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

February 4, 2009

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

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-09037

Subject: MHI's Responses to US-APWR DCD RAI No.146-1804 Revision 0

Reference: 1) "REQUEST FOR ADDITIONAL INFORMATION NO. 146-1804 REVISION 0, SRP Section: 16 - Technical Specifications Application Section: TS Section 3.4, QUESTIONS for Technical Specification Branch (CTSB)" dated January 9, 2009.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Responses to Request for Additional Information No.146-1804 Revision 0."

Enclosed is the responses to Questions 16-66 through 16-99 that are contained within Reference 1.

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 below.

Sincerely,

Y. Ogata

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

Enclosure:

1. Responses to Request for Additional Information No.146 Revision 0

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

D081
NRO

<u>Contact Information</u>

C. Keith Paulson, Senior Technical Manager
Mitsubishi Nuclear Energy Systems, Inc.
300 Oxford Drive, Suite 301
Monroeville, PA 15146
E-mail: ck_paulson@mnes-us.com
Telephone: (412) 373-6466

Enclosure 1

UAP-HF-09037
Docket No. 52-021

Responses to Request for Additional Information No.146-1804
Revision 0

February 2009

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-66

TS Section 3.4 (EDITORIAL).

The following typographical or editorial errors were noted in US-APWR TS LCO 3.4 and associated BASES:
1. Page 3.4.7-1, LCO 3.4.7 NOTES 2: The phrase "the other RHR loop is" should be "the other two RHR loops are."

2. Page 3.4.7-2, Condition A statement: the logical connector "OR" should be indented and the logical connector "AND" should be flushed with the text letf margin.

3. Page 3.4.8-1, LCO 3.4.8 NOTES 2: The phrase "One RHR loop" should be "One required RHR loop."

4. Page 3.4.12-4, SURVEILLANCE REQUIREMENTS, SR 3.4.12.5, FREQUENCY: The connectors "AND" and "OR" should be underlined per the Improved Technical Specification Writers Guide, TSTF-GG-05-01.

5. Page 3.4.16-1, Required Action A.1: Insert the sign "<" after "I-131"

6. Page B 3.4.3-1, BACKGROUND, 5th Paragraph, 2nd Sentence: Incomplete sentence.

7. Page B 3.4.3-4, ACTIONS, A.1 and A.2, 1st Paragraph, 2nd Sentence: The word "parameter" should be "parameters."

8. Page B 3.4.7-3, LCO, 5th Paragraph (top of page): "Note 2" should be "Note 3."

9. Page B 3.4.7-3, APPLICABILITY, 1st Paragraph, 2nd Sentence: The phrase "Two Loops3" should be "Two loops."

10. Page B 3.4.8-1, LCO, 1st Paragraph, 4th Sentence: The word "loop" should be "loops."

11. Page B 3.4.9-1, BACKGROUND, Second Paragraph, Second Sentence: The phrase "pressurizer power operated relief valves (PORVs)" should be "safety depressurization valves

(SDVs)"

12. Page B 3.4.11-3, ACTIONS, B.1, B.2, and B.3, First Sentence: The word "PORV" should be "SDV".

13. Page B 3.4.11-5, ACTIONS, F.1, delete "F.2 and F.3" since only F.1 exists in TS 3.4.11.

14. Page B 3.4.11-5, SURVEILLANCE REQUIREMENTS, SR 3.4.11.2, First Sentence: The word "PORV" should be "SDV".

15. Page B 3.4.12-4, APPLICABLE SAFETY ANALYSES, Heat Input Type Transients, Last Paragraph, 1st Sentence: The phrase "two RHR suction relief valve" should be "two RHR suction relief valves."

16. Page B 3.4.12-4, APPLICABLE SAFETY ANALYSES, RHR Suction Relief Valve Performance, 2nd Sentence: The phrase "that RHR" should be "that one RHR."

17. Page B 3.4.12-8, SURVEILLANCE REQUIREMENTS, SR 3.4.12.1/2/3, First Paragraph: the word "incapable" should be "capable."

18. Page B 3.4.12-9, SURVEILLANCE REQUIREMENTS, SR 3.4.12.6, 1st Sentence: The phrase "by testing it" should be "by testing them."

19. Page B 3.4.12-9, SURVEILLANCE REQUIREMENTS, SR 3.4.12.6, 2nd Sentence: "SR 3.4.12.2" should be "SR 3.4.12.4."

20. Page B 3.4.12-9, REFERENCES, Reference 4: "ASME, Section XI" should be "ASME Code for Operation and Maintenance of Nuclear Plants"

21. Page B 3.4.14-2, BACKGROUND, Last Sentence of Fifth Paragraph: Add the word "in" after the word "listed"

22. Page B 3.4.14-4, ACTIONS, C.1, 1st Sentence: The phrase "incapable preventing" should be "incapable of preventing."

23. Page B 3.4.14-6, SURVEILLANCE REQUIREMENTS, SR 3.4.14.2, First Sentence of First Paragraph: Add the word "beyond" after the word "system"

24. Page B 3.4.14-6, SURVEILLANCE REQUIREMENTS, SR 3.4.14.2, 1st Paragraph, 1st Sentence: The word combination "of900 psig," should be "of 900 psig."

25. Page B 3.4.14-6, SURVEILLANCE REQUIREMENTS, SR 3.4.14.2, 1st Paragraph, 4th Sentence: The word combination "24month," should be "24 month."

26. Page B 3.4.15-1, BACKGROUND, 3rd Paragraph, Second Sentence: The word combination "monitorare," should be "monitor are."

27. Page B 3.4.16-3, ACTIONS, A.1 and A.2, Third Paragraph, Third Sentence: A line break was incorrectly inserted after the word "conservatism."

ANSWER:

TS 3.4 and related Bases are revised to incorporate the comments in QUESTION NO.16-66 in items 1 through 3, 5 through 15, and 18 through 27.

4. The format of SR applying Surveillance Frequency Control Program is prescribed in TSTF-425, which was approved by NRC. We followed this prescription, in which "OR" is neither underlined nor indented.

16. In US-APWR design, two RHR suction relief valves should be in operation during LTOP events. Therefore, LCO 3.4.12 BASES, Applicable Safety Analyses, RHR Suction Relief Valve Performance will be revised to read the required number of RHR suction relief valve is two.

17. In this SR, two of four safety injection pumps and one of two charging pumps should be confirmed to be incapable of injecting. Therefore, the description of this SR seems to be appropriate.

Impact on DCD

1. LCO 3.4.7 NOTES 2 will be revised as follows:

   2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loops are~~is~~ OPERABLE and in operation.

2. LCO 3.4.7 Condition A will be revised as follows:

   A. One required RHR Loop inoperable.

      OR

      One or more required SGs with secondary side water level not within limit

      AND

      Two RHR loops OPERABLE and in operation.

3. LCO 3.4.8 NOTES 2 will be revised as follows:

   2. One required RHR loop may be inoperable for = 2 hours for surveillance testing provided that the other two RHR loops are OPERABLE and in operation.

4. There is no impact on the DCD.

5. Please see Response to Question No. 16-97.

6. LCO 3.4.3 BASES, Background, 5th paragraph, 2nd sentence will be revised as follows:

   Reference 1 requires a~~A~~n adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests.

7. LCO 3.4.3 BASES, ACTION A.1 and A.2, 5th paragraph, 2nd sentence will be revised as follows:

   Restoration of P/T parameters to the analyzed range reduces the RCPB stress.

8. LCO 3.4.7 BASES, LCO, 5th paragraph, 1st sentence will be revised as follows:

   Note 3~~2~~ requires that the secondary side water temperature of each SG be = 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature = Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR.

9. LCO 3.4.7 BASES, Applicability, 1st paragraph, 2nd sentence will be revised as follows:

   Two loops~~3~~ of RHR provides sufficient circulation for these purposes.

10. LCO 3.4.8 BASES, LCO, 1st paragraph, 4th sentence will be revised as follows:

   A minimum of two running CS/RHR pumps meets the LCO requirement for two loops in operation.

11. LCO 3.4.9 BASES, Background, 2nd paragraph, 2nd sentence will be revised as follows:

   Pressurizer safety valves and safety depressurization valves~~pressurizer power operated relief valves~~ are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Safety Depressurization Valves (SDVs)~~Pressurizer Power Operated Relief Valves (PORVs)~~," respectively.

12. LCO 3.4.11 BASES, ACTIONS, B.1, B.2 and B.3, 1st sentence will be revised as follows:

   If one SDV~~PORV~~ is inoperable and not capable of being manually cycled, it must be either restored, or isolated by closing the associated block valve and removing the power to the associated block valve.

13. LCO 3.4.11 BASES, ACTIONS, F.1, F.2 and F.3 will be revised as follows:

   F.1~~, F.2, and F.3~~

14 LCO 3.4.11 BASES, Surveillance Requirements, SR 3.4.11.2, 1st sentence will be revised as follows:

   SR 3.4.11.2 requires a complete cycle of each SDV~~PORV~~.

15. LCO 3.4.12 BASES, Applicable Safety Analyses, Heat Input Type Transients, last paragraph, 1st sentence will be revised as follows:

   Since neither two RHR suction relief valves nor the RCS vent can handle the pressure transient need from accumulator injection, when RCS temperature is low, the LCO also requires the accumulators isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

16. LCO 3.4.12 BASES, Applicable Safety Analyses, RHR Suction Relief Valve Performance will be revised as follows:

   The RHR suction relief valves do not have variable pressure and temperature lift setpoints. Analyses must show that two RHR suction relief valves lifting at its specified setpoint will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting LTOP event, two~~an~~ RHR suction relief valves will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation = 10% of the rated lift setpoint.

17. There is no impact on the DCD.

18. LCO 3.4.12 BASES, Surveillance Requirements, SR 3.4.12.6, 1st sentence will be revised as follows:

   The RHR suction relief valves shall be demonstrated OPERABLE by verifying that both RHR suction isolation valves in one flow path are open and by testing them~~it~~ in accordance with the Inservice Testing Program.

19. LCO 3.4.12 BASES, Surveillance Requirements, SR 3.4.12.6, 2nd sentence will be revised as

follows:

(Refer to SR 3.4.12.4~~2~~ for the RHR suction isolation valve Surveillance.)

20. Please see the response for Question No. 16-88.

21. LCO 3.4.14 BASES, Background, 5th paragraph, last sentence will be revised as follows:

The PIVs are listed in Chapter 3. (Ref. 6).

22. LCO 3.4.14 BASES, ACTIONS, C.1, 1st sentence will be revised as follows:

The inoperability of the RHR suction valve interlock renders the RHR suction isolation valves incapable of preventing inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure.

23. and 24. LCO 3.4.14 BASES, Surveillance Requirements, SR 3.4.14.2, 1st paragraph, 1st sentence will be revised as follows:

Verifying that the RHR suction valve interlock is OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond its design pressure of 900 psig.

25. LCO 3.4.14 BASES, Surveillance Requirements, SR 3.4.14.2, 1st paragraph, 4th sentence will be revised as follows:

[The 24 month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage.

26. LCO 3.4.15 BASES, Background, 3rd paragraph, 2nd sentence will be revised as follows:

The containment sump used to collect unidentified LEAKAGE and air cooler condensate flow rate monitor are instrumented to alarm for increases of greater than or equal to 1.0 gpm in the normal flow rates.

27. Please see Response to Question No. 16-97.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-67

TS 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits.

Provide further explanation in the second paragraph of the LCO 3.4.1 Bases discussion of LCO.

The Bases state "RCS total flow contains a measurement error based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators." The explanation should explicitly identify and describe the source of the measurement error. For example, the STS, NUREG-1431, Bases provide a discussion of this measurement error due to fouling of the feedwater venturi used in the operating plants. Discuss, in the US-APWR TS LCO 3.4.1, BASES LCO section, impacts of detected and undetected fouling of the feedwater flow venturi on performing a precision heat balance, or clarify why this issue does not need to be addressed. NUREG-1431, Rev. 3.1 BASES, LCO section for LCO 3.4.1 indicates that potential fouling of the feedwater venturi could bias the precision heat balance value for total RCS flow rate.

This information will be used to ensure completeness of information provided in the TS Bases.

ANSWER:

In the US-APWR design, the feedwater flow for the thermal design is measured by the ultrasonic flowmeter. Therefore, the discussion about fouling of the feedwater venturi is not needed.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-68

TS 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits.

Compare the minimum total flow rate of 460000 gpm specified in LCO 3.4.1 to the the RC Pump design data shown in FSAR Table 5.4.1. Correct or justify any inconsistencies.

FSAR Table 5.4.1-1 lists a pump design flow of 112000 gpm per pump which is equivalent to a total flow of only 448000 gpm. LCO 3.4.1 and the associated bases show a minimum total flow rate of 460000 gpm which accounts for a maximum of 10% SG tube plugging.

This information is needed to ensure TS requirements are consistent with referenced information provided in the APWR FSAR.

ANSWER:

The description in LCO 3.4.1 provides an allowable minimum value for the measured RCS flow rate, which may contain measurement errors. This limit flow rate is called as Minimum Measured Flow (MMF), and is defined as 460,000 gpm (4 x 115,000 gpm/loop) in DCD Table 5.1-3.
As long as the measured flow rate is greater than MMF, the net RCS flow is assured to be greater than Thermal Design Flow (TDF) at 10% SG tube plugging, 448,000 gpm (4 x 112,000 gpm/loop), which is also defined in the same table.

In the DCD Table 5.4.1-1, the pump flow rate of 112,000 gpm and the corresponding requirement for the developed head are listed as a set of RCP design parameters to ensure TDF at 10% tube plugging. MMF is evaluated from the RCS resistance and the Q-H characteristic curve of RCP, which satisfies the above design parameter set, considering design uncertainties and measurement errors.

Thus, the flow rate conditions in LCO 3.4.1 and DCD Table 5.4.1-1 are chosen for their purposes individually, and there is no discrepancy between them.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-69

TS 3.4.3, RCS P/T Limits.

Explain the inclusion of the term "criticality" in the first paragraph of the TS bases B 3.4.3, LCO section.

In the discussion of TS 3.4.3 LCO, the first paragraph states "The two elements of this LCO are: (a) The limit curves for heat up, cool down, and In Service Leak Hydro (ISLH) testing and criticality; and (b) Limits on the rate of change of temperature." The operational limits for criticality is covered in Safety Limit 2.1, TS 3.4.1 and TS 3.4.2. which are more restrictive than TS 3.4.3. Moreover, jusify applying TS 3.4.3 to criticality. Provide a Surveillance Requirement for verifications against these limits.

This information is needed to ensure supporting information in the TS bases is consistent with TS requirements.

ANSWER:

The term "criticality" is included in the first paragraph of the TS BASES 3.4.3 LCO as the core critical operating condition is listed as a P/T limit requirement in Table-1 of 10 CFR 50 Appendix G. In addition, NUREG-1431 Vol.1 TS 5.6.6 paragraph a. also specifies P/T limits for criticality.

Concerning possible conflict with other operating limits for criticality, this is described in the second paragraph of TS BASES 3.4.3 LCO.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-70

TS 3.4.3, RCS P/T Limits.

Clarify the specific methodology for determining the P/T limits in the US-APWR that are discussed in US-APWR TS LCO 3.4.3 BASES, APPLICABLE SAFETY ANALYSES section.

NUREG-1431, Rev. 3.1, TS LCO 3.4.3 BASES identifies a topical report that defines the methodology, but US-APWR TS BASES ommitted the references for the requirements from 10 CFR 50, Appendix G. MHI omitted the quoted Reference 1 in the STS.

NUREG 1431, Rev 3.1, TS LCO 3.4.3 BASES, APPLICABLE SAFETY ANALYSES indicates that the methodology for determining the P/T limits is referenced in WCAP-7924-A, April 1975. If a similar reference is available for the US-APWR, that reference should be identified in the US-APWR TS LCO 3.4.3 BASES.

ANSWER:

The methodology for determining the US-APWR P/T limits that are discussed in US-APWR TS LCO 3.4.3 BASES, APPLICABLE SAFETY ANALYSES section are described in US-APWR DCD Chapter 5 subsection 5.3.2.1.

US-APWR TS LCO 3.4.3 BASES will be revised so that it references the US-APWR DCD Chapter 5 for the methodology to determine the P/T limits.

Impact on DCD

DCD Chapter 16, TS 3.4.3, BASES, Applicable Safety Analyses, first paragraph will be revised as follows:

> The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 7~~1~~ establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

DCD Chapter 16, TS 3.4.3, BASES, References will be revised to add a new reference as follows:

7. Subsection 5.3.2.1

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-71

TS 3.4.6, RCS Loops - MODE 4.

Confirm when RHR pumps provide circulation of RCS flow through the core, that only two RHR loops are required to be operable and one RHR loop to be in operation to satisfy LCO 3.4.6 requirements regarding decay heat removal. Revise LCO 3.4.6 and related information in the TS bases B 3.4.6, as appropriate.

The APWR LCO 3.4.6 text repeats the STS LCO 3.4.6 text. The APWR design, however, includes four 50% RHR trains for decay heat removal functions while the Westinghouse design in the STS reflects two 100% RHR trains. Consideration of single failure criteria is required when establishing LCO requirements.

This information is needed to ensure LCO requirements are consistent with RHR system design described in the APWR FSAR.

ANSWER:

As the US-APWR design includes four 50% RHR trains for decay heat removal functions, LCO 3.4.6 and associated Bases will be revised.

Impact on DCD

DCD Chapter 16, TS 3.4.6 and BASES will be revised as follows:

LCO

LCO 3.4.6 ~~Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.~~

Two RCS loops shall be OPERABLE and one RCS loop shall be in operation

OR

Three Residual Heat Removal (RHR) loops shall be OPERABLE and two RHR loops shall be in operation

ACTIONS, A.2 Required Action NOTE

Only required if two RHR loops are ~~is~~ OPERABLE

ACTIONS, B, Condition
Two or more required loops inoperable
OR
Required loop(s) not in operation.

Surveillance Requirements
SR 3.4.6.1 Verify required RHR or RCS loops are ~~is~~ in operation.

BASES
Background, last paragraph
In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or two ~~one~~ RHR loops for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that additional ~~two~~ paths be available to provide redundancy for decay heat removal.

LCO, first paragraph
The purpose of this LCO is to require that at least two RCS loops or three RHR loops are ~~be~~ OPERABLE in MODE 4 and that one of the RCS loops or two of the RHR loops are ~~these loops be~~ in operation. ~~The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops.~~ Any one RCS loop or two RHR loops in operation provide~~s~~ enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Applicability, first paragraph
In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of ~~either~~ RCS or two loops of RHR provides sufficient circulation for these purposes. However, additional loops consisting ~~two loops consisting of any combination~~ of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Actions, A.2
If restoration is not accomplished and two ~~an~~ RHR loops are ~~is~~ OPERABLE, the unit must be brought to MODE 5 within 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only two ~~one~~ RHR loops OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 rather than MODE 4. The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if two ~~a~~ RHR loops are ~~is~~ OPERABLE. With no RHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on the restoration of a RHR loop, rather than a cooldown of extended duration.

Actions, B.1 and B.2
If two or more required loops are inoperable or a required loop(s) are ~~is~~ not in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until the required loop(s) are ~~one loop is~~ restored to OPERABLE status and

operation.

Surveillance Requirements, SR 3.4.6.1, first sentence
This SR requires verification that the required RCS or RHR loops are ~~is~~ in operation.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-72

TS 3.4.6, RCS Loops - MODE 4.

Provide the methodology for determining the SG secondary side water level "13%" limit required in SR 3.4.6.2and the location of this limit in the FSAR.

The information will be used to ensure that all TS specific operating parameters are verified as correct based upon values stated in the FSAR.

ANSWER:

To maintain the heat sink necessary for removal of decay heat in MODE 4, the steam generator secondary side water level is required to be above the low steam generator water level trip setpoint of 13%. The low steam generator water level trip setpoint is 13% of span as shown in Table 7.2-3 (Sheet 2) of the US-APWR DCD. The low steam generator water level trip protects the reactor from the loss of its heat sink. This reactor trip is assumed in the following events in Chapter 15 of the US-APWR DCD:

15.2.7 Loss of Normal Feedwater Flow
15.2.8 Feedwater System Pipe Break Inside and Outside Containment

The reactor trip analytical limit for the steam generator water level is assumed to be 0% for these events as shown in Table 15.0-4 in Chapter 15 of the DCD.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-73

TS 3.4.9, Pressurizer.

Confirm that the US-APWR pressurizer heaters are permanently powered by Class 1E power supplies in US-APWR TS LCO 3.4.9.

NUREG-1431, Rev 3.1, Surveillance Requirements section notes that SR 3.4.9.3 is not applicable if the pressurizer heaters are permanently powered by Class 1E power supplies. SR 3.4.9.3 is not included in the US-APWR TS, and no substantiating material appears in the US-APWR BACKGROUND section, nor is any FSAR chapter other than Chapter 15 referenced.

ANSWER:

As shown in DCD Figure 8.3.1-1, pressurizer heaters (back-up group A, B, C and D), which are used to maintain natural circulation in hot standby condition in conformance with the requirement of 10CFR50.34 (f) (2)(xiii), are respectively powered via their own train of Class 1E 480V load centers of Class 1E ac power distribution system. It is not necessary to switch from Non-Class 1E power distribution system. MHI believes that SR 3.4.9.3 of NUREG-1431 is not applicable due to this power distribution arrangement for pressurizer heaters.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-74

TS 3.4.10, Pressurizer Safety Valves.

Revise SR 3.4.10.1 and related information in TS bases B 3.4.10 to reconcile the lift setpoint requirements.

LCO 3.4.10 specifies the allowable range for OPERABILITY of the Pressurizer Safety Valves to be from 2435 psig to 2485 psig (2460 psig +/- 1%). SR 3.4.10.1 requires verification that the lift setting to be within +/- 1%. The basis for SR 3.4.10.1, however, states that "the pressurizer safety valve setpoint is +/- 3% for OPERABILITY, and the valves are reset to +/- 1% during the Surveillance to allow for drift." Also, it should be noted that the +/- 1% tolerance is based on ASME Code, Section III, NB 7500 requirements which state, in part, "the set pressure tolerance plus or minus shall not exceed the following: 2 psi (15 kPa) for pressures up to and including 70 psi (480 kPa), 3% for pressures from 70 psi (480 kPa) to 300 psi (2 MPa), 10 psi (70 kPa) for pressures over 300 psi (2 MPa) to 1,000 psi (7 MPa), and 1% for pressures over 1,000 psi (7 MPa). The set pressure tolerance shall apply unless a greater tolerance is established as permissible in the Overpressure Protection Report (NB-7200)."

ANSWER:

The Bases for TS SR 3.4.10.1 will be revised to indicate +/- 1% OPERABILITY range for the pressurizer safety valve lift settings, to be consistent with SR 3.4.10.1.

Reference 1 in the Bases for TS 3.4.10 will be revised to indicate "NB 7500" of the ASME Code.

Impact on DCD

DCD Chapter 16, the Bases for TS 3.4.10 will be revised as follows:

SR 3.4.10.1
The pressurizer safety valve setpoint is ± ~~3~~1% for OPERABILITY; ~~however,~~and the valves are reset to remain within ± 1% during the Surveillance to allow for drift.

REFERENCES
1. ASME, Boiler and Pressure Vessel Code, Section III, NB 7500.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-75

TS 3.4.10, Pressurizer Safety Valves.

Clarify the location of the overpressure protection analysis for the US-APWR.

NUREG-1431, Rev 3.1, TS LCO 3.4.10 BASES, APPLICABLE SAFETY ANALYSES section refers to the overpressure protection analysis in a separate topical report (WCAP-7769, Rev 1, June 1972) as a basis for the operation of three pressurizer safety valves. US-APWR TS LCO 3.4.10 BASES has no such reference to a separate overpressure protection analysis of the four pressurizer safety valves included in the design.

US-APWR Chapters 5 and 15 are cited in lieu of any separate analyses. Identify the specific Chapter locations that provide the appropriate analysis.

ANSWER:

As stated in US-APWR DCD Chapter 16 B 3.4.10, the accidents and safety analyses that require pressurizer safety valve actuation and could result in overpressurization if not properly terminated are as follows:

a. Loss of external electrical load,
b. Loss of normal feedwater flow,
c. Reactor coolant pump shaft break,
d. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low-power startup condition, and
e. Spectrum of rod cluster control assembly ejection accidents

These accidents are classified as expected system pressure transient conditions as defined in ASME NB-7000.

US-APWR DCD Section 5.2.2.1.1 describes the pressurizer safety valve sizing methodology based on an analysis of a complete loss of steam flow to the turbine with the reactor operating at 102% of the design nuclear steam supply system thermal power. In the analysis, the reactor is maintained at full power by not taking credit for the first reactor trip signal and conservatively ignoring the second reactor trip signal during the short duration of the transient. The detailed analysis conditions are further described in the response to Question No. 05.02.02-2 in MHI letter UAP-HF-08303, "MHI's Responses to US-APWR RAI No.103-1448 Revision 0," dated December

25, 2008. This analysis condition is classified as an unexpected system pressure transient condition as defined in ASME NB-7000.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-76

TS.3.4.11 - SDVs and Block Valves.

Justify not including verification the proposed SR that the SDVs and Block Valves are capable being powered from emergency power supplies.

This RAI is needed to confirm that SDV is powered from a safety-related AC power source, thus capable being powered from the emergency power supplies.

ANSWER:

As mentioned below, safety depressurization valves and block valves are powered from the Class 1E power distribution system and it is not necessary to switch from the Non-Class 1E power distribution system. MHI believes that SR 3.4.11.4 of NUREG-1431 is not applicable due to this power distribution arrangement for each valve.

RCS-MOV-117A and 117B, safety depressurization valves, are respectively powered from Class 1E 480V motor control center A1 and D1. As shown in DCD Figure 8.3.1-1, Class 1E 480V motor control centers A1 and D1 are respectively connected to Class 1E 480V load centers A1 and D1. The Class 1E 480V load center A1 is normally connected to train A of Class 1E ac power distribution system. Similarly, Class 1E 480V load center D1 is normally connected to train D of the Class 1E ac power distribution system.

RCS-MOV-116A and 116B, block valves, are respectively powered from train B and C of Class 1E dc power distribution system via MOV inverter B and C. As shown in DCD Figure 8.3.2-1, 480V AC input power to each battery charger in Class 1E dc power distribution system is respectively supplied from Class 1E 480V motor control center B and C.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 146-1804 REVISION 0**

SRP SECTION: **16 - TECHNICAL SPECIFICATIONS**

APPLICATION SECTION: **TS SECTION 3.4**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO.: 16-77

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Discuss the NOTE regarding the applicability of LCO 3.0.4.b when entering MODE 4 in the ACTIONS Table for US-APWR TS LCO 3.4.12.

NUREG-1431 TS LCO 3.4.12 BASES, ACTIONS Section explains the NOTE accompanying the TS Action Table. The comparable section of the US-APWR TS BASES does not contain a similar explanation of the TS Action Table NOTE.

The US-APWR TS LCO 3.4.12 BASES eliminates an explanation of a NOTE contained in the STS (NUREG-1431) that appears to be applicable, without providing an alternate explanation.

ANSWER:

The US-APWR TS LCO 3.4.12 BASES will be revised to include an explanation of a NOTE in ACTIONS, which is same as STS description.

Impact on DCD

DCD TS 3.4.12 BASES, ACTIONS will be revised as follows:

ACTIONS (Inserted before first paragraph)
A NOTE prohibits the application of LCO 3.0.4.b to an inoperable LTOP System. There is an increased risk associated with entering MODE 4 from MODE 5 with LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 146-1804 REVISION 0**

SRP SECTION: **16 - TECHNICAL SPECIFICATIONS**

APPLICATION SECTION: **TS SECTION 3.4**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO.: 16-78

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Provide the US-APWR accumulator pressure that cannot exceed the LTOP limits if the accumulators are fully injected when the RCS is above the LTOP arming temperature specified in the PTLR.

NUREG-1431 identifies this pressure in the BASES, ACTIONS Section, C.1, D.1, and D.2. The comparable section of the US-APWR TS BASES does not specify a pressure, but leaves a blank space where the pressure should be inserted.

ANSWER:

The accumulator pressure is lacking in the US-APWR TS BASES, ACTIONS, C.1, D.1 and D.2 section. TS BASES, ACTIONS, C.1, D.1 and D.2 section will be revised to include the accumulator pressure.

Impact on DCD

The second paragraph in TS BASES, ACTIONS, C.1, D.1 and D.2 section will be revised as follows:

> If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to >LTOP arming temperature specified in the PTLR, an accumulator pressure of <u>695 psig</u> cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-79

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Justify required Actions E.1 and E.2 and their assigned Completion Times. Revise TS Bases B 3.4.12, as appropriate.

TS 3.4.12 Condition E is for one (out of two) RHR Suction relief valve inoperable. The specified Actions and Completion times (12 hours) are different from a comparable Condition in the STS (7 days). Single failure criteria is clearly addressed in the STS TS bases. No equivalent discussion is provided in the APWR TS bases.

This information is needed to ensure adequacy of specified TS requirements and completeness of supporting information in the TS bases.

ANSWER:

The RHR Suction relief valves are considered passive components since these valves are a spring-loaded type. Therefore, there is no need to consider single active component failure. Since Actions E.1 and E.2 are for the actions when the LTOP system lost its capability for LTOP, Completion time of E.1 and E.2 is selected to 12 hours. In STS, the Completion time when all LTOP system function is lost is 12 hours (Completion time of Action G). Therefore, the Completion time of E.1 and E.2 is based on the same intent of STS.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-80

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Describe the analysis performed to support the required vent size of 2.6 sq. inch. Provide the valve in the reactor coolant system that is equivalent to this vent size and will be used by plant procedure to achieve system depressurization.

This RAI is needed to confirm the design information in the APWR FSAR to support TS requirements.

ANSWER:

The vent size is determined based on pressure drop calculation. When an overpressure event occurs, RCS water is discharged from the RCS vent. In this condition, RCS pressure depends on the pressure drop of the vent portion. This vent size enables to relieve maximum mass input (actuation of safety injection pumps) and the RCS pressure within the PTLR limit. Vent size will be revised to be 4.7 sq. inch since 2.6 sq. inch is incorrect. The pressure drop of the vent portion, which size is 4.7 sq. inch, is approx. 540 psi with assuming maximum mass input. This means the RCS pressure becomes approx. 540 psig, and so it is below the LTOP limit (See DCD Figs. 5.3-2 and 5.3-3.)

In actual plant operation, a pressurizer safety valve (6B) will be removed or pressurizer manway will be opened to achieve system depressurization for LTOP.

Impact on DCD

TS 3.4.12 LCO will be revised as follows:

b. The RCS depressurized and an RCS vent of = 4.7~~2.6~~ square inches.

TS 3.4.12 ACTIONS, Required Action E.2 will be revised as follows:

E.2 Depressurize RCS and establish RCS vent of = 4.7~~2.6~~ square inches

TS 3.4.12 SURVEILLANCE REQUIREMENTS, SR 3.4.12.5 will be revised as follows:

SR 3.4.12.5 Verify required RCS vent = 4.7~~2.6~~ square inches open.

TS 3.4.12 BASES, Applicable Safety Analyses, RCS Vent Performance, first sentence will be revised as follows:

With the RCS depressurized, analyses show a vent size of 4.7~~2.6~~ square inches is capable of mitigating the allowed LTOP overpressure transient.

TS 3.4.12 BASES, LCO, 4th paragraph, b. will be revised as follows:

b. A depressurized RCS and an RCS vent.
An RCS vent is OPERABLE when open with an area of = 4.7~~2.6~~ square inches.

TS 3.4.12 BASES, SURVEILLANCE REQUIREMENTS, SR 3.4.12.5, 1st sentence will be revised as follows:

The RCS vent of = 4.7~~2.6~~ square inches is proven OPERABLE by verifying its open condition [either:

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-81

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Confirm that the safety analyses exist to demonstrate that the US-APWR reactor vessel is adequately protected from exceeding 10CFR50, Appendix G P/T limits.

NUREG-1431 TS LCO 3.4.12 BASES, APPLICABLE SAFETY ANALYSES Section references FSAR, Chapter 15 as supporting analyses. The corresponding section of the US-APWR TS BASES does not supply a supporting reference.

The US-APWR TS BASES state that the reactor vessel is adequately protected against exceeding the 10CFR50, Appendix G P/T limits. Provide supporting documentation for this statement.

ANSWER:

The summary of analysis for LTOP is shown in RAI #103 responses, UAP-HF-08303, dated December 25, 2008. This summary shows that the reactor vessel is adequately protected against exceeding the 10 CFR 50, Appendix G P/T limits. Please confirm this analysis result.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-82

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Clarify the temperature band of concern for RCS overpressurization as a result of accumulator discharge during US-APWR low temperature plant conditions.

NUREG-1431 TS LCO 3.4.12 BASES, APPLICABLE SAFETY ANALYSES Section identifies a narrower range of temperature concern for the effects of an accumulator discharge than the LCO. The comparable section of the US-APWR TS BASES does not discuss this temperature band.

NUREG-1431 BASES identifies a band of [175]°F and below as the temperature band of concern for an accumulator discharge while the TS LCO identifies a band of [275]°F and below. The US-APWR TS LCO 3.4.12 BASES provides no discussion regarding a narrower temperature band of concern for accumulator discharge or any supporting analyses. The LTOP arming temperature for the US-APWR is specified in the PTLR. However, this does not preclude an amplified discussion of the accumulator discharge.

ANSWER:

Description of the temperature band of concern for RCS overpressurization as a result of accumulator discharge during US-APWR low temperature plant condition will be added in TS BASES as shown below.

Impact on DCD

DCD Chapter 16, TS BASES Applicable safety analyses will be revised to add the following paragraph after 7th paragraph.

<u>The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions. The analyses show the effect of accumulator discharge is over a narrower RCS temperature range (195°F and below) than that of the LCO (LTOP arming temperature and below).</u>

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-83

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Identify the analyses used to establish the temperature for US-APWR LTOP applicability.

NUREG-1431 TS LCO 3.4.12 BASES, APPLICABLE SAFETY ANALYSES Section states that fracture mechanics analyses are used to establish the temperature of LTOP applicability. The comparable section of the US-APWR TS BASES does not identify any similar analyses.

The US-APWR TS LCO 3.4.12 BASES eliminates a statement contained in the STS (NUREG-1431) that appears to be applicable, without providing an alternate explanation.

ANSWER:

Fracture mechanics analyses are used to establish the temperature of LTOP Applicability in US-APWR. Therefore, the US-APWR TS BASES will include the statement such as STS.

Impact on DCD

DCD Chapter 16, TS BASES Applicable safety analyses will be revised to add the following paragraph before RHR Suction Relief Valve Performance section.

> Fracture mechanics analyses established the temperature of LTOP Applicability at LTOP arming temperature specified in the PTLR.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-84

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Resolve the following inconsistency in the discussion of RHR Suction Relief Valves in the TS Bases B 3.4.12, Applicable Safety Analyses section.

The last sentence of the first paragraph states "overpressure prevention is provided by two RHR suction relief valves." However, in the eighth paragraph one RHR suction relief valve is said to maintain RCS pressure to within the valve rated lift setpoint.

ANSWER:

The statement in the eighth paragraph of TS BASES B 3.4.12, Applicable Safety Analyses section will be revised to be consistent with other statements.

Impact on DCD

The last sentence of the eighth paragraph of TS BASES B 3.4.12, Applicable Safety Analyses section will be revised as follows:

> Assuming all relief flow requirements during the limiting LTOP event, ~~an~~ RHR suction relief valves will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation = 10% of the rated lift setpoint.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-85

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Confirm that the US-APWR RHR suction relief valves are considered active components.

NUREG-1431 TS LCO 3.4.12 BASES, APPLICABLE SAFETY ANALYSES Section states that the RHR suction relief valves are considered active components and constitute the worst case single active failure. The comparable section of the US-APWR TS BASES does not contain a similar statement.

The US-APWR TS LCO 3.4.12 BASES eliminates a statement contained in the STS (NUREG-1431) that appears to be applicable, without providing an alternate explanation.

ANSWER:

The RHR suction relief valves are considered passive components since these valves are simple spring-loaded type as described in DCD Subsection 5.2.2 and 5.4.7. Therefore, they do not constitute the worst case single active failure.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-86

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Discuss the Required Action B.1 in the TS bases B 3.4.12.

The APWR TS bases B 3.4.12 discussion of Actions A.1 and B.1 addresses only SI pumps in Condition A but not the charging pump in Condition B.

This information is needed to ensure supporting information in the TS bases is complete.

ANSWER:

DCD TS bases B 3.4.12 will be revised to add the description of the charging pumps in Action A.1 and B.1.

Impact on DCD

DCD Chapter 16, TS BASES Actions will be revised as follows:

A.1 and B.1
With three or more safety injection pumps, or two or more charging pumps capable of injecting into the RCS, RCS overpressurization is possible.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-87

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Explain the safety injection pumps being rendered incapable of injecting into the RCS, including alternate methods of LTOP control for SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3.

NUREG-1431 TS LCO 3.4.12 BASES, SURVIELLANCE REQUIREMENTS Section discussion of SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 provides an explanation of the actual means of rendering the required components incapable of injecting into the RCS. The comparable section of the US-APWR TS BASES does not contain a similar discussion.

The US-APWR TS LCO 3.4.12 BASES eliminates SR discussion contained in the STS (NUREG-1431) that appears to be applicable, without providing an alternate explanation.

ANSWER:

TS 3.4.12 BASES, Surveillance Requirements section will be revised to add the description for the actual means of rendering the required components incapable of injecting into the RCS.

Impact on DCD

The following paragraph will be added after first paragraph in TS 3.4.12 BASES, Surveillance Requirements section. SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3.

> The safety injection pumps and charging pump are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in pull to lock and at least one valve in the discharge flow path being closed.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-88

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Clarify the reference to the ASME, Boiler and Pressure Vessel Code, Section XI (Reference 4) cited twice in the US-APWR TS BASES, APPLICABLE SAFETY ANALYSES Section.

The identical language cited in NUREG-1431 actually refers to the FSAR, Chapter 15 analyses. There is no Chapter 15 reference listed in the US-APWR TS BASES, REFERENCE Section. The reference referred to twice for the US-APWR in the TS BASES, APPLICABLE SAFETY ANALYSES Section as Reference 4 is incorrect.

NOTE: In addition to the above mixed up in Reference 4 in the bases, the ASME Code Section XI has been replaced by the ASME Code for Operation and Maintenance (OM) for Nuclear Power Plants for IST requirements.

ANSWER:

The reference to the ASME, Boiler and Pressure Vessel Code, Section XI will be revised to be the correct reference. The reference to the ASME Code Section XI will be revised to ASME OM Code.

Impact on DCD

First sentence of DCD TS 3.4.12 BASES, Applicable Safety Analyses section will be revised as follows:

> Safety analyses (Ref. 4) demonstrate that t~~T~~he reactor vessel is adequately protected against exceeding the Reference 1 P/T limits.

The last sentence of DCD TS 3.4.12 BASES, Surveillance Requirements section, SR 3.4.12.6 will be revised as follows:

> The ASME Code (Ref. 5~~4~~), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

References of DCD TS 3.4.12 BASES will be revised as follows:

4. Subsection 5.2.2 ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~
5. ASME, Code for Operation and Maintenance of Nuclear Power Plants.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-89

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System.

Demonstrate that the US-APWR conforms to 10 CFR 50.46 and 10 CFR 50, Appendix K regarding the consequences of a small break loss of coolant accident (LOCA) by limiting the number of OPERABLE safety injection (SI) pumps and charging pumps when SI actuation is enabled.

NUREG-1431 TS LCO 3.4.12 BASES, APPLICABLE SAFETY ANALYSES Section states that the consequences of a small break LOCA conforms to the above requirements by limiting the number of OPERABLE high-pressure injection pumps and charging pumps when SI actuation is enabled. The comparable section of the US-APWR TS BASES does not contain a similar statement.

The US-APWR TS LCO 3.4.12 BASES eliminates a statement contained in the STS (NUREG-1431) that appears to be applicable, without providing an alternate explanation.

ANSWER:

The description regarding a small break loss of coolant accident (LOCA) in LTOP MODE 4 will be added in TS Bases B 3.4.12, APPLICABLE SAFETY ANALYSES.

Impact on DCD

Editorial: After the second paragraph on page B 3.4.12-4 of the DCD Revision 1, the following paragraph will be added.

> The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6), requirements by having a maximum of two SI pumps and one charging pump OPERABLE and SI actuation enabled.

Editorial: On page B 3.4.12-9 of the DCD Revision 1, the following two references will be added.

> 5. 10 CFR 50, Section 50.46.
> 6: 10 CFR 50, Appendix K.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-90

LCO 3.4.13, RCS Operational LEAKAGE.

Clarify the differences in the US-APWR TS LCO 3.4.13 BASES, APPLICABLE SAFETY ANALYSES Section from the STS (NUREG-1431) regarding operational primary to secondary leakage.

NUREG-1431 LCO 3.4.13 BASES, APPLICABLE SAFETY ANALYSES Section assumes that operational primary to secondary leakage from all steam generators is 1 gallon per minute (1,440 gallons per day) or increases to 1 gallon per minute as an initial accident condition that ultimately results in steam discharge to the atmosphere. The comparable BASES Section in the US-APWR TS assumes that operational primary to secondary leakage from all steam generators is 600 gallons per day, which is less conservative than the STS.

The STS LCO 3.4.13 statement is identical to the US-APWR LCO 3.4.13 statement, but the discussion of operational primary to secondary leakage is substantially different. The US-APWR discussion also makes the statement that leakage through any one steam generator that is limited to less than or equal to 150 gallons per day is equivalent to the conditions assumed in the safety analysis. However, in NUREG-1431 this condition is described as significantly less than the conditions assumed in the safety analysis.

NUREG-1431 also contains information regarding RCS operational leakage associated with the steam line break accident and steam generator tube rupture as described in the FSAR, Chapter 15. The US-APWR TS has no such description.

ANSWER:

The US-APWR limits the operational primary to secondary leakage to 150 gpd per steam generator in TS LCO 3.4.13. This same value of operational primary to secondary leakage is used for the safety analysis. Although STS limits the operational primary to secondary leakage to 150 gpd per steam generator as an LCO, STS states that the safety analysis assumes 1 gpm per steam generator as a conservative value. In comparison to these assumptions in the STS, the US-APWR assumption is less conservative. However, it is understood that no safety problems are anticipated since the operational primary to secondary leakage used in the safety analyses is equal to the limit specified in this TS.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-91

LCO 3.4.13, RCS Operational LEAKAGE.

Indicate in the US-APWR TS LCO 3.4.13 BASES where FSAR, Chapter 15 is referenced.

FSAR, Chapter 15 is identified as Reference 3, but this reference is not identified in the body of the BASES text.

ANSWER:

The reference to Chapter 15 in TS LCO 3.4.13 BASES was omitted. TS LOC 3.4.13 BASES will be revised to clearly include the reference to Chapter 15 in the APPLICABLE SAFETY ANALYSIS section.

Impact on DCD

TS LCO 3.4.13 BASES will be revised as indicated below:

APPLICABLE SAFETY ANALYSES	Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is 600 gallons per day. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is equivalent to the conditions assumed in the safety analysis (Ref. 3). The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-92

TS 3.4.14, RCS PIV Leakage.

Justify not including Condition C for the inoperability of the RHR suction valve interlock.

The omission of Condition C appears to be an editorial error. A discussion of Required Action C.1 and its associated Completion Time of 4 hours is provided in the TS Bases B 3.4.14. In addition, SR 3.4.14.2 is assigned to verify the operability of this interlock.

ANSWER:

TS 3.4.14 ACTIONS will be revised to add Condition C.

Impact on DCD

DCD Chapter 16, TS 3.4.14 ACTIONS will be revised to add Condition C as follows:

CONDITION	REQUIED ACTION	COMPLETION TIME
C. RHR suction valve interlock function inoperable	C.1 Isolate the affected penetration by use of one closed manual or deactivated automatic valve	4 hours

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: NO. 146-1804 REVISION 0

SRP SECTION: 16 - TECHNICAL SPECIFICATIONS

APPLICATION SECTION: TS SECTION 3.4

DATE OF RAI ISSUE: 1/9/2009

QUESTION NO.: 16-93

TS 3.4.14, RCS PIV Leakage.

Clarify the statement in the TS Bases B 3.4.14, Surveillance Requirements, SR 3.4.14.2 second paragraph "these SRs are modified by Notes allowing the RHR autoclosure function to be disabled."

This statement is a repeat of a statement in the STS Bases 3.4.14 for the discussion of SR 3.4.14.3 which verify the RHR autoclosure function in the Westinghouse design. STS SR 3.4.14.3 is not included in the APWR GTS and the autoclosure function is not described in APWR FSAR Section 5.4.7.

This is needed to ensure consistent information are provided in the TS Bases and the FSAR.

ANSWER:

The autoclosure interlock for RHR suction valves is not installed in the US-APWR design. This is based on B 10 of BTP 5-2. Therefore, since the statement is not needed for US-APWR TS Bases, this statement will be deleted.

Impact on DCD

DCD Chapter 16, TS Bases B 3.4.14 Surveillance Requirements, SR 3.4.14.2 second paragraph will be deleted as follows:

> ~~These SRs are modified by Notes allowing the RHR autoclosure function to be disabled when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.~~

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-94

TS 3.4.14, RCS PIV Leakage.

Identify the failure consequences that could be associated with overpressure of the low pressure piping or components.

NUREG-1431 LCO 3.4.14 BASES, BACKGROUND Section indicates that the failure consequences could be a loss of coolant accident (LOCA) outside containment degrading the ability for low pressure injection. The comparable section of the USAPWR TS BASES does not contain a similar discussion.

The US-APWR design may preclude a LOCA outside containment. However, the USAPWR TS LCO 3.4.14 BASES eliminates discussion regarding the ability for low pressure injection contained in the STS (NUREG-1431) that appears to be applicable, without providing an alternate explanation.

ANSWER:

The description regarding the failure consequences will be added in TS Bases B 3.4.14, BACKGROUND.

Impact on DCD

DCD Chapter 16, TS Bases B 3.4.14 BACKGROUND, the following sentence will be added in the third paragraph.

> <u>Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for containment spray.</u>

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-95

TS 3.4.14, RCS PIV Leakage.

Justify that the PIV leakage at 0.5 gpm per nominal inch of valve size is acceptable. NUREG-1431 LCO 3.4.14 BASES, LCO Section describes the reasoning behind establishing the PIV leakage based on valve size. The comparable section of the USAPWR TS BASES does not contain a similar discussion.

The US-APWR TS LCO 3.4.14 BASES eliminates discussion regarding the PIV leakage limit contained in the STS (NUREG-1431) that appears to be applicable, without providing an alternate explanation.

ANSWER:

Same description as STS for the reasoning behind establishing the PIV leakage based on valve size will be added.

Impact on DCD

DCD Chapter 16, TS Bases B 3.4.14 LCO, the second paragraph will be revised as follows:

> The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. <u>The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.</u>

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-96

TS 3.4.14, RCS PIV Leakage.

Clarify the discussion regarding the 900 psig design of the low pressure portion of the RHR system preventing any overpressurization failure of the RHR low pressure line, thereby preventing an intersystem LOCA.

NUREG-1431 TS LCO 3.4.14 BASES, APPLICABLE SAFETY ANALYSES Section allows for the possibility of an overpressurization failure, an intersystem LOCA, and subsequent risk for core melt

The US-APWR operates above 900 psig. The BASES discussion should be more specific as to the prevention of an overpressurization failure of the low pressure portion of the RHR system. The statement in the US-APWR BASES is not supported.

ANSWER:

The description regarding the 900 psig design of the low pressure portion of the RHR system will be added in TS Bases B 3.4.14.

Impact on DCD

DCD Chapter 16, TS Bases B 3.4.14 APPLICABLE SAFETY ANALYSES, the last sentence in first paragraph will be revised as follows:

> Because the low pressure portion of the RHR System is designed for 900 psig, <u>and 900 psig design is able to bear the RCS pressure without pipe rupture</u>, overpressurization failure of the RHR low pressure line is prevented, thus preventing a LOCA outside containment and subsequent risk of core melt.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-97

TS 3.4.16, RCS Specific Activity.

Justify the the APWR GTS, Section 3.4.16 not fully implementing TSTF-490, Revision 1; or revise Section 3.4.16 to fully reflect proper implementation of TSTF-490, if determined to be applicable.

Discussions in the TS bases indicate that TSTF-490 is incorporated into APWR GTS, but it does not seem to be fully implemented.This additional information will be used to ensure that the applicable LCO correctly considered TSTF-490, as appropriate.

ANSWER:

Section 3.4.16 will be revised in order to fully implement TSTF-490, appropriately.

Impact on DCD

TS Section 3.4.16 will be revised as indicated below.

LCO 3.4.16 ~~The specific activity of the reactor coolant shall be within limits.~~ <u>RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits</u>

APPLICABILITY: MODES 1, ~~and~~ 2, 3 and 4.

~~MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit. ~~> 1.0 μCi/gm.~~	--------------------NOTE------------------- LCO 3.0.4.c is applicable. ---	
	A.1 Verify DOSE EQUIVALENT I-131 $\leq$ 60 μCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit. ~~> 300 μCi/gm.~~	--------------------NOTE------------------- LCO 3.0.4.c is applicable. ---	
	B.1 Restore DOSE EQUIVALENT Xe-133 to within limit. ~~Be in MODE 3 with T_{avg} < 500°F.~~	48~~6~~ hours
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 > 60 μCi/gm.	C.1 Be in MODE 3. ~~with T_{avg} < 500°F.~~ AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	------------------------------NOTE------------------------------ Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity = 300 μCi/gm.	[7 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.4.16.2	------------------------------NOTE------------------------------ Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity = 1.0 μCi/gm.	[14 days OR In accordance with the Surveillance Frequency Control Program] AND Between 2 and 6 hours after a THERMAL POWER change of = 15% RTP within a 1 hour period

The BASES of TS Section 3.4.16 will be revised as indicated below.

BASES

BACKGROUND	The maximum ~~total effective~~ dose ~~equivalent~~ that an individual at the ~~site~~ exclusion area boundary can receive for 2 hours ~~during~~ following an accident, or at low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50. 34 (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are ~~held to a small fraction of the 10 CFR 50.34~~ appropriately limited ~~limits~~ during analyzed transients and accidents. The RCS specific activity LCO limits the allowable concentration level of ~~iodines and noble gases~~ radionuclides in the reactor coolant. The LCO limits are established to minimize the ~~offsite radioactivity~~ dose consequences in the event of a steam system piping failure or a steam generator tube rupture (SGTR) accident. The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133~~-specific activity~~. ~~The LCO limits are established by assuming 1 % failed fuel.~~ The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 2).
APPLICABLE SAFETY ANALYSES	The LCO limits on the specific activity of the reactor coolant ensures that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a steam system piping failure or SGTR accident. ~~2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 50.34 dose guideline limits following a SGTR accident.~~ The ~~SGTR~~ safety analyses ~~analysis~~ (Refs. 3 and 4) assumes the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 600 gpd exists. The safety analyses ~~analysis~~ assumes the specific activity of the secondary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.14, "Secondary Specific Activity." The analysis for the steam system piping failure and SGTR accidents establishes the acceptance limits for RCS specific activity. Reference to these ~~this~~ analyses ~~analysis~~ is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits. The safety analyses consider ~~analysis is for~~ two cases of reactor coolant iodine specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately ~~I-131 activity in the reactor coolant~~ after a steam system piping failure (by a factor of 500), or SGTR (by a factor of 335), respectively. ~~the accident.~~ The second case assumes the initial reactor coolant iodine activity at 60 μCi/gm DOSE EQUIVALENT I-131 due to an ~~a pre-accident~~ iodine spike caused by a reactor or an RCS transient prior to the accident.

~~an RCS transient.~~ In both cases, the noble gas specific activity ~~in the reactor coolant~~ is assumed to be 300 μCi/gm DOSE EQUIVALENT XE-133. ~~These limits are established by assuming 1% failed fuel.~~

The SGTR analysis also assumes a loss of offsite power at the same time as the reactor trip. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal. If the reactor trip system has not automatically tripped the reactor, operators are assumed to manually trip the reactor.

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the main steam relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the Residual Heat Removal (RHR) system is placed in service.

The steam system piping failure radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 60.0 μCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

The specific iodine activity in the reactor coolant is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 300 μCi/gm DOSE EQUIVALENT XE-133 ~~specific activity in the reactor coolant is limited to 300 μCi/gm~~. These limits on specific activity ensure that offsite and control room ~~the~~ doses ~~to an individual at the site boundary during a Design Basis Accident (DBA)~~ will meet the appropriate SRP acceptance criteria (Ref. 2) ~~be a small fraction of the limits specified in 10 CFR 50.34~~.

The steam system piping failure and SGTR accident analyses ~~analysis~~ (Ref. 3 and 4) shows that the calculated ~~offsite~~ doses ~~levels~~ are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a steam system piping failure or ~~an~~ SGTR, lead to ~~site boundary~~ doses that exceed the SRP acceptance criteria (Ref. 2) ~~10 CFR 50.34 dose guideline limits~~.

APPLICABILITY

In MODES 1, ~~and~~ 2, 3, and 4 ~~in MODE 3 with RCS average~~

~~temperature = 500°F,~~ operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is ~~specific activity are~~ necessary to limit ~~contain~~ the potential consequences of a steam system piping failure or ~~an~~ SGTR to within the SRP acceptance criteria (Ref. 2) ~~acceptable site boundary dose values~~.

In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required. ~~For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.~~

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity ~~DOSE EQUIVALENT I-131~~ is = 60 µCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is ~~done to~~ continued every 4 hours to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a steam system piping failure or SGTR occurring during this time period. ~~based on a reasonable time for normal iodine spikes to decay back to within the LCO limit. If the concentration cannot be restored to within the LCO limit within 48 hours, then the LCO violation did not result from normal iodine spiking.~~

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), ~~while~~ relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. ~~the ACTIONS~~. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

B.1

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a steam system piping failure or SGTR occurring during this time period. ~~-specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.~~

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Actions B.1 while the DOSE EQUIVALENT Xe-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

~~The change within 6 hours to MODE 3 and RCS average temperature ≤ 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.~~

C.1 and C.2

If the ~~a~~ Required Action and the associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > 60 μCi/gm, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. ~~with RCS average temperature ≤ 500°F within 6 hours.~~ The allowed Completion Time are reasonable ~~of 6 hours is reasonable~~, based on operating experience, to reach the required plant conditions ~~MODE 3 below 500°F~~ from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. ~~a quantitative measure of radionuclides with half lives longer than 15 minutes.,~~ This Surveillance provides an indication of any increase in the ~~release of~~ noble gas specific activity. ~~from fuel clad defects.~~

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. ~~The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F.~~ [The 7 day Frequency considers the low probability ~~unlikelihood~~ of a gross fuel failure during this ~~the~~ time. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT Xe-133 is not detected, it should be assumed to be present at the minimum detectable activity.

A Note modifies the SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allow the Surveillance to be performed in those MODES, prior to entering MODE 1.

SR 3.4.16.2

This Surveillance is performed ~~in MODE 1 only~~ to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. ~~increased releases of iodine from fuel defects are more apt to occur.~~ [The 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.] The Frequency, between 2 and 6 hours after a power change = 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; ~~fuel failure;~~ samples at other times would provide inaccurate results.

The Notes modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

REFERENCES

1. 10 CFR 50.34.
2. Standard Review Plan (SRP) Section 15.0.3 "Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors."
3~~2~~. Subsection ~~Chapter~~ 15.1.5.
4. Subsection 15.6.3.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-98

TS 3.4.16, RCS Specific Activity.

List a reference to 10 CFR 100.11 within the US-APWR TS LCO 3.4.16 BASES. The references cited in US-APWR TS LCO BASES are not consistent with the references cited in LCO BASES for LCO 3.4.16 in NUREG 1431.

10 CFR 50.34(b)(1) provides a reference to part 100 within the discussion of the FSAR content. However, a direct reference to the part 100, "Reactor Site Criteria," is appropriate for this LCO.

ANSWER:

10 CFR 100 states that the criteria for radiological dose consequences of postulated accidents for applications after April 1997 are found in 10 CFR 50.34(a)(1). SRP 15.0.3 also specifies that the criteria for off-site dose are found in 10 CFR 50.34(a)(1). Therefore, the reference in this section is made to 10 CFR 50.34 rather than 10 CFR 100.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/4/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 146-1804 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**TS SECTION 3.4**
DATE OF RAI ISSUE:	**1/9/2009**

QUESTION NO.: 16-99

TS 3.4.4, RCS Loops - MODES 1 and 2.

Justify not including the Westinghouse STS 3.4.19 which establishes exceptions to STS 3.4.4 requirements during Physics Testing at lower power below 25%.

In APWR Technical Report MUAP-07039, "Justifications for Deviations Between NUREG-1431 Rev. 3.1 and US-APWR Technical Specifications," MHI states "Natural Circulation Test is required at low power. This test is necessary for first plant of USAPWR. However, the Generic FSAR doesn't include this requirement," and further indicates that the natural circulation test is described in FSAR section 14.2.12.2.3.9.

Requirements of STS 3.4.19 should be provided in the APWR GTS, and thus in the PTS, to allow for exceptions to TS 3.4.4 during the performance of the natural circulation test eventhough it is only needed for the first APWR plant.

ANSWER:

MHI agrees with adding STS 3.4.19. STS 3.4.19 will be added as US-APWR TS 3.4.18 RCS Loops - Test Exceptions. Also, Technical Report MUAP-07039 will be revised reflecting this DCD modification.

Impact on DCD

DCD Chapter 16 TS 3.4.18 RCS Loops – Test Exceptions and related Bases will be added, See Attachment 1.

Impact on COLA

There are impacts on the COLA to incorporate the DCD change.

Impact on PRA

There is no impact on the PRA.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 RCS Loops – Test Exceptions

LCO 3.4.18 The requirements of LCO 3.4.4, "RCS Loops - MODES 1 and 2," may be suspended with THERMAL POWER < P-7.

APPLICABILITY: MODES 1 and 2 during startup and PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ P-7.	A.1 Open reactor trip breakers.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.18.1	Verify THERMAL POWER is < P-7.	1 hour
SR 3.4.18.2	Perform a COT for each power range neutron flux - low channel, intermediate range neutron flux channel, P-10, and P-13.	Prior to initiation of startup and PHYSICS TESTS
SR 3.4.18.3	Perform an ACTUATION LOGIC TEST on P-7.	Prior to initiation of startup and PHYSICS TESTS

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 RCS Loops - Test Exceptions

BASES

BACKGROUND

The primary purpose of this test exception is to provide an exception to LCO 3.4.4, "RCS Loops - MODES 1 and 2," to permit reactor criticality under no flow conditions during certain PHYSICS TESTS (natural circulation demonstration, station blackout, and loss of offsite power) to be performed while at low THERMAL POWER levels. Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the power plant as specified in GDC 1, "Quality Standards and Records" (Ref. 2).

The key objectives of a test program are to provide assurance that the facility has been adequately designed to validate the analytical models used in the design and analysis, to verify the assumptions used to predict plant response, to provide assurance that installation of equipment at the unit has been accomplished in accordance with the design, and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, and following low power operations.

The tests will include verifying the ability to establish and maintain natural circulation following a plant trip at low power, performing decay heat removal via natural circulation, and during the natural circulation condition, showing that pressure can be controlled using auxiliary spray and pressurizer heaters powered from the emergency power sources.

APPLICABLE SAFETY ANALYSES

The tests described above require operating the plant without forced convection flow and as such are not bounded by any safety analyses. However, operating experience has demonstrated this exception to be safe under the present applicability.

As describe in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

BASES

LCO	This LCO provides an exemption to the requirements of LCO 3.4.4. The LCO is provided to allow for the performance of PHYSICS TESTS in MODE 2 (after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without the LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODES 1 and 2), and the appropriate tests could not be performed. In MODE 2, where core power level is considerably lower and the associated PHYSICS TESTS must be performed, operation is allowed under no flow conditions provided THERMAL POWER is ≤ P-7 and the reactor trip setpoints of the OPERABLE power level channels are set ≤ 25% RTP. This ensures, if some problem caused the plant to enter MODE 1 and start increasing plant power, the Reactor Trip System (RTS) would automatically shut it down before power became too high, and thereby prevent violation of fuel design limits. The exemption is allowed even though there are no bounding safety analyses. However, these tests are performed under close supervision during the test program and provide valuable information on the plant's capability to cool down without offsite power available to the reactor coolant pumps.
APPLICABILITY	This LCO is applicable when performing low power PHYSICS TESTS without any forced convection flow. This testing is performed to establish that heat input from nuclear heat does not exceed the natural circulation heat removal capabilities. Therefore, no safety or fuel design limits will be violated as a result of the associated tests.
ACTIONS	<u>A.1</u> When THERMAL POWER is ≥ the P-7 interlock setpoint 10%, the only acceptable action is to ensure the reactor trip breakers (RTBs) are opened immediately in accordance with Required Action A.1 to prevent operation of the fuel beyond its design limits. Opening the RTBs will shut down the reactor and prevent operation of the fuel outside of its design limits.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.18.1

Verification that the power level is ≤ the P-7 interlock setpoint (10%) will ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The Frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Plant operations are conducted slowly during the performance of PHYSICS TESTS and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.

SR 3.4.18.2

The power range and intermediate range neutron detectors. P-10, and the P-13 interlock setpoint must be verified to be OPERABLE and adjusted to the proper value. The Low Power Reactor Trips Block, P-7 interlock, is actuated from either the Power Range Neutron Flux, P-10, or the Turbine Inlet Pressure, P-13 interlock. The P-7 interlock is a logic Function with train, not channel identity. A COT is performed prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The SR 3.3.1.7 Frequency is sufficient for the power range and intermediate range neutron detectors to ensure that the instrumentation is OPERABLE before initiating PHYSICS TESTS, because the RTS is self-tested on a continuous basis from digital side of all input modules to the digital side of all output modules.

SR 3.4.18.3

The Low Power Reactor Trips Block, P-7 interlock, must be verified to be OPERABLE in MODE 1 by LCO 3.3.1, "Reactor Trip System Instrumentation." The P-7 interlock is actuated from either the Power Range Neutron Flux, P-10, or the Turbine Inlet Pressure, P-13 interlock. The P-7 interlock is a logic Function. An ACTUATION LOGIC TEST is performed to verify OPERABILITY of the P-7 interlock prior to initiation of startup and PHYSICS TESTS. This will ensure that the RTS is properly functioning to provide the required degree of core protection during the performance of the PHYSICS TESTS.

BASES

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50, Appendix A, GDC 1, 1988.