ArevaEPRDCPEm Resource

From:	Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
Sent:	Wednesday, January 14, 2009 1:44 PM
To:	Getachew Tesfaye
Cc:	PANNELL George L (AREVA NP INC); DELANO Karen V (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC)
Subject:	Response to U.S. EPR Design Certification Application RAI No. 57, Supplement 1
Attachments:	RAI 57 Supplement 1 Response USEPRDC.pdf

Getachew,

AREVA NP Inc. provided responses to 9 of the 20 questions of RAI No. 57 on October 13, 2008. The attached file, "RAI 57 Supplement 1 Response USEPRDC.pdf," provides technically correct and complete responses to 7 of the remaining 11 questions, as committed.

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 57 Question 07.07-15.

The following table indicates the respective page(s) in the response document, "RAI 57 Supplement 1 Response USEPRDC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 57 - 07.07-2	2	2
RAI 57 - 07.07-3	3	3
RAI 57 - 07.07-6	4	4
RAI 57 - 07.07-7	5	5
RAI 57 - 07.07-13	6	6
RAI 57 - 07.07-15	7	7
RAI 57 - 07.07-17	8	11

The schedule for technically correct and complete responses to the remaining 4 questions is unchanged and provided below:

Question #	Response Date
RAI 57 - 07.07-1	March 3, 2009
RAI 57 - 07.07-8	March 31, 2009
RAI 57 - 07.08-1	March 31, 2009
RAI 57 - 07.08-3	March 31, 2009

Sincerely,

Ronda Pederson
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From: Pederson Ronda M (AREVA NP INC)
Sent: Wednesday, November 26, 2008 3:41 PM
To: 'Getachew Tesfaye'
Cc: DELANO Karen V (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); PANNELL George L (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 57, FSAR Ch 7, Revised Schedule

Getachew,

On October 13, 2008, AREVA NP provided technically correct and complete responses to 9 of the 20 questions and a schedule for the remaining 11 questions of RAI No. 57. On October 22, 2008, a public meeting was held between AREVA NP Inc. and the NRC to discuss the U.S. EPR FSAR Chapter 7 and RAI No.'s 56 through 61.

A revised schedule for a technically correct and complete response to each of the remaining 11 questions is provided below.

Question #	Response Date
RAI 57 - 07.07-1	March 3, 2009
RAI 57 - 07.07-2	January 15, 2009
RAI 57 - 07.07-3	January 15, 2009
RAI 57 - 07.07-6	January 15, 2009
RAI 57 - 07.07-7	January 15, 2009
RAI 57 - 07.07-8	March 31, 2009
RAI 57 - 07.07-13	January 15, 2009
RAI 57 - 07.07-15	January 15, 2009
RAI 57 - 07.07-17	January 15, 2009
RAI 57 - 07.08-1	March 31, 2009
RAI 57 - 07.08-3	March 31, 2009

Sincerely,

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From: WELLS Russell D (AREVA NP INC)
Sent: Monday, October 13, 2008 6:00 PM
To: 'Getachew Tesfaye'

Cc: 'John Rycyna'; Pederson Ronda M (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 57, FSAR Ch 7

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 57 Response US EPR DC.pdf" provides technically correct and complete responses to 9 of the 20 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 57 Questions 07.07-14, and 07.08-2.

The following table indicates the respective pages in the response document, "RAI 57 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 57 — 07.07-1	2	2
RAI 57 — 07.07-2	3	3
RAI 57 — 07.07-3	4	4
RAI 57 — 07.07-4	5	5
RAI 57 — 07.07-5	6	6
RAI 57 — 07.07-6	7	7
RAI 57 — 07.07-7	8	8
RAI 57 — 07.07-8	9	9
RAI 57 — 07.07-9	10	10
RAI 57 — 07.07-10	11	11
RAI 57 — 07.07-11	12	12
RAI 57 — 07.07-12	13	15
RAI 57 — 07.07-13	16	16
RAI 57 — 07.07-14	17	17
RAI 57 — 07.07-15	18	18
RAI 57 — 07.07-16	19	19
RAI 57 — 07.07-17	20	20
RAI 57 — 07.08-1	21	21
RAI 57 — 07.08-2	22	22
RAI 57 — 07.08-3	23	23

A complete answer is not provided for 11 of the 20 questions. The schedule for a technically correct and complete response to this question is provided below.

Question #	Response Date
RAI 57 — 07.07-1	December 4, 2008
RAI 57 — 07.07-2	December 4, 2008
RAI 57 — 07.07-3	December 4, 2008
RAI 57 — 07.07-6	December 4, 2008
RAI 57 — 07.07-7	December 4, 2008
RAI 57 — 07.07-8	December 4, 2008
RAI 57 — 07.07-13	December 4, 2008
RAI 57 — 07.07-15	December 4, 2008
RAI 57 — 07.07-17	December 4, 2008
RAI 57 — 07.08-1	December 4, 2008
RAI 57 — 07.08-3	December 4, 2008

Sincerely,

(Russ Wells on behalf of)

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From: Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Friday, September 12, 2008 5:52 PM
To: ZZ-DL-A-USEPR-DL
Cc: Kenneth Mott; Terry Jackson; Michael Canova; Joseph Colaccino; John Rycyna
Subject: U.S. EPR Design Certification Application RAI No. 57 (951,972),FSAR Ch 7

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on August 26, 2008, and on September 5, 2008, 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: 116

Mail Envelope Properties (5CEC4184E98FFE49A383961FAD402D3197A958)

Subject: Response to U.S. EPR Design Certification Application RAI No. 57, Supplement 1
Sent Date: 1/14/2009 1:43:41 PM
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MESSAGE	6899	1/14/2009 1:43:44 PM
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Reply Requested: No
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Response to

Request for Additional Information No. 57 Supplement 1 (951,972), Revision 0

9/12/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 07.07 - Control Systems
SRP Section: 07.08 - Diverse Instrumentation and Control Systems
Application Section: FSAR Ch 7
ICE1 Branch

Question 07.07-2:

Are the rod position indicators that are discussed in Section 4.6.1 of the U.S. EPR DC-FSAR the same as the rod cluster control assembly (RCCA) lower end position sensors that are discussed in Section 7.2.1.3.6? If yes, what is the reason for using different terminology? If no, how are these two items/components different?

Response to Question 07.07-2:

The rod position indicators addressed in U.S. EPR FSAR Tier 2, Section 4.6.1 are not the same as the lower end position sensors addressed in U.S. EPR FSAR Tier 2, Section 7.2.1.3.6.

The rod position indicators can represent any value of RCCA insertion between 0% and 100%. These indicators are used in the low departure from nucleate boiling ratio (DNBR) reactor trip function addressed in U.S. EPR FSAR Tier 2, Section 7.2.1.2.1.

The lower end position indicators provide output only when the RCCA is fully inserted. These indicators are used in generating the P8 permissive signal as addressed in U.S. EPR FSAR Tier 2, Section 7.2.1.3.6.

U.S. EPR FSAR Tier 2, Figure 3.9.4-1, “Control Rod Drive Mechanism Assembly,” illustrates both the rod position measurement coil and the lower end position indicator coil.

FSAR Impact:

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

Question 07.07-3:

DC FSAR, Tier 2, Section 4.6.1, states that the safety-related analog measurement is addressed in Section 7.2. Where (section and page) exactly is the operation of the RCCA assemblies analog measurement discussed in detail in Section 7.2? Where are the schematics and the detailed functional diagrams for the analog measurement located? If they currently are not in the DC FSAR, please provide.

Response to Question 07.07-3:

The Response to RAI 57 Question 07.07-15 provides a description of analog rod position measurement operation.

U.S. EPR FSAR Tier 2, Figure 7.2-5 provides the functional diagram for logic processing of the analog measurement.

FSAR Impact:

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

Question 07.07-6:

DC FSAR, Tier 2, Section 7.7.1.1, page 7.7-2, states: "The logic that generates the control current comes from the RCSL System." How does the RCSL [Reactor Control, Surveillance and Limitation] system perform logic processing? Provide detailed explanation and provide at a minimum a schematic of process components with associated signal inputs and outputs and a logic diagram.

10 CFR Part 50, Appendix A, General Design Criterion 10, requires in part, that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any control of normal operation, including the effects of anticipated operational occurrences. The staff needs additional information on the RCSL system.

Response to Question 07.07-6:

The Response to RAI 57 Question 07.07-12 provides an explanation of RCSL logic processing.

FSAR Impact:

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

Question 07.07-7:

DC FSAR, Tier 2, Section 7.7.2.3.6, page 7.7-13, it states: "Four (i.e., one per PS division) RCCA drop detection logic signals are acquired in RCSL and voted one out of four." Explain and describe in detail how voting is accomplished in the RCSL. What is the design basis for voting "one out of four"? Provide schematic of process components with associated signal inputs and outputs.

10 CFR Part 50, Appendix A, General Design Criterion 13, requires, in part, appropriate controls to be provided to maintain variables that affect the fission process within prescribed operating ranges. The staff needs additional details regarding the RCCA drop detection logic to complete the review.

Response to Question 07.07-7:

The protection system (PS) is responsible for detecting dropped rods. This detection logic is described in U.S. EPR FSAR Tier 2, Section 7.2.1.2.1. Each division of the PS monitors a different quadrant of the reactor core for dropped rods. The division of the PS that detects a dropped rod sends a signal to the corresponding reactor control, surveillance and limitation (RCSL) division. This signal is acquired by the acquisition units (AU) and transferred to the control units (CU) in RCSL Divisions 1 and 4. In each CU, one out of four voting logic is performed to determine if any AUs received a dropped rod signal.

One out of four voting is used because a dropped rod in any one of the four quadrants affects core power level and distribution.

The Response to RAI 57 Question 07.07-12 provides a detailed explanation and schematic of RCSL process components with associated signal inputs, outputs, and logic diagram.

FSAR Impact:

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

Question 07.07-13:

Provide a list of non-safety control system failures, their effect on the plant, and the expected response of safety-related instrumentation and control systems.

10 CFR Part 50, Appendix A, General Design Criteria 13 requires, in part, that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

DC FSAR, Tier 2, Section 7.7.2.10, states that the non-safety related RCSL control system uses signal selection algorithms to calculate a process representation value which is then used by the non-safety control system to take action. The calculated value reflects all the input signals and not a specific signal value. This technique provides fewer challenges to safety-related I&C systems due to non-safety control system failure. The staff requests additional information to understand the non-safety control system failure modes and their effects on the plant and safety systems.

Response to Question 07.07-13:

In the U.S. EPR FSAR Tier 2, Chapter 15 safety analysis, the failure of non-safety-related control systems is considered as event initiators. The details of the effect of the failure on the plant and the safety system response are also described in U.S. EPR FSAR Tier 2, Chapter 15.

U.S. EPR FSAR Tier 2, Table 15.0-1—U.S. EPR Initiating Events contains the list of postulated accidents (PA) and anticipated operational occurrences (AOO) that are analyzed in the safety analysis. Detailed analysis for each event is separately provided in sections throughout U.S. EPR FSAR Tier 2, Chapter 15 under the heading “Identification of Causes and Event Descriptions.” The content of these sections include the cause of the event, the effect of the event on the plant, and the safety system response to the event.

For example, U.S. EPR FSAR Tier 2, Section 15.1.1.1 identifies the cause of the decrease in feedwater temperature to be an inadvertent opening of a feedwater heater bypass valve. The inadvertent opening of the feedwater heater bypass valve could be caused by either a failure of the mechanical system that contains the valve or a failure of the control system that controls the valve. The precise cause of failure of the valve is not relevant for the analysis, however, the failure of the control system that controls the valve is considered as a possible event initiator. U.S. EPR FSAR Tier 2, Section 15.1.1.1 also provides a description of the event, its effect on the plant, and the response to the safety related I&C system, which in this event a reactor trip provides protection to the plant.

FSAR Impact:

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

Question 07.07-15:

DC FSAR, Tier 2, Section 4.6.2, states that Control Rod Drive Mechanisms (CRDMs) are equipped with a digital and analog position indication system. The same section also states that the analog position indication system is part of the Protection System. Describe the physical location of the analog position indication system and how it operates.

Response to Question 07.07-15:

The CRDMs are equipped with position indicator coils. The protection system computers acquire the measurements provided by these coils.

The position indicator coils are located within each CRDM as illustrated in U.S. EPR FSAR Tier 2, Figure 3.9.4-1, “Control Rod Drive Mechanism Assembly.” The protection system equipment that acquires these measurements is located in the I&C cabinet rooms in the four Safeguards Buildings.

The analog position measurement is provided by a secondary coil coupled to a primary coil by the drive rod portion of the CRDM. As the drive rod moves up or down, the coupling between the primary coil and secondary coil changes. A constant electrical current is applied to the primary coil. The change in coupling allows measurement of current in the secondary coil to detect changes in the rod position.

For clarification, the terminology in U.S. EPR FSAR Tier 2, Section 4.6.1 and Section 4.6.2 will be revised to be more consistent with the descriptions in U.S. EPR FSAR Tier 2, Chapter 7.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 4.6.1 and Section 4.6.2 will be revised as described in the response and indicated on the enclosed markup.

Question 07.07-17:

Explain how the Process Automation System performs the calculations, setpoint comparisons, generate actuation signal outputs (i.e. to open or close valve) and perform logic processing. Provide detailed explanation and provide schematic(s) of process components with associated signal inputs and outputs.

10 CFR Part 50, Appendix A, General Design Criteria 13 requires, in part, that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges. The staff does not have sufficient information to determine adequacy of the PAS.

Response to Question 07.07-17:

Based on the type of information requested, the portion of GDC 13 in question appears to be the final sentence which states:

> *"Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges."*

NUREG-0800 Section 7.7.III identifies specific topics that should be emphasized in the review of control systems. The first bulleted item in this section, "Design Bases," is the review topic relevant to process automation system (PAS) compliance with the portion of GDC 13 in question. This item states:

> *"The review should confirm that the control systems include the necessary features for manual and automatic control of process variables within prescribed normal operating limits."*

U.S. EPR FSAR Tier 2, Section 7.7.2.2 describes individual control functions performed by the PAS that control variables of the type described in GDC 13. The descriptions of these control functions are intended to demonstrate that the PAS includes the functional features necessary to control process variables within their normal operating limits. These functional descriptions address:

- The purpose of the control function.
- The plant modes of operation in which the function is utilized.
- The primary variable to be controlled by the function.
- The input variables used in the control function.
- The plant actuators controlled by the function to affect the primary control variable.
- The relationship of the control band to other setpoints for the same variable.
- Plant conditions causing the control function to be bypassed.

- Functionality of the automatic portions of the control.
- Functionality of the manual portions of the control.
- For variable control setpoints, the conditions that determine the value of the setpoint.

In addition to the functional descriptions provided in U.S. EPR FSAR Tier 2, Section 7.7, other information has been provided in the U.S. EPR FSAR regarding the overall design and quality of the PAS. U.S. EPR FSAR Tier 2, Section 7.1.1.4.6 describes the PAS subsystem architectures including redundancy applied to control functions, the nature of the equipment used, the nature of data communications used, and the power supplies for the system.

AREVA NP has developed documentation that describes the QAP for non-safety-related products and services supplied by AREVA NP relative to the U.S. EPR design and deployment activities. The PAS design, procurement, fabrication, installation, and testing are performed in accordance with this program to ensure the quality of the system. The requirements within the documentation were modeled from the requirements of Addendum A of the AREVA NP Topical Report ANP-10266A Rev. 1, "AREVA NP Inc. Quality Assurance Plan for Design Certification of the U.S. EPR" (Reference 1). Topical Report ANP-10266A has been reviewed and approved by the NRC. .

AREVA NP believes that the information provided in the U.S. EPR FSAR relative to the PAS design and its significant control functions supports a conclusion that the PAS includes the necessary features for manual and automatic control of process variables within prescribed normal operating limits, consistent with the relevant review procedure in NUREG-0800, Section 7.7.III.

This RAI question requests a detailed explanation of how the PAS performs the calculations, performs setpoint comparisons, generates actuation signal outputs, and performs logic processing. A request was also made for schematic(s) of process components with associated signal inputs and outputs. This type of detailed information is dependent on:

- Selection of a computer platform, manufacturer, and vendor.
- Detailed application software design.
- Detailed system hardware design.

These aspects of the non-safety-related control system design are addressed later in the decision process. Further, AREVA NP does not understand why this level of detailed information is needed to support a reasonable assurance finding.

AREVA NP notes that:

NUREG-0800, Section 7.0.III, "Review Scope and Content," states:

> *"Regardless of the type of application under consideration, the fundamental purpose of the NRC review is to determine whether the facility and equipment, the proposed use of the equipment, the operating procedures, the processes to be performed, and other technical requirements provide reasonable assurance that the applicant/licensee will*

comply with the regulations of 10 CFR 1–199 (Chapter I), "Nuclear Regulatory Commission," and that public health and safety will be protected.

It is not intended that the review, audit, or inspection activities by the reviewer include a complete evaluation of all aspects of the design and implementation of the I&C system. The review scope need only be sufficient to allow the reviewer to reach the conclusion of reasonable assurance described above."

NUREG-0800, Section 7.0A.3.B, "Identification of Review Topics," states:

"The level of review depends upon the importance to safety of the system under review. Control systems receive a limited review as necessary to confirm that control system failures cannot have an adverse effect on safety system functions and will not pose frequent challenges to the safety systems."

These NUREG-0800 excerpts indicate that detailed design information for non-safety-related control systems, such as that requested in this RAI question, is not needed to reach a reasonable assurance finding. Instead, the review of non-safety-related control systems should be focused on the effects of control system failure on plant safety and independence of safety-related systems from the control systems.

U.S. EPR FSAR Tier 2, Section 7.7.2.6 addresses the effects of PAS actions and inaction on accidents and AOOs.

U.S. EPR FSAR Tier 2, Section 7.7.2.7 describes features of the PAS that minimize the effects of control system failures.

The Response to Question 07.07-13 describes consideration of non-safety-related system failures in the U.S. EPR FSAR Tier 2, Chapter 15 plant safety analyses.

U.S. EPR FSAR Tier 2, Section 7.1.1.6.4 describes the measures taken to establish independence between safety-related and non-safety-related I&C systems.

Reference:

1. AREVA NP Topical Report ANP-10266A Rev. 1, "AREVA NP Inc. Quality Assurance Plan for Design Certification of the U.S. EPR."

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

- SIS.

4.6.1 Information for Control Rod Drive System

The U.S. EPR contains 89 electromagnetic jack ~~designed~~ type CRDMs, each consisting of a drive rod, pressure housing, latch unit, and coil housing assembly. The CRDMs use natural air circulation, convection cooling; therefore a separate, dedicated liquid or forced air cooling system is not required. Natural convection cooling maintains the temperature of the CRDMs below design operating temperature. CRDM equipment is designed and qualified to operate in the reactor vessel cavity environment. Details of these CRDM components and how the components operate are provided in Section 3.9.4, and a diagram of the CRDM assembly is shown in Figure 3.9.4-1~~—Control Rod Drive Mechanism Assembly~~. An overview of the CRDM penetrations into the reactor pressure vessel is provided in Figure 3.9.5-1~~—Reactor Pressure Vessel General Arrangement~~, and the layout of RCCA control and shutdown banks within the core is provided in Figure 4.3-34~~—Rod Cluster Control Assembly Pattern~~. The RCCAs are described in Section 4.2. The instrumentation and control (I&C) systems providing rod control are described in Section 7.7, which includes the CRDCS and RCSL systems.

The CRDMs are mounted on top of the reactor pressure vessel head and are protected from potential tornado-generated missile damage by being housed in a Seismic Category 1 structure~~,~~ (i.e.~~:~~, containment). The ~~CDRM~~ CRDMs are protected from internally generated missiles by the concrete secondary shield wall and by reinforced concrete missile shield slabs mounted above the reactor vessel. The CRDMs are seismically restrained by the reactor pressure vessel closure head equipment as addressed in Section 5.4.14.

07.07-15

~~The CRDMs are equipped with a digital and analog position indication system so the RCCA position is measured over the height of the core by two diverse methods:~~

The I&C systems associated with RCCA control count CRDM movement steps to provide a digital measurement of RCCA position. The CRDMs are also equipped with position indicator coils that provide analog RCCA position measurements. As such, the RCCA position is measured over the height of the core by two diverse methods:

- The digital measurement~~, as addressed in Section 7.7,~~ is non-safety related.
- The analog measurement, ~~as addressed in Section 7.2~~using position indicator coils, is safety related.

Additionally, an upper and lower rod position indicator provides indication when the RCCA is at the top or bottom position.

Section 7.2 describes the PS, including I&C for CRDS trip functions.

- Reliability.
- Common cause failure.

07.07-15

As described in Section 7.2, the PS is designed to fail into a safe state or into a state that has been demonstrated to be acceptable in accordance with GDC 23. Each protective function has different requirements and therefore different criteria are used to achieve a fail safe state. The PS divisions are physically separated in their respective Safeguard Buildings. The four divisionally separated rooms containing the PS equipment are in different fire zones. Therefore, the consequences of internal hazards, such as fire, would impact only one PS division. The ~~analog~~ position indicator~~ion~~ coils, which provide input measurements to ~~system, which is part of~~ the PS, is the only instrumentation required of the CRDM and supporting systems to safely operate. Failure of the ~~analog rod~~ position indicator~~s~~ coils to operate properly would not prevent the RCCAs from being insert~~ing~~ed into the core or result~~ing~~ in inadvertent withdrawal from the core. The PS has also been evaluated in the probabilistic risk assessment (PRA) and determined to be of high reliability because of its diverse signals and redundant channels and divisions. Chapter 19 provides a summary of the PRA. The PS is environmentally and seismically qualified to perform its designed safety functions while exposed to normal, abnormal, test, and post-event environmental conditions, as addressed in Section 7.2. As noted in Sections 7.1 and 7.7, there is independence between safety-related equipment of the PS and non-safety-related equipment, and failure of the non-safety-related portions of the CRDCS can not affect the safety-related function of the trip contactors.

A failure modes and effects analysis of the PS, as described in Section 7.2, verifies that the PS will initiate a reactor trip when required even with a credible failure of a single active component.

4.6.3 Testing and Verification of the Control Rod Drive System

The CRDS operability assurance program is described in Section 3.9.4.4. Testing of the CRDS verifies system operability and is conducted in several stages:

- Prototype tests and manufacturer tests prior to initial installation.
- Preoperational and initial startup tests.
- Inservice tests.
- Tests following maintenance and fuel movement.

Abstracts of CRDS tests performed as part of the initial test program are provided in Section 14.2. Also, the Technical Specifications and Section 3.1 provide requirements for surveillance and testing of reactivity control systems.