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TOKYO, JAPAN

February 20, 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-09058

Subject: MHI's Responses to US-APWR DCD RAI No.158-1814 Revision 0

Reference: 1) "REQUEST FOR ADDITIONAL INFORMATION NO. 158-1814 REVISION 0, SRP Section: 16 - Technical Specifications Application Section: 16, QUESTIONS for Technical Specification Branch (CTSB)" dated January 21, 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.158-1814 Revision 0."

Enclosed is the responses to Questions 16-100 through 16-114 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.158 Revision 0

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

D081
NRO

Contact Information

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Enclosure 1

UAP-HF-09058
Docket No. 52-021

Responses to Request for Additional Information No.158-1814
Revision 0

February 2009

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/20/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 158-1814 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**16**
DATE OF RAI ISSUE:	**1/21/2009**

QUESTION NO.: 16-100

Justify the setting of 2735 psig as the maximum RCS pressure allowed.

Ch. 5 of the FSAR contains Table 5.2.2-1 which states the design pressure of the RCS system is 2485 psig. The setting of 2485 psig yields a 110% value (maximum allowed for RCS pressure vessel by ASME Code, Section III) of 2733.5 psig. Therefore a less than or equal to 2735 psig would exceed 110% design value if that setting were reached.

This justification is needed to preclude the plant from exceeding ASME Code.

ANSWER:

The setting of 2735 psig as the maximum RCS pressure allowed will be revised to 2733.5 psig in accordance with ASME Code, Section III.

Impact on DCD

The description relating to the RCS pressure SL in TS will be revised as follows:

TS 2.1.2, Reactor Coolant System (RCS) Pressure SL on page 2.0-1
In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained = ~~2735~~2733.5 psig.

TS B 2.1.2, SAFETY LIMITS, the second sentence on page B 2.1.2-2
Therefore, the SL on maximum allowable RCS pressure is ~~2735~~2733.5 psig.

TS B 3.3.1, BACKGROUND on page B 3.3.1-3
3. The RCS pressure SL of ~~2735~~2733.5 psig shall not be exceeded.

TS B 3.4.10, BACKGROUND, the last sentence of the first paragraph on page B 3.4.10-1
The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), ~~2735~~2733.5 psig, which is 110% of the design pressure.

Impact on COLA

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

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 16-101

Justify the peak fuel centerline temperature of 5072 degrees F in Safety Limit (SL) 2.1.1.2. As stated in paragraph 4.2.1.2.1 of the US-APWR FSAR, the melting temperature for fuel not containing gadolinia is 5072 F and 4892 F for fuel which does contain gadolinia.

Per Table 4.2-1 of the FSAR, the US-APWR core can contain as much as 10% gadolinia in the fuel. In order to address both types of fuel, both melting temperatures should be evaluated to incorporate the most limiting.

The reduction in melting temperature based on burnup (58 degrees F per 10,000 MWD/MTU) was consistent with both the FSAR and STS used (NUREG 1431 Rev. 3.1) and is acceptable.

ANSWER:

As discussed in the DCD Subsection 4.4.1.2, the evaluation shows that the temperature of gadolinia fuel rods is not critical because the peak linear heat rate for gadolinia fuel rods is decreased by reduced UO_2 enrichment content.
Therefore, the SL 2.1.1.2 provides the fuel centerline temperature limits only for UO_2 fuel, which must be maintained in MODE 1 and 2 operations and represents a necessary and sufficient design requirement for establishing the Reactor Protection System (RPS) trip set points.

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.

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QUESTION NO.: 16-102

Clarify and revise the correlation used in defining Safety Limit (SL) 2.1.1.1. SL 2.1.1.1 only lists the WRB-2 DNB correlation, while the FSAR Ch. 4 also discusses the use of W-3 DNB correlation.

The pressure range for the WRB-2 DNB correlation is 1440-2490 psia per Ch. 4 of the FSAR. Therefore, the data obtained via this correlation is only valid during accident analysis during which the system pressure remains in this range. However, Ch. 4 further discusses that accident analysis during which pressure falls below this applicable range would rely on W-3 DNB correlation, even stating the limits for certain pressure ranges (i.e. 1.45 for 500-1000 psia and 1.30 for 1000-1440 psia)

ANSWER:

As discussed in the APPLICABILITY section of Bases 2.1.1, SL 2.1.1 only applies in MODE 1 and 2 because these are the only MODES in which the reactor is critical.

Pressure range of WRB-2 correlation covers the operation range in MODE 1 and 2, in which the reactor core is protected by the Low Pressurizer Pressure Reactor Trip and High Pressurizer Pressure Reactor Trip. Therefore the discussion for W-3 correlation is not needed here.

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.

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QUESTION NO.: 16-103

ANSWER:
The rod drop time from start the rod drop to the dashpot entry can not be measured. As for the alternative method to measure the rod drop time, the time from the beginning of decay of stationary gripper coil voltage to dashpot entry is applied in the SR 3.1.4.3.
The criteria of the rod drop time from start the rod drop to the dashpot is 3 seconds which is same with states in Subsection 15.0.0.2.5. However, considering measurement on the site, the response of the control rod drive mechanism should be considered which is identified in DCD Subsection 3.9.4.2.1. Therefore allowable response of the control rod drive mechanism, 0.15 seconds, is added on the rod drop time criteria,3 seconds, for the criteria of SR 3.1.4.3.

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.

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QUESTION NO.: 16-104

Justify deviation from American National Standard ANSI/ANS-19.6.1-2005 in regard to test required under the Physics Test program.

Mitsubishi Heavy Industries (MHI) supplied a document listing the justification for deviations between the US-APWR FSAR and the Standard Technical Specifications (STS) used, which in most cases was NUREG 1431 Rev. 3.1. In this document, it was stated that the NUREG 1431 Rev. 1 was used for Physics Test - Mode 1. The NUREG 1431 Rev. 3.1 does not contain a section for Mode 1 Physics Tests, only Mode 2.

NUREG 1431 Rev. 1 states the required tests to be performed on page B3.1-53. The first test listed is the Neutron Flux Symmetry test. The NUREG states that this test can be done in Mode 1 or 2, because the power limit for the test (less than or equal to 30% RTP) can be achieved in either Mode.

ANSI/ANS-19.6.1-2005, referenced in US-APWR section B3.1.8, contains Table 1 on page 3 which states the Flux symmetry test as one of the required tests to be performed during Physics Testing, and describes the test in detail on page 7, including the power level limit of less than or equal to 30% RTP.

The US-APWR FSAR does not contain the Flux Symmetry test in either the Physics Test - Mode 1 (B3.1.8), page B3.1.8-2 of DCD, or Physics Test - Mode 2 (B3.1.9), page B3.1.9-2 of FSAR. The omission of this test from the Physics Test program would not allow the program to be carried out in accordance with the American National Standard.

This justification is required to ensure that the Physics Testing is carried out in accordance with the American National Standard.

ANSWER:

The US-APWR DCD Chapter 16, TS 3.1.8 BASES includes the test "The Power Distribution - Intermediate Power Test" as described on page B 3.1.8-2. The test is performed at least once before achieving 30% RTP.
This test covers the requirements for "The Neutron Flux Symmetry test". Therefore, the Physics

Test program of the US-APWR satisfies the requirement of ANSI/ANS-19.6.1-2005.

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.

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QUESTION NO.: 16-105

Justify deviating from American National Standard ANSI-ANS-19.6.1-2005 regarding a physics test to be performed in Mode 1.

The US-APWR FSAR on page B3.1.8-2 states that the "Critical Boron Concentration-Full Power" is to be performed.

The Standard ANSI-ANS-19.6.1-2005 lists all tests to be performed in Table 1 on page 3. The final test is listed as "HZP to HFP reactivity difference", and is discussed in detail on page 7. Previous revisions of ANSI-ANS-19.6.1 regard the test as "Critical Boron Concentration-Full Power".

The 2005 Standard is the reference cited in the US-APWR FSAR, and this revision of the Standard incorporates the new name for the test.

This information is needed to ensure testing is completed in accordance with the most recent applicable American National Standard.

ANSWER:

The description "Critical Boron Concentration-Full Power" on page B 3.1.8-2 will be replaced with "HZP to HFP reactivity difference" in accordance with ANSI-ANS-19.6.1-2005.

Impact on DCD

The DCD Chapter 16, TS 3.1.8 BASES, the third paragraph on page B 3.1.8-2 will be revised as follows:

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 1 are listed below:

a. Power Distribution – Intermediate Power,
b. Power Distribution – Full Power, and
c. HZP to HFP reactivity difference. ~~Critical Boron Concentration – Full Power.~~

Impact on COLA

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

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 16-106

Justify deviation from American National Standard ANSI-ANS-19.6.1-2005 regarding the conduct of the Power Distribution - Intermediate Power Test.

The US-APWR FSAR on page B3.1.8-2 states that the test will be conducted "at least one time by 30% RTP."

ANSI-ANS-19.6.1-2005, which is referenced in the US-APWR FSAR, states in paragraph 6.6.3 on page 7 that for this test: "reactor power shall be at a stable power level between 40 and 80% of full power."

This information is required to ensure that the Physics Testing will be conducted in accordance with the American National Standard.

ANSWER:

MHI will revise BASES 3.1.8 of US-APWR DCD to conduct "the Power Distribution - Intermediate Power Test" also at a stable power level between 40% and 80% of RTP in accordance with ANSI-ANS-19.6.1-2005.

Impact on DCD

The DCD Chapter 16, TS 3.1.8 BASES, the last paragraph on page B 3.1.8-2 will be revised as follows.

a. The Power Distribution – Intermediate Power Test measures the power distribution of the reactor core at intermediate power levels at least one time by 30% RTP <u>and between 40% and 80% RTP</u>. This test uses the incore flux detectors to measure core power distribution.

Impact on COLA

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

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 16-107

ANSWER:

MHI agrees with this comment. The description of the reduction of the required channels in LCO 3.1.9 should have been applied only for functions which require all 4 channels during Modes 1 and 2. The required channels for Function 6 in RTS Instrumentation, Overtemperature delta T, of the US-APWR is 3. Therefore MHI will delete Function 6 from LCO 3.1.9.

Impact on DCD

The third paragraph in LCO 3.1.9 of section 3.1.9 on DCD Chapter 16 will be revised as follows:

> may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3~~, 6~~ and 15.c, may be reduced to 3 required channels, provided:

Impact on COLA

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

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 16-108

Justify deviation from American National Standard ANSI-ANS-19.6.1-2005 regarding the physics tests performed. The US-APWR FSAR states on page B3.1.9-2 that one of the Mode 2 tests to be performed is the Critical Boron Concentration - Control Rods Inserted.

The Standard ANSI-ANS-19.6.1-2005 contains Table 1 on page 3 that lists the required physics tests. The Critical Boron Concentration with Control Rods Inserted is not listed on that table.

Previous revisions of the ANSI-ANS-19.6.1 have two Critical Boron Concentration tests, one with control rods withdrawn and one with control rods inserted. The 2005 ANSI document only lists one test of this nature entitled "Critical Boron Concentration". Paragraph 6.2.1 on page 5 of ANSI-ANS-19.6.1-2005 describes the test as being performed with all control rods withdrawn.

This information is required to ensure that the Physics Tests are being performed in accordance with the American National Standard.

ANSWER:

MHI will revise the BASES 3.1.9 of US-APWR DCD in accordance with ANSI-ANS-19.6.1-2005.

Impact on DCD

The DCD Chapter 16, TS 3.1.9 BASES, BACKGROUND on page B 3.1.9-2 to B 3.1.9-3 is revised as follows.

BACKGROUND (continued)

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

a. Critical Boron Concentration - Control Rods Withdrawn,

~~b. Critical Boron Concentration - Control Rods Inserted,~~

b~~c~~. Control Rod Worth, and

c~~d~~. Isothermal Temperature Coefficient (ITC)

These tests are performed in MODE 2. These and other supplementary tests may be required to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

a. The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (keff = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

~~b. The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% Δk/k when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," or LCO 3.1.6, "Control Bank Insertion Limits."~~

b~~c~~. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6.

c~~d~~. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity

change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the...

Impact on COLA

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

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 16-109

The following comments are of the editorial nature:
1. Correct the typographical error discussed below.
On page B2.1.1-2 of the FSAR, the second full paragraph states "...by the limit DNBR values stated in SL2.1.1.1, which are...". There is no space between the 'L' and the '2'.

2. Correct the typographical error discussed below.
On page B 3.1.6-1 of the US-APWR bases, a line in the third paragraph ends with the word 'group' broken up on two lines. The letters 'gr' are ending a line and the letters 'oup' begin the next.

ANSWER:

MHI will revise BASES 2.1.1 and BASES 3.1.6 of the US-APWR DCD to incorporate the comments in QUESTION NO.16-109.

Impact on DCD

1. The DCD Chapter 16, TS 2.1.1 BASES, a part of second paragraph on page B 2.1.1-2 is revised as follows:

> The above criterion a. is represented by the limit DNBR values stated in SL 2.1.1.1~~SL2.1.1.1~~,

2. The DCD Chapter 16, TS 3.1.6 BASES, a part of third paragraph on page B 3.1.6-1 is revised as follows:

> The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two or more groups to provide for precise reactivity control. A group ~~gr oup~~ consists of two or more RCCAs that are electrically paralleled to step simultaneously.

Impact on COLA

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

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 16-110

Clarify the explanation of the transient discussed for maximum RCS pressure setting. STS (NUREG 1431 Rev 3.1) states in B 2.1.2 on page B 2.1.2-2 that during the transient, no control actions are assumed, except the secondary safety valves are assumed to open and nominal feedwater supply is maintained. The same bases for US-APWR omits the final statement about nominal feedwater supply being maintained, which could have significant impact on the transient and/or system response.

ANSWER:

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. Steam relief through the main steam safety valves is considered as described on page B 2.1.2-2. However, nominal feedwater flow is assumed to be lost and therefore the statement about nominal feedwater supply being maintained is omitted from the bases on page B 2.1.2-2 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.

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.

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QUESTION NO.: 16-111

Clarify the explanation of the Safety Limit (SL) reference in the bases for SL 2.1.2.

The STS (NUREG 1431 Rev 3.1) states the bases of the RCS pressure limit is 110% of design pressure due to that number being the most restrictive between the 110% figure (ASME Code for RCS Vessel) and 120% figure (USAS Section B31.1 for RCS piping, valves, and fittings).

The SL is the for the entire RCS system, therefore the deviation from the STS (NUREG 1431) as presented in the US-APWR bases on page B 2.1.2-2 does not fully justify the pressure limit because only the 110% of RCS pressure vessel, and not the 120% for RCS piping, valves, and fittings, is discussed in the US-APWR bases.

ANSWER:

The pressure safety limit for RCS piping, valves, and fittings which comprise RCPB is governed by the ASME Code, Section III. The maximum transient pressure allowed under the ASME Code, Section III, is 100% of design pressure.

Impact on DCD

The DCD Chapter 16, TS B 2.1.2, SAFETY LIMITS, the first sentence is revised as follows:

The maximum transient pressure allowed in the RCS pressure vessel<u>, piping, valves, and fittings</u> under the ASME Code, Section III, is 110% of design pressure.

Impact on COLA

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

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 16-112

Clarify the explanation of plant conditions discussed in bases for 3.1.2.

The third paragraph of the STS (NUREG 1431 Rev 3.1) on page B3.1.2-1 contains the following:" When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by...".

The same line in the US-APWR bases omits the plant condition description of being at RTP and moderator temperature. The omission of these plant conditions impacts the intent of the paragraph by altering the amount of negative reactivity in the core and corresponding compensation.

ANSWER:

MHI will revise the third paragraph of page B 3.1.2-1 so that the description is identical to the STS (NUREG-1431 Rev 3.1).

Impact on DCD

The DCD Chapter 16, TS 3.1.2 BASES, the second paragraph on page B 3.1.2-1 will be revised as follows.

> In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

Impact on COLA

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

Impact on PRA

There is no impact on the PRA.

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DATE OF RAI ISSUE:	**1/21/2009**

QUESTION NO.: 16-113

Justify the omission of the maximum upper limit placed on LCO 3.1.3 as prescribed in STS (NUREG 1431 Rev. 3.1). NUREG 1431 also contains Figure 3.1.3-1 showing MTC vs. % RTP and this Figure was also omitted.

The basis in NUREG 1431 on page B 3.1.3-4 states that the LCO establishes a maximum positive value which cannot be exceeded, and that BOC and EOC limits are established in the COLR to allow specifying limits for each particular cycle. These statements were also omitted from the bases for US-APWR.

While bounding values were used in the accident analysis contained in Ch. 15 of the FSAR, the incompleteness of the specification does not support normal operation, particularly when transitioning from one fuel cycle to the next.

ANSWER:

As described in BACKGROUND in BASES 3.1.3 of US-APWR DCD, both initial and reload cores are designed so that the MTC is less than zero in MODE1 and MODE2 with keff = 1.0. A positive or power-dependent MTC is not assumed in the safety analysis in Chapter 15 of US-APWR DCD. Therefore the related sentences and figure were not described in TS 3.1.3 and BASES 3.1.3 of US-APWR 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

2/20/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 158-1814 REVISION 0**
SRP SECTION:	**16 - TECHNICAL SPECIFICATIONS**
APPLICATION SECTION:	**16**
DATE OF RAI ISSUE:	**1/21/2009**

QUESTION NO.: 16-114

Revise all references of ***10 CFR 50.36(d)(2)(ii)*** to read ***10 CFR 50.36(c)(2)(ii)***
This correction is required following the NRC final ruling in August 2008 to restore original paragraph designation as existed before the 2007 Part 52 rulemaking.

ANSWER:

MHI will revise all references of ***10 CFR 50.36(d)(2)(ii)*** to read ***10 CFR 50.36(c)(2)(ii)*** in DCD Chapter 16.

Impact on DCD

In DCD Chapter 16, all references of ***10 CFR 50.36(d)(2)(ii)*** will be revised to read ***10 CFR 50.36(c)(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.