ArevaEPRDCPEm Resource

From:	BRYAN Martin (EXT) [Martin.Bryan.ext@areva.com]
Sent:	Tuesday, April 27, 2010 2:33 PM
To:	Tesfaye, Getachew
Cc:	DELANO Karen V (AREVA NP INC); ROMINE Judy (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); WELLS Russell D (AREVA NP INC)
Subject:	Response to U.S. EPR Design Certification Application RAI No. 365, FSAR Ch. 5 OPEN ITEM
Attachments:	RAI 365 Response US EPR DC.pdf

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 365 Response US EPR DC.pdf" provides a technically correct and complete response to 5 of the 6 questions.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 365 Questions 05.02.05-9, 05.02.05-11, and 05.02.05-14.

The following table indicates the respective pages in the response document, "RAI 365 Response US EPR DC.pdf" that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 365 — 05.02.01.01-5	2	4
RAI 365 — 05.02.05-9	5	5
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The schedule for technically correct and complete responses to the remaining 1 question is provided below:

Question #	Response Date
RAI 365 — 05.02.01.01-5C	May 28, 2010

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To: ZZ-DL-A-USEPR-DL
Cc: Wu, Cheng-Ih; Hawkins, Kimberly; Li, Chang; Segala, John; Lee, Samuel; Jenkins, Joel; Terao, David; Roy, Tarun
Subject: U.S. EPR Design Certification Application RAI No. 365(4317,4318,4319), FSAR Ch. 5 OPEN ITEM

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on February 3, 2010, and discussed with your staff on March 2, 2010. Draft RAI Questions 05.02.01.01-5a was modified as a result of that discussion, and the staff has added 05.02.01.01-5b and 05.02.01.01-5c to address similar concerns that was not part of that discussion. The questions in this RAI are OPEN ITEMS in the safety evaluation report for Chapter 5 for Phases 2 and 3 reviews. As such, the schedule we have established for your application assumes technically correct and complete responses prior to the start of Phase 4 review. For any RAI that cannot be answered prior to the start of Phase 4 review, it is expected that a date for receipt of this information will be provided so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
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Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 1354

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Response to

Request for Additional Information No. 365(4317, 4318, 4319), Revision 0

3/31/2010

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 05.02.01.01 - Compliance With the Codes and Standards Rule, 10 CFR 50.55a
SRP Section: 05.02.05 - Reactor Coolant Pressure Boundary Leakage Detection
SRP Section: 05.03.01 - Reactor Vessel Materials
Application Section: Chapter 5

QUESTIONS for Engineering Mechanics Branch 1 (AP1000/EPR Projects) (EMB1)
QUESTIONS for Balance of Plant Branch 2 (ESBWR/ABWR) (SBPB)
QUESTIONS for Component Integrity, Performance, and Testing Branch 1 (AP1000/EPR Projects) (CIB1)

Question 05.02.01.01-5:

OPEN ITEM

Follow-up to RAI 51, Question 05.02.01.01-3

A. The applicant did not address how 10 CFR 50.55a(b)(1)(ii) is satisfied for the U.S. EPR design. The staff noted that the code of record for U.S. EPR design is the 2004 Edition of ASME code, and the piping stress analysis and seismic design are performed in accordance with the 1993 Addenda to the 1992 Edition to meet the requirement of 10 CFR 50.55a(b)(1)(iii). However, the use of either the 2004 Edition or the 1993 Addenda is disallowed by 10 CFR 50.55a(b)(1)(ii). AREVA is requested to provide the technical basis of how 10 CFR 50.55a(b)(1)(ii), "weld leg dimensions" is addressed while using the 2004 Edition and 1993 Addenda.

B. 10 CFR 50.55a requires that the Code edition and addenda to be applied to ASME Class 1, 2 and 3 piping and components must be determined by the rules of the ASME Section III paragraph NCA-1140(2), which disallows use of Code Edition and Addenda in the Design Specifications that is (a) earlier than three years prior to the date the construction permit application is docketed or (b) earlier than the latest Edition and Addenda endorsed by the regulatory authority at the time the construction permit application is docketed. This requirement is not satisfied by the code of record for U.S. EPR. For instance, the code used for U.S. EPR DC is the 2004 Edition with no Addenda while COL application date for Calvert Cliffs 3 and 4 is March 13, 2008. This implies a violation of NCA-1140(2)(a). To resolve the issue, ASME approved a Code Case N-782 in January 2009 which allows that the Code Edition and Addenda endorsed in a design certified or licensed by the regulatory authority may be used for systems and components as an alternative rule to NCA-1140(2)(a) and (2)(b). However, 10 CFR 50.55a requires that the optional ASME Code cases must be those listed in the NRC Regulatory Guide (RG) 1.84 that is incorporated by reference in paragraph 50.55a (b)(4). Code Case N-782 is not listed for acceptance in Revision 34 of RG 1.84. In order to apply the alternative rule to requirements of NCA-1140, the applicant is requested to provide justification for inclusion of Code Case N-782 in U.S. EPR DCD in accordance with 10 CFR 50.55a(3)(i) and (ii).

C. In its response to RAI 51, AREVA confirmed that the base code for USEPR design of piping systems, components and their supports is the 2004 Edition with no Addenda of the ASME Section III Code. As a result, AREVA is requested to revise its topical reports to base on the 2004 Edition for consistency with the EPR design as it relates to Code Edition and Addenda of U.S. EPR design certification. It is noted that if other Code Editions and Addenda than the 20004 Edition must be used for design of EPR safety related components, AREVA is requested to provide justification to reconcile the use of the other Code Edition and Addenda to the requirements of the 2004 Edition in accordance with NCA-1140 and 10 CFR 50.55a (a)(3). The staff notes that Section 3.12 of USEPR DCD identifies the Code Edition and Addenda by referring to the topical report ANP-10264NP where the 2001 Edition with the 2003 Addenda is used for design and analysis of piping and its supports. This implies that there are two Code Editions used by a USEPR Design. AREVA is also requested to discuss how the use of multiple code editions and addenda for a DCD design to satisfy the Section III Subsection NCA-1140(a)(1) which states that all items of a nuclear power plant may be constructed to a single Code Edition and Addenda, or each item may be constructed to individually specified Code Editions and Add

Response to Question 05.02.01.01-5:

A. U.S. EPR FSAR Tier 2, Section 5.2.1.1, “Compliance with 10 CFR 50.55a,” states:

> “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.”

Additionally, U.S. EPR FSAR Tier 2, Section 3.12.2, “Codes and Standards,” states:

> “Applicable codes and standards for piping and pipe supports are detailed in Section 2.0 and in Section 6.1 of Reference 1.”

Reference 1 is AREVA NP topical report ANP-10264NP-A. Section 2.1 of topical report ANP-10264NP-A states:

> “Piping analysis and pipe support design for the U.S. EPR addressed in this topical report use the 2001 ASME Code, Section III, Division 1, 2003 addenda[2] as the base code with limitations identified in the Code of Federal Regulations, 10 CFR 50.55a(b)(1). Accordingly, the 2001 Edition of the ASME Code, 2003 addenda, will be the design code for Class 1, 2, and 3 piping with the restriction that the treatment of dynamic loads, including seismic loads, in the pipe stress analyses will be according to sub-articles NB/NC/ND-3650 of the 1993 Addenda of the ASME Code[3].”

The reference made to 50.55a includes the appropriate subsections of 50.55a(b)(1,) including 10 CFR 50.55a(b)(1)(ii), "weld leg dimensions", with the exception of seismic design of piping which, as noted in the above paragraph, is performed in accordance with the 1993 Addenda to the 1992 Edition to meet the 10 CFR 50.55a(b)(1)(iii) requirement.

Please note that the Question inaccurately states:

> “However, the use of either the 2004 Edition or the 1993 Addenda is disallowed by 10 CFR 50.55a(b)(1)(ii).”

10 CFR 50.55a(b)(1)(ii) states:

> “Weld leg dimensions. When applying the 1989 Addenda through the latest edition, and addenda incorporated by reference in paragraph (b)(1) of this section, applicants or licensees may not apply paragraph NB–3683.4(c)(1), Footnote 11 to Figure NC–3673.2(b)–1, and Figure ND–3673.2(b)–1.”

10 CFR 50.55a(b)(1) incorporates, by reference, Section III of the ASME Boiler and Pressure Vessel Code, including the 1963 Edition through 1973 Winter Addenda, and the 1974 Edition (Division 1) through the 2004 Edition (Division 1). Accordingly, the use of the 2004 Edition or the 1993 Addenda is permitted by 10 CFR 50.55a(b)(1)(ii), subject to the limitations and modifications described in 10 CFR 50.55a(b)(1).

B. The U.S EPR conforms to the requirements of NCA-1140, specifically NCA-1140(a)(2) states:

> "In no case shall the Code Edition and Addenda dates established in the Design Specifications be earlier than:
>
> (a) 3 years prior to the date that the nuclear power plant construction permit application is docketed; or
>
> (b) the latest edition and addenda endorsed by the regulatory authority having jurisdiction at the plant site at the time the construction permit application is docketed."

The Question states that the Code Edition and Addenda dates established in the Design Specifications are required to comply with either paragraph (a) or (b) of NCA-1140(a)(2), but not both. The U.S. EPR design certification was docketed on February 25, 2008 by NRC Accession Number ML080380357. The Code Edition and Addenda that were endorsed by the NRC when the U.S. EPR design certification was docketed were the 2001 Edition with the 2003 Addenda. The code of record, as specified in the U.S. EPR design certification (and also the design specifications), is the 2004 Edition with no Addenda. Therefore, the Code Edition and Addenda for the U.S. EPR is not earlier than the latest Code Edition and Addenda endorsed by the regulatory authority with jurisdiction at the time that the U.S EPR design certification was docketed. Therefore, the U.S EPR is in compliance with NCA-1140(a)(2)(b), and Code Case N-782 is not required for the U.S. EPR. Furthermore, as noted in the Question, the NRC has not yet accepted Code Case N-782.

C. A response to this question will be provided by May 28, 2010.

FSAR Impact:

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

Question 05.02.05-9:

OPEN ITEM

Follow-up to RAI 244, Question 05.02.05-6

Even though the responsibility for the operating and emergency operating procedures is with the COL applicant, the design certification application is incomplete without identifying a COL information item specifically for the procedures relating to the conversion of instrument indicators and alarm setpoint. AREVA is requested to identify a COL information item.

Response to Question 05.02.05-9:

As noted in the Response to RAI 244, Question 05.02.05-6, the requested information is addressed in RG 1.45, Revision 1. A COL information item will be added to U.S. EPR FSAR Tier 2, Table 1.8-2 and U.S. EPR FSAR Tier 2, Section 5.2.5.5 stating that a COL applicant that references the U.S. EPR design certification will develop procedures in accordance with RG 1.45, Revision 1.

FSAR Impact:

U.S. EPR FSAR, Tier 2 Table 1.8-2 and Section 5.2.5.5 will be revised as described in the response and indicated on the enclosed markup.

Question 05.02.05-10:

OPEN ITEM

Follow-up to RAI 244, Question 05.02.05-7

Even though the responsibility for the operating and emergency-operating procedures is with the COL applicant, the design certification application is incomplete without identifying a COL information item specifically for the procedures relating to operator actions to manage the long-term low-level RCS leakage. AREVA is requested to identify a COL information item.

Response to Question 05.02.05-10:

See the Response to Question 05.02.05-9 and the associated markups to U.S. EPR FSAR Tier 2, Table 1.8-2 and Section 5.2.5.5.

Question 05.02.05-11:

OPEN ITEM

Follow-up to RAI 244, Question 05.02.05-8

The applicant agreed to include the RCPB leakage detection system in the ITAAC. The staff reviewed the marked-up pages for the revised ITAAC and found that the verification of the RCPB leakage detection sensitivity, response time, and alarm limits for the RCPB leakage detection instrument is not included in the proposed ITAAC. AREVA is requested to include the RCPB leakage detection sensitivity, response time, and alarm limits for the RCPB leakage detection instrument in the ITAAC.

Response to Question 05.02.05-11:

U.S. EPR FSAR Tier 1, Table 2.4.8-1, Table 2.4.9-3, and Table 2.9.5-2, will be revised as requested, consistent with the guidance of RG 1.45, Revision 1.

FSAR Impact:

U.S. EPR FSAR Tier 1, Table 2.4.8-1, Table 2.4.9-3, and Table 2.9.5-2 will be revised as described in the response and indicated on the enclosed markup.

Question 05.03.01-14:

OPEN ITEM

Follow-up to RAI 232, Question 05.03.01-10c

The applicant's response did not address the minimum qualified thickness of cladding qualified as buttering, and because cladding thickness has implications for heat input to the RPV during subsequent welding operations, AREVA is requested to specify the minimum thickness of the cladding when qualified as weld buttering.

Response to Question 05.03.01-14:

The nominal thickness requirement of 0.295 inches for cladding, as specified in U.S. EPR FSAR Tier 2, Table 5.3-7, is applicable to deposited weld metal performing a cladding function, regardless of how the weld is qualified. This clarifies the Response to RAI 232, Question 05.03.01-10c which stated that "where the cladding is also qualified as weld buttering, additional thickness requirements may apply." As discussed in the Response to RAI 304, Question 05.03.01-12, the requirements of QW-283 in ASME Code Section IX are applicable to the weld procedure qualification of the buttering process, and determine the required thickness of the deposited weld metal to support subsequent welding without post weld heat treatment. Since an ASME N-Type Certificate Holder is responsible for the qualification of welding procedures used by their organization, the minimum qualified thickness of buttering will be dependent on where and by whom the item is fabricated.

U.S. EPR FSAR Tier 2, Section 5.3.1.2 will be revised to state that deposited weld metal performing a cladding function, whether qualified as cladding, structural weld, or weld buttering, meets the cladding thickness specified in U.S. EPR FSAR Tier 2, Table 5.3-7.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 5.3.1.2 will be revised as described in the response and indicated on the enclosed markup.

Question 05.03.01-15:

OPEN ITEM

Follow-up to RAI 232, Question 05.03.01-11b

The transition ring with the radial-key attachment welds is post-weld heat-treated in the RPV construction sequence. Therefore, the radial-key attachment welds receive an indirect, post-weld heat-treatment as a result of the heat treatment of the RPV. Because the option exists for the applicant to fabricate the radial-key attachment welds without a subsequent post-weld heat treatment, and because the applicant did not confirm that a low-heat-input weld process will be used, AREVA is requested to confirm that a low-heat-input weld process will be used for sequences where the radial-key attachment welds are made without subsequent post-weld heat treatment.

Response to Question 05.03.01-15:

The design requirement for the weld between the radial key and the transition ring is to obtain a structurally sound weld meeting all requirements of ASME Sections III and IX which does not induce adverse affect on the base materials (heat affected zone) being joined. The structural integrity of the weld and the underlying base metal is based on qualification of the weld process which demonstrates through testing of coupons that both the deposited weld and the heat affected zone of the base materials develop or maintain the required mechanical properties including impact testing (see response to RAI 304, Question 05.03.01-12 for additional discussion).

Heat input is controlled by the weld procedure qualification process to meet ASME Sections III and IX as stated above. As required by QW-283.2, the variables for the buttering and the subsequent weld shall be in accordance with QW-250. For any of the welding processes (e.g., SMAW, SAW, GMAW, GTAW, PAW) using the weld filler materials identified for this specific application in U.S. EPR Tier 2, Section 5.3 (SFA-5.11 or SFA-5.14), ASME Section IX specifies heat input as a supplementary essential variable which is required for materials for which notch toughness (impact) tests are required. Since impact testing is required for the base material in this specific weld application, any increase in heat input above that of the procedure qualification would require a new qualification (QW-409.1). Therefore, where the weld is performed within the parameters of a qualified welding procedure meeting the requirements of ASME Sections III and IX as currently required by U.S. EPR Tier 2, Section 5.2.3, the heat input associated with the weld is adequately controlled.

Use of a low heat input weld process by itself does not ensure the underlying heat affected zone of the base metal remains structurally sound. Furthermore, the term ‘low heat input’ is not a clearly defined term with specific measurable acceptance criteria and may be subject to interpretation. Therefore, the inclusion of a specific requirement to utilize a ‘low heat input’ weld process is not necessary.

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

Table 2.4.8-1—Leakage Detection System ITAAC

	Commitment Wording	Inspections, Tests, Analyses	Acceptance Criteria
2.1	Reactor Building fan cooler condensate collector level indication is provided in the MCR.	Testing will be performed for the Reactor Building condensate collector level indications.	Condensate collector level change is indicated in the MCR on the Reactor Building condensate collector level indications. • Reactor Building Fan Cooler Level Condensate Levels JYH11 CF001 JYH14 CF001 JYH21 CF001 JYH22 CF001 JYH23 CF001 JYH24 CF001 JYH22 CF003 JYH22 CF004 JYH23 CF003 JYH23 CF004 • The system can detect 1.0 gpm condensate flow within 1 hour.

05.02.05-11

Next File

Table 2.9.4-3—Sampling Activity Monitoring System ITAAC (2 Sheets)

	Commitment Wording	Inspections, Tests, Analyses	Acceptance Criteria
4.3	Reactor Building radiation is indicated in the MCR.	A testing will be performed to verify radiation level indication in the MCR. 05.02.05-11	a. Radiation level indication is provided in the MCR for the Reactor Building radiation monitors listed in Table 2.9.4-1. b. The monitor can detect 10^{-9} µCi/cc.
5.1	The components designated as Class 1E in Table 2.9.4-2 are powered from a Class 1E division in a normal or alternate feed condition.	a. Testing will be performed for components designated as Class 1E in Table 2.9.4-2 by providing a test signal in each normally aligned division. b. Testing will be performed for components designated as Class 1E in Table 2.9.4-2 by providing a test signal in each division with the alternate feed aligned to the divisional pair.	a. The test signal provided in the normally aligned division is present at the respective Class 1E component identified in Table 2.9.4-2. b. The test signal provided in each division with the alternate feed aligned to the divisional pair is present at the respective Class 1E component identified in Table 2.9.4-2.
6.1	MCR Ventilation Intake Radioactivity Monitors listed in Table 2.9.4-1 initiate isolation of the MCR ventilation and initiation of supplemental filtration upon receipt of high radioactivity levels.	A test will be performed to verify that the MCR ventilation isolation and supplemental filtration is initiated upon radiation levels exceeding a preset limit.	The monitors listed in Table 2.9.4-1 initiate MCR ventilation isolation and supplemental MCR filtration when radiation level exceeds a preset limit.

Next File

Table 2.9.5-2—Nuclear Island Drain and Vent System ITAAC (2 Sheets)

	Commitment Wording	Inspections, Tests, Analyses	Acceptance Criteria
2.1	The location of the sump level sensors is as listed in Table 2.9.5-1.	An inspection will be performed to verify the location of the sump level sensors listed in Table 2.9.5-1.	The location of the sump level sensors is as listed in Table 2.9.5-1. 05.02.05-11
3.1	Displays listed in Table 2.9.5-1 are retrievable in the MCR.	Tests will be performed for MCR displays listed in Table 2.9.5-1.	a. Displays listed in Table 2.9.5-1 are retrievable in the MCR. b. The system can detect 1.0 gpm inflow within one hour.
3.2	The sump level sensor in a Safeguard Building trips the ESWS pump and closes the pump discharge valve in response to a flooding signal.	a. A test will be performed on the SB 1 sump level sensor (30KTE20CL001) listed in Table 2.9.5-1. b. A test will be performed on the SB 2 sump level sensor (30KTE20CL003) listed in Table 2.9.5-1. c. A test will be performed on the SB 3 sump level sensor (30KTE20CL005) listed in Table 2.9.5-1. d. A test will be performed on the SB 4 sump level sensor (30KTE20CL007) listed in Table 2.9.5-1.	a. ESWS pump 1 trips and ESWS pump 1 discharge valve closes on a SB 1 sump level signal. b. ESWS pump 2 trips and ESWS pump 2 discharge valve closes on a SB 2 sump level signal. c. ESWS pump 3 trips and ESWS pump 3 discharge valve closes on a SB 3 sump level signal. d. ESWS pump 4 trips and ESWS pump 4 discharge valve closes on a SB 4 sump level signal.
4.1	The sump level sensors designated as Class 1E in Table 2.9.5-1 are powered from the Class 1E division listed in Table 2.9.5-1.	Tests will be performed for sump level sensors designated as Class 1E in Table 2.9.5-1 by providing a test signal to the aligned Class 1E division.	The test signal provided in the aligned Class 1E division is present at the sump level sensors identified in Table 2.9.5-1.

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	~~Y~~	
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	~~Y~~	
5.2-4	A COL applicant that references the U.S. EPR design certification will develop procedures in accordance with RG 1.45, Revision 1.	5.2.5.5	05.02.05-9	
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	~~Y~~	
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		~~Y~~
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		~~Y~~
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	~~Y~~	

- Leakage from the letdown line heat exchangers to the CCWS is detected by radiation monitors and flow sensors which indicate and alarm in the MCR. In the unlikely event of a tube rupture, CCWS flow to the letdown line heat exchanger automatically isolates.

These methods are supplemented by radiation monitors, process sampling, and laboratory analysis, which indicate increased CCWS system activity from small leaks. Section 9.2.2 and Section 11.5 further address the control of RCS leakage into the CCWS.

5.2.5.4 Inspection and Testing Requirements

The leakage detection systems are designed to permit operability testing and calibration during plant operation. Refer to Chapter 16 (SR 3.4.14) for surveillance requirements. Periodic testing of the floor drainage system verifies that it is free of blockage.

5.2.5.5 Instrumentation Requirements

The leakage detection systems provide data to the instrumentation and control systems for indication, alarm, and archival. Operators in the MCR are provided with the leakage rate (gpm) from each detection system and a common leakage equivalent (gpm) from both identified and unidentified sources. Alarms indicate that leakage has exceeded predetermined limits. The instrumentation system is described in Section 7.1. A COL applicant that references the U.S. EPR design certification will develop procedures in accordance with RG 1.45, Revision 1.

~~Leakage conversion procedures are developed as part of operating and emergency operating procedures described in Section 13.5.2.1 to convert various indications to an identified and unidentified common leakage equivalent and leakage rate of change. Leakage management procedures are also included in the operating and emergency operating procedures described in Section 13.5.2.1. These procedures includes means to identify the leak source, monitor and trend leak rate and evaluate various corrective action plans in response to prolonged low leakage conditions that exceed normal leakage rates, but do not exceed the Technical Specification (TS) limit to provide the operator sufficient time to take corrective actions before the leakage exceeds the TS limit value.~~

05.02.05-9

5.2.5.5.1 RCDT Indications

The RCDT collects continuous flow during operation from PZR degassing and the RCP seals' leakoff. This flow is quantified from tank level and pump run time indications and a baseline normal in-leakage rate is established. Changes in this rate indicate leakage from additional components whose discharge is routed to the RCDT. Such leakage can be identified through indications from these components and, once quantified, can be monitored as identified leakage.

attachment configuration. The cladding at all interfaces between the stainless steel and Ni-Cr-Fe cladding is deposited with Ni-Cr-Fe weld filler material.

As the radial keys perform a structural function in support of the reactor vessel internals, the Ni-Cr-Fe cladding directly below the radial key attachment is qualified as part of the radial key full penetration structural weld in accordance with ASME Sections III and IX. Where the radial keys are welded to the cladding without subsequent post-weld heat treatment, the cladding is qualified as weld buttering in accordance with ASME Sections III and IX. Deposited weld metal performing a cladding function, whether qualified as cladding, structural weld, or weld buttering, meets the cladding thickness specified in Table 5.3-7.

05.03.01-14

5.3.1.3 Special Methods for Nondestructive Examination

The non-destructive examination (NDE) of the RPV and its appurtenances is conducted in accordance with ASME Section III requirements. Full penetration weld preparations for pressure retaining materials are examined in accordance with ASME Section III, NB-5130, prior to welding.

The cladding on the sealing surfaces and load-bearing surfaces of the RPV flange and the closure head flange are ultrasonically examined for the complete volume for both bond and defects. All cladding is ultrasonically examined for bond. Surfaces to be clad are examined using magnetic particle or liquid penetrant techniques in accordance with ASME Section III NB-2545 or NB-2546, respectively, prior to cladding.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

Welding of ferritic and austenitic stainless steels is addressed in Section 5.2.3, which addresses conformance to the guidance of RG 1.31, RG 1.34, RG 1.37, RG 1.43, RG 1.44, RG 1.50, and RG 1.71 regarding welding, composition, heat treatments, and similar processes. In addition, RG 1.99 is addressed in Section 5.3.1.5 and RG 1.190 is addressed in Section 5.3.1.6.

5.3.1.5 Fracture Toughness

RCPB ferritic materials provide adequate fracture toughness in accordance with ASME Section III, NB-2300 and 10 CFR Part 50, Appendix G.

The initial Charpy V-notch minimum upper-shelf fracture energy levels for the RPV beltline materials (in the transverse direction for base materials), including welds (along the weld), is 75 ft-lbs, as required by 10 CFR Part 50, Appendix G. The maximum initial nil-ductility reference temperature, $RT_{NDT,}$ of the RPV is -4°F. Materials are evaluated with regard to the effects of chemistry (copper content), initial upper shelf energy, and neutron fluence to assure that 50 ft-lbs upper-shelf energy, as required by 10 CFR Part 50, Appendix G, is maintained throughout the life of the