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MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, KONAN 2-CHOME, MINATO-KU

May 20, 2009

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffery A. Ciocco,

Docket No. 52-021
MHI Ref: UAP-HF-09259

**Subject:** MHI's Response to US-APWR DCD RAI No. 316

**Reference**: 1) "Request for Additional Information No. 316-2296 Revision 0, SRP Section: 04.06 – Functional Design of Control Rod Drive System

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") documents as listed in Enclosure.

Enclosed is the response to 1 RAI contained within Reference 1.

As indicated in the enclosed materials, this submittal contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted with the information identified as proprietary redacted and replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittals. His contact information is provided below.

Sincerely,

Y. Ogata

Yoshiki Ogata,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. "Response to Request for Additional Information No. 316-2296, Revision 0" (Proprietary Version)
3. "Response to Request for Additional Information No. 316-2296, Revision 0" (Non-Proprietary Version)

CC: J. A. Ciocco
C. K. Paulson

Contact Information
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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-09259

**MITSUBISHI HEAVY INDUSTRIES, LTD.**

**AFFIDAVIT**

I, Yoshiki Ogata, state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, LTD ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Response to Request for Additional Information No. 316-2296, Revision 0", dated May 2009, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages contain proprietary information are identified with the label "Proprietary" on the top of the page, and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The information identified as proprietary in the enclosed document has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that it describes the unique design parameters developed by MHI for the Reactor Internals and Core Support Structures.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the

information contained in the referenced document would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

A. Loss of competitive advantage due to the costs associated with the development of the unique design parameters.

B. Loss of competitive advantage of the US-APWR created by the benefits of the Control Rod Drive Mechanism operation.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 20th day of May 2009.

Y. Ogata

Yoshiki Ogata,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosure 3

UAP-HF-09259
Docket No. 52-021

Response to Request for Additional Information No. 316-2296, Revision 0

May 2009
(Non-Proprietary)

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/20/2009

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

**RAI NO.:** NO. 316-2296 REVISION 0

**SRP SECTION:** 04.06 FUNCTIONAL DESIGN OF CONTROL ROD DRIVE SYSTEM

**APPLICATION SECTION:** 4.6

**DATE OF RAI ISSUE:** 4/2/09

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**US-APWR Design Certification- 04.06 - Functional Design of Control Rod Drive System**
[Review performed against revision 0 of the US-APWR DCD Tier 2.]

**QUESTION NO. : RAI 2296-4.6-1**

GDC 4 requires that the CRDS should remain functional and provide reactor shutdown capabilities under adverse environmental conditions. In FSAR Section 4.6, the applicant states that failure in the CRDM cooling system will, in the worst case, result in an individual control rod trip or a full reactor trip. However, the staff was unable to find a discussion or reference that included a consideration of other environmental conditions, such as humidity, vibration, or possible pipe fracture releasing fluid onto the CRDMs that could affect the CRDS functional capabilities to provide a safe reactor shutdown.

Provide a discussion or reference(s) that identifies the environmental conditions that were evaluated to demonstrate the CRDS capabilities to operate and perform its design function in the reactor vessel cavity under adverse environmental conditions such that the requirements in GDC 4 are satisfied. Specifically, address the seismic qualification or failure of non-safety related equipment in vicinity of the CDRS (e.g., failure of an in core instrumentation line failure on CRDS), a description or reference to the equipment qualification program and how it relates to CDRS safety related performance during AOOs and PAs.

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**ANSWER:**

The CRDMs meet the requirements of GDC 4. Environmental conditions in the reactor vessel cavity can affect electrical components such as coil stacks. The CRDM coil stack is designed for an environmental temperature under 392 °F. The design provides cooling to maintain acceptable environmental temperatures. Furthermore if CRDM coil stack fails, CRDM latches will open and RCCAs will drop into the core to shut down the reactor with respect to an electrical fault the design of the CRDM is a fail safe design.

The CRDMs are also seismically qualified as identified in the DCD Subsection 3.9.4.3. The CRDM housings are supported to satisfy displacement requirement, under 1.18 inches, during seismic condition, In the allowable displacement of the CRDM housing, the RCCA can drop into the core to shut down the reactor.

The CRDMs are included in the equipment qualification program. Rod drop test will be performed during preoperational testing (DCD Subsection 14.2.12.2.1.5). In-service condition, monitoring include rod drop tests performed at start up after an outage. Periodic in-service testing is described in DCD Subsection 4.6.3.

**Impact on DCD**

There is no impact on DCD.

**Impact on COLA**

There is no impact on COLA.

**Impact on PRA**

There is no impact on PRA.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**5/20/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

**RAI NO.:** **NO. 316-2296 REVISION 0**

**SRP SECTION:** **04.06 FUNCTIONAL DESIGN OF CONTROL ROD DRIVE SYSTEM**

**APPLICATION SECTION:** **4.6**

**DATE OF RAI ISSUE:** **4/2/09**

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**QUESTION NO. : RAI 2296-4.6-2**

The instrumentation and controls for the non-safety-related reactor control is described in DCD Section 7.7 with Subsections 7.7.1.3 and 7.7.1.4 referring to the digital and analog control rod position indication systems respectively. In addition, a supplemental discussion of the two independent systems is presented in Chapter 16, Technical Specification (TS) BASES Section B 3.1.7, "Rod Position Indication." The BASES discussion of the two methods of position indication is described in detail that clearly illustrates their diverse and independent system design to measure axially the RCCA position. In the TS, digital measurement is identified as Bank Demand Position Indication System and the analog measurement is referred to as the Rod Position Indication System. However, the term, "bank demand position indication system", is not identified in Subsection 7.7.1.3 but is described in terms of "motion demand signals". Also, in Figure 7.1-1, it is unclear as to whether the bank demand position indication system is part of the Rod Position Indication System diagram block or the CRDM Control System block. For consistency with TS Section 3.1.7, provide clarification in Subsection 7.7.1.3 to include the term "bank demand position indication system" and its relationship to Figure 7.1-1. Also, provide justification for not including the term in the ACRONYMS AND ABBREVIATIONS listing.

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**ANSWER:**

The Bank Demand Position Indication System which is commonly called the group step counter is part of the Control Rod Drive Mechanism Control System (CRDM-CS). Therefore, we will add the description about that.

**Impact on DCD**

The description which is underlined will be added to the 6th paragraph of the Technical Specification BASES Section B3.1.7 of Chapter 16 of the DCD as below:

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) <u>which is included in the CRDM control system</u> and the Rod Position Indication (RPI) System.

**Impact on COLA**

There is no impact on COLA.

**Impact on PRA**

There is no impact on PRA.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**5/20/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

**RAI NO.:** **NO. 316-2296 REVISION 0**

**SRP SECTION:** **04.06 FUNCTIONAL DESIGN OF CONTROL ROD DRIVE SYSTEM**

**APPLICATION SECTION:** **4.6**

**DATE OF RAI ISSUE:** **4/2/09**

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**QUESTION NO. : RAI 2296-4.6-3**

As discussed in Section 9.4.6, "Containment Ventilation System", the control rod drive mechanism (CRDM) cooling system is part of the containment ventilation system which is classified as a non-safety related and non-seismic Category I system. The CRDM cooling system maintains the temperature of the CRDMs below design operating temperature by satisfying the following design bases: (1) Removal of heat dissipated by the CRDMs, (2) Continuity and reliability of operation with 100% standby capacity for system fans, and (3) During a loss of offsite power (LOOP) condition, the CRDM cooling fans are served by the alternate AC power source. However, references were not provided in this section to document the specific analysis and qualification of the CRDM cooling system to satisfy the design bases. Provide the reference(s) that identifies the analysis which supports qualification of the CRDM cooling system compliance to the design bases.

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**ANSWER:**

The CRDM cooling system is designed to maintain cooling air flow at the CRDM coil stack assembly area. The required cooling air flow rate per CRDM is about [ ] cfm based on existing plants. Required total flow rate is [ ] cfm which is decided considering the space in the CRDM cooling shroud which is set on the reactor vessel head. On the other hand, the CRDM cooling system's capacity is 71,000 cfm, which is identified in Table 9.4.6-1 of the DCD Subsection 9.4.6. Thus, the US-APWR CRDM cooling system has enough capacity compared with requirement. If the CRDM cooling function fails, the temperature of the CRDM coil stack will increase. High temperature condition lead to loss of CRDM coil stack function. Loss of CRDM coil stack function is meaning loss of RCCA holding function which leads to drop a RCCA in to the core to shutdown the reactor. Thus safety reactivity control function such as reactor shutdown is maintained after loss of CRDM cooling function.

**Impact on DCD**

There is no impact on DCD.

**Impact on COLA**

There is no impact on COLA.

**Impact on PRA**

There is no impact on PRA.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/20/2009

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

**RAI NO.:** NO. 316-2296 REVISION 0

**SRP SECTION:** 04.06 FUNCTIONAL DESIGN OF CONTROL ROD DRIVE SYSTEM

**APPLICATION SECTION:** 4.6

**DATE OF RAI ISSUE:** 4/2/09

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**QUESTION NO. : RAI 2296-4.6-4**

In FSAR Subsection 4.6.2, the applicant states that the protection of the essential components of the CRDS from the effects of postulated moderate and high energy waterline breaks and associated generated missiles is described in Subsection 3.5.1.2. Primary missile protection is provided by locating the credible missile sources behind concrete walls and floors, and/or locating the SSCs outside the zones of postulated missile strikes. However, the discussion was a general description of the protection features without directly addressing the specific features incorporated to protect the CRDS components. Provide a reference or detail discussion that addresses the specific features incorporated to protect the CRDS components including the analysis that confirms the protection features satisfy the requirements of GDC 4.

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**ANSWER:**

The load against postulated moderate and high energy water line breaks is considering pressure housing design condition. The stress analysis results of the CRDM housing is described in the technical report MUAP-09009-P/NP R0 "Summary of Stress Analysis Results for the US-APWR Control Rod Drive Mechanism" which was submitted to NRC.

Any missile source is not occurred in the containment vessel which is identified in the DCD Subsection 3.5.1.2.2.1.

**Impact on DCD**

There is no impact on DCD.

**Impact on COLA**

There is no impact on COLA.

**Impact on PRA**

There is no impact on PRA.

**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

5/20/2009

**US-APWR Design Certification**

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**APPLICATION SECTION:** **4.6**

**DATE OF RAI ISSUE:** **4/2/09**

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**QUESTION NO. : RAI 2296-4.6-5**

In FSAR Section 4.6, the applicant states that GDC 23 is satisfied without providing a discussion or reference of an evaluation performed to come to this conclusion. The applicant also states that in the worst case, failure of the CRDM cooling system will result in an individual control rod trip or a full reactor trip. The staff considers the failure of the CRDM cooling system as an adverse condition by which the CRDS fails in an acceptable condition that prevents damage to the fuel cladding and excessive reactivity changes during failure thereby satisfying GDC 23. However, the staff believes that the one adverse condition identified by the applicant is insufficient to conclude that GDC 23 is met without providing an evaluation of the adverse conditions considered. Provide a discussion or reference of an evaluation performed to confirm that GDC 23 is satisfied.

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**ANSWER:**

The CRDM system is fail safe. A failure of the CRDM results in a control rod drop, or insertion, resulting in a decrease in core reactivity. Loss of electrical power results in fail safe operation. The CRDMs are environmentally and seismically qualified and are protected from high energy line breaks and missiles. Failure of normal cooling does not negate fail safe features. Hence the design meets GDC 23.

Other failure modes of the CRDS will be addressed in DCD subsection 3.9.4 in response to RAI 107-1293 revision 0 as follows.

The ASME Code requirements do not apply to non-pressurized components such as latch mechanism, the drive rod and the coil assembly. These non-pressurized components are classified as non-safety components. This is based on a fail safe design with scram principle utilizing gravity. If the coil assembly or electric device of the CRDM fails, the control rods are dropped/inserted into the core by gravity and reduce the reactivity. If the drive rod fails, the control rods drop into the core and reduce the reactivity.

**Impact on DCD**

DCD Revision 2 will incorporate the following changes as resolving RAI 107-1293 and RAI 316-2296:

Replace the 1st sentence of 2nd paragraph in Subsection 3.9.4.2.3 with the following: "The ASME Code requirements do not apply to non-pressurized components such as latch mechanism, the drive rod and the coil assembly. These non-pressurized components are classified as non-safety components. This is based on a fail safe design with the scram principle utilizing gravity. If the coil assembly or electric device of the CRDM fails, the control rods are dropped/inserted into the core by gravity and reduce the reactivity. If the drive rod fails, the control rods drop into the core and reduce the reactivity."

**Impact on COLA**

There is no impact on COLA.

**Impact on PRA**

There is no impact on PRA.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**5/20/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

**RAI NO.:** **NO. 316-2296 REVISION 0**

**SRP SECTION:** **04.06 FUNCTIONAL DESIGN OF CONTROL ROD DRIVE SYSTEM**

**APPLICATION SECTION:** **4.6**

**DATE OF RAI ISSUE:** **4/2/09**

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**QUESTION NO. : RAI 2296-4.6-6**

CRDS testing is categorized in several phases: (1) Prototype component testing; (2) Production testing of components; (3) On-site preoperational testing; and (4) Periodic in service testing. In Section 4.6.3, reference is made to the CRDS operability assurance programs, described in Section 3.9.4.4, which were formed to confirm the functional performance of the CRDMs both statically and dynamically. This section states that the structural integrity of the RCS pressure boundary is confirmed by stress analysis performed in accordance with ASME Code, Section III. However, the section did not provide a discussion or a reference to a report of the analysis that demonstrates compliance with the ASME Code. Provide a discussion or a reference of the analysis that supports compliance with the ASME Code.

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**ANSWER:**

A stress report of the CRDM pressure boundary has been submitted to the NRC and is contained in MUAP-09009-P/NP.

**Impact on DCD**

There is no impact on DCD.

**Impact on COLA**

There is no impact on COLA.

**Impact on PRA**

There is no impact on PRA.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**5/20/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

**RAI NO.:** **NO. 316-2296 REVISION 0**

**SRP SECTION:** **04.06 FUNCTIONAL DESIGN OF CONTROL ROD DRIVE SYSTEM**

**APPLICATION SECTION:** **4.6**

**DATE OF RAI ISSUE:** **4/2/09**

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**QUESTION NO. : RAI 2296-4.6-7**

Although the Technical Specifications' (TS) Section 3.1 provides requirements for surveillance and testing of reactivity control systems, it was excluded from Section 4.6 as part of the CRDS operability assurance programs. Provide the justification to exclude TS Section 3.1 with respect to the CRDS operability assurance programs.

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**ANSWER:**

Surveillance and testing requirements in TS Section 3.1 are intended to check plant conditions. On the other hand, the CRDS operability assurance program assures CRDS function. Therefore, surveillance and testing requirements in TS Section 3.1 are excluded from Subsection 4.6.

**Impact on DCD**

There is no impact on DCD.

**Impact on COLA**

There is no impact on COLA.

**Impact on PRA**

There is no impact on PRA.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**5/20/2009**

**US-APWR Design Certification**

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**DATE OF RAI ISSUE:** **4/2/09**

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**QUESTION NO. : RAI 2296-4.6-8**

In Section 4.6.5, the applicant restates the description given in FSAR Section 4.6.4 that only a limited number of postulated events assume the availability of two reactivity control systems to prevent or mitigate the accident such as the SLB and LOCA. The applicant did not discuss or provide a reference to demonstrate that the combined performance of the two reactivity control systems is in compliance with GDCs 27 and 28. Provide a discussion or a reference of the analysis that confirms the combined performance of the two reactivity control systems is in compliance with GDCs 27 and 28.

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**ANSWER:**

A discussion or reference of the analysis that confirms the combined performance of two reactivity control system will be submitted by the 16th of June '09.

**Impact on DCD**

There is no impact on DCD.

**Impact on COLA**

There is no impact on COLA.

**Impact on PRA**

There is no impact on PRA.

# RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**5/20/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

**RAI NO.:** **NO. 316-2296 REVISION 0**

**SRP SECTION:** **04.06 FUNCTIONAL DESIGN OF CONTROL ROD DRIVE SYSTEM**

**APPLICATION SECTION:** **4.6**

**DATE OF RAI ISSUE:** **4/2/09**

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**QUESTION NO. : RAI 2296-4.6-9**

During the review of FSAR Section 4.6, the staff found an editorial error between the ACRONYMS AND ABBREVIATIONS sections of Tier 1 and Tier 2 in regard to the SIS definition. Tier 1 refers to SIS as safety injection pump; whereas, Tier 2 defines SIS as safety injection system. It is generally acknowledged that the conventional abbreviation for safety injection system is SIS. For consistency, correct the editorial error.

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**ANSWER:**

The DCD revision 1 of Tier 1 corrected the editorial error identified in ACRONYMS AND ABBREVIATIONS as follows.

SIS as safety injection ~~pump~~ system

**Impact on DCD**
There is no impact on the DCD.

**Impact on COLA**

There is no impact on COLA.

**Impact on PRA**

There is no impact on PRA.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**5/20/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

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**APPLICATION SECTION:** **4.6**

**DATE OF RAI ISSUE:** **4/2/09**

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**QUESTION NO. : RAI 2296-4.6-10**

During the review of initial startup tests in respect to FSAR Section 4.6, the staff identify that the prerequisite section of test 14.2.12.1.9, "Reactor Control, Rod Control, and Rod Position Indication Preoperational Test", has an editorial error in the list where the last prerequisite is mislabeled and added onto prerequisite 4. For consistency, correct the editorial error.

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**ANSWER:**

The DCD revision 1 of Tier 1 corrected the editorial error in Subsection 14.2.12.1.9 as follows.

B. Prerequisites
1. Required construction testing is completed.
2. Component testing and instrument calibration is completed.
3. Test instrumentation is available and calibrated.
4. Required support systems are available.
~~2.~~5. Required electrical power supplies and control circuits are operational.

**Impact on DCD**

There is no impact on DCD.

**Impact on COLA**

There is no impact on COLA.

**Impact on PRA**

There is no impact on PRA.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**5/20/2009**

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**APPLICATION SECTION:** **4.6**

**DATE OF RAI ISSUE:** **4/2/09**

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**QUESTION NO. : RAI 2296-4.6-11**

During the review of the initial startup tests in respect to FSAR Section 4.6, the staff identify that the acceptance criterion section D of test 14.2.12.2.1.8, "Rod Position Indication Test", refers to "the required indication and alarm functions, as described in Subsection 7.7.1.4"; however, FSAR Section 7.7.1.4 does not discuss the required indication and alarm functions. Identify the FSAR section that supports the acceptance criterion of section D. A discussion of the control rod position monitor alarm functions is presented in FSAR Subsection 7.7.1.1.4. If this is not a complete list, provide a complete list or reference where the alarms can be found.

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**ANSWER:**

As described in the question, the alarm function is presented in section 7.7.1.1.4. We will add the reference description in section D of test 14.2.12.2.1.8 as follows:

**Impact on DCD**

The description which is underlined will be added to the section D of test 14.2.12.2.1.8 as below:

The rod position indication system performs the required indication and alarm functions, as described in Subsection <u>7.7.1.1.4 and</u> 7.7.1.4, and each rod operates over its entire range of travel.

**Impact on COLA**

There is no impact on COLA.

**Impact on PRA**

There is no impact on PRA.