- Database of dosimetry measurements.
- Benchmark database comparing calculations to measurements.
- Sensitivity evaluation with fabrication and operational tolerances.

5.3.1.6.2 Plant Specific Monitoring

The uncertainty evaluations noted in BAW-2241P-A provide calculations, with well-defined uncertainties, for RPV fluence in operating light water reactors. While it is expected that the calculations for the U.S. EPR will have similar accuracy and random uncertainties, measured data from the material surveillance program will supplement the calculated predictions. A COL applicant that references the U.S. EPR design certification will provide plant-specific surveillance capsule data to benchmark BAW-2241P-A and demonstrate applicability to the specific plant. The capsule withdrawal and reporting requirements will follow 10 CFR Part 50, Appendix H. The recommended withdrawal schedule is outlined in Table 5.3-6—Surveillance Specimen Withdrawal Schedule Per ASTM E185-82.

04.03-27

Calculations are used to estimate the initial fluence to the vessel materials. Once operation has commenced, plant specific dosimetry measurements are evaluated to demonstrate that fluence uncertainties are consistent with historical data. Showing

Response to

Request for Additional Information No. 367

3/11/2010

U.S. EPR Standard Design Certification

AREVA NP Inc.

Docket No. 52-020

SRP Section: 04.06 - Functional Design of Control Rod Drive System

Application Section: 04.06

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)

DRAFT

Question 04.06-14:

OPEN ITEM:

In general terms of the U.S. EPR Tier 2 FSAR Subsections 4.4.2.9.5, 4.4.4.3, 4.4.4.5.3, 4.4.4.5.4, 4.4.6.1, 4.4.6.4, and 4.4.6.5, the applicant provides a brief description of the fixed in-core SPND neutron flux measurements in relationship to the two types of in-core trips with regard to uncertainties in the calculations, influence of power distributions, high linear power density functions, low DNBR I&C functions, and analysis of steady and transient conditions. In these subsections, the applicant refers to the topical report ANP-10287P, "In-core Trip Setpoint and Transient Methodology for the U.S. EPR Topical Report" for a more detail discussion.

However, after review of these subsections and the topical report, the staff has determined that more information is necessary to determine how the methods described in ANP-10287P will be implemented and verified in regard to instrumentation and control systems.

Therefore, the staff requests that the applicant provide a description on how the methodologies contained in ANP-10287P will be implemented and verified for the U.S. EPR design relating to instrumentation and control systems. In particular, define what will be checked and verified by COL applicants prior to the first cycle core loading and as part of reload analysis to satisfy the following approval limitations stated in the SE for ANP-10287P:

1. LIMITATION NO. 1 – MIXED CORES

Section 4.8 contains an evaluation of the applicability of the U.S. EPR setpoint methodology to mixed core configurations. The following limitation applies to the application of ANP-10287P, Revision 0:

The setpoint methodology documented in ANP-10287P, Revision 0 [1] is only acceptable to cores that consist entirely of hydraulically compatible fuel assemblies, i.e, a single package of assembly specific CHF correlation.

2. LIMITATION NO. 2-CYCLE SPECIFIC UNCERTAINTY VALUES

Since the actual uncertainties and setpoint values are not part of this review and are not available to the staff, any transient analyses taking credit for the in-core setpoint system can only be approved when actual values of these uncertainties and setpoints are conservatively applied following this methodology, or it has been demonstrated that the uncertainties can be conservatively bounded. The following limitation applies to the application of ANP-10287P, Revision 0:

Applications of the setpoint methodology documented in ANP-10287P, Revision 0 [1] must include a review of the applicable uncertainty values used to generate the setpoint values used in the analyses.

3. APPLICATION SPECIFIC ITEM PRIOR TO THE FIRST CYCLE OPERATION

The methodology to confirm that the static setpoint values provide adequate protection during transient events described in Section 9 of Topical Report ANP-10287P depends on identifying and characterizing the limiting transient. The limiting transient is defined as the event that

results in the minimum difference between uncompensated DNBR and LPD results and the SAFDLs. The following limitation applies to the application of ANP-10287P, Revision 0:

Prior to the first cycle operation, confirmatory evaluation has to be performed for every AOO using the procedure described in Section 9 of Topical Report ANP-10287P to identify the limiting transient of the plant as built. Based on the confirmatory evaluation results, analyses have to be performed for AOOs that have significant differences between the assumed input conditions and the as-built conditions, if the differences can not be conservatively bounded by the assumed uncertainty values. For the most limiting transient that relies on in-core trip for protection, the applicant shall provide for staff review the analysis results demonstrating that the uncompensated DNBR and LPD satisfies SAFDL with a 95/95 assurance.

4. APPLICATION SPECIFIC ITEM FOR RELOAD ANALYSIS

The methodology described in Section 9 of Topical Report ANP-10287P is vague on which transient events are used to confirm that the static setpoint values provide adequate protection during transient events. Therefore, the following limitation applies to the application of ANP-10287P, Revision 0:

During reload analysis, it has to be confirmed and appropriately documented using the methodology described in Section 9 of ANP-10287P that the static setpoint value provides adequate protection for at least the three most limiting AOOs identified by Item 3 above.

Response to Question 04.06-14

It is anticipated that the NRC safety evaluation for the AREVA NP Topical Report ANP-10287P will include the limitations specified in this question. Because these will be limitations on the approved methodology defined in the topical report, the plant and cycle specific implementation of the methodology, whether performed by AREVA NP Inc. (AREVA NP) or a COL applicant, must satisfy the approved methodology. AREVA NP considers that the approved methodology consists of details defined in the body of the topical report, RAI responses incorporated into the approved version of the report, and limitations defined in the NRC safety evaluation.

Therefore the implementation of this methodology will require that the following be verified with respect to the limitations defined in the NRC safety evaluation.

1. Characteristics of the fuel assemblies will be examined to verify that they are all hydraulically compatible based on the criterion that a single package of assembly specific critical heat flux (CHF) correlations can be used to evaluate the assembly performance.
2. Uncertainties used in the setpoint analyses will be verified to be appropriate for the plant and cycle being analyzed.
3. A COL applicant that references the U.S. EPR design certification will provide for staff review, prior to the first cycle of operation, analysis results demonstrating that the uncompensated departure from nucleate boiling ratio (DNBR) and linear power density (LPD) satisfies the specified acceptable fuel design limits (SAFDL) with a 95/95 assurance.
4. Current analysis results will be reviewed for subsequent cycles to confirm that the static setpoint value provides adequate protection for at least the three limiting anticipated

operational occurrences (AOO). This review will be documented in the quality assurance records.

Table 1.8-2 of the U.S. EPR FSAR Tier 2 will be revised to include a COL item in Section 15.0.0.3.9. Section 15.0.0.3.9 will be revised to state that a COL applicant that references the U.S. EPR design certification will provide for staff review, prior to the first cycle of operation, analysis results demonstrating that the uncompensated departure from nucleate boiling ratio (DNBR) and linear power density (LPD) satisfies the specified acceptable fuel design limits (SAFDL) with a 95/95 assurance.

FSAR Impact:

U.S. EPR FSAR Tier 2, Table 1.8-2 and U.S. EPR FSAR Tier 2, Section 15.0.0.3.9 will be revised as described in the response and indicated on the enclosed markup.

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U.S. EPR Final Safety Analysis Report Markups

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Table 1.8-2—U.S. EPR Combined License Information Items
Sheet 44 of 49

Item No.	Description	Section	Action-Required by COL Applicant	Action-Required by COL Holder
<u>14.2-12</u>	<u>A COL applicant that references the U.S. EPR design certification will provide site-specific test abstract information for plant laboratory equipment.</u>	<u>14.2.12</u>		
14.3-1	A COL applicant that references the U.S. EPR design certification will provide ITAAC for emergency planning, physical security, and site-specific portions of the facility that are not included in the Tier 1 ITAAC associated with the certified design (10 CFR 52.80(a)).	14.3	Y	
14.3-2	A COL applicant that references the U.S. EPR design certification will describe the selection methodology for site-specific SSC to be included in ITAAC, if the selection methodology is different from the methodology described within the FSAR, and will also provide the selection methodology associated with emergency planning and physical security hardware.	14.3	Y	
<u>14.3-3</u>	<u>A COL applicant that references the U.S. EPR design certification will identify a plan for implementing DAC. The plan will identify 1) the evaluations that will be performed for DAC, 2) the schedule for performing these evaluations, and 3) the associated design processes and information that will be available to the NRC for audit.</u>	<u>14.3</u>		
15.0-1	A COL applicant that references the U.S. EPR design certification will provide for staff review, prior to the first cycle of operation, the analysis results demonstrating that the uncompensated DNBR and LPD satisfies the SAFDL with a 95/95 assurance in accordance with ANP-10287P.	<u>15.0.0.3.9</u>		

04.06-14 ↗

- Spectrum of Rod Ejection Accidents.
- Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary.

Transient Analysis with Incore Trips

The transient analysis is performed with incore trip models decoupled from the system simulation code, S-RELAP5. The incore trip models are generically referred to as the “algorithm” or separately as the Low DNB Channel algorithm and High LPD Channel algorithm. The core boundary conditions for the algorithm are generated in S-RELAP5 and power distributions are generated in the nodal neutronics code, PRISM.

The Low DNB Channel and High LPD Channel algorithms are simulated to predict times at which the incore trip setpoints are reached, and to demonstrate the adequacy of the dynamic compensation on the trips. Table 15.0-7 lists the incore trip setpoints used in the accident analyses. The methodology for confirming the dynamic compensation is described in Section 9.4 of Reference 2.

The Low DNB Channel and High LPD Channel algorithms use the following measurements:

- The reactor power distributions derived from the SPNDs, which are part of the nuclear incore instrumentation.
- The primary system pressure derived from the primary pressure sensors.
- The core flow derived from the reactor coolant pump (RCP) speed sensors and the calibrated volumetric flow from a surveillance measurement.

04.06-14

- The reactor inlet temperature derived from the cold-leg temperature sensors.

• A COL applicant that references the U.S. EPR design certification will provide for staff review, prior to the first cycle of operation, the analysis results demonstrating that the uncompensated DNBR and LPD satisfies the SAFDL with a 95/95 assurance in accordance with ANP-10287P.

15.0.0.3.10 Plant Design Changes

The information presented in Section 15.0 represents the current U.S. EPR design. Some of the analyses presented in this section used slightly different values. In these cases the differences have been evaluated and found to have a negligible or conservative impact on the results and conclusions.

15.0.1 Radiological Consequence Analysis

This section is not applicable to new plants. The radiological consequences analyses are addressed in Section 15.0.3.

Response to

Request for Additional Information No. 366, Supplement 1

2/17/2010

U.S. EPR Standard Design Certification

AREVA NP Inc.

Docket No. 52-020

SRP Section: 04.06 - Functional Design of Control Rod Drive System

Application Section: 04.06

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)

DRAFT

Question 04.06-13:**OPEN ITEM:**

FSAR Tier 2, Section 4.6.4 states that in the safety analyses in FSAR Tier 2, Chapter 15, except for the large break loss of coolant accident, no credit is taken for reactivity control systems other than reactor trip to mitigate the events to achieve a stable plant condition. The staff notes that in FSAR Tier 2, Section 15.1.5 appears to indicate that boron addition via the SIS is credited to mitigate large steam line breaks from hot zero power conditions. Please clarify this apparent discrepancy.

Response to Question 04.06-13:

U.S. EPR FSAR Tier 2, Table 15.1-15, indicates that medium head safety injection (MHSI) is actuated for the most limiting main steam line break (MSLB) event. Although boron tracking is activated in the S-RELAP5 analysis of this event, the reactivity contribution of the boron (shown in U.S. EPR FSAR Tier 2, Figure 15.1-54) is not necessary for the event mitigation. U.S. EPR FSAR Tier 2, Table 15.1-15 shows that the peak return to power for the limiting MSLB event occurs at 273.2 seconds with a value of approximately 23 percent of rated power (see U.S. EPR FSAR Tier 2, Figure 15.1-55). However, borated MHSI fluid does not begin entering the reactor coolant system (RCS) until after 720 seconds (see U.S. EPR FSAR Tier 2, Table 15.1-15), well beyond the return to power peak. At this point in the transient, reactor power is approximately 3 percent of rated power (see U.S. EPR FSAR Tier 2, Figure 15.1-55). The power excursion is caused by moderator reactivity feedback resulting from the reduction in core inlet temperature due to secondary side overcooling. The power excursion is terminated when the affected steam generator dries out (see U.S. EPR FSAR Tier 2, Figure 15.1-49) and significant primary-to-secondary heat transfer ceases (see U.S. EPR FSAR Tier 2, Figure 15.1-48).

Boration resulting from MHSI actuation does not contribute to the mitigation of this event and the statement in U.S. EPR FSAR Tier 2, Section 4.6.4 is accurate.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.