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TOKYO, JAPAN

August 31, 2011

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-11276

Subject: MHI's Response to US-APWR DCD RAI No. 785-5885 Revision 3 (SRP 15.4.8)

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "MHI's Response to US-APWR DCD RAI No. 785-5585 Revision 3 (SRP 15.4.8)". The enclosed material provides MHI's response to the NRC's "Request for Additional Information (RAI) 785-5585 Revision 3," dated July 26, 2011.

As indicated in the enclosed materials, Enclosure 2 contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted in this package (Enclosure 3). In the non-proprietary version, the proprietary information, bracketed in the proprietary version, is replaced by the designation "[ ]".

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

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

Sincerely,

Y. Ogata

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

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. MHI's Response to US-APWR DCD RAI No. 785-5885 Revision 3 (SRP 15.4.8) (proprietary)
3. MHI's Response to US-APWR DCD RAI No. 785-5885 Revision 3 (SRP 15.4.8) (non-proprietary)

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

<u>Contact Information</u>

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ENCLOSURE 1

Docket No. 52-021

MHI Ref: UAP-HF-11276

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, depose and state as follows:

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

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "MHI's Response to US-APWR DCD RAI No. 785-5885 Revision 3 (SRP 15.4.8)", dated August 31, 2011, and have determined that the document contains proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The basis for holding the referenced information confidential is that it describes the unique design of the safety analysis, developed by MHI (the "MHI Information").

4. The MHI Information is not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it required the performance of research and development and detailed design for its software and hardware extending over several years. Therefore public disclosure of the materials would adversely affect MHI's competitive position.

5. The referenced information has in the past been, and will continue to be, held in confidence by MHI and is always subject to suitable measures to protect it from unauthorized use or disclosure.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information.

7. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of supporting the NRC staff's review of MHI's application for certification of its US-APWR Standard Plant Design.

8. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without the costs or risks associated with the design and testing of new systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI in the U.S. nuclear plant market.

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

Executed on this 31st day of August, 2011.

Y. Ogata

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

ENCLOSURE 3

UAP-HF-11276
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 785-5885 Revision 3
(SRP 15.4.8)

August 2011

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/31/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 785-5885 REVISION 3**
SRP SECTION:	**15.04.08 - SPECTRUM OF ROD EJECTION ACCIDENTS (PWR)**
APPLICATION SECTION:	**15.04.08**
DATE OF RAI ISSUE:	**7/26/2011**

QUESTION NO.: 15.04.08-11

In the DCD rod ejection analysis, the analytical limit for power range neutron flux (high setpoint) is 118%, which includes the nominal setpoint of 109% plus 9% additional uncertainty. As described in MUAP-09022 and RAIs associated with MUAP-07010-P, this 9% bounds the uncertainty in power distribution effects for AOOs, but may not bound the uncertainty for rapid reactivity insertions such as control rod ejection. Justify 9% uncertainty as being appropriate for the rod ejection analysis, or determine what the appropriate uncertainty should be and revise the rod ejection analysis accordingly.

ANSWER:

MHI submitted the Non-LOCA Methodology Topical Report, MUAP-07010 Revision 2 to the NRC via MHI letter UAP-HF-11277 on August 31, 2011. This revision includes an update of the rod ejection analysis methodology. Based on the revised rod ejection analysis methodology, MHI performed a sensitivity analysis of the uncertainty in the power distribution.

Figures 15.4.8-11.1 through 15.4.8-11.6 show the comparison of the sensitivity case and the DCD case. While the current DCD case assumes that the reactor trip occurs when the actual reactor power reaches 118% (setpoint of 109% plus 9%, the sensitivity case assumes that the reactor trip occurs when the 3rd-highest measured ex-core detector power reaches considering a single failure in an ex-core NIS detector. The uncertainty of which excludes the power distribution effect, is calculated as follows. (The methodology is described in Appendix G of MUAP-07010-P Revision 2.)

[ ]

This gives:

[ ]

The results indicate that using measured ex-core detector power for the rod ejection transient has a small impact on the time of reactor trip and therefore a small impact on the cladding and fuel temperatures and fuel enthalpy.

The Non-LOCA Methodology Topical Report, MUAP-07010, Revision 2 incorporates the methodology changes described above as well as changes in the rod ejection number of rods in DNB analysis methodology. Therefore, MHI performed the sensitivity analysis of the number of rods in DNB based on the updated methodology.

Three cases of the DNB evaluation (short-term period, long-term / rapid depressurization and long-term / slow depressurization) are performed based on MUAP-07010 Revision 2. For the short-term period case, the results indicate that DNB remains above the safety analysis limit and therefore there are no rods in DNB, as shown in Table 15.4.8-11.1 and Figure 15.4.8-11.8 for BOC and Figure 15.4.8-11.10 for EOC. For the long-term / rapid depressurization case, the final result is that the maximum number of fuel rods in DNB is [ ] for the most severe case, as shown in Table 15.4.8-11.2. For the long-term / slow depressurization case, the resulting maximum number of fuel rods in DNB is [ ] for the most severe slow depressurization case, as shown in Table 15.4.8-11.3 and Figure 15.4.8-11.11. The detailed methodologies for these evaluations are provided in MUAP-07010 Revision 2.

DCD Section 15.4.8 is revised as described in the "Impact on DCD" section below to incorporate the results of the sensitivity cases for the uncertainty in power distribution effects and rods in DNB cases.

Table 15.4.8-11.1 Analysis Conditions and Results for Short-Term Period Case

Table 15.4.8-11.2 Analysis Condition and Result for Long-Term Period / Rapid Depressurization Case

Table 15.4.8-11.3 Analysis Condition and Result for Long-Term Period / Slow Depressurization Case

Figure 15.4.8-11.1 **Reactor Power versus Time**
Rod Ejection (HFP, BOC)

Figure 15.4.8-11.2 Fuel and Cladding Temperature versus Time Rod Ejection (HFP, BOC)

Figure 15.4.8-11.3 **Radial Average Fuel Enthalpy versus Time Rod Ejection (HFP, BOC)**

Figure 15.4.8-11.4 **Reactor Power versus Time**
Rod Ejection (HFP, EOC)

Figure 15.4.8-11.5 **Fuel and Cladding Temperature versus Time Rod Ejection (HFP, EOC)**

Figure 15.4.8-11.6 **Radial Average Enthalpy versus Time Rod Ejection (HFP, EOC)**

Figure 15.4.8-11.7 Reactor Power – First Cycle, BOC, HFP Case (Short-Term Period)

Figure 15.4.8-11.8 Minimum DNBR – First Cycle, BOC, HFP Case (Short-Term Period)

Figure 15.4.8-11.9 Reactor Power – First Cycle, EOC, HFP Case (Short-Term Period)

Figure 15.4.8-11.10 Minimum DNBR – First Cycle, EOC, HFP Case (Short-Term Period)

Figure 15.4.8-11.11 **Core Census for Rods in DNB for Long Term Period (0% to 5% Range) RCCA Ejection (EOC HFP, No Flux Trip)**

Impact on DCD

DCD Section 15.4.8 is revised as indicated in mark-up in Attachment 1. To maintain consistency with DCD Section 15.4.8, DCD Table 15.0-1 will also be revised as indicated in Attachment 2.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

This completes MHI's response to the NRC's question.

15.4.8 Spectrum of Rod Ejection Accidents

15.4.8.1 Identification of Causes and Frequency Classification

This accident is defined as the mechanical failure of a control rod drive mechanism (CRDM) housing, which results in the ejection of a~~n~~ rod cluster control assembly (RCCA) and its drive shaft. The consequence of this RCCA ejection is a rapid positive reactivity insertion with an increase of core power peaking, possibly leading to localized fuel rod failure. The nuclear excursion is terminated by Doppler reactivity feedback from increased fuel temperature, and the core is shut down by the high power range neutron flux, over temperature ΔT, or low pressurizer pressure reactor trip ~~(high and low setpoint for hot full power (HFP) and hot zero power (HZP), respectively)~~. DCD_15.04.08-11

This event is classified as a postulated accident (PA) as defined in Section 15.0.0.1. Historically, these events have been classified as Condition IV events as defined in ANSI N18.2 (Ref. 15.4-1). Additional event-specific acceptance criteria are described in Section 15.4.8.2.5.

15.4.8.2 Sequence of Events and Systems Operation

This postulated accident is initiated by the failure of the CRDM housing. Sudden ejection of an RCCA adds positive reactivity to a localized region of the core in a very short period of time. This RCCA ejection results in a power excursion in the region near the affected fuel assembly. With the reactivity feedback, the core power eventually reaches an equilibrium state, which is characterized by highly asymmetric power distribution in the radial dimension. This adverse power distribution subsequently leads to overheating of the affected fuel assemblies and possible fuel damage.

The sequence and timing of major events for the spectrum of rod ejection accidents is described in the results section.

The following automatic trip signals are assumed to be available to provide protection from this transient:

- High power range neutron flux (high setpoint)
- High power range neutron flux (low setpoint)
- High power range neutron flux rate
- Over temperature ΔT trip DCD_15.04.08-11
- Low pressurizer pressure trip

In the safety analysis, the high power range neutron flux rate trip is conservatively ignored.

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3.

This event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could, in turn, cause a loss of offsite power, which could, in turn, cause a reactor coolant pump (RCP) coastdown. As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of peak radial average fuel enthalpy, peak fuel temperature and peak reactor coolant pressure so that these maximum values for the entire transient are the same whether offsite power is available or unavailable. Since the two cases have equally limiting peak radial average fuel enthalpy, peak fuel temperature and peak reactor coolant pressure, the case where offsite power is unavailable is not presented.

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

15.4.8.2.1 Nuclear Design

The US-APWR reactivity control functions are provided by two independent mechanisms: adjusting the boron concentration in the reactor coolant system (RCS) (chemical shim) and maneuvering the RCCAs.

Chemical shim is used to compensate slow reactivity changes such ~~u~~as fuel depletion. It also provides sufficient negative reactivity to bring the reactor to cold shutdown. DCD_15.04.08-11

The RCCAs are typically used for rapid reactivity changes, such as changes in power demand or temperature transients. During normal operation, the RCCAs can be inserted up to their insertion limits, as specified in the Technical Specifications. Therefore, the control banks are assumed to be at their respective insertion limits prior to the rod ejection accident. The most limiting ejected rod location is identified for each core condition.

15.4.8.2.2 Mechanical Design

Since rod ejection is potentially a PA, mechanical design and certain quality control programs are implemented to prevent its occurrence:

- The structural reliability of the CDRM housing for the US-APWR is increased by the elimination of the canopy seals.
- All CRDM pressure housings are performed hydrostatic test in accordance with ASME code Section III.
- All CRDM pressure housings are individually hydrotested after they are attached to the reactor vessel head.
- The latch mechanism housing and the rod travel housing are single piece forged stainless steel. This material has demonstrated excellent notch toughness at temperatures anticipated to be encountered during the reactor operating life time.

- Anticipated system transients have little effect on the stress levels in CRDM housings. Moments induced by the design basis earthquake are within the allowable range specified by the ASME code, Section III.

15.4.8.2.3 Reactor Protection

The automatic features of the RTS in an RCCA ejection incident include the high power range neutron flux ~~trip~~ (high and low setpoints), ~~and~~ the high power range neutron flux rate, the over temperature ΔT, and low pressurizer pressure reactor trips. The reactor trip functions are described in Section 7.2. DCD_15.04.08-11

Under the conditions created by the rod ejection accident, the reactor is shut down by ~~the high power range neutron flux trip~~one of the reactor trips listed above. The high power range neutron flux rate trip is conservatively ignored in the safety analyses. DCD_15.04.08-11

15.4.8.2.4 Effects on Neighboring Control Rod Housings

It is assumed that the break of the CRDM housing occurs at a weld. The broken CRDM housing is ejected vertically upward because it is guided by the drive rod, and the driving force from the reactor coolant is vertical. However, the travel of the ejected CRDM housing is limited by the missile shield, which dissipates its kinetic energy. The broken part of the CRDM housing rebounds after impact with the missile shield. However, the broken CRDM contains the drive rod inside, and the top end plates of the rod position indicator coil assemblies prevent it from hitting a second CRDM housing. Even if the rebounding CRDM directly hits an adjacent CRDM housing~~.~~, its kinetic energy would be too low to cause the mechanical failure of a second CRDM housing. Therefore, the adjacent control rod housing failure does not further increase the severity of the accident. DCD_15.04.08-11

15.4.8.2.5 Acceptance Criteria

For the rod ejection accident, the objective is to eliminate or minimize the potential for fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. In an effort to accomplish this goal, this analysis applies the following additional acceptance criteria (beyond those for a typical PA):

- Peak reactor coolant pressure is less than that could cause stresses, which exceed the "Service Limit C" as stipulated by the ASME code (SRP 15.4.8).

- The total number of failed fuel rods that must be considered in the radiological assessment is equal to the sum of all of the fuel rods failing each of the criteria below. The fuel rods that are predicted to fail more than one of the criteria are not double counted (SRP 4.2 Appendix B).

  a. The high cladding temperature failure criterion for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure, or 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure. For full power conditions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g. DNBR).

b. The pellet/cladding mechanical interaction (PCMI) failure criterion is an increase in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure 15.4.8-1.

In addition to the fuel failure and boundary criteria above, the following criteria from SRP 4.2 Appendix B apply to core coolability:

- Peak radial average fuel enthalpy must remain below 230 cal/g.
- Peak fuel temperature must remain below incipient melting conditions.

15.4.8.3 Core and System Performance

15.4.8.3.1 Evaluation Model

The TWINKLE-M code (Ref. 15.4-2) is used to determine the core transient including core average and local power behavior following a RCCA ejection. An increase of local power and the Doppler feedback due to an increase of fuel effective temperature are calculated in each spatial mesh.

The three-dimensional method is applied to the hot zero power (HZP) condition in order to conform to the PCMI fuel failure criteria. The applied core mesh division is 2 x 2 meshes per assembly in the radial direction. For the hot full power (HFP) ~~case~~analysis of peak fuel and cladding temperature and fuel enthalpy, a one-dimensional method is applied and an external reactivity insertion is simulated by changing the eigenvalue of the neutron kinetics. A small Doppler weighting factor is used to compensate for collapsing the 3-D problem into a 1-D axial model. The suitability and conservatism of this approach is confirmed in Appendix C of Reference 15.4-2. The measured reactor power at the ex-core detectors is calculated using three-dimensional detector weighting factors from each assembly. Changes made by MHI to increase the number of meshes and the use of TWINKLE-M for transient calculations are further described in Reference 15.4-2. DCD_15.04.08-11 DCD_15.04.08-11 DCD_15.04.08-11

The VIPRE-01M code (Ref. 15.4-3) calculates fuel temperature, fuel enthalpy, and DNBR at the hot spot during the transient using two interface files created by the TWINKLE-M code. One of the interface files is a time-dependant history of the core average power and the other is a time-dependant history of the hot channel factor. The hot channel factor time history is used for the three-dimensional calculation only. The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation. Additional details regarding the VIPRE-01M methodology are available in Reference 15.4-2.

The static method is applied for the HFP rods-in-DNB calculation. For the case where the reactor power does not reach the high power range neutron flux reactor trip analytical limit, the limiting conditions occur when the reactor does not trip until the over temperature ΔT reactor trip occurs. For this analysis, it is assumed that fuel rods with an $F_{\Delta H}^{N}$ greater than the design limit at the time the over temperature ΔT reactor trip occurs, if any, are failed. The $F_{\Delta H}^{N}$ census for the rods-in DNB calculation is calculated using the ANC code. DCD_15.04.08-11

Additional details on the overall evaluation methodology for the rod ejection accident analysis can be found in the MHI Non-LOCA Methodology topical report (Ref. 15.4-2). Analyses of the spectrum of rod ejection accidents are performed for the following cases:

- Hot full power initial condition at beginning-of-cycle (HFP BOC)
- Hot full power initial conditions at end-of-cycle (HFP EOC)
- Hot zero power initial condition at beginning-of-cycle (HZP BOC)
- Hot zero power initial condition at end-of-cycle (HZP EOC)

15.4.8.3.2 Input Parameters and Initial Conditions

Plant initial conditions are given in Section 15.0.0.2. The following assumptions are utilized in order to calculate conservative transient results for the peak fuel and cladding temperature and fuel enthalpy analyses~~four previously described rod ejection accident cases. Analysis assumptions and calculation conditions for the core kinetics follow Regulatory Guide 1.77 Appendix A (Ref. 15.4-5). Table 15.4.8-2 tabulates the parameters used in the rod ejection analysis~~. DCD_15.04.08-11

- Initial condition assumptions are based on a typical 24 month equilibrium core at the beginning-of-cycle (BOC) and end-of-cycle (EOC) to address refueling cycles of up to 24 months duration.
  - HFP with initial uncertainty for fuel temperature evaluation (102% of the licensed core thermal power level with initial reactor coolant temperature 4°F above the nominal value and the pressurizer pressure 30 psi below the nominal value)~~, and without initial uncertainty for rods in DNB evaluation (consistent with the use of the RTDP)~~. The nominal value of core power, reactor coolant temperature, and RCS pressure conditions are described in Table 15.0-3. DCD_15.04.08-11
  - HZP for fuel enthalpy evaluation (the initial values of reactor coolant average temperature and RCS pressure used in VIPRE-01M are assumed to be 4°F above and 30 psi below the values corresponding to hot standby conditions).
- A conservative large reactivity, chosen at the design limit, is inserted within 0.1 seconds.
  - In the three-dimensional methodology case, the most reactive RCCA ejection is selected. The inserted reactivity is directly simulated by the change of the absorption cross section caused by the ejection of the most reactive RCCA. The deficit of the inserted reactivity compared with the design limit is made up for by changing the eigenvalue of the neutron kinetics.
  - In the one-dimensional methodology case, the reactivity design limit is externally added to the core within 0.1 seconds.

- The Doppler feedback is conservatively estimated by multiplying the fast absorption cross section for the given change in the calculated fuel effective temperature by a conservative multiplier. In the MHI one-dimensional methodology, a small Doppler weighting factor is used to compensate for collapsing the 3-D problem into a 1-D axial model. The suitability and conservatism of this approach is confirmed by a comparison between the three-dimensional and one-dimensional kinetic results presented in Appendix C of Reference 15.4-2. Additional details regarding the Doppler feedback are discussed in Table 15.0-1.

- Moderator reactivity feedback has a relatively minor contribution during the initial phase of the transient. The reason is that the heat transfer between the fuel and moderator takes much longer than the neutron response time. However, after the initial neutron flux peak occurs, the moderator reactivity feedback slows the decrease of neutron power. The moderator reactivity is conservatively estimated by multiplying the moderator slowing down cross section by a conservative multiplier. Additional details regarding moderator reactivity feedback are discussed in Table 15.0-1.

- For the hot spot fuel calculation using the VIPRE-01M code, the film heat transfer coefficient is calculated using the Dittus-Boelter correlation for single phase heat transfer, the Thom correlation for nucleate boiling heat transfer, and the Bishop-Sandberg-Tong correlation for film boiling heat transfer after DNB. Hot spot DNB is conservatively assumed to start at the beginning of the accident. Additional details regarding film heat transfer are available in Reference 15.4-3.

- Conservative assumptions for the trip simulation (trip reactivity, rod drop time, RTS signal processing delays) are used in the analysis. The reactor trip is simulated by dropping partially and fully withdrawn rod banks into the core. Maximum time delay from reactor trip signal to rod motion and a conservative RCCA insertion curve are simulated as described in Table 15.0-4 and Section 15.0.0.2.5, respectively. The trip reactivity used is the design limit, which is -4%Δk/k for the hot full power case and -2%Δk/k for the HZP case, respectively.

- For the HZP cases, ~~T~~the reactor is assumed to be automatically tripped on the high power range neutron flux low setpoint signal. ~~The reactor trips on the high setpoint for the full power cases and the low setpoint in the zero power cases.~~ Table 15.0-4 summarizes the reactor trip analytical limits assumed in the analysis. DCD_15.04.08-11

- For the HFP cases, the reactor trip occurs when the measured neutron flux considering a single failure of one ex-core detector channel reaches the high power range neutron flux high setpoint plus uncertainty. DCD_15.04.08-11

- Minimum delayed neutron fraction and minimum neutron lifetime are used.

- In the case of three-dimensional methodology, a history of hot channel factor is calculated by the TWINKLE-M code. For conservatism, the maximum value of the hot channel factor used in the VIPRE-01M code is adjusted to the design limit.

- In the case of one-dimensional methodology, the hot channel factor used in the VIPRE-01M code is assumed to instantaneously increase to the design limit and is conservatively assumed to remain constant, ignoring feedback effects during the transient.

- Initial conditions of hot spot fuel temperature are consistent with the results of the fuel design code FINE (Ref. 15.4-6). According to the evaluation purpose, the following assumptions are applied conservatively to pellet and cladding gap conductance in the transient analysis using the VIPRE-01M code.

  - Remains constant for fuel temperature and enthalpy analysis
  - Instantaneously decreases to zero for the adiabatic fuel enthalpy analysis
  - Rapidly increases to the maximum value for the cladding temperature analysis
  - Realistically increases for the ~~DNB rods and~~ RCS pressure analysis DCD_15.04.08-11

The following assumptions are utilized in order to calculate conservative steady state results for the rods-in-DNB analysis.

- Initial condition assumptions are based on a typical 24 month equilibrium core at the beginning-of-cycle (BOC) and end-of-cycle (EOC) to address refueling cycles of up to 24 months duration.

  - HFP without initial uncertainty for rods-in-DNB evaluation (consistent with the use of the RTDP). The nominal value of core power, reactor coolant temperature, and RCS pressure conditions are described in Table 15.0-3.

Analysis assumptions and calculation conditions for the core kinetics follow Regulatory Guide 1.77 Appendix A (Ref. 15.4-5). Table 15.4.8-2 tabulates the parameters used in the rod ejection analysis.

15.4.8.3.3 Results

Analyses are performed for RCCA ejection at the BOC and EOC with HFP and HZP. For all cases, the RCCAs are inserted to their insertion limits before the rod ejection occurs. The reactor power, fuel and cladding temperature, and radial average fuel enthalpy transients for the HFP BOC case are presented in Figures 15.4.8-2 through 15.4.8-4. The same transient parameter information for the HFP EOC case is in Figures 15.4.8-5 and 15.4.8-7. The reactor power and fuel enthalpy transients for HZP cases are presented in Figures 15.4.8-8 through 15.4.8-10 for the BOC case and Figures 15.4.8-11 through 15.4.8-13 for the EOC case, respectively. The calculated sequence of events corresponding to these limiting events is provided in Table 15.4.8-1. These analytical results are discussed in the following paragraphs:

- Beginning-of-cycle, full power

  For the HFP BOC case, control bank-D is assumed to be inserted to its insertion limit when the rod ejection occurs. A bounding maximum ejected rod worth of

110 pcm and a design hot channel factor of 5.0 are assumed to provide margin for future cores. The reactivity insertion causes a rapid increase in power, and the power increase is terminated by Doppler feedback. The reactor trip is initiated by high power range neutron flux (high setpoint) and the reactor returns to subcritical following the trip. The peak fuel centerline temperature is ~~4220~~4232°F, which remains below the fuel melting temperature limit. DCD_15.04.08-11

~~The rods-in-DNB analysis confirmed that the number of rods predicted to be in DNB is less than 10% of the core, which is the value used in the radiological consequence analysis.~~ DCD_15.04.08-11

- Beginning-of-cycle, zero power

  For the HZP BOC case, control bank-D is assumed to be fully inserted and the others inserted to their insertion limit when the rod ejection occurs. A bounding maximum ejected rod worth of 600 pcm and a hot channel factor of 14.0 are assumed to provide margin for future cores. The reactivity insertion causes a rapid increase in power, and the power excursion is terminated by Doppler feedback. The reactor trip is initiated by high power range neutron flux (low setpoint) and the reactor returns to subcritical following the trip. The peak fuel enthalpy is 97.5 cal/g (the increase of the peak fuel enthalpy from its initial condition is 49.0 cal/g). The number of PCMI failed fuel rods is zero.

- End-of-cycle, full power

  For the HFP EOC case, control bank-D is assumed to be inserted to its insertion limit when the rod ejection occurs. A bounding maximum ejected rod worth of 120 pcm and a design hot channel factor of 6.0 are assumed to provide margin for future cores. The reactivity insertion causes a rapid increase in power, and the power increase is terminated by Doppler feedback. The reactor trip is initiated by high power range neutron flux (high setpoint) and the reactor returns to subcritical following the trip. The peak fuel centerline temperature is ~~4325~~4343°F, which remains below the fuel melting temperature limit. DCD_15.04.08-11

  ~~The rods-in-DNB analysis confirmed that the number of rods predicted to be in DNB is less than 10% of the core, which is the value used in the radiological consequence analysis.~~ DCD_15.04.08-11

- End-of-cycle, zero power

  For the HZP EOC case, Control Bank-D is assumed to be fully inserted, and the others inserted to their insertion limits when the rod ejection occurs. A bounding maximum ejected rod worth of 800 pcm and a hot channel factor of 35.0 are assumed to provide margin for future cores. The reactivity insertion causes a rapid increase in power, and the power excursion is terminated by Doppler feedback. The reactor trip is initiated by high power range neutron flux (low setpoint) and the reactor returns to subcritical following the trip. The hot spot peak fuel enthalpy is 72.7 cal/g and the prompt fuel enthalpy rise is 50.8 cal/g. The number of PCMI failed fuel rods is zero.

For all four cases analyzed, the average fuel pellet enthalpy at the hot spot remains significantly below 230 cal/g.

In the BOC HZP and EOC HZP cases, the hot spot peak fuel enthalpy is well below the high cladding temperature failure criterion. Therefore, high cladding temperature failure does not occur.

In the BOC HZP and EOC HZP cases, the prompt fuel enthalpy rise is less than 60 cal/g, which is the lowest criterion of the PCMI failure depicted in Figure 15.4.8-1. Additionally, the oxide/wall thickness rate is less than 0.2 (described in Section 4.2.3.3.6). Therefore, the PCMI failure does not occur in either case.

In the BOC HFP and EOC HFP cases, there is no DNB occurrence due to the prompt increase in the reactor power and local power peaking. However, DNBR continues to decrease due to RCS depressurization even after the reactor power and local power peaking reach stable conditions. The reactor is shut down by the low pressurizer pressure trip when the RCS depressurization is rapid or the over temperature ΔT trip when the RCS depressurization is slow. The rods-in-DNB analysis confirmed that the number of rods predicted to be in DNB is less than 10% of the core considering RCS depressurization, which is the value used in the radiological consequence analysis. DCD_15.04.08-11

If a water-logged fuel rod is assumed to exist near the hot spot, this fuel rod may fail at a lower enthalpy rise than the intact fuel rods. However, the probability that a water-logged fuel rod exists, and the probability that such a fuel rod is near the hot spot are both extremely low; thus, the probability of fuel failure in a water-logged fuel rod is negligible.

The rod ejection accident creates an opening in the reactor coolant system. Following the RCCA ejection, the plant response is the same as a small-break loss-of-coolant accident (LOCA). The effects and consequences of a small break LOCA are discussed in Section 15.6.5.

15.4.8.4 Barrier Performance

15.4.8.4.1 Evaluation Model

The evaluation for the peak RCS pressure analysis is similar to the model used for peak fuel and cladding temperature and fuel enthalpy~~hot spot~~ analysis described in Section 15.4.8.3.1. The TWINKLE-M code is used to analyze the core average power histories following a rod ejection accident. The VIPRE-01M code generates a time-dependent core total void fraction and core heat flux interface file which is used by the MARVEL-M code to calculate the RCS pressure transient. Additional details regarding this methodology are provided in Reference 15.4-2. DCD_15.04.08-11

15.4.8.4.2 Input Parameters and Initial Conditions

The barrier performance case for peak RCS pressure is similar to the ~~hot spot~~peak fuel and cladding temperature and fuel enthalpy analysis described in Section 15.4.8.3.2 with the following differences: DCD_15.04.08-11

Table 15.4.8-1
Time Sequence of Events for Rod Ejection

Accident	Event	Time (seconds)	
Case 1: HFP Beginning-of-cycle	Rod ejection occurs	0.0	
	High power range neutron flux (high setpoint) ~~analytical limit~~ reached*1	0.~~07~~10	DCD_15.04.08-11
	Peak reactor power occurs	0.11	
	Reactor trip initiated (rod motion begins)	0.~~67~~70	DCD_15.04.08-11
	Maximum fuel temperature occurs	2.5	
	Maximum fuel enthalpy occurs	2.5	
Case 2: HFP End-of-Cycle	Rod ejection occurs	0.0	
	High power range neutron flux (high setpoint) ~~analytical limit~~ reached*1	0.~~06~~10	DCD_15.04.08-11
	Peak reactor power occurs	0.11	
	Reactor trip initiated (rod motion begins)	0.~~66~~70	DCD_15.04.08-11
	Maximum fuel temperature occurs	2.6	
	Maximum fuel enthalpy occurs	2.5	
Case 3: HZP Beginning-of-cycle	Rod ejection occurs	0.0	
	High power range neutron flux (low setpoint) analytical limit reached	0.24	
	Peak reactor power occurs	0.28	
	Reactor trip initiated (rod motion begins)	0.84	
	Maximum fuel enthalpy occurs	1.8	
Case 4: HZP End-of-Cycle	Rod ejection occurs	0.0	
	High power range neutron flux (low setpoint) analytical limit reached	0.15	
	Peak reactor power occurs	0.16	
	Reactor trip initiated (rod motion begins)	0.75	
	Maximum fuel enthalpy occurs	1.2	

*1 The reactor trip occurs when the measured neutron flux considering a single failure of one ex-core detector channel reaches the high power range neutron flux high setpoint plus uncertainty. DCD_15.04.08-11

Table 15.4.8-2
Parameters Used In Rod Ejection Analysis
(Peak Fuel and Cladding Temperature and Fuel Enthalpy Analysis)

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Parameter	HFP		HZP	
	BOC	EOC	BOC	EOC
RCCA Ejection Time	0.1 sec			
Initial Hot Channel Factor	2.60	2.60	(N/A)	(N/A)
Peak Hot Channel Factor	5	6	14	35
Ejected RCCA Worth	110 pcm	120 pcm	600 pcm	800 pcm
Doppler Weighting Factor	1.31	1.28	(N/A)	(N/A)
Minimum scram reactivity	-4 % Δ k/k		-2 % Δ k/k	
Delayed Neutron Fraction (β_{eff})	0.49 %	0.44 %	0.49 %	0.44 %
Neutron Lifetime	8 μsec	8 μsec	8 μsec	8 μsec

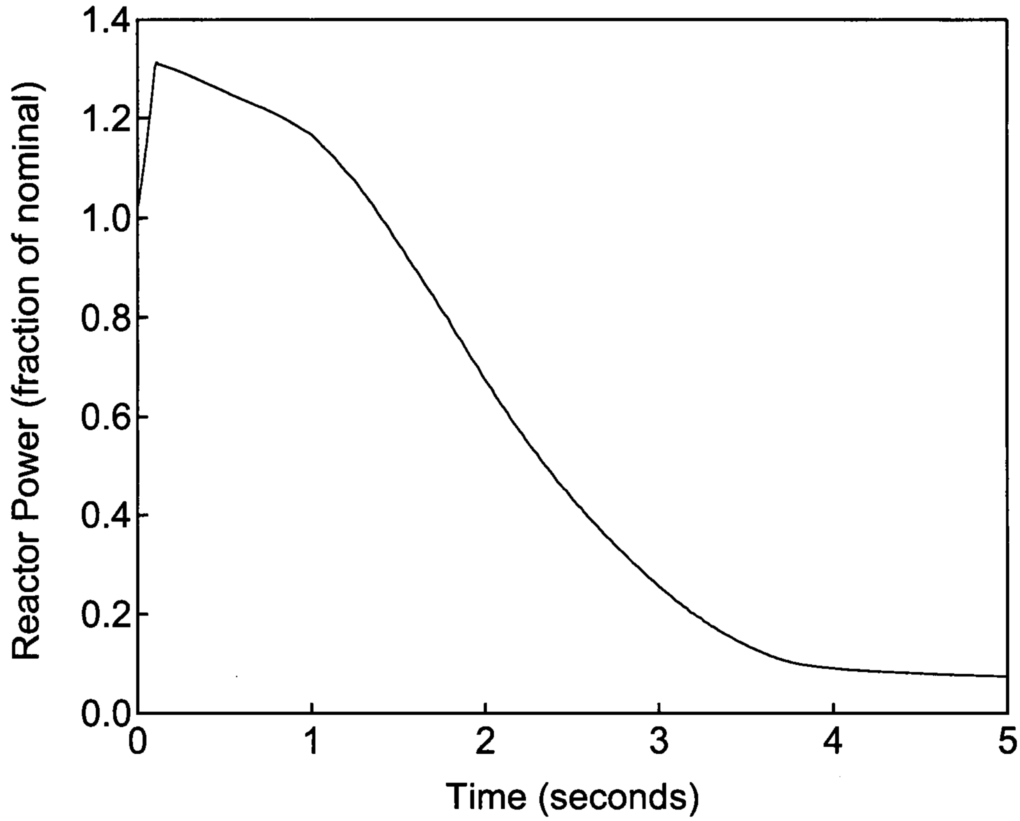

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Figure 15.4.8-2 Reactor Power versus Time

Rod Ejection (HFP, BOC)

DCD_15.04.08-11

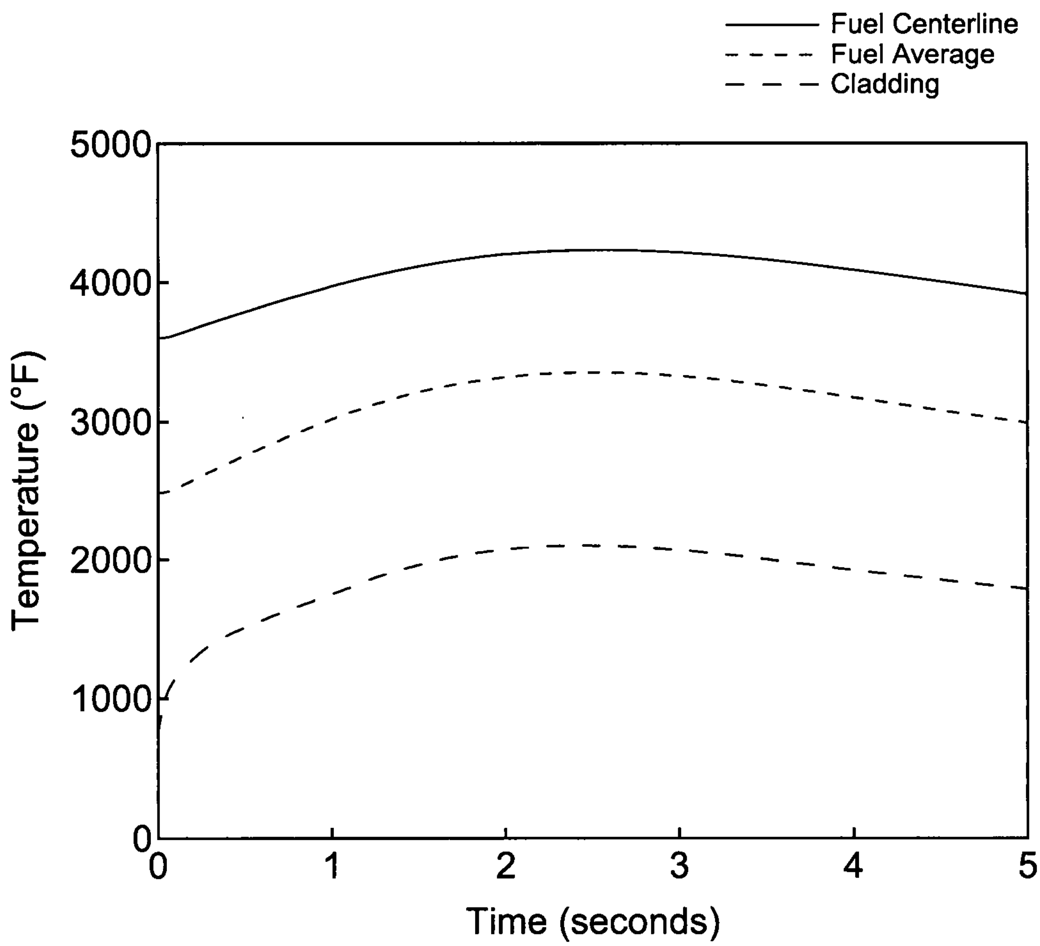

Figure 15.4.8-3 Fuel and Cladding Temperature versus Time

Rod Ejection (HFP, BOC)

DCD_15.04.08-11

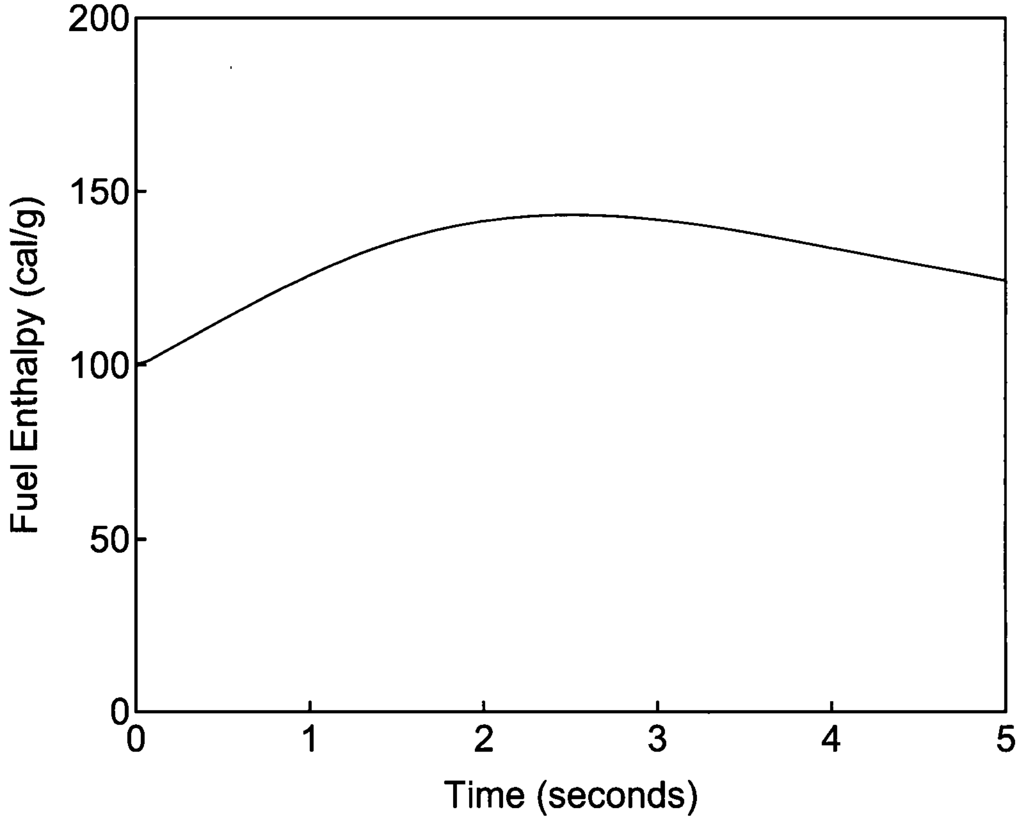

Figure 15.4.8-4 **Radial Average Fuel Enthalpy versus_Time Rod Ejection (HFP, BOC)**

DCD_15.04.08-11

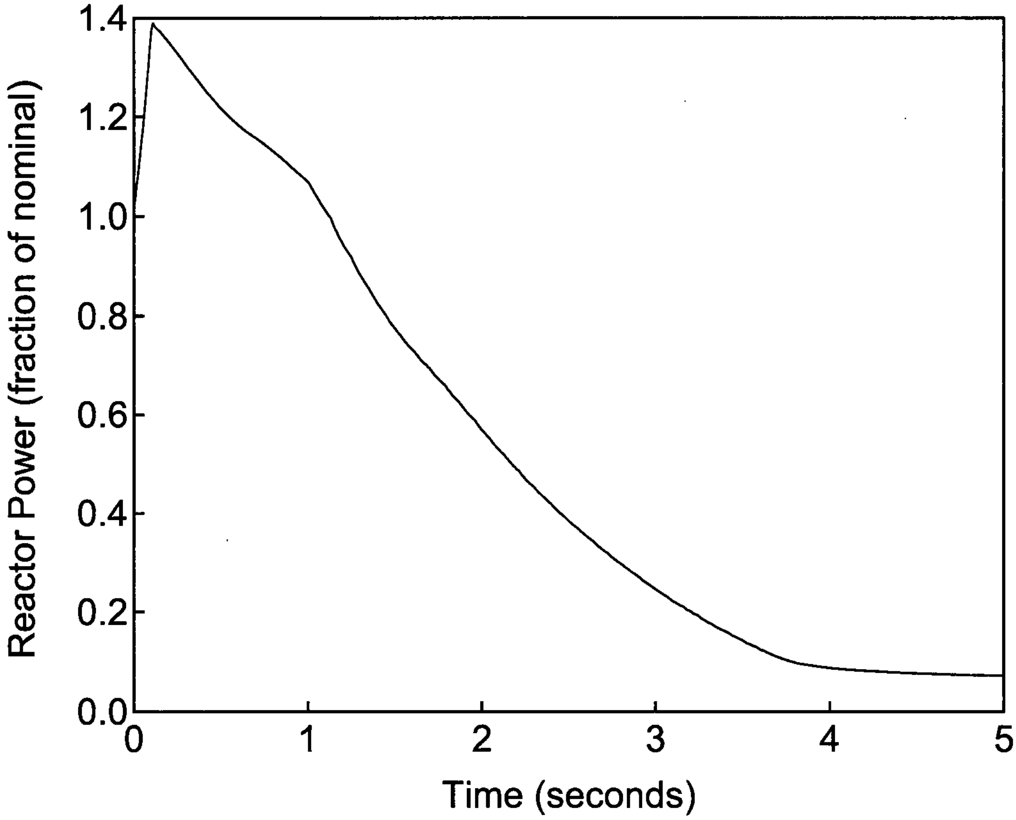

Figure 15.4.8-5 **Reactor Power versus Time**

Rod Ejection (HFP, EOC)

DCD_15.04.08-11

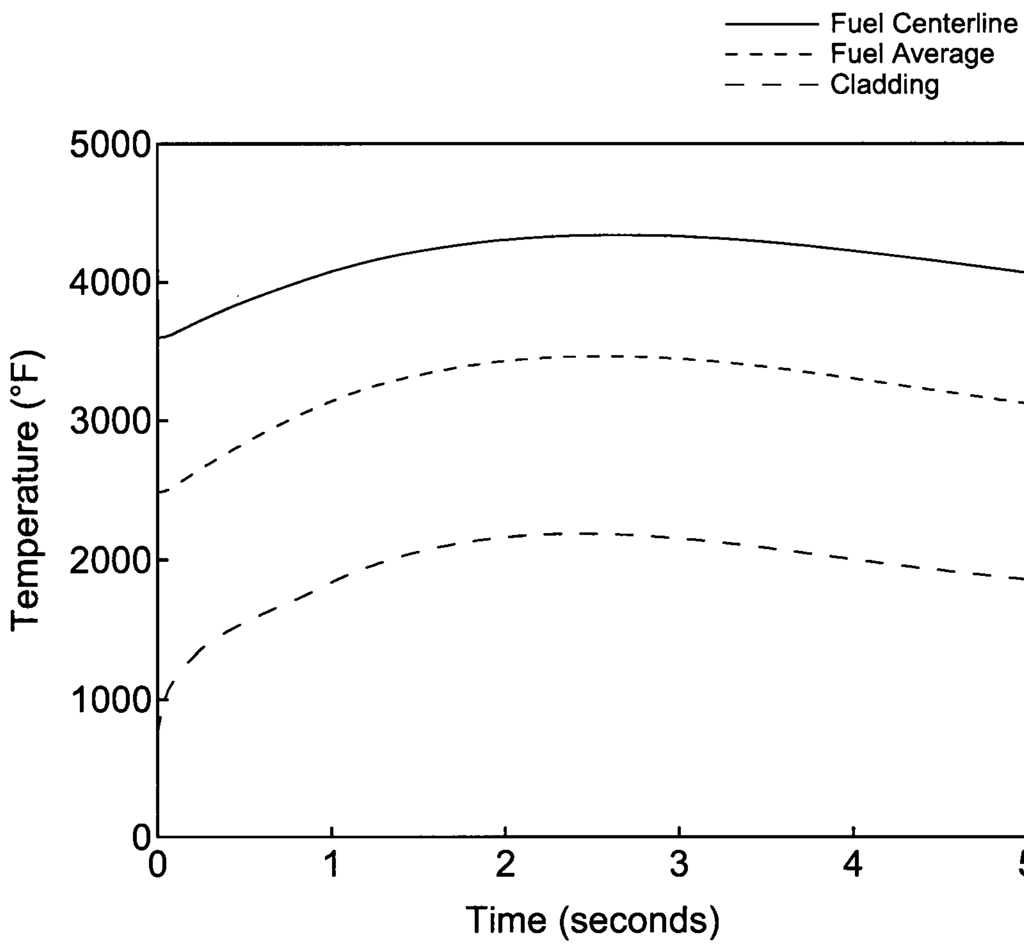

Figure 15.4.8-6 Fuel and Cladding Temperature versus Time

Rod Ejection (HFP, EOC)

DCD_15.04.08-11

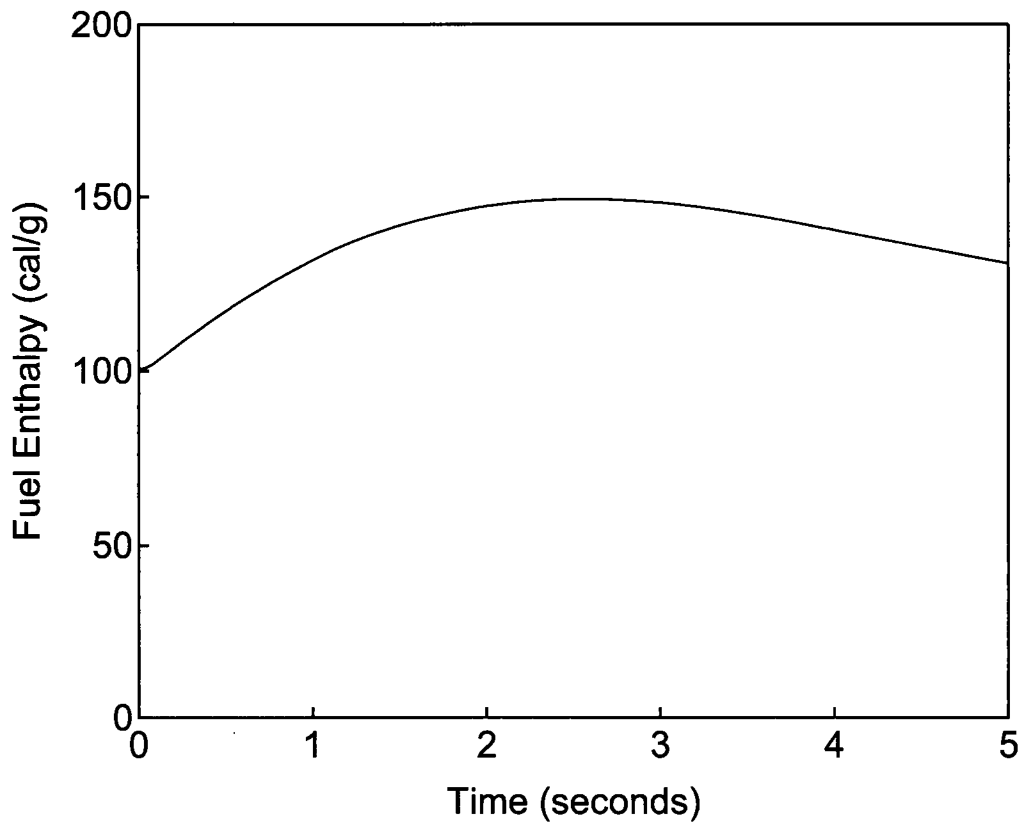

Figure 15.4.8-7 Radial Average Enthalpy versus Time

Rod Ejection (HFP, EOC)

Attachment 2

Table 15.0-1
Summary of Event Classification, Initial Conditions and Computer Codes (Sheet 3 of 4)

DCD_15.0.0-30

Section	Event	Category	Computer Code(s) Utilized	Reactivity Coefficients Assumed			Initial Power Output (MW_t)
				Moderator Density	Moderator Temperature (pcm/°F)	Doppler*7	
15.4.4	Startup of an inactive loop or recirculation loop at an incorrect temperature	AOO	N/A	--	--	--	--
15.4.5	Flow controller malfunction causing an increase in BWR recirculation loop	N/A to US-APWR					
15.4.6	Inadvertent decrease in boron concentration in the RCS	AOO	N/A	--	--	--	0 and 4466
15.4.7	Inadvertent loading and operation of a fuel assembly in an improper Position	PA	ANC	--	--	--	--
15.4.8	Spectrum of rod ejection accidents	PA	TWINKLE-M, VIPRE-01M, MARVEL-M, ANC	--	Temperature coefficient -20% from design	Temperature coefficient -20% from design	0 and 4540*3
15.5.1	Inadvertent operation of ECCS that increases reactor coolant inventory	AOO	N/A	--	--	--	--
15.5.2	CVCS malfunction that increases reactor coolant inventory	AOO	MARVEL-M	min	--	min feedback Figure 15.0-2	4555*2
15.6.1	Inadvertent opening of a PWR pressurizer pressure relief valve	AOO	MARVEL-M	min	--	max feedback Figure 15.0-2	4466

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