## ArevaEPRDCPEm Resource

**From:** Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
**Sent:** Wednesday, August 12, 2009 5:46 PM
**To:** Tesfaye, Getachew
**Cc:** BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); KOWALSKI David J (AREVA NP INC)
**Subject:** Response to U.S. EPR Design Certification Application RAI No. 237, FSARCh. 9, Supplement 1
**Attachments:** RAI 237 Supplement 1 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. provided a response to RAI No. 237 on July 8, 2009 which provided a schedule for submittal of a response to the question. The attached file, "RAI 237 Supplement 1 Response US EPR DC.pdf" provides a technically correct and complete response to the question, as committed.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which supports the response to RAI 237 Question 09.03.02-16.

The following table indicates the respective pages in the response document, "RAI 237 Supplement 1 Response US EPR DC.pdf," that contain AREVA NP's response to the subject question.

| Question # | Start Page | End Page |
|---|---|---|
| RAI 237 — 09.03.02-16 | 2 | 8 |

This concludes the formal AREVA NP response to RAI 237, and there are no questions from this RAI for which AREVA NP has not provided responses.

Sincerely,

*Ronda Pederson*

ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
**AREVA NP Inc.**
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

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**From:** Pederson Ronda M (AREVA NP INC)
**Sent:** Wednesday, July 08, 2009 6:26 PM
**To:** 'Getachew Tesfaye'
**Cc:** BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); KOWALSKI David J (AREVA NP INC)
**Subject:** Response to U.S. EPR Design Certification Application RAI No. 237, FSARCh. 9

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 237 Response US EPR DC.pdf" states that a complete answer cannot be currently provided for this question.

The following table indicates the pages in the response document, "RAI 237 Response US EPR DC.pdf," that contain AREVA NP's response to the subject question.

| Question # | Start Page | End Page |
|---|---|---|
| RAI 237 — 09.03.02-16 | 2 | 5 |

The schedule for a technically correct and complete response to this question is provided below.

| Question # | Response Date |
|---|---|
| RAI 237 — 09.03.02-16 | August 14, 2009 |

Sincerely,

*Ronda Pederson*

ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
**AREVA NP Inc.**
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

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**From:** Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
**Sent:** Friday, June 05, 2009 5:59 PM
**To:** ZZ-DL-A-USEPR-DL
**Cc:** Jeffrey Poehler; David Terao; Peter Hearn; Joseph Colaccino; ArevaEPRDCPEm Resource
**Subject:** U.S. EPR Design Certification Application RAI No. 237 (2907), FSARCh. 9

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on May 21, 2009, and on June 5, 2009, you informed us that the RAI is clear and no further clarification is needed. As a result, no change is made to the draft RAI. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

**Hearing Identifier:** AREVA_EPR_DC_RAIs
**Email Number:** 725

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**From:** Pederson Ronda M (AREVA NP INC)

**Created By:** Ronda.Pederson@areva.com

**Recipients:**
"BENNETT Kathy A (OFR) (AREVA NP INC)" <Kathy.Bennett@areva.com>
Tracking Status: None
"DELANO Karen V (AREVA NP INC)" <Karen.Delano@areva.com>
Tracking Status: None
"KOWALSKI David J (AREVA NP INC)" <David.Kowalski@areva.com>
Tracking Status: None
"Tesfaye, Getachew" <Getachew.Tesfaye@nrc.gov>
Tracking Status: None

**Post Office:** AUSLYNCMX02.adom.ad.corp

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**Response to**

**Request for Additional Information No. 237, Supplement 1**

**6/05/2009**

**U. S. EPR Standard Design Certification**
**AREVA NP Inc.**
**Docket No. 52-020**
**SRP Section: 09.03.02 - Process and Post-Accident Sampling Systems**
**Application Section: 9.3.2**

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

**Question 09.03.02-16:**

Background

10 CFR 52.47(a)(8) requires that standard design certification applications address the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v).

10 CFR 50.34(f)(2)(xxvi) requires that applicants, "Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. (III.D.1.1)"

III.D.1.1 refers to the TMI Action Plan requirement which is detailed and further clarified in NUREG-0737, "Clarification of TMI Action Plan Requirements." Systems listed by Item III.D1.1 as potentially in scope of the requirement are residual heat removal (RHR), containment spray recirculation, high-pressure injection recirculation, containment and primary coolant sampling, reactor core isolation cooling, makeup and letdown (PWRs only), and waste gas (includes headers and cover gas system outside of containment in addition to decay or storage system)

The US-EPR FSAR, Tier 2, lists 10 CFR 50.34(f)(2)(xxvi) among the design bases of several systems, including RHR, ECCS, PSS, and CVCS. However, the level of design information provided to demonstrate meeting the criterion is inconsistent for the systems. Further, the interface requirements with the COL applicants with respect to the programmatic aspects of the criterion appear to be incomplete. The criterion is addressed for specific systems as described below.

Residual Heat Removal System (RHRS)

US-EPR FSAR Tier 2, Table 1.9-3, "U.S. EPR Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)," indicates that 10 CFR 50.34(f)(2)(xxvi) is addressed in FSAR Tier 2 Section 5.4.7 (Residual Heat Removal System). FSAR Tier 2 Section 5.4.7.1, "Design Bases," states that the Safety Injection System/Residual Heat Removal System (SIS/RHRS) is designed to control and detect leakage outside containment following an accident. The section of the SIS/RHRS that is located outside the containment can be isolated from the containment in the event of a break in the SIS/RHRS piping (10 CFR 50.34(f)(2)(xxvi)). Leakage from the system is detected, monitored, and controlled by plant operating procedures and programs.

Emergency Core Cooling Systems (ECCS)

For the ECCS systems, FSAR Tier 2 Section 6.3.1, "Design Bases," states that the ECCS is designed with the capability for leakage detection and control to minimize the leakage from those portions of the SIS outside of the containment that may contain radioactive material following an accident (10 CFR 50.34(f)(2)(xxvi)). FSAR Tier 2 Section 6.3.2.2.1, "System Overview," further states that leakage from the SIS in the Safeguards Buildings (SBs) is detected and monitored by operating procedures and programs. Each SB has sump level indication to detect SIS/RHRS leakage.

Process Sampling System (PSS)

For the PSS, Section 9.3.2.1, “Design Bases,” states that non-safety-related portions of the process sampling systems are designed to have provisions for a leakage detection and control program to minimize the leakage from those portions of the process sampling systems outside of the containment that contain or may contain radioactive material following an accident (10 CFR 0.34(f)(2)(xxvi)). Further, Section 9.3.2.3, “Safety Evaluation,” states:

“The design of the process sampling systems satisfies 10 CFR 50.34(f)(2)(xxvi) regarding having provisions for a leakage detection and control program to minimize the leakage from those portions of the process sampling systems outside of the containment that contain or may contain radioactive material following an accident.

- The NSS (Nuclear Sampling System) samples the RCS (Reactor Coolant System) to provide information necessary to assess and control the plant under accident conditions.
- The SASS (Severe Accident Sampling System) obtains and analyzes gaseous samples from the containment atmosphere following a severe accident for the purpose of confirming whether the containment atmosphere contains airborne activity.
- The NSS and SASS contains proper equipment to prevent unnecessary high exposures to workers and minimize leakage from the system to ALARA.
- Safety-related CIVs (containment isolation valves) close on receipt of a CIS (containment isolation signal) and contain radioactive material inside the RB (reactor building). Refer to Section 6.2.4.”

Chemical and Volume Control System (CVCS)

For the CVCS system, Section 9.3.4.1, “Design Bases,” states that safety-related portions of the CVCS are designed to have provisions for a leakage detection and control program to minimize the leakage from those portions of the CVCS outside of the containment that contain or may contain radioactive material following an accident (10 CFR 50.34(f)(2)(xxvi)).

Section 9.3.4.3, “Safety Evaluation,” states that the design of safety-related portions of the CVCS satisfies 10 CFR 50.34(f)(2)(xxvi) regarding detection of reactor coolant leakage outside containment by providing leakage control and detection systems in the CVCS and implementation of appropriate leakage control program.

- The CVCS isolates components or piping so that the CVCS safety function is not compromised. Design provisions include the capability to identify and isolate the leakage or malfunction, and to isolate the non-safety-related portions of the system.

Section 9.3.4.2.3.5, “Accident Conditions,” contains the following information with regard to detecting system leaks outside containment:

Postulated System Leaks in the Fuel Building - In the event of a CVCS or RCP seal water system leak in the Fuel Building, reactor coolant with temperatures of approximately 120°F is released.

Due to the loss of reactor coolant, the following alarms are also generated:

- VCT (volume control tank) low water level.
- Sump high water level in the FB (Fuel Building) vent and drain system.

Gaseous Waste Management System

For the gaseous waste management system, 10 CFR 50.34(f)(2)(xxvi) is not listed in the FSAR among the design bases, although provisions to minimize system leakage are described in FSAR Tier 2 Section 11.3. This system would be the equivalent to the waste gas system listed as one of the systems potentially in scope of TMI Action Item III.D.1.1.

Technical Specifications

Additionally, FSAR Tier 2 Chapter 16, Technical Specification 5.5.2, "Primary Coolant Sources Outside Containment," states the following:

"This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Low Head Safety Injection, Medium Head Safety Injection, and Nuclear Sampling. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system once per 24 months."

This appears to describe a program intended to fulfill the requirements of 10 CFR 50.34(f)(2)(xxvi). However, Technical Specification 5.5.2 does not contain all the elements required by NUREG-0737 Item III.D.1.1, and the list of systems is inconsistent with those systems that list 10 CFR 50.34(f)(2)(xxvi) as a design bases. Finally, the initial and periodic tests required by Item III.D1.1 would be performed by the COL holder. However, the US-EPR FSAR does not identify a COL information item to ensure the COL holder has a leakage control program, and the initial leak test is not addressed in the initial test program information included in the FSAR.

Requested Information

1. Other than RHR, ECCS, PSS, and CVCS, identify any other systems in scope of 10 CFR 50.34(f)(2)(xxvi). If any systems expected to contain radioactive materials after an accident are excluded from the leakage detection program, justify the exclusion of these systems.
2. For the PSS, describe the design provisions that facilitate minimization and detection of leakage for each of the systems in accordance with 10 CFR 50.34(f)(2)(xxvi). Specifically, describe the "proper equipment" mentioned in Section 9.3.2.3 in more detail. Confirm that the PSS credits the SB building sump level indication as in the case of the CVCS system.
3. For those systems that credit building sump level indication/alarms for leakage detection, describe the identification process for the location of the specific leakage.
4. Discuss the need to include a COL information item in the FSAR to ensure the COL holder develops a program for leakage monitoring and prevention to fulfill the requirements of 10 CFR 50.34(f)(2)(xxvi).

5. Clarify whether proposed Technical Specification 5.5.2 is intended to fulfill the requirements of 10 CFR 50.34(f)(2)(xxvi). If so, these criteria should be referenced in the technical specification.
6. In FSAR Tier 1 and Tier 2, provide the initial test program information for leakage control and detection for all systems outside containment that contain (or might contain) accident source term radioactive materials following an accident.

**Response to Question 09.03.02-16:**

10 CFR 50.34(f)(2)(xxvi) requires that a programmatic approach be used to detect and control system leakage to ensure that systems that can potentially contain highly radioactive fluids following a beyond design basis event (BDBE) will not result in unacceptable levels of airborne activity. The systems that are within the scope of 10 CFR 50.34(f)(2)(xxvi) potentially include safety-related and non-safety-related, permanent plant systems that can be reasonably expected to be placed in-service following a BDBE. In general, the target total leakage from all systems containing highly radioactive fluids is on the order of a few drops per minute. Typically, components located in 10 CFR 50.34(f)(2)(xxvi) systems with visible leakage or boron crystals are identified for repair at the next scheduled opportunity.

The level of acceptable leakage that meets 10 CFR 50.34(f)(2)(xxvi) program requirements can be contrasted with leakage programs and instrumentation that satisfy the requirements of U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications (TS) 3.4.12, RCS Operational Leakage; and instrumentation that is intended to identify catastrophic failure of systems containing reactor coolant system (RCS) coolant. The sump level instrumentation that is designed to provide these design functions is too gross to meet 10 CFR 50.34(f)(2)(xxvi) leakage monitoring goals.

1. The U.S. EPR systems that are within the scope of 10 CFR 50.34(f)(2)(xxvi) include:
    - Low head safety injection.
    - Medium head safety injection.
    - Severe accident heat removal.
    - Nuclear sampling.
    - Severe accident sampling.
    - Hydrogen monitoring.
    - Chemical and volume control.
    - Gaseous waste processing.

    The fluid systems will be tested by operating the systems at normal operating pressure and temperature. Identified leaks will be repaired, as necessary. Systems that cannot be tested in the previously identified manner will be pressurized with air and a tracer gas, and leaks will be identified and repaired.

2. The process sampling systems (PSS) use the building sumps to identify leakage from a catastrophic failure of a sampling system, but not to determine leakage from specific components to meet 10 CFR 50.34(f)(2)(xxvi) leakage monitoring goals. The following discussion clarifies information in the U.S. EPR FSAR that describes the detection of gross system leakage from the sampling systems:

   Primary Samples

   The primary samples are routed from the Reactor Building (RB) into the adjoining FB, where they are cooled and depressurized. A pressure relief valve is located downstream of the pressure reducing valve in the FB. The discharge of the pressure relief valve goes into the nuclear island drain/vent system (NIDVS). The samples are then routed to the sample panel in the Nuclear Auxiliary Building (NAB). Cooling and depressurizing as close to the Containment as possible reduces the high energy pathway, which minimizes leakage. Refer to U.S. EPR FSAR Tier 2, Figure 9.3.2-1—Nuclear Sampling System.

   The primary sample collection tank can be routed to the coolant treatment system (CTS) or an NIDVS sump. The NIDVS sump is in the NAB. If the fluid is highly radioactive, the fluid can be routed back into the RB sump for storage. The re-injection reduces the risk to personnel. Refer to U.S. EPR FSAR Tier 2, Figure 9.3.3-1—Nuclear Island Drain and Vent System.

   The primary sample lines are the size of the smallest practical bore that will still facilitate flushing, minimize conditioning requirements, reduce lag time and changes in sample composition, and provide adequate velocity and turbulence. As a result, the flow is restricted and the leakage is minimized.

   Severe Accident Samples

   The samples are taken from the RB. The location of the process modules are in the Safeguard Building 4. This is as close to Containment as possible to minimize leakage. The sampling station is in the FB and is enclosed in a shielded glove box, which minimizes risk to personnel.

   The gas sampling module can dilute the gaseous phase taken from the containment atmosphere by a factor of 1:10, 1:100 or 1:1000. This minimizes the risk of personnel handling highly radioactive samples.

   The primary sample lines are the size of the smallest practical bore that will still facilitate flushing, minimize conditioning requirements, reduce lag time and changes in sample composition, and provide adequate velocity and turbulence. As a result, the flow is restricted and the leakage is minimized. The portion of the samples not used for analysis, as well as the purge media (nitrogen, water) for the system, are collected in a vessel and re-injected into the Containment.

   In summary, radiation exposure is minimized in the design by reducing exposure by shielding, dilution of samples, and re-injection of samples into the Containment.

U.S. EPR FSAR Tier 2, Section 9.3.2.3 will be revised to clarify that radiation exposure is minimized in the NSS and SASS.

3. The leakage control program that is developed to meet the requirements of 10 CFR 50.34(f)(2)(xxvi) and NUREG-0737, Item III.D.1.1 will maintain and confirm that leakage levels are below those that could be detected by sump level indication and alarms. The leakage control program that is developed by the COL applicant in accordance with COL Information Item No. 13.5-1 in U.S. EPR FSAR Tier 2, Table 1.8-2—U.S. EPR Combined License Information Items (refer to the Response to Question 09.03.02-16, Part 4) relies on establishing a low threshold for leakage and providing reasonable assurance that maintenance activities maintain leakage below analyzed limits.

4. U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications (TS) 5.5.2, Primary Coolant Sources Outside Containment, describes a program that provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. TS 5.5.2 is intended to fulfill the requirements of 10 CFR 50.34(f)(2)(xxvi) and NUREG-0737, Item III.D.1.

   The Response to Question 09.03.02-16, Part 5 states that TS Section 5.5.2 will be revised to reflect the plant systems that are in the leakage control program.

   This leakage control program is addressed in COL Information Item No. 13.5-1 in U.S. EPR FSAR Tier 2, Table 1.8-2. This COL item states:

   > "A COL applicant that references the U.S. EPR design certification will provide site-specific information for administrative, operating, emergency, maintenance, and other operating procedures."

   Therefore, it is not necessary to include an additional COL information item in the U.S. EPR FSAR to provide reasonable assurance that the COL holder develops a program for leakage monitoring and prevention to fulfill the requirements of 10 CFR 50.34(f)(2)(xxvi).

5. On July 2, 1980, the NRC issued Generic Letter (GL) 80-61. This GL requested that licensees propose additional technical specifications and license conditions as a follow-up to NUREG-0578 TMI-2 Lessons Learned. Enclosure 2 of GL 80-61 included a model license condition for Systems Integrity. This model license condition implemented TMI Item III.D.1.1. This model license condition was later included in the Improved Technical Specifications (ITS) as Section 5.5.2 – Primary Coolant Sources Outside Containment. The same specification is included in each of the ITS (NUREGs – 1430, -1431, -1432, -1433, and -1434). U.S. EPR FSAR Tier 2, Chapter 16 Technical Specification 5.5.2 incorporated the same language as used in the ITS, and therefore, addresses 10CFR50.34(f)(2)(xxvi) to the same extent as for currently operating plants.

   U.S. EPR FSAR Tier 2, Chapter 16 Technical Specification 5.5.2 will be revised to include each of the systems identified in the Response to Question 09.03.02-16, Part 1.

6. The initial test program will be revised to include a test abstract to test systems impacted by the leakage control program. The U.S. EPR FSAR Tier 2, Section 14.2 testing will be

performed in conjunction with hot functional testing and activities prior to loading fuel. The leakage control program that is developed to meet the requirements of 10 CFR 50.34(f)(2)(xxvi) and NUREG-0737, Item III.D.1.1 can evaluate the results of the U.S. EPR FSAR Tier 2, Section 14.2 testing and maintenance activities that have been performed since the preoperational testing, and evaluate the suitability of those tests during Cycle 1.

Inspection, Test, Analysis and Acceptance Criteria (ITAAC) for Three Mile Island (TMI) Item 10 CFR 50.34(f)(2)(xxvi) are not required to be included in U.S. EPR FSAR Tier 1. Information in U.S. EPR FSAR Tier 2 is screened as described in U.S EPR FSAR Tier 2, Section 14.3.2, to determine if it is safety-significant. A part of the screening approach involves an expert review panel that identifies safety-significant features based on assumptions and insights from key safety and integrated plant safety analyses in U.S. EPR FSAR Tier 2, where plant performance is dependent on contributions from multiple systems. The expert review panel is based on guidance in Standard Review Plan (SRP) 14.3, page 14.3-21, and one of the areas reviewed by the expert review panel includes the TMI items in 10 CFR 50.34(f). Results of the expert review panel meetings are provided in U.S. EPR FSAR Tier 2, Table 14.3-1—Design Basis Accident Analysis (Safety-Significant Features) through Table 14.3-7—Licensing (Safety-Significant Features). The expert review panel reviewed TMI Item 10 CFR 50.34(f)(2)(xxvi) and did not judge this item to be safety-significant for U.S. EPR FSAR Tier 1.

**FSAR Impact:**

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

2. U.S. EPR FSAR Tier 2, Section 9.3.2.3 will be revised as described in the response and indicated on the enclosed markup.

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

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

5. U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications will be revised as described in the response and indicated on the enclosed markup.

6. U.S. EPR FSAR Tier 2, Chapter 14.2 Test #153 and Table 14.2-1 will be revised as described in the response and indicated on the enclosed markup.

# U.S. EPR Final Safety Analysis Report Markups

## 5.5 Programs and Manuals

### 5.5.2 Primary Coolant Sources Outside Containment

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This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Low Head Safety Injection, Medium Head Safety Injection, ~~and~~ Nuclear Sampling, Severe Accident Heat Removal, Severe Accident Sampling, Chemical and Volume Control, Gaseous Waste Processing, and Hydrogen Monitoring. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system once per 24 months.

The provisions of SR 3.0.2 are applicable.

### 5.5.3 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

3.0 TEST METHOD

3.1 Verify the performance of the MSRT by simulating a safety injection signal and verifying that the MSRT setpoint is reduced.

3.2 Verify power-operated valves fail upon loss of motive power as designed.

3.3 Check electrical independence and redundancy of power supplies for safety-related functions by selectively removing power and determining loss of function.

3.4 Verify that the MSRCV positions to 40 percent open based on a thermal power level of 0 percent.

4.0 DATA REQUIRED

4.1 Valve position indication as a function of time.

4.2 RCS temperature and pressure as a function of time.

4.3 RCS depressurization rate as a function of time.

4.4 SG pressure and level as a function of time.

4.5 Position response of MSRT valves to loss of motive power.

5.0 ACCEPTANCE CRITERIA

5.1 The main steam system provides a depressurization path through the MSRT valves and associated silencers to atmosphere, as designed (refer to Section 10.3.2.2).

5.2 The MSRT setpoint is reduced upon receipt of a safety injection signal, as designed (refer to Sections 6.3.3.1, 10.3.2.2, and 16 B3.3.1).

5.3 Verify that safety-related components meet electrical power supply independence and redundancy requirements, as designed (refer to Section 8.1.4.2).

5.4 The MSRCV positions to 40 percent based on a thermal power of 0 percent, as designed (refer to Section 10.3.2.2).

5.5 The MSRIV positions as required to control the rate of steam pressure reduction with minimal overshoot.

5.6 Verify the response of the partial cooldown function to a SIS signal.

5.6.1 Table 14.3-1, Item 1-44.

09.03.02-16

### 14.2.12.12.7 ~~Reserved (Test #153)~~Integrity of Systems Likely to Contain Radioactive Material (Test #153)

1.0 OBJECTIVE

1.1 This system integrity test is applicable to portions of systems that are located outside of containment that could contain radioactive material

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following a serious transient or accident. The objective of this test is to limit the exposure to personnel from leakage.

1.2 To verify the system integrity of the following impacted systems:

1.2.1 Low Head Safety Injection

1.2.2 Severe Accident Heat Removal System

1.2.3 Medium Head Safety Injection

1.2.4 Nuclear Sampling System

1.2.5 Severe Accident Sampling System

1.2.6 Hydrogen Monitoring

1.2.7 Chemical and Volume Control System (makeup and letdown)

1.2.8 Gaseous Waste Processing

2.0 PREREQUISITES

2.1 Construction activities on the impacted systems have been completed.

2.2 System hydrostatic testing on the impacted systems has been completed prior to performing leakage testing.

2.3 Support systems required for operation of the impacted systems are completed and functional.

2.4 Initial preoperational testing scheduled prior to hot functional testing on the impacted systems has been completed.

2.5 Test instrumentation for detecting the presence of helium/$SF_6$ is available and calibrated.

2.6 An adequate supply of helium/$SF_6$ is available, as required.

3.0 TEST METHOD

3.1 Verify that the impacted systems that contain liquids are leak tight when pressurized at normal system operating pressure and temperature, as applicable.

3.2 Identified leaks should be repaired and retested, as applicable.

3.3 Verify that impacted systems that contain gases are pressurized to normal operating pressure with a mixture of compressed air and a suitable tracer gas, such as helium/$SF_6$.

3.4 Leaking portions of impacted systems can be corrected by maintenance activities and retested by either of the methods described above.

4.0 DATA REQUIRED

4.1 Walkdown inspection reports completed by qualified personnel.

4.2 Helium/$SF_6$ equipment calibration references.

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5.0 ACCEPTANCE CRITERIA

5.1 The leakage from impacted systems meets the requirements of NUREG-0578, Recommendation 2.1.6.a.

5.2 The leakage from impacted systems meets the requirements of NUREG-0660, Item III.D.1.1.

5.3 The leakage from impacted systems meets the requirements of NUREG-0664, Part 2.

#### 14.2.12.12.8 Remote Safe Shutdown (Test #154)

1.0 OBJECTIVE

1.1 To demonstrate the proper operation of the remote safe shutdown function.

2.0 PREREQUISITES

2.1 The instrumentation used during safe shutdown has been calibrated and is operating satisfactorily prior to performing the following test.

2.2 External test instrumentation is available and calibrated.

2.3 Support systems required for testing safe shutdown are functional.

3.0 TEST METHOD

3.1 Verify that safe shutdown control signals override lower priority signals.

3.2 ~~Activate manual trips and monitor operation~~Verify functionality of hard-wired functions at the RSS.

3.2.1 RSS transfer switches are repositioned to allow RSS control.

3.2.2 Manual reactor trip is available at the RSS.

3.3 Simulate safe shutdown scenarios and observe actuation of the appropriate trip circuit and associated alarms.

3.4 Exercise the control functions to the safety depressurization and shutdown cooling system to verify as designed operations.

~~3.5 Activate manual trips and observe relay operation.~~

4.0 DATA REQUIRED

4.1 Power supply voltages.

4.2 Circuit breaker and indicator operation.

4.3 Safety parameter trends during testing.

4.4 Reactor trip and actuation path response.

The design of the process sampling systems satisfies 10 CFR 50.34(f)(2)(xxvi) regarding having provisions for a leakage detection and control program to minimize the leakage from those portions of the process sampling systems outside of the containment that contain or may contain radioactive material following an accident.

- The NSS samples the RCS to provide information necessary to assess and control the plant under accident conditions.
- The SASS obtains and analyzes gaseous samples from the containment atmosphere following a severe accident for the purpose of confirming whether the containment atmosphere contains airborne activity.
- The NSS and SASS contains proper equipment to prevent unnecessary high exposures to workers and minimize leakage from the system to maintain exposure ALARA.
- Safety-related CIVs close on receipt of a CIS and contain radioactive material inside the RB. Refer to Section 6.2.4.

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### 9.3.2.4 Inspection and Testing Requirements

Components in the process sampling systems are inspected and tested during plant startup. Refer to Section 14.2 (test abstract #071, #092, #100 and #204) for initial plant startup test program. The components are designed to permit periodic testing and inservice inspections during plant operation. System components are monitored during operation to demonstrate satisfactory functioning of the equipment. A description of the inservice testing program and inservice inspection program is provided in Section 3.9.6 and Section 6.6, respectively.

### 9.3.2.5 Instrumentation Requirements

During normal plant operation, continuous sampling of the reactor coolant and steam generator blowdown is performed by online monitors. These sample lines are automatically isolated on a CIS.

Normal plant process condition indication (e.g., pressure, temperature and flow) are used by plant operations personnel to verify system status before manual samples are taken.

### 9.3.2.6 References

1. EPRI Report 1008224, “Pressurized Water Reactor Secondary Water Chemistry Guidelines,” Revision 6, Electric Power Research Institute, December 2004.

**Table 14.2-1—List of Initial Tests for the U.S. EPR**
**Sheet 9 of 14**

| Test # | Test Name | FSAR or COLA Test | Applicable Section of RG 1.68, Revision 3 | Other RG | ITAAC |
|---|---|---|---|---|---|
| 140 | Remote Shutdown Station | FSAR | Appendix A, 1.j.(19) | RG 1.68.2 | |
| 141 | Incore Instrumentation System | FSAR | Appendix A, 1.j.(13) | | |
| 142 | Excore Instrumentation System | FSAR | Appendix A, 1.j.(13) | | |
| 143 | Radiation Monitoring System | FSAR | Appendix A, 1.k.(1) | | |
| 144 | Process and Effluent Radiological Monitoring System | FSAR | Appendix A, 1.k.(1) | | |
| 145 | Hydrogen Monitoring System | FSAR | Appendix A, 1.j.(23) | | |
| 146 | Protection System | FSAR | Appendix A, 1.c. | | |
| 147 | Reactor Control, Surveillance ~~&~~ and Limitation System | FSAR | Appendix A, 1.j.(8) | | |
| 148 | Main Steam Relief Trains | FSAR | Appendix A, 1.d.(3) & 1.e.(4) | | |
| 149 | Steam Generator Level Control | FSAR | Appendix A, 1.j.(2) | | |
| 150 | ~~Reactor~~ Partial Trip | FSAR | Appendix A, 1.c. | | |
| 151 | Primary Depressurization | FSAR | Appendix A, 1.a.(2)(d) & 1.h.(2) | | |
| 152 | Partial Cooldown | FSAR | Appendix A, 1.a.(2)(d) & 1.h.(2) | | |
| 153 | Integrity of Systems Likely to Contain Radioactive Material | FSAR | Appendix A, 5.cc | NUREG-0578, 0660, and 0664 | |
| 154 | Remote Safe Shutdown | FSAR | Appendix A, 1.c | | |
| 155 | Post-accident Monitoring Instrumentation | FSAR | Appendix A, 1.j.(22) | | |

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