Enclosure 3

UAP-HF-08143, Rev.0

MHI's Response to NRC's Request for Additional Information on "Design Data for TRACE Input Deck Preparation by ERI"

August 2008
(Non Proprietary)

MHI's Response to NRC's

Request for Additional Information

on

"Design Data for TRACE Input Deck Preparation by ERI"

Non-Proprietary Version

August 2008

INTRODUCTION

This report documents MHI's response to the NRC's Request for Additional Information (RAI) on "Design Data for TRACE Input Deck Preparation by ERI" dated July 9, 2008.

MHI provides the requested input data for all 19 items; however, input data specific to the non-LOCA analyses is not included in this response. The input data uniquely applicable to the non-LOCA analyses will be provided to the NRC by separate correspondence no later than September 8, 2008 (60 days after the issuance of the formal RAI). The non-LOCA specific input data pertains to RAIs 13 through 14 and 16 through 19.

REQUEST-1

Detailed engineering drawings including various dimensions (e.g., length, thickness, elevation, etc.) of the reactor pressure vessel and its internals, pressurizer and steam generators. These drawings should also provide elevations for different penetrations in the above components.

RESPONSE

(1) Reactor Vessel
Section 5.3 of the US-APWR Design Control Document (DCD) provides details regarding the US-APWR reactor vessel (RV) design. Figures 1-1, 1-2, and 1-3 (reproduced from DCD Chapter 5) provide dimensions of the reactor vessel and its internal components. Table 1-1 provides elevations that correspond to the numbered RV subregions depicted in Figure 1-1(1/2). Table 1-2 summarizes the reactor vessel design data used in the DCD Chapter 15 safety analysis. Table 1-3 shows the number of upper core internals assembly components used in the safety analysis corresponding to items shown in Figure 1-3. Flow areas and volumes of the reactor vessel subregions are provided in Table 4-1 as part of the response for Request-4.

(2) Pressurizer
Section 5.4.10 of the US-APWR DCD provides details regarding the US-APWR pressurizer design. Figure 1-4 provides a schematic of the US-APWR pressurizer including the basic design dimensions.

(3) Steam Generator
Section 5.4.2 of the US-APWR DCD provides details regarding the US-APWR steam generator design. Figure 1-5 provides a schematic of the US-APWR steam generator. Table 1-4 provides the length and volume of the subregions corresponding to those depicted in Figure 1-5. The main design data for the steam generator shown in Table 1-5 is reproduced directly from US-APWR DCD Table 5.4.2-1.

Table 1-1 Reactor Vessel Elevations Corresponding to Figure 1-1 (1/2) Subregions

[ ]

Table 1-2 Reactor Vessel Design Data Used in the Safety Analysis

[ ]

Table 1-3 Number of Upper Core Internals Assembly Components Used in the Safety Analysis

[ ]

Table 1-4 Volume and Length of the Steam Generator Subregions Used in the Safety Analysis

Table 1-5 Steam Generator Design Data
(Source: DCD Table 5.4.2-1)

Design pressure, primary side (psig)	2,485
Design pressure, secondary side (psig)	1,185
Design pressure, primary to secondary (psi)	1600
Design temperature, primary side (°F)	650
Design temperature, secondary side (°F)	568
Heat transfer area (ft^2/SG)	91,500
Maximum moisture carryover (weight %)	0.10
Overall height (ft-in.)	71-3
Number of U-tubes (/SG)	6,747
U-tube nominal outside diameter (in.)	0.75
Tube-wall nominal thickness (in.)	0.043
Tube pitch (in.)	1.00 (Triangular)

Figure 1-1 Reactor Vessel Side View (1/2)

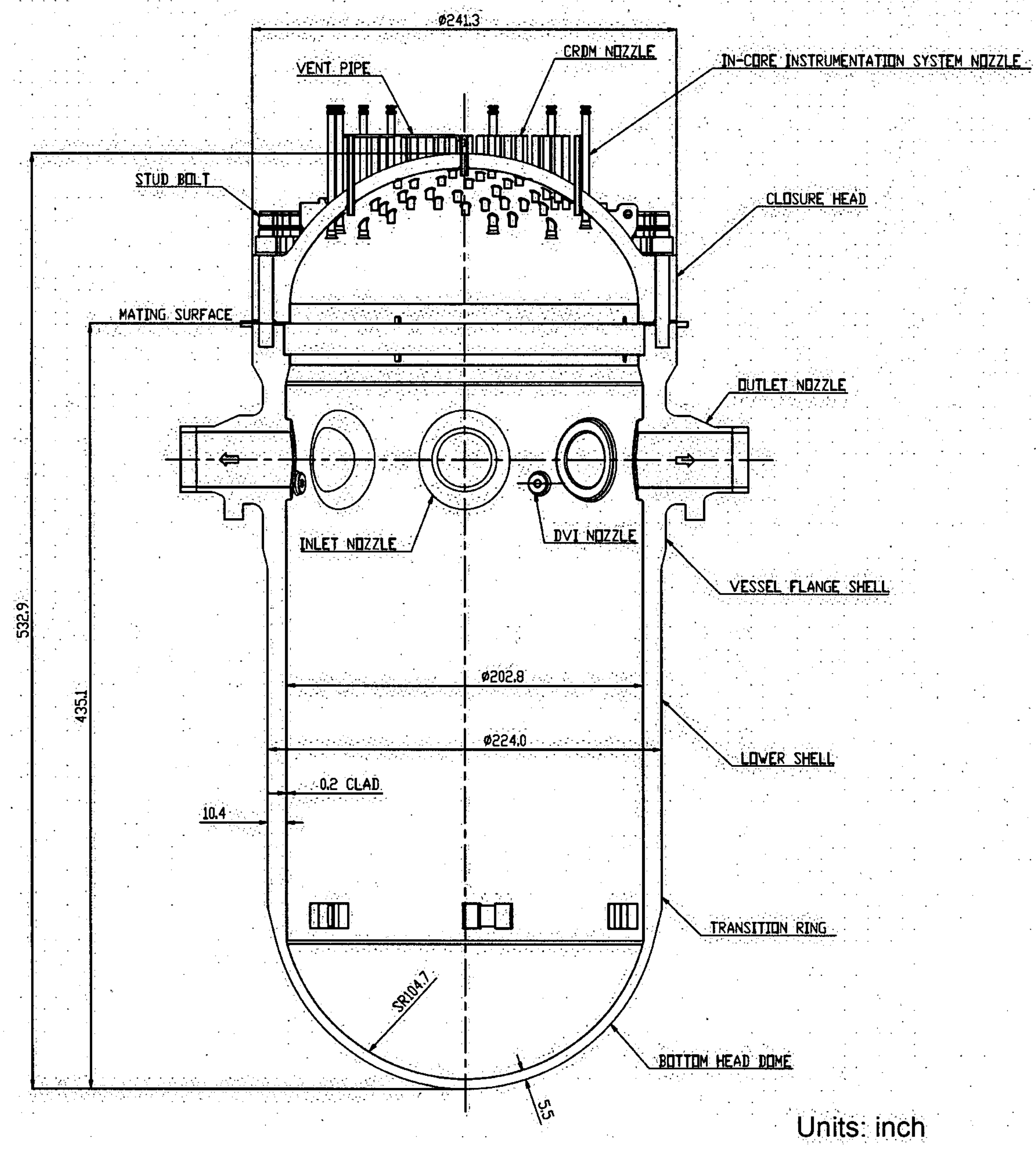

Figure 1-1 Reactor Vessel Side View (2/2)
(Source: DCD Figure 5.3-4)

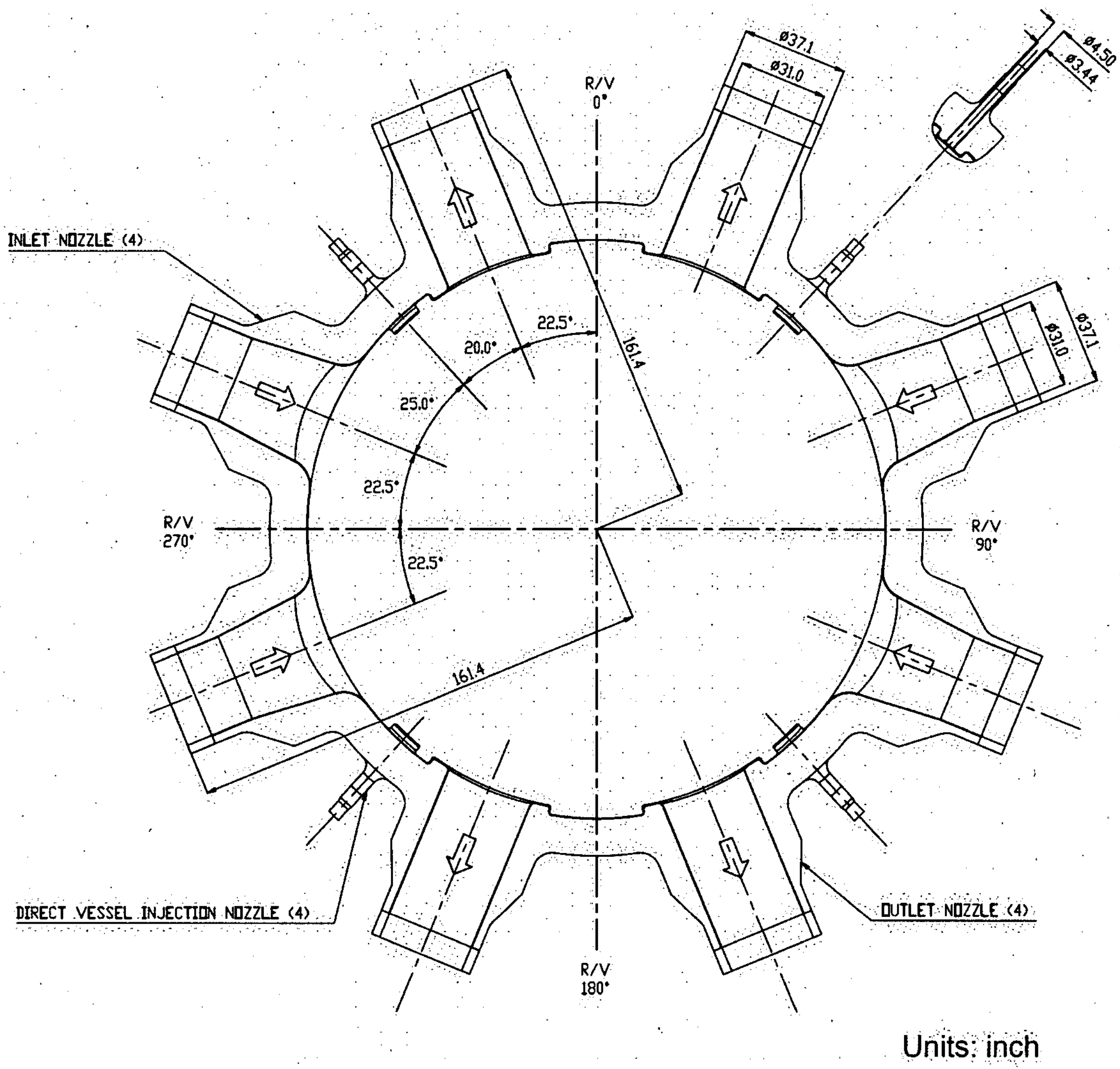

Figure 1-2 Reactor Vessel Cross-Section View at Inlet and Outlet Nozzle Center (Source: DCD Figure 5.3-5)

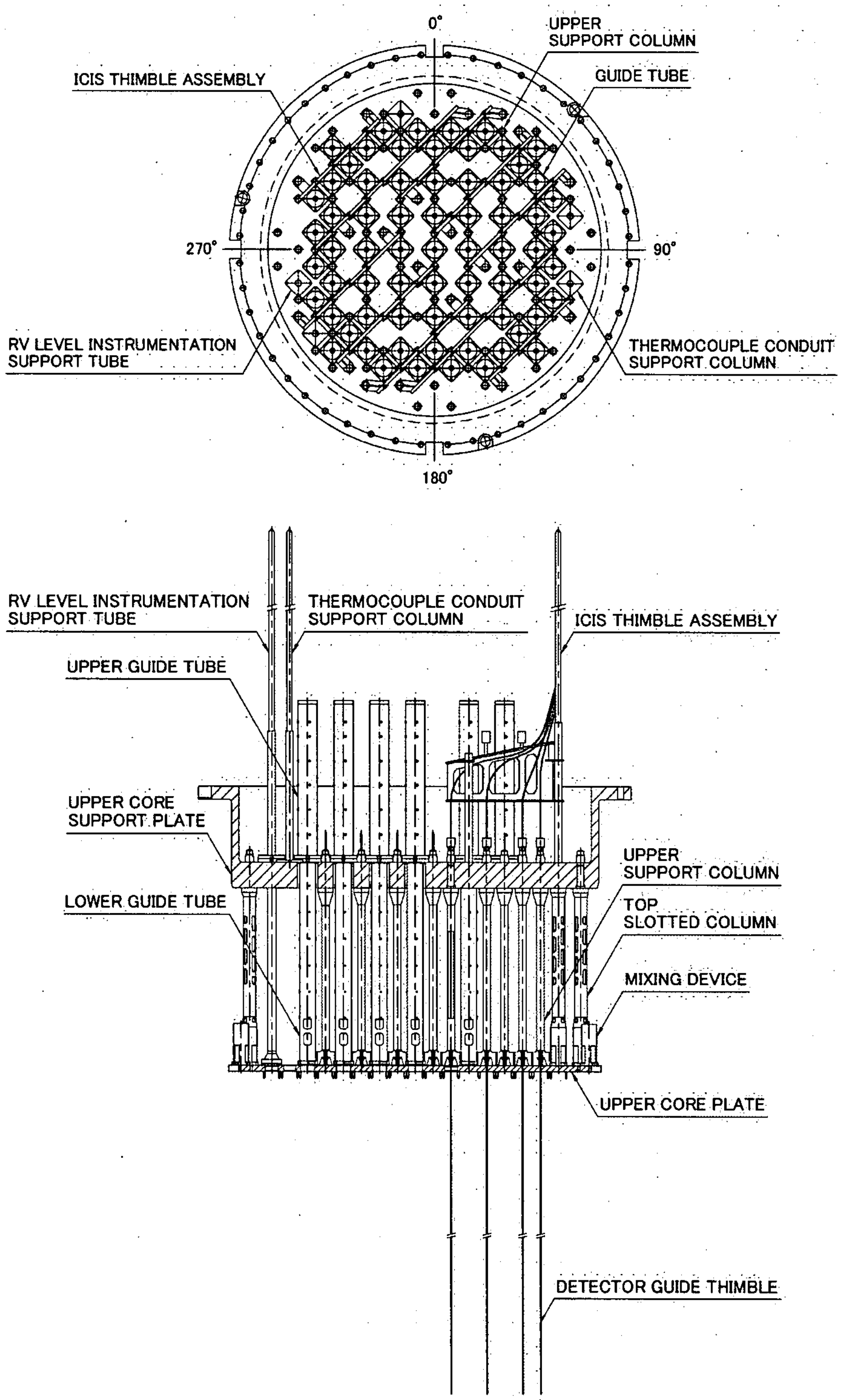

Figure 1-3 Upper Reactor Internals Assembly
(Source: DCD Figure 3.9-5)

Figure 1-4 Pressurizer
(drawing units are mm)

Figure 1-5 Steam Generator

REQUEST-2

Legible and scaled drawings of the accumulators, including various dimensions (e.g., length, thickness, elevation, etc.).

RESPONSE

Section 6.3.2 of the US-APWR Design Control Document (DCD) provides details regarding the US-APWR accumulator design. Figure 2-1 provides a schematic of the US-APWR accumulator including the relevant design dimensions. Table 2-1 shows the accumulator design data used for the DCD Chapter 15 safety analysis.

Table 2-1 Design Data for the Accumulator Used in the Safety Analysis
(Source: DCD Table 6.3-5)

[ ]

Figure 2-1 Accumulator
(drawing units are mm)

REQUEST-3

Piping diagrams giving details, including lengths, of the arrangement of the reactor coolant system (RCS) and emergency core cooling system (ECCS) piping.

RESPONSE

Figures 3-1 and 3-2 show the arrangement of the reactor coolant system (RCS) for the US-APWR. Figures 3-3 and 3-4 provide schematics of US-APWR pressurizer surge line. The lengths, thicknesses, and materials for the various portions of the RCS piping are provided in Table 3-1. Inner diameters, flow areas, and volumes for the RCS piping are shown in Table 3-2. Reactor coolant pump data used in the safety analysis is shown in Table 3-3. The properties of the piping of the injection line of the accumulators are shown in Table 3-4. The direct vessel injection line is not modeled for the LBLOCA and SBLOCA analysis. Elevations of the subregions of the RCS used in the safety analysis are shown in Table 3-5.

Table 3-1 Length, Thickness, and Material of the Coolant System Piping Used in the Safety Analysis

Table 3-2 Inner Diameter, Flow Area, and Volume of the Coolant System Piping Used in the Safety Analysis

Table 3-3 Reactor Coolant Pump Data Used in the Safety Analysis

Table 3-4 Properties of the Emergency Core Cooling System Piping Used in the Safety Analysis

[ ]

Table 3-5 Elevations of the Subregions of the Reactor Coolant System Used in the Safety Analysis

[ ]

Figure 3-1 Reactor Coolant System Loop Layout
(Source: DCD Figure 5.1-3)

Figure 3-2 Reactor Coolant System Elevations
(Source: DCD Figure 5.1-4)

Figure 3-3 Pressurizer Surge Line (1)

Figure 3-4 Pressurizer Surge Line (2)

REQUEST-4

Flow areas and free volumes in the lower plenum, core support plate, reactor fuel region (core), the upper plenum and the vessel upper and lower heads.

RESPONSE

Figure 4-1 depicts the subdivided regions of the reactor vessel internal volumes. Table 4-1 provides the corresponding flow areas, free volumes, and hydraulic diameters for cold conditions. Figure 4-2 defines the subregions for the volumes and heat transfer areas of the metal structure. The corresponding volumes and heat transfer areas are shown in Table 4-2.

Table 4-1 Flow Area, Free Volume, and Hydraulic Diameter in the Reactor Vessel

Table 4-2 Metal Structure Volume and Heat Transfer Area of Reactor Vessel and Core Internals

Figure 4-1 Illustration of Subregions for Volumes in the Reactor Vessel

Figure 4-2 Illustration of Subregions for Metal Structure Volume and Heat Transfer Area in the Reactor Vessel

REQUEST-5

Breakdown of the bypass flow through each of the bypass components mentioned in the FSAR.

RESPONSE

Section 4.4.1.3 of the US-APWR Design Control Document (DCD) discusses the US-APWR core flow. Per DCD Section 4.4.1.3.2 and DCD Table 4.4-3, the estimated minimum value of the effective core cooling flow is 91.0 percent of the RCS flow rate, which is the total RCS flow minus the design core bypass flow of 9% (which includes uncertainty).

DCD Section 4.4.4.2 describes five sources of core bypass flow that is unavailable for providing core cooling. Table 5-1 provides a best estimate breakdown of the design core bypass flow.

[ ]

Table 5-1 Breakdown of the Bypass Flows

REQUEST-6

Pressure drops in the lower plenum, core support plate, fuel assemblies (core), reflector, control rod guide tubes, upper plenum and the upper vessel head.

Note: The pressure drop values requested can be substituted with information regarding the inlet and exit form losses for the corresponding components.

RESPONSE

The pressure drops in the reactor vessel are shown in Table 6-1. The values in Table 6-1 are based on thermal design flow with 10% SG tube plugging.

Table 6-1 Pressure Drops in the Reactor Vessel

[ ]

REQUEST-7

Total length of the control rod guide tubes.

RESPONSE

[ ]

REQUEST-8

Pressure drops in the RCS piping components such as the hot legs, cold legs and the cross over legs.

RESPONSE

The pressure drops in the RCS piping components are shown in Table 8-1. The values in Table 8-1 are based on thermal design flow with 10% SG tube plugging.

Table 8-1 Pressure Drops in the RCS Piping

[ ]

REQUEST-9

Pressure drop values for the primary side of the steam generators (e.g. inlet and outlet plenums, U-tubes, etc.).

RESPONSE

The pressure drops for the primary side of the steam generators are shown in Table 9-1. The values in Table 9-1 are based on thermal design flow with 10% SG tube plugging.

Table 9-1 Pressure Drops for the Primary Side of the Steam Generator

REQUEST-10

Pump characteristics (head versus flow), preferably in tabular format, for the Reactor Coolant Pumps (RCPs) for single-phase water and under degraded conditions, including rated head and the rated torque.

RESPONSE

The pump characteristics are shown by homologous curves instead of head-versus-flow data and are as follows:

(a) Single-phase homologous curves Shown in Table 10-1.
(b) Fully degraded homologous curves Shown in Table 10-2 and Table 10-3.
(c) Rated head [ ]
(d) Rated torque

[ ]

Reference 10-1
Large Break LOCA Applicability Report for US-APWR, MUAP-07011 (Proprietary) and MUAP -07011 NP (Non-Proprietary), July 2007.

Table 10-1 Single-phase Head/Torque Homologous Curves (1/2)

[ ]

$\alpha=\omega/\omega_r$, $v=Q/Q_r$, $h=H/H_r$, $\beta=T/T_r$.

where, ω: Rotational velocity, Q: Flow rate, H: Head, T: Torque, and the subscript "r" denotes "rated".

State: H A N

- N: Normal, Dissipation, Turbine, Reverse
- A: $|v/\alpha| \leq 1$ (A) or $|v/\alpha| > 1$ (V)
- H: Head (H) or Torque (B)

Table 10-1 Single-phase Head/Torque Homologous Curves (2/2)

Table 10-2 Fully Degraded Head Homologous Curves

Table 10-3 Fully Degraded Torque Homologous Curves

REQUEST-11

The pump impeller moment of inertia and the rated speed, preferably in tabular format, for the RCPs.

RESPONSE

DCD Table 5.4.1-1 provides design data for the US-APWR reactor coolant pumps. The pump moment of total inertia of and the rated speed are shown below.

(a) Moment of total inertia 115,330 (lbm-ft^2)
(b) Rated speed [ ]

REQUEST-12

Pump characteristics (head versus flow) for the High Head Injection System, including rated head and the rated torque.

RESPONSE

High head safety injection flow characteristics for minimum and maximum safeguards are provided for the system in Table 12-1 and Table 12-2. The data of Table 12-1 is also shown in Figure 12-1, which is taken directly from Figure 6.3-15 in Chapter 6 of the DCD for the US-APWR. And the data of Table 12-2 is also shown in Figure 12-2, which is taken directly from Figure 6.3-16 in Chapter 6 of the DCD for the US-APWR.

Table 12-1 High Head Safety Injection Flow Characteristic Curve (Minimum Safeguards)

[ ]

Table 12-2 High Head Safety Injection Flow Characteristic Curve (Maximum Safeguards)

[ ]

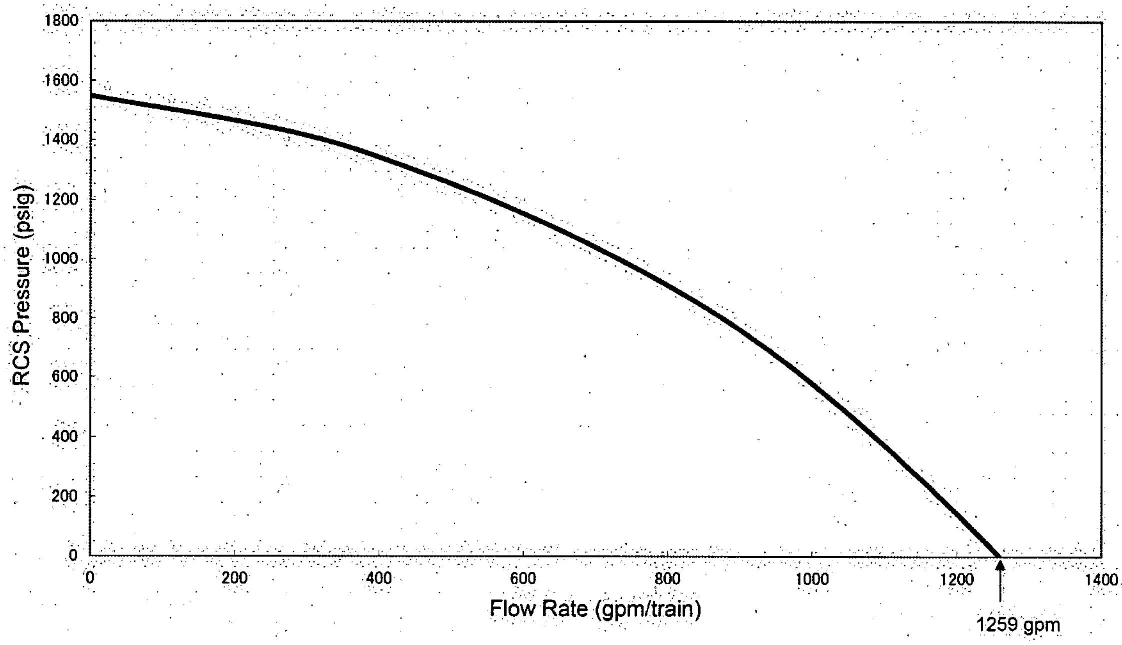

Figure 12-1 High Head Safety Injection Flow Characteristic Curve (Minimum Safeguards)
(Source: DCD Figure 6.3-15)

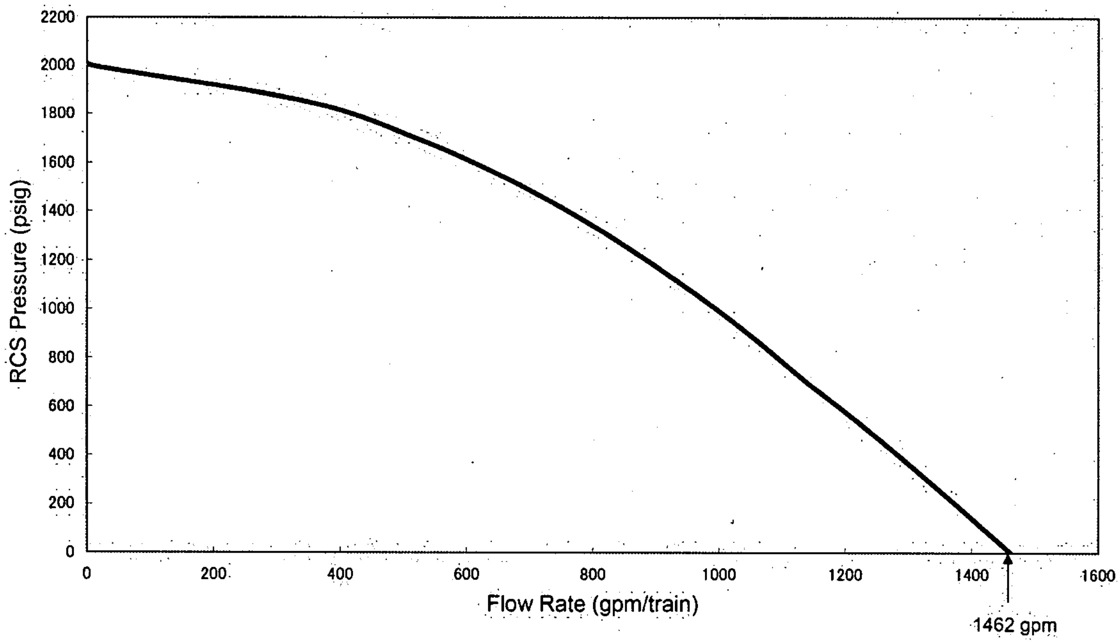

Figure 12-2 High Head Safety Injection Flow Characteristic Curve (Maximum Safeguards)
(Source: DCD Figure 6.3-16)

REQUEST-13

Heater power input and control (including actuation set point) for the pressurizer.

RESPONSE

(1) LBLOCA
No pressurizer heater operation is assumed for the LBLOCA analysis.

(2) SBLOCA
No pressurizer heater operation is assumed for the SBLOCA analysis.

(3) Non-LOCA
The non-LOCA data will be provided separately, as described in the Introduction.

REQUEST-14

The pressurizer spray header flow area, rated flow rate (and corresponding driving pressure difference), actuation set point, and control logic.

RESPONSE

(1) LBLOCA
No pressurizer spray system is expected for the LBLOCA analysis.

(2) SBLOCA
No pressurizer spray system is expected for the SBLOCA analysis.

(3) Non-LOCA
The non-LOCA data will be provided separately, as described in the Introduction.

REQUEST-15

Plant protection system logic and associated equations and delay times for reactor scram.

RESPONSE

DCD Figure 7.2-2 sheets 1 through 7, 9, 11, and 13 provide the functional diagrams for the reactor trip system logic. The equations for the overtemperature ΔT and overpower ΔT trips are defined on DCD Figure 7.2-2 sheet 5. Non-trip plant protection system logic (i.e., feedwater isolation, EFW actuation & isolation, ECCS actuation, containment isolation, etc.) is provided on DCD Figure 7.2-2 sheets 8, 10, 11, 12, 13, and 14.

DCD Table 15.0-4 summarizes the reactor trip and ESF actuation analytical limits and time delays assumed for the Chapter 15 safety analyses. For convenience, a summary of these values is provided in tabular format below.

Protective Function Description	Maximum Response Time (sec)	Analytical Limit
High Power Range Neutron Flux (low setpoint) Reactor Trip	0.6	35% RTP
High Power Range Neutron Flux (high setpoint) Reactor Trip	0.6	118% RTP
Overtemperature ΔT Reactor Trip	6.0	Variable (See DCD Figure 15.0-1)
Overpower ΔT Reactor Trip	6.0	Variable (See DCD Figure 15.0-1)
Low Reactor Coolant Flow Reactor Trip	1.8	87% of rated flow
Low Reactor Coolant Pump Speed Reactor Trip	0.6	95% of rated pump speed
High Pressurizer Pressure Reactor Trip	1.8	2425 psia
Low Pressurizer Pressure Reactor Trip	1.8	1860 psia
Low Steam Generator Water Level Reactor Trip	1.8	0% of narrow range level span
High-High Steam Generator Water Level Reactor Trip	1.8	75% of narrow range level span
High Pressurizer Water Level Reactor Trip	1.8	100% of level span
ECCS Signal Reactor Trip	3.3	--
ECCS Signal on Low Main Steam Line Pressure	3.0	500 psia
ECCS Signal on Low Pressurizer Pressure	3.0	1760 psia
ECCS Signal on High Containment Pressure	3.0	24.0 psia

Protective Function Description	Maximum Response Time (sec)	Analytical Limit
Main Steam Line Isolation Signal on Low Steam Line Pressure	3.0	500 psia
Main Steam Line Isolation Signal on High-High Containment Pressure	3.0	39.9 psia
Containment Spray Signal on High-3 Containment Pressure	3.0	51.2 psia
Emergency Feedwater Actuation on Low Steam Generator Water Level	3.0	0% of narrow range level span
Emergency Feedwater Isolation Signal on Low Main Steam Line Pressure	3.0	500 psia
Emergency Feedwater Isolation Signal on High Steam Generator Water Level	3.0	55% of narrow range level span

Overtemperature ΔT

1. DNB Protection

$$\left(\frac{\Delta T}{\Delta T^{nom}}\right)\frac{(1+\tau_7 s)}{(1+\tau_8 s)}\left(\frac{1}{1+\tau_9 s}\right)=K_1-K_2\frac{1+\tau_2 s}{1+\tau_3 s}\left(T_{avg}-T_{avg0}\right)+K_3\left(P-P_0\right)-f_1(\Delta I)$$

Where:

ΔT is measured RCS ΔT, °F
s is the Laplace transform operator, sec^{-1}
T_{avg} is the measured RCS average temperature, °F
T_{avg0} is the nominal T_{avg} at RTP, ≤ [ ] °F

P is the measured pressurizer pressure, psig
P_0 is the nominal RCS operating pressure, ≥ [ ] psig

K_1 ≤ [ ] K_2 ≥ [ ] $°F^{-1}$ K_3 ≥ [ ] psi^{-1}
τ_2 ≥ [ ] sec τ_3 ≤ [ ] sec τ_7 ≥ [ ] sec
τ_8 ≤ [ ] sec τ_9 ≤ [ ] sec

2. Core Exit Boiling Limit

$$\left(\frac{\Delta T}{\Delta T^{nom}}\right)\frac{(1+\tau_7 s)}{(1+\tau_8 s)}\left(\frac{1}{1+\tau_9 s}\right)=K_4-K_5\frac{1+\tau_4 s}{1+\tau_5 s}\left(T_{avg}-T_{avg0}\right)+K_6\left(P-P_0\right)$$

Where:

ΔT is measured RCS ΔT, °F
s is the Laplace transform operator, sec^{-1}
T_{avg} is the measured RCS average temperature, °F

T_{avg0} is the nominal T_{avg} at RTP, ≤ [ ] °F

P is the measured pressurizer pressure, psig
P_0 is the nominal RCS operating pressure, ≥ [ ] psig

K_4 ≤ [ ] K_5 ≥ [ ] $°F^{-1}$ K_6 ≥ [ ] psi^{-1}
τ_4 ≥ [ ] sec τ_5 ≤ [ ] sec τ_7 ≥ [ ] sec
τ_8 ≤ [ ] sec τ_9 ≤ [ ] sec

<u>Overpower ΔT</u>

$$\left(\frac{\Delta T}{\Delta T^{nom}}\right)\frac{(1+\tau_{13}s)}{(1+\tau_{14}s)}\left(\frac{1}{1+\tau_{15}s}\right) = K_7 - K_8\frac{\tau_6}{1+\tau_6 s}T_{avg} - K_9\left(T_{avg} - T_{avg0}\right) - f_2(\Delta I)$$

Where:

ΔT is measured RCS ΔT, °F
s is the Laplace transform operator, sec^{-1}
T_{avg} is the measured RCS average temperature, °F
T_{avg0} is the nominal T_{avg} at RTP, ≤ [ ] °F

K_7 ≤ [ ]
K_8 ≥ [ ] $°F^{-1}$ for increasing T_{avg}
[ ] $°F^{-1}$ for decreasing T_{avg}
K_9 ≥ [ ] $°F^{-1}$ when $T_{avg} > T_{avg0}$
[ ] $°F^{-1}$ when $T_{avg} \le T_{avg0}$
τ_6 ≥ [ ] sec
τ_{13} ≥ [ ] sec
τ_{14} ≤ [ ] sec
τ_{15} ≤ [ ] sec

DCD Figure 15.0-4 shows the negative reactivity addition as a function of time that is used in the Chapter 15 safety analyses to model RCCA insertion following a reactor trip. Table 15-1 provides a tabular version of the data (prior to normalization) utilized for the creation of DCD Figure 15.0-4.

Table 15-1 Scram Reactivity

REQUEST-16

Information (preferably with drawings) of the safety system related valves, valve characteristics (e.g., flow versus area), the corresponding loss coefficients, and the fully open flow areas.

RESPONSE

(1) LBLOCA
The safety system related valves are not modeled for the LBLOCA analysis.

(2) SBLOCA
Only the main steam safety valve (MSSV) is assumed to operate in the SBLOCA analysis as safety system related valves. The MSSV type is spring-loaded. Table 16-1 shows specifications of the MSSV. No pressurizer safety valve (including the safety depressurization valve) is modeled for SBLOCA analysis, because it would not operate in SBLOCA situations. In SBLOCA analysis, the MSSV flow area was set to an empirical value, and the discharge coefficient was adjusted so that the flow rate under saturated steam and choked flow conditions at the design pressure agreed with the main steam flow rate. The main steam flow rate was used because it was less than the MSSV design capacity and therefore conservative. As a consequence of this, the loss coefficients were set to zero.

(3) Non-LOCA
The non-LOCA data will be provided separately, as described in the Introduction.

Table 16-1 Specifications on the MSSV

REQUEST-17

Control logic for opening/closing of the safety system related valves along with corresponding delay times.

RESPONSE

(1) LBLOCA
The safety system related valves are not modeled for the LBLOCA analysis.

(2) SBLOCA
Control logic for the MSSV in SBLOCA is as follows:
(a) Opening/closing set point
(b) Delay time
[ ]

(3) Non-LOCA
The non-LOCA data will be provided separately, as described in the Introduction.

REQUEST-18

The core radial and axial power distributions (preferably in tabular form).

RESPONSE

(1) LBLOCA
The core axial power distribution for the LBLOCA analysis is shown in Table 18-1. The values in Table 18-1 are used with the reference case of the LBLOCA analysis of Section 15.6.5 of the DCD for the US-APWR. The radial power distribution used with the reference case of the LBLOCA analysis of Section 15.6.5 of the DCD for the US-APWR is shown in Table 18-2.

(2) SBLOCA
The core axial power distribution for the hot rod in SBLOCA analysis is shown in Table 18-3. The values in Table 18-3 are used with the SBLOCA analysis in the DCD for the US-APWR. A heat flux peaking factor, F_Q, of 2.6 and an enthalpy rise hot channel factor, $F_{\Delta H}$, of 1.78 were used.

(3) Non-LOCA
The non-LOCA data will be provided separately, as described in the Introduction.

Table 18-1 Core Radial Power Distribution in LBLOCA Analysis

Table 18-2 Core Axial Power Distribution in the LBLOCA Analysis (1/2)

[ ]

Table 18-2 Core Axial Power Distribution in the LBLOCA Analysis (2/2)

[ ]

Table 18-3 Core Axial Power Distribution in the SBLOCA Analysis

REQUEST-19

The neutron kinetics parameters (e.g., prompt neutron generation time, delayed neutron fractions, decay constants, for six groups, Doppler reactivity coefficient, void reactivity coefficient, etc.).

RESPONSE

(1) LBLOCA
The built-in model used with the LBLOCA analysis of Section 15.6.5 of the DCD for the US-APWR is the same as WCOBRA/TRAC MOD7A Rev. 6.

(2) SBLOCA

(a) Prompt neutron generation time	[ ] (μsec)
(b) Delayed neutron fraction	[ ] (%)
(c) Decay constants for six groups	Eleven group decay constants, shown in Table 19-1, were used for SBLOCA analysis.
(d) Doppler reactivity coefficient	[ ] (pcm/°F)
(e) Void reactivity coefficient	Density reactivity coefficients, shown in Table 19-2, were used for SBLOCA analysis instead of void reactivity coefficients.

(3) Non-LOCA
The non-LOCA data will be provided separately, as described in the Introduction.

Table 19-1 Eleven Group Decay Constants in the SBLOCA analysis

Energy yields and decay constants were obtained by fitting 1971 ANS standard curve with eleven groups.

Table 19-2 Density Reactivity Coefficient in the SBLOCA Analysis