US-APWR
Sump Strainer
Downstream Effects

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Abstract

The intent of this technical report is to assess the US-APWR systems and components downstream of the containment sump strainers to ensure that that these systems and components will operate as designed under post Loss-of Coolant Accident Conditions (LOCA). Downstream systems and components include the Emergency Core Cooling System (ECCS), Containment Spray System (CSS) and the reactor core. This report evaluates the effects of operating with debris-laden, post-LOCA fluid.

This review incorporates the lessons learned as part of the USNRC Generic Safety Issue 191 (GSI-191) and addresses the component and system related concerns identified in Generic Letter (GL) 2002-04. This report has been prepared in accord with NEI 04-07 and published USNRC staff expectations. The report meets the intent of WCAP-16793-NP, Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid. Information and guidance used has been extracted from the noted references and adapted to the US-APWR design.

This technical report is divided into two (2) subject areas: Ex-Vessel and In-Vessel Evaluations.

This report concludes that the US-APWR Emergency Core Cooling System, Containment Spray System and their components are fully capable of performing their intended functions under post-LOCA operating conditions. The ECCS and CSS are fully capable of providing adequate core cooling to ensure the reactor core is maintained in a safe, stable condition following a Loss-of-Coolant Accident (LOCA).

The report concludes that debris-laden post-LOCA fluid will not plug or block the reactor core such that cooling flow is reduced below the required flow to maintain the core in a long-term coolable geometry. The report also shows that chemical induced local blockages or scale formation on the fuel cladding surface on reactor fuel and cladding will not affect the ability to provide adequate decay heat removal. Cladding temperatures are maintained below those required by Section 50.46 of Title 10 of the Code of Federal Regulations (10 CFR).

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List of Acronyms

Al	Aluminum
APWR	Advanced Pressurized Water Reactor
BWG	British Wire Guage
Ca	Calcium
CSS	containment spray system
CVCS	Chemical and Volume Control System
DBA	Design Basis Accident
DCD	Design Control Document
DHR	Decay Heat Removal
DVI	Direct Vessel Injection
ECCS	Emergency Core Cooling System
ECC/CS	Emergency Core Cooling and Containment Spray
ESF	Engineered Safety Features
GL	Generic Letter
GSI	Generic Safety Issue
HHIS	High Head Injection System
HVAC	Heating, Ventilating and Air Conditioning
ICET	Integrated Chemical Effects Testing
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LTCC	Long-term Core Cooling
MCR	Main Control Room
MHI	Mitsubishi Heavy Industries, Ltd.
NaTB	Sodium Tetra Borate, decahydrates $Na_2B_4O_7 \cdot 10H_2O$
NEI	Nuclear Energy Institute
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PCT	Peak Cladding Temperature
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
RCS	Reactor Coolant System
RHR	Residual Heat Removal

RMI	Reflective Metal Insulation
RWSP	Refueling Water Storage Pit
SBLOCA	Small Break Loss of Coolant Accident
S-singal	Safety Injection Signal
SI	Safety Injection
Si	Silicon
SIS	Safety Injection System
US-APWR	U.S. Advanced Pressurized Water Reactor
US.NRC	US.Nuclear Regulatory Commission
ZOI	Zone of Influence

1.0 INTRODUCTION

Generic Letter (GL) 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors (Ref. 3-1) was issued by the USNRC requesting holders of operating reactor licenses to evaluate their emergency core cooling system (ECCS) and containment spray system (CSS) recirculation functions in light of events regarding the blockage of containment sump strainers. The GL notes that:

> "Debris could also plug or wear close-tolerance components within the ECCS or CSS systems. This plugging or wear might cause a component to degrade to the point where it could not perform its designated function (i.e., pump fluid, maintain system pressure, or pass and control system flow.)
>
> …Third, debris blockage at flow restrictions within the ECCS recirculation flowpath downstream of the sump screen is a potential concern for PWRs. Debris that is capable of passing through the recirculation sump screen may have the potential to become lodged at a downstream flow restriction, such as a high-pressure safety injection (HPSI) throttle valve or fuel assembly inlet debris screen. Debris blockage at such flow restrictions in the ECCS flow-path could impede or prevent the recirculation of coolant to the reactor core, thereby leading to inadequate core cooling. Similarly, debris blockage at flow restrictions in the CSS flowpath, such as a containment spray nozzle, could impede or prevent CSS recirculation, thereby leading to inadequate containment heat removal. Debris may also accumulate in close-tolerance subcomponents of pumps and valves. The effect may be either to plug the subcomponent, thereby rendering the component unable to perform its function, or to wear critical close tolerance subcomponents to the point at which component or system operation is degraded and unable to fully perform its function. Considering the recirculation sump screen's design function of intercepting potentially harmful debris, it is essential that the screen openings be adequately sized and that the sump screen's current configuration be free of gaps or breaches which could compromise the ECCS and CSS recirculation functions. It is also essential that system components be designed and evaluated to be able to operate as necessary with debris laden fluid post-LOCA"

The GL requested the following specific information be provided.

> "(v) The basis for concluding that inadequate core or containment cooling would not result

due to debris blockage at flow restrictions in the ECCS and CSS flowpaths downstream of the sump screen, (e.g., a HPSI throttle valve, pump bearings and seals, fuel assembly inlet debris screen, or containment spray nozzles). The discussion should consider the adequacy of the sump screen's mesh spacing and state the basis for concluding that adverse gaps or breaches are not present on the screen surface.

(vi) Verification that close-tolerance subcomponents in pumps, valves and other ECCS and CSS components are not susceptible to plugging or excessive wear due to extended post-accident operation with debris-laden fluids"

In response to the GL, the Nuclear Energy Institute, NEI, issued Guidance Report NEI 04-07 (Ref. 3-16) delineating a generic, consistent approach to address the NRC concerns identified in GL 2004-02. The USNRC, in turn, issued a Safety Evaluation clarifying those items to be addressed and endorsing an approach that would be acceptable to the USNRC staff (Ref. 3-2). With respect to downstream effects, the SE states:

1. Licensees should consider that some particles larger than the flow openings in a sump screen will deform and flow through or orient axially and flow through the screen, and determine what percentage of debris would likely pass through the sump screen and be available for blockage at downstream locations.

2. Licensees should consider term of system operating lineup (short or long), conditions of operation, and mission times.

3. Licensees should consider wear and abrasion of pumps and rotating equipment, piping, spray nozzles, instrumentation tubing, and HPSI throttle valves. The potential for wear to alter system flow distribution and/or form plating of slurry materials (in heat exchangers) should be included.

4. An overall ECC or CS system evaluation should be performed considering the potential for reduced pump/system capacity resulting from internal bypass leakage or through external leakage.

5. Licensees should consider flow blockage associated with core grid supports, mixing vanes, and debris filter, and its effect on fuel rod temperature.

And that "… licensees should address chemical effects on a plant-specific basis."

For Ex-Vessel evaluations, the USNRC further clarified specific areas to be addressed with the issuance of an NRC letter entitled "Audit Plan for Verifying the Adequacy of Licensee Responses to Generic Letter 2004-02" (Ref. 3-3). This audit plan was intended to fully address the USNRC concerns identified in Reference 3-2, Final Safety Evaluation for NEI Guidance Report 04-07, regarding the evaluation of ex-vessel components downstream of the containment sump during post-LOCA operation. There are no other regulatory guidance documents that specifically pertain to the evaluation of ex-vessel downstream components. Therefore, addressing the issues identified in the "Audit Guidelines" fully addresses the concerns identified in the GL.

USNRC accepted guidelines and methods for the evaluation the reactor vessel and fuel will be contained in Topical Report TR-WCAP-16793-NP, Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid. The USNRC has not yet fully reviewed Topical Report (TR) WCAP-16793-NP, REVISION 1. Once the USNRC has issued their Safety Evaluation on the Topical, MHI will review this report and revise it as needed.

This technical report fully meets the intent of and addresses the technical concerns and considerations identified in the current USNRC guidance documents.

2.0 OBJECTIVE

The objective of ex-vessel downstream effects evaluation is to assess the US-APWR Emergency Core Cooling system (ECCS) and Containment Spray System (CSS) to ensure that these systems and their components will operate as designed under post Loss-of Coolant Accident Conditions (LOCA).

The objective of in-vessel downstream effects evaluation is to provide that long-term core cooling (LTCC) would be established and maintained properly during post-LOCA considering the presence of debris in the recirculating coolant delivered to the RCS and core and would be achieved to satisfy the requirements of 10 CFR 50.46 for the US-APWR plant.

3. EX-VESSEL DOWNSTREAM EFFECTS

3.1 System Descriptions

The US-APWR engineered safety features (ESF) includes an Emergency Core Cooling System (ECCS) / Safety Injection System (SIS) and a Containment Spray System (CSS). The ECCS is automatically initiated by a safety injection signal (S-signal) and the CSS is automatically initiated by a containment spray signal. Both systems take suction from refueling water storage pit (RWSP). Four ECC/CS strainers are installed in the RWSP; one for each of the four ECC/CS trains. These systems include an accumulator system, pumps, valves, heat exchangers, piping, fittings and other components. These systems and components may be affected by debris that passes through the containment sump strainer during recirculation following a LOCA.

Figure 3.1-1 shows a schematic flow diagram of the ECCS and CSS.

3.1.1 Emergency Core Cooling System

The primary function of the ECCS is to remove stored and fission product decay heat from the reactor core following an accident. The ECCS is designed to meet the acceptance criteria of 10CFR50.46(b).

In meeting 10CFR50.46(b), the ECCS is designed to perform the following major safety-related functions:

- Provides safety Injection flow to the reactor core following a LOCA
- Maintains the reactor in a safe shutdown condition
- Assists in maintaining pH control of the post-LOCA fluid.

The US-APWR ECCS consists of an accumulator system, a high head injection system (HHIS) and an emergency letdown system. The ECCS injects borated water into the RCS following a postulated LOCA to cool the reactor core to prevent damage to the fuel cladding, and to limit the fuel cladding zirconium-water reaction. The accumulator and emergency letdown systems are not active during post-LOCA long-term operation. Therefore, for the purpose of this report, only the HHIS will be discussed.

The HHIS consists of four independent safety trains, each designed as a 50% capacity train. Each train includes a safety injection (SI) pump suction isolation valve, a dedicated, 50% capacity SI pump, a safety injection pump discharge containment isolation valve, a direct vessel safety injection line isolation valve, and a hot leg injection isolation valve. The SI pumps are aligned to take suction from the RWSP and deliver borated water directly to the reactor vessel downcomer. The RWSP is located within the lowest portion of the containment vessel and collects the water from the postulated break and from the containment spray wash down. The RWSP provides a continuous borated water source for the SI pumps, thus avoiding the need to switch the pump suction from a storage water tank to the containment recirculation sump.

The SI pumps are automatically initiated on an S-signal and supply borated water (at approximately 4,000 ppm boron) from the RWSP to the reactor vessel through direct vessel injection (DVI) lines.

The HHIS is realigned to shift the RCS injection from the DVI line to the hot leg injection line after a LOCA in order to prevent boron precipitation. Therefore, both injection lines are in the flow-paths of the water through the ECC/CS strainers. Safety injection pump minimum flow lines are also in these flow-paths, and are always is use when the SI pumps are operating.

The SI pumps continue to supply borated water during long-term cooling. During long-term cooling, core temperatures are reduced to long term, steady state levels associated with the dissipation of residual heat generation. During long term cooling, the HHIS injects into both the RCS hot legs and the reactor vessel to avoid an unacceptably high concentration of boric acid (H_3BO_3) in the core. (Ref. 3-9, Section 6.3)

3.1.2 Containment Spray System

The containment spray system is a dual-function ESF system. The system provides containment spray for fission product removal and containment cooling. The system also provides residual heat removal for normal plant shutdown and refueling operations. The CSS and the Residual Heat Removal System (RHRS) share the containment spray/residual heat removal (CS/RHR) pumps, CS/RHR heat exchangers and some system piping and valves. For the purposes of this report only the specific components used during CSS operation were evaluated.

The CSS has four 50% capacity trains of containment spray, including four CS/RHR RWSP suction lines, four CS/RHR spray pumps, four CS/RHR heat exchangers, and a spray ring header. The spray rings are supplied from the four trains of containment spray.

The CSS is designed to perform the following functions:

- Provide containment spray to assist in containment heat removal.
- Provide fission product removal though atmospheric scrubbing

The CSS is designed to limit and control post-LOCA containment pressure, so that the peak containment accident pressure is kept well below the containment design pressure. With CSS operation, containment pressure is reduced to less than 50% of the peak calculated pressure during the design basis LOCA within 24 hours after the postulated accident.

Following a DBA, the containment pressure approaches atmospheric pressure. When the containment pressure is reduced sufficiently and the operator determines that containment spray is no longer required, the operator terminates containment spray. The operator closes the containment spray header isolation valves and aligns system flow through the full flow test line. The RWSP water is then recirculated and cooled. (Ref. 3-9, Subsection 6.2.2)

The CS/RHR pumps are motor-driven centrifugal pumps with mechanical seals. The pumps are sized to deliver 3,000 gpm at a discharge head of 410 ft. The 100% capacity design flow rate (two of four 50% capacity CS/RHR pumps) is based on a 15.2 gpm flow per nozzle and 348 nozzles in the ring header. The two-pump 100% flow rate is, therefore, 6,000 gpm.

The CS/RHR heat exchangers provide long term cooling by removing heat from the recirculated post-LOCA fluid. The reduced temperature of the RWSP fluid aids in the further reduction of containment pressure.

3.2 Design inputs / Evaluation Assumptions

3.2.1 LOCA Scenarios

This report addresses ECCS and CSS operation under small-break, and large-break LOCA conditions. Figure 3.1-1 shows a schematic flow diagram of the ECCS and CSS during post-LOCA operation.

The two operational conditions during post-LOCA long-term cooling that are addressed are the maintenance of long-term decay heat removal and the potential for boric acid (H_3BO_3) precipitation. After the quenching of the core at the end of reflood phase, the ECCS supplies borated water from the RWSP to remove decay heat and to keep the core subcritical. Borated water from the RWSP is initially injected through DVI lines (reactor vessel (RV) injection mode). If left uncontrolled, boric acid (H_3BO_3) concentration in the core may increase due to boiling and reach the precipitation concentration in the case of cold leg break. Boric acid precipitation in the core could affect the core cooling. To prevent the boric acid precipitation, the operator switches over the operating DVI lines to the hot leg injection line (simultaneous RV and hot leg injection mode).

3.2.1.1 Small Break LOCA Operational Description

Refer to the simplified ECCS Piping and Instrument Drawing in Figure 3.1-1.

The small break LOCA (SBLOCA) is assumed to occur in the cold leg piping located between the outlet of the RCP and the corresponding RV inlet nozzle, as this break places the most severe performance requirement on the ECCS.

Compared with the large break, the phases of the SBLOCA prior to recovery occur over a longer time period. A SBLOCA can be divided into five phases: blowdown, natural circulation, loop seal clearance, boil-off, and core recovery. The duration of each phase depends on the break size and the performance of the ECCS.

For the purpose of this report the SBLOCA is bounded by the large break LOCA (LBLOCA), recirculation and post-LOCA long-term cooling. The ECCS flows during a SBLOCA are considerably smaller than during a LBLOCA. Also, the debris source term is expected to be much smaller during a SBLOCA. Therefore, the SBLOCA is bounded by the conditions of the LBLOCA with respect to the evaluation of downstream components.

3.2.1.2 Large Break LOCA Operational Description

Refer to the simplified ECCS Piping and Instrument Drawing in Figure 3.1-1.

The pipe break for the LBLOCA is assumed to occur in the cold leg piping located between the outlet of the reactor coolant pump (RCP) and the corresponding RV inlet nozzle, as this break places the most severe performance requirement on the ECCS. The double-ended cold leg guillotine (DECLG) and split breaks are considered. The LBLOCA is generally divided into three phases; the blowdown phase, refill phase, and the reflood phase.

During a LBLOCA, coolant flow initially is from the four accumulators. Once, accumulator volume is depleted, flow to the core becomes dominant through the SIS. Initially flow is only through DVI. After four hours, flow is through both DVI and the hot-leg injection. The Safety Injection pumps, SIS-RPP-001A, B, C, D, take suction from the RWSP and inject directly into the reactor vessel via SIS-MOV-009A, B, C, D and SIS-MOV-011A, B, C, D or the Hot Leg via MOVs SIS-MOV-009A, B, C, D and SIS-MOV-014A, B, C, D.

The CS/RHR pumps supply the CSS. The CSS flow-path is from the discharge of a CS/RHR Pump, RHS-RPP-001A, B, C, D, through a CS/RHR Heat Exchanger, RHS-RHX-001A, B, C, D to the containment spray header. Recirculation of the post-LOCA fluid is continuous from the start of the event. The RWSP is located in lowest part of containment and is refilled as a result of flow from both the break and CSS operation.

3.2.2 Mission Time

Mission time is defined as the period for which a System, Structure or Component (SSC) is required or credited in performing its safety related function. Mission time, in this context, is not the engineering design or purchase specification operating time. It is the DCD and/or accident analysis credited time.

For the purpose of these evaluations, the mission time for the ECCS and CSS, including post-LOCA long-term operation is defined as 30 days. A 30 day mission time bounds the descriptions and discussions contained in the US-APWR Design Control Document (DCD) Chapters 6 and 15 (Ref. 3-9 and 3-10).

The duration of the ECCS and CSS operation as indicated in the safety analysis evaluation of the Chapter 15 (Ref. 3-10) events is generally only long enough to assure that the appropriate acceptance criteria have been met and does not typically include the transition to shutdown conditions. Therefore, the event-specific discussion does not typically address long-term cooling.

3.2.3 Component List

Table 3.2-1 lists all components and flow-paths within the scope of the downstream evaluation(s). The tables are organized by LOCA Scenario and System Line-up.

Each table also includes the component materials, hardness values of all wetted surfaces (piping, orifice, heat exchanger, throttle valve plug and seat materials, etc), actual and assumed flow velocities, other information used in the piping, pump, heat exchanger and system evaluations.

Material hardness data is provided in Table 3.2-2. (Ref. 3-18)

3.2.4 Post-LOCA Fluid Constituents

The nominal diameter of the sump strainer holes is equal or less than 0.066”.

Tables 3.2-3, 4 and 5 list the constituents, quantities and properties of the post-LOCA fluid (abrasiveness, solids content and size, fiber content and size, chemical properties, etc).

This analysis assumes 100% latent debris bypass, 50% fiber bypass and 5% RMI bypass through the containment sump strainers. The specific quantity, sizes and material properties are referenced either to the DCD Section, technical report to published data, textbook data or vendor data as appropriate.

3.2.5 ECC and CSS System Flows and Flow Velocities

The range of system flow and local velocities expected within the ECC and CS piping systems is provided in tabular form in Table 3.2-6 based on LBLOCA conditions. As stated previously, SBLOCA conditions are bounded by LBLOCA due to the higher flows creating more wear and generating a greater debris load.

The US-APWR is a fixed resistance system under valve wide-open conditions. Emergency Operating Procedures do allow for operator action to throttle flow based on main control room (MCR) indication. The range of operation is therefore assumed to be from shutoff head conditions to runout conditions.

Safety Injection Pump flow is assumed to be 200 gpm for the purposes of calculating settling velocities. Flow is assumed to be 2,000 gpm for the purpose of component wear rate evaluations. Engineering design range of flow is 265 gpm at shutoff and 1,540 gpm at runout.

CS/RHR Pump flow is assumed to be 300 gpm for the purposes of calculating settling velocities. Flow is assumed to be 4,000 gpm for the purpose of component wear rate evaluations. Engineering design range of flow is 355 gpm at shutoff and 3,650 gpm at runout.

These values allow for variations during component procurement and define engineering margin for analysis.

The “as procured” Safety Injection Pump runout flow will be verified to be less than 2,000 gpm. Confirmation Item 3.7.1.

The “as procured” CS/RHR Pump runout flow will be verified to be less than 4,000 gpm. Confirmation Item 3.7.2.

Reliability of the ECCS and CSS are considered in the design, procurement, and installation/layout of components. Verification that the pumps meet the flow requirements is considered part of the COL (Ref. 3-11, Subsection 17.4.9).

3.2.6 Summary of Analysis Conservatisms

This section summarizes the significant conservative assumptions contained within this evaluation. It is not intended to be all inclusive.

3.2.6.1 Safety Injection Pump flow is assumed to be 200 gpm for the purposes of calculating settling velocities. Flow is assumed to be 2,000 gpm for the purpose of component wear rate evaluations. Engineering design range of flow is 265 gpm at shutoff and 1,540 gpm at runout.

3.2.6.2 CS/RHR Pump flow is assumed to be 300 gpm for the purposes of calculating settling velocities. Flow is assumed to be 4,000 gpm for the purpose of component wear rate evaluations. Engineering design range of flow is 355 gpm at shutoff and 3,650 gpm at runout.

3.2.6.3 Wear is calculated from “time zero”, i.e. start of the event. Worst case fluid properties are assumed to be present. This assumption is conservative since it does not credit debris transport or the slow increase of fluid properties due to long term mixing.

3.2.6.4 Fluid velocity through a single CS/RHR heat exchanger tube is assumed to be 15 ft/s. A nominal design and operating heat exchanger velocity range is 3 to 10 ft/s. Therefore the use of 15 ft/s is conservative from a heat exchanger design perspective and bounds the heat exchanger design and procurement specifications.

3.2.6.5 This analysis assumes 100% latent debris bypass, 50% fiber bypass and 5% RMI bypass through the containment sump strainers. It is noted that these quantities are greater than those assumed in the in-vessel evaluation and are therefore conservative with respect to the overall downstream evaluation. Reference Chapter 4, In-Vessel Evaluation.

3.3 Piping, Valve and Heat Exchanger Evaluations

This section evaluates ECCS and CSS piping, valves and heat exchangers with respect to wear and blockage.

Reliability of the ECCS and CSS are considered in the design, procurement, and installation/layout of components. Verification that the materials of construction of the piping, valves and heat exchangers meet the requirements specified below is considered part of the COL (Ref. 3-11, Subsection 17.4.9).

3.3.1 Wear Rate Evaluation Summary

Table 3.3-1 contains a summary of the piping and orifice wear calculations. This calculation assumes a 3-Body (free-flowing) wear model.

Piping and component wear is tabulated for 30 day durations.

3.3.2 Heat Exchanger Evaluation

The CS/RHR heat exchangers are designed to cool the reactor coolant during RHR operation. They remove residual heat during normal shutdown, during shutdown in case of loss of external power sources and during safe shutdown. They assist in long-term cooling operation by cooling post-LOCA fluid prior to discharge through the CSS.

The CS/RHR heat exchangers are specified as shell and U-tube units. The heat exchangers are comprised of ¾" OD, BWG 18 (0.049 in.), 304 SS tubes (Ref. 3-8, Table 5.4.7-2). A single unit is provided in each of the four CSS trains.

The reactor coolant discharged from the CS/RHR pump is circulated through the tube side of the CS/RHR heat exchanger, while cooling is provided by circulating Component Cooling Water through the shell side. The tubes are welded to the tube sheet to prevent leakage of the reactor coolant.

The heat exchanger plugging, fouling and wear evaluation are done in the context of the equipment specification. For velocity, a maximum tube velocity of 15 ft/s is assumed. A nominal design and operating heat exchanger velocity range is 3 to 10 ft/s. Therefore the use of 15 ft/s is conservative from a heat exchanger design perspective and bounds the heat exchanger design and procurement specification(s).

3.3.2.1 Heat Exchanger Plugging

The heat exchanger tubes are ¾" OD, BWG 18 wall. The strainer hole size is 0.066". The heat exchanger tubes are significantly larger than the largest expected particle size. Therefore, a heat exchanger tube will not be plugged or blocked by post-LOCA debris. The flow velocity within a heat exchanger tube is significantly greater than the flow velocity transporting debris to the ECCS inlet piping. Therefore, the particles in solution will remain in solution and not settle out and plug a CS/RHR heat exchanger.

These conclusions are consistent with the referenced NRC Safety Evaluation on WCAP 16406 (Ref. 3-4).

3.3.2.2 Heat Exchanger Performance and Wear

The CS/RHR heat exchanged are sized and specified considering a fouling factor of 0.0005 h ft^2 °F/Btu for closed cycle condensate water (Ref. 3-15). Post-LOCA fluid does contain small amounts latent debris. However, fouling is considered a long-term phenomenon. CS/RHR heat loads are greatest at the start of the event and decrease rapidly over the first 24 hours. Heat removal capacity is not degraded over this short period. Any potential reduction in capability over the 30 day mission time will be gradual and is well within the nominal heat exchanger design. The CS/RHR heat exchangers are sized considering maximum heat load including fouling. Therefore, the CS/RHR heat exchanges are fully capable of performing their intended function using post-LOCA fluid as the process fluid.

The CS/RHR heat exchanger tubes are specified to be constructed of 304 stainless steel. Stainless steel is appropriate for use as heat exchanger tubing and is standard for use in mildly abrasive applications. The tube material will not significantly degrade considering operation with post-LOCA fluid over an intended mission time of 30 days.

3.3.3 Valve Wear Evaluation

Valve and valve trim materials are specified to be wear resistant. The valve procurement specification will note the constituents of the post-LOCA fluid and require that the valve be able to operate reliably under those conditions for a minimum of 30 days.

Reliability of the ECCS and CSS are considered in the design, procurement, and installation/layout of components. Verification that the system valves will meet the fluid wear resistance requirements is considered part of the COL (Ref. 3-11, Subsection 17.4.9). Confirmation Item 3.7.3.

Direct Vessel Safety Injection Line Isolation Valves

Flow balance through the ECCS is controlled through the use of orifice plates and balancing through 4” globe valve SIS-MOV-011A, B, C, D. Due to the presence of downstream flow balancing orifices, the throttle valves are expected to be throttled to a minimum of 1” open between the valve disc and seat. Any potential wear, the opening of a throttle valve and decreasing flow resistance, may be compensated for by throttling the valve from the MCR. Final system start-up testing will confirm that the system will not be in run-out conditions with all system valves wide open.

CS/RHR Pump Full-flow Test Line Stop Valves

A single motor-operated 8” globe valve, RHS-MOV-025 A, B, C, D, with a throttling capability is placed in each of the four RHR return lines. These valves are positioned from the MCR. Any potential wear, the opening of a throttle valve and decreasing flow resistance, may be compensated for by throttling the valve from the MCR. Final system start-up testing with full-flow test line-up condition will confirm that the system will not be in run-out conditions with all system valves wide open.

CS/RHR heat exchanger outlet flow control valves

8” air-operated butterfly valves, RHS-FCV-601 and RHS-FCV-631, are placed in each of two CS/RHR heat exchanger outlet lines. These valves are only adjusted during shutdown cooling operation and are not throttles during post-LOCA operation.

The rate of the opening of the valves can be manually adjusted from the MCR, and the valves fail in the ”open” position to ensure a flow-path of RHRS and CSS. These valves provide the capability to control the flow rates through the heat exchangers by operator’s action based on the RCS temperature changes during plant cool down. Any potential wear, the opening of a throttle valve and decreasing flow resistance, may be compensated for by throttling the valve from the MCR. Final system start-up testing will confirm that the system will not be in run-out conditions with all system valves wide open.

3.3.4 Piping and Valve Blockage and Debris Settling Evaluation

The strainer hole size is 0.066”. Therefore, when the gap of the components is 0.066”+0.007 (10%) or 0.074“ or less than this value, the flow-path or component may be blocked. This is consistent with Reference 3-4. Components that are in the flow-paths during accidents are listed in Table 3.2-1.

Piping

The piping, by design, minimizes low flow areas. The system low points are at the RHR suction. The fluid is generally fully turbulent, lessening the possibility and probability of debris settling.

Pipe diameters are significantly larger than the strainer hole size. Flow velocities in all cases are above the settling velocities of the post-LOCA fluid. Refer to Table 3.2-6. Debris settling is a longer term phenomena and has no short term impact on flow. Therefore, the potential of piping plugging or blockage and it’s impact on system operation is very low. Reliability of the SIS is considered in the design, procurement, and installation/layout of components. DCD Chapter 17 discusses Quality Assurance (QA) during design, construction and operation. Verification that the system valves will meet the fluid wear resistance requirements is considered part of the COL (Ref. 3-11, Subsection 17.4.9).

Valves

The valve types that are used in the flow-path during an accident are gate, check, globe and butterfly valves, see Table 3.2-1.

Gate valves

Gate valves are used full-open or full-close. In the US-APWR, gate valve sizes are above 4'', see Table 3.2-1. Flow velocities in all cases are above the settling velocities of the post-LOCA fluid. Refer to Table 3.3-2. Reference 3-7, NUREG/CR-6902, states that valve openings significantly larger than the debris size will not clog. The strainer hole size is 0.066". The 4" valve opening is considerably larger than any expected particle passes through the sump strainer. Therefore, the valves will not clog due to post-LOCA insulation debris.

Check valves

Check valves in the US-APWR are used with sufficient flow rate, and check valve sizes are above 4'', see Table 3.2-1. Flow velocities in all cases are above the settling velocities of the post-LOCA fluid. Refer to Table 3.3-2. Reference 3-7, NUREG/CR-6902, states that valve openings significantly larger than the debris size will not clog. The strainer hole size is 0.066". The 4" valve opening is considerably larger than any expected particle passes through the sump strainer. Therefore the valves will not clog due to post-LOCA insulation debris.

Globe valves

ECCS and CSS flow is controlled though a combination of orifices and throttled valves. Globe valves normally are full open but may be used for throttling system flow. ECCS and CSS pressure and flow are monitored in the MCR. In general, if a globe valve is in a throttled position and it begins to clog, system flow will decrease. Operator action may be taken to open the valve, thus clearing the potential clog. In the US-APWR, globe valve sizes are above 2'', see Table 3.2-1. Flow velocities in all cases are above the settling velocities of the post-LOCA fluid. Refer to Table 3.2-6. Reference 3-7, NUREG/CR-6902, states that valve openings significantly larger than the debris size will not clog. The strainer hole size is 0.066". Throttle valves are expected to be throttled to a minimum of 1" open between the valve disc and seat. The 1" valve opening is considerably larger than any expected particle passes through the sump strainer. Therefore the valves will not clog due to post-LOCA insulation debris.

Butterfly valves

Butterfly valves are used at the outlet of CS/RHR heat exchanger. These valves are used full open and valve sizes are 8''. Flow velocities in all cases are above the settling velocities of the post-LOCA fluid. Refer to Table 3.2-6. Reference 3-7, NUREG/CR-6902, states that valve openings significantly larger than the debris size will not clog. The strainer hole size is 0.066". The 8" butterfly valve opening is considerably larger than any expected particle passes through the sump strainer. Therefore the valves will not clog due to post-LOCA insulation debris.

Orifice

ECCS and CSS flow is controlled though a combination of orifices and throttled valves. Orifices are used for throttling system flow. ECCS and CSS pressure and flow are monitored on the MCB. In the US-APWR, orifice sizes are above ½". Flow velocities in all cases are above the settling velocities of the post-LOCA fluid (Table 3.3-2). Therefore, the potential of orifice plugging is very low.

Spray Nozzles

The containment spray nozzles have an inlet orifice 0.375" in diameter. This orifice is the smallest portion of spray nozzle. The strainer hole size is 0.066". Reference 3-7, NUREG/CR-6902, states that valve openings significantly larger than the debris size will not clog. Containment spray nozzles are significantly larger than the strainer hole size. Their one-piece design provides a large, unobstructed flow passage that resists clogging by particles. Therefore, the potential of spray nozzle plugging is very low.

3.3.5 Instrument Clogging Evaluation

Per the USNRC review guidance (Ref. 3-3) instrument connection should be reviewed to determine their susceptibility to clog or plug. Reliability of the ECCS and CSS is considered in the design, procurement, and installation/layout of components. All connections, by design, are either at the horizontal or above. Flow velocities in all cases are above the settling velocities of the post-LOCA fluid (Table 3.3-2). Therefore, the potential for instrument and instrumentation tubing plugging is very low.

3.3.6 Chemical Effects Evaluation

Precipitants and other chemical forms present as a result of the chemical effects testing have no effect on the plugging or wear evaluations.

Chemicals and precipitants are typically soft, non-abrasive, low-shear and readily stay in solution due to the fully developed turbulent flow conditions present within the piping system(s). As such, they do not contribute to plugging or change wear characteristics of piping, pump, heat exchangers or valves downstream of the containment sump (Ref. 3-13 and 3-14).

3.4 ECCS and CSS Pump Evaluations

Reliability of the SIS is considered in the design, procurement, and installation/layout of components.

3.4.1 Review of Design/ Procurement Specification(s)

3.4.1.1 ECCS Pumps

Engineering Design Requirements for the ECCS pumps are contained in Appendix A.

The SI pumps are motor-driven horizontal, multistage, centrifugal pumps with mechanical seals. The pumps are sized to deliver 1,540 gpm at a discharge head of 2,756 ft.

The ECCS pumps will be specified to meet the intent of American Petroleum Institute (API) Standard 610 (Ref. 3-17) in the context of rotor dynamic analysis. API-610 provides a standard analysis method for severe service pumps and is a recognized design standard for the design and operation of centrifugal pumps for petroleum, heavy duty chemical and gas industry services (Ref. 3-4). Details will be provided in the procurement specifications. Verification that specification requirements are considered part of the COL (Ref. 3-11, Subsection 17.4.9). Confirmation Item 3.7.4.

The mission time for the ECCS, including post-LOCA long-term operation is defined as 30 days.

The fluid characteristics listed in Table 3.2-3, 4 and 5 conservatively represent the post-LOCA fluid conditions that an SI pump will experience. At the procurement stage, the pump vendor will provide a table listing the materials and hardness's of all wetted pump surfaces (wear rings, pump internals, bearings, casings, etc). Confirmation Item 3.7.5.

At the procurement stage, the pump vendor will provide a table listing the design or specified opening sizes and internal running clearances. Confirmation Item 3.7.5.

MHI does not intend to specify equipment strainers, cyclone separators or other filtering components as part of the pump design or procurement specification. If a pump vendor supplies a pump with such equipment, the manufacturer will be required to confirm that stated that their design is not susceptible to plug or clog during post-LOCA long-term operation and that there will be no negative impact on pump performance or reliability. - Confirmation Item 3.7.5.

The pump purchase specification will state that there will be no changes in system or equipment operation caused by wear (i.e. pump vibration and rotor dynamics) such that the pump will not be able to perform as specified.   Confirmation Item 3.7.5.

Rotor dynamic studies and bearing load models will be required to be submitted as part of the purchase specification and procurement documents. The pump vendor will confirm that internal bypass flow increases due to impellor or casing wear will not affect the ability of the pump to meet its intended function.   Confirmation Item 3.7.5.

3.4.1.2 CSS Pumps

Engineering Design Requirements for the CSS pumps are contained in Appendix B.

The CS/RHR pumps are motor-driven centrifugal pumps with mechanical seals. The pumps are sized to deliver 3,000 gpm at a discharge head of 410 ft. The 100% capacity design flow rate (two of four 50% capacity CS/RHR pumps) is based upon a 15.2 gpm flow per nozzle and 348 nozzles in the ring header. The CS/RHR pumps will be specified to meet the intent of American Petroleum Institute (API) Standard 610 (Ref. 3-17) in the context of rotor dynamic analysis. API-610 provides a standard analysis method for severe service pumps and is a recognized design standard for the design and operation of centrifugal pumps for petroleum, heavy duty chemical and gas industry services (Ref. 3-4). Details will be provided in the procurement specifications.   Verification that specification requirements are considered part of the COL (Ref. 3-11, Subsection 17.4.9).   Confirmation Item 3.7.4

The mission time for the CSS, including post-LOCA long-term operation is defined as 30 days.

The fluid characteristics listed in Table 3.2-3, 4 and 5 conservatively represent the post-LOCA fluid conditions that a CS/RHR pump will experience. At the procurement stage, the pump vendor will provide a table listing the materials and hardness's of all wetted pump surfaces (wear rings, pump internals, bearings, casings, etc).   Confirmation Item 3.7.5.

At the procurement stage, the pump vendor will provide a table listing the design or specified opening sizes and internal running clearances. Confirmation Item 3.7.5.

At the procurement stage, the pump vendor will provide a table listing the design or specified opening sizes and internal running clearances. Confirmation Item 3.7.5.

MHI does not intend to specify equipment strainers, cyclone separators or other filtering components as part of the pump design or procurement specification. If a pump vendor supplies a pump with such equipment, the manufacturer will be required to confirm that their design is not susceptible to plug or clog during post-LOCA long-term operation and that there will be no negative impact on pump performance or reliability. - Confirmation Item 3.7.5.

The pump purchase specification will state that there will be no changes in system or equipment operation caused by wear (i.e. pump vibration and rotor dynamics) such that the pump will not be able to perform as specified. Confirmation Item 3.7.5.

Rotor dynamic studies and bearing load models will be required to be submitted as part of the purchase specification and procurement documents. The pump vendor will confirm that internal bypass flow increases due to impellor or casing wear will not affect the ability of the pump to meet its intended function. Confirmation Item 3.7.5.

3.4.2 Affects of Air Entrainment

The scope of this evaluation does not include vortexing at the strainer or system suction. This evaluation discusses the ability of the ECC and CSS pumps to operate under voided conditions.

Reliability of the ECCS and CSS are considered in the design, procurement, and installation/layout of components. DCD Chapter 17 discusses Quality Assurance (QA) during design, construction and operation. Verification that programmatic controls are in place regarding air entrainment or that design specifications refer to and specify high point vents, continually up sloping lines etc is considered part of the COL (Ref. 3-11, Subsection 17.4.9 and Ref. 3-6).

ECCS / CSS pump suction lines are designed to prevent degradation of pump performance through air ingestion and other adverse hydraulic effects (e.g., circulatory flow patterns, high intake head losses).

RWSP suction strainers are submerged under a minimum of approximately 4 ft. of water during a LOCA. The RWSP recirculation supply is sufficient to preclude adverse hydraulic effects (e.g., vortex formation and high suction head loss). A low approach velocity at the strainer surface also mitigates the risk of vortexing.

3.4.3 Seal Leakage

Both the ECCS and CS/RHR pumps are specified to maintain a leak rate of less than [ ]. Under complete seal failure conditions, the leak rate is specified to be less than 50 gpm. The pump seal vendor will confirm that their design meets or exceeds these conditions. Confirmation item 3.7.6.

The CSS is provided with a leakage detection system to minimize the leakage from those portions of the system outside of the containment that contain or may contain radioactive material following an accident.

A pit (sump) with a leak detector installed in each safeguard component area and alarms to MCR to prevent significant leakage of radioactive recirculation water from the high head injection system to the reactor building. The high head injection system is designed with sufficient redundancy and independence to prevent loss of core cooling function during an accident assuming the isolation of the leaked train after leakage is detected.

The [ ] leak rate is bounded by the Environmental Qualification (EQ) profile.

3.4.4 Chemical Effects Evaluation

Precipitants and other chemical forms present as a result of the chemical effects testing have no effect on the plugging, wear or pump performance evaluations.

Chemicals and precipitants are typically soft, non-abrasive, low-shear and readily stay in solution due to the fully developed turbulent flow conditions present within a pump. As such, they do not contribute to plugging or change wear characteristics (Ref. 3-13 and 3-14).

3.5 ECCS and CSS Performance Evaluations

3.5.1 ECCS Performance Evaluation

Based upon the piping wear and pump operation evaluations, it is concluded that the system piping and component flow resistances will change minimally during the course of the LOCA. Therefore flow balances and system performance is not affected in an appreciable manner.

The resulting flows and pressures are consistent or conservative with respect to the accident analysis. The minor resistance changes do not affect the system flow calculations and Design Bases analysis.

The ECCS contains instrumentation to monitor system performance. The following system parameters are monitored in the MCR:

1. SI pump discharge flow rate
2. SI pump minimum flow rate
3. SIS flow rate
4. SI pump suction and discharge pressure indication

Motor operated valves on the SI pump discharge may be throttled as needed to maintain an SI pump in the desired operating range. Therefore, any changes as a result of long-term wear (30 days) or operation will be detected and system performance adjusted as needed.

3.5.2 CSS Performance Evaluation

Based upon the piping wear and pump operation evaluations, it is concluded that the system piping and component flow resistances will change minimally during the course of the LOCA. Therefore flow balances and system performance is not affected in an appreciable manner.

The resulting flows and pressures are consistent or conservative with respect to the accident analysis. The minor resistance changes do not affect the system flow calculations and Design Bases analysis

The CSS/ RHRS contains instrumentation to monitor system performance. The following system parameters are monitored in the MCR:

1. CS/RHR pump discharge flow rate
2. CS/RHR pump minimum flow rate
3. CS/RHR flow rate
4. CS/RHR heat exchanger inlet and outlet temperature indication
5. CS/RHR pump suction and discharge pressure indication

Therefore, any changes as result of long-term wear (30 days) or operation will be detected.

3.6 Regulatory Summary

Generic Letter (GL) 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors (Ref. 3-1) was issued by the USNRC requesting holders of operating reactor licenses to evaluate their emergency core cooling system (ECCS) and containment spray system (CSS) recirculation functions in light of events regarding the blockage of containment sump strainers. The GL notes that:

“Debris could also plug or wear close-tolerance components within the ECCS or CSS systems. This plugging or wear might cause a component to degrade to the point where it could not perform its designated function (i.e., pump fluid, maintain system pressure, or pass and control system flow.)

Third, debris blockage at flow restrictions within the ECCS recirculation flow-path downstream of the sump screen is a potential concern for PWRs. Debris that is capable of passing through the recirculation sump screen may have the potential to become lodged at a downstream flow restriction, such as a high-pressure safety injection (HPSI) throttle valve or fuel assembly inlet debris screen. Debris blockage at such flow restrictions in the ECCS flow-path could impede or prevent the recirculation of coolant to the reactor core, thereby leading to inadequate core cooling. Similarly, debris blockage at flow restrictions in the CSS flow-path, such as a containment spray nozzle, could impede or prevent CSS recirculation, thereby leading to inadequate containment heat removal. Debris may also accumulate in close-tolerance subcomponents of pumps and valves. The effect may be either to plug the subcomponent, thereby rendering the component unable to perform its function, or to wear critical close tolerance subcomponents to the point at which component or system operation is degraded and unable to fully perform its function. Considering the recirculation sump screen’s design function of intercepting potentially harmful debris, it is essential that the screen openings be adequately sized and that the sump screen’s current configuration be free of gaps or breaches which could compromise the ECCS and CSS recirculation functions. It is also essential that system components be designed and evaluated to be able to operate as necessary with debris laden fluid post-LOCA”

The GL requested the following specific information be provided.

“(v) The basis for concluding that inadequate core or containment cooling would not result due to debris blockage at flow restrictions in the ECCS and CSS flow-paths downstream of the sump screen, (e.g., a HPSI throttle valve, pump bearings and seals, fuel assembly inlet debris screen, or containment spray nozzles). The discussion should consider the adequacy of the sump screen's mesh spacing and state the basis for concluding that adverse gaps or breaches are not present on the screen surface.

(vi) Verification that close-tolerance subcomponents in pumps, valves and other ECCS and CSS components are not susceptible to plugging or excessive wear due to extended post-accident operation with debris-laden fluids"

In response to the GL, the Nuclear Energy Institute, NEI, issued Guidance Report NEI 04-07 delineating a generic, consistent approach to address the NRC concerns identified in GL 2004-02. The USNRC, in turn, issued a Safety Evaluation clarifying those items to be addressed and endorsing an approach that would be acceptable to the USNRC staff (Ref. 3-2).

The USNRC further clarified specific areas to be addressed with the issuance of an NRC letter entitled "Audit Plan for Verifying the Adequacy of Licensee Responses to Generic Letter 2004-02" (Ref. 3-3). This audit plan was intended to fully address the USNRC concerns identified in Reference 3-2, Final Safety Evaluation for NEI Guidance Report 04-07, regarding the evaluation of ex-vessel components downstream of the containment sump during post-LOCA operation. There are no other regulatory guidance documents that specifically pertain to the evaluation of ex-vessel downstream components. Therefore, addressing the issues identified in the "Audit Guidelines" will fully address the concerns identified in the GL.

The sections of Table 3.6-1 are excerpted from the Audit Guidelines (Ref. 3-3). Following each section is a reference to the report section where it is addressed and a summary of evaluation.

3.7 Confirmation Items

Reliability of the ECCS and CSS are considered in the design, procurement, and installation/layout of components. Verification that the system components will meet their design specifications is considered part of the COL (Ref. 3-11, Subsection 17.4.9).

3.7.1 Verify the “as procured” Safety Injection Pump runout flow is less than 2,000 gpm.

3.7.2: Verify the “as procured” CS/RHR Pump runout flow is less than 4,000 gpm.

3.7.3 Valve and valve trim materials will be specified to be wear resistant. The valve procurement specification will note the constituents of the post-LOCA fluid and require that the valve be able to operate reliably under those conditions for a minimum of 30 days.

3.7.4 The ECCS and CSS pumps will be specified to meet the intent of American Petroleum Institute (API) Standard 610 in the context of rotor dynamic analysis. API-610 provides a standard analysis method for severe service pumps and is a recognized design standard for the design and operation of centrifugal pumps for petroleum, heavy duty chemical and gas industry services. Details will be provided in the procurement specifications.

3.7.5 ECCS and CSS pump wetted materials and seals will be specified to be wear resistant. The pump(s) procurement specification will note the constituents of the post-LOCA fluid and require that the pump(s) be able to operate reliably under those conditions for a minimum of 30 days. The pump vendor will supply a list of materials, material hardness and design and maximum running clearances.

3.7.6 The ECCS and CS/RHR pumps seal specification will state that the seals be designed to maintain a leak rate of less than [ ]. Under complete seal failure conditions, the leak rate will be specified to be less than 50 gpm.

3.8 Summary of Results

The intent of this technical report is to assess the US-APWR Emergency Core Cooling (ECC) and Containment Spray (CSS) systems to ensure that that these systems and their components will operate as designed under post Loss-of Coolant Accident Conditions (LOCA).

This review incorporates the lessons learned as part of the USNRC Generic Safety Issue 191 (GSI-191) and addresses the concerns identified in Generic Letter (GL) 2002-04 (Ref.3-1). This report has been prepared in accord with NEI 04-07 (Ref. 3-16) and published USNRC staff expectations. Information and guidance used has been extracted from the noted references and adapted to the US-APWR design.

This report concludes that the US-APWR ECC / CS Systems and components are fully capable of performing their intended functions under post-LOCA operating conditions with regard to ex-vessel downstream effects.

Table 3.2-1 Components in the Flow Path during a LBLOCA

Components	Remark
Pumps	
SIS-RPP-001A,B,C,D	Multi-stage centrifugal type
RHS-RPP-001A,B,C,D	Centrifugal type
Heat Exchangers	
RHS-RHX-001A,B,C,D	Shell & tube type
Valves	
SIS-MOV-001A,B,C,D	Gate, 10’’
SIS-VLV-004A,B,C,D	Check, 4’’
SIS-MOV-009A,B,C,D	Gate, 4’’
SIS-VLV-010A,B,C,D	Check, 4’’
SIS-MOV-011A,B,C,D	Globe, 4’’
SIS-VLV-012A,B,C,D	Check, 4’’
SIS-VLV-013A,B,C,D	Check, 4’’
SIS-MOV-014A,B,C,D	Globe, 4’’
SIS-VLV-015A,B,C,D	Check, 4’’
SIS-VLV-023A,B,C,D	Globe, 2’’
CSS-MOV-001A,B,C,D	Gate, 14’’
CSS-VLV-002A,B,C,D	Gate, 10’’
CSS-MOV-004A,B,C,D	Gate, 8’’
CSS-VLV-005A,B,C,D	Check, 8’’
RHS-VLV-004A,B,C,D	Check, 16’’
RHS-VLV-013A,B,C,D	Globe, 3’’
RHS-HCV-603	Butterfly, 8’’
RHS-HCV-633	Butterfly, 8’’
RHS-MOV-021A,B,C,D	Gate, 8’’
RHS-VLV-022A,B,C,D	Check, 8’’
RHS-MOV-025A,B,C,D	Globe, 8’’
Orifice	
SI pump outlet flow instrument orifice SI pump minimum flow orifice Direct vessel injection line orifice Hot leg injection line orifice CS/RHR pump outlet flow instrument orifice CS/RHR pump minimum flow instrument orifice CS/RHR pump minimum flow line orifice Containment spray ring orifice	Hole size of 1/2” is assumed as the smallest of these orifices.
Spray Nozzle	
Containment Spray Nozzle	Orifice size 0.375 in.

Table 3.2-2 Material Hardness Data

Material	**Grade or Type**	**Component**	**Brinell Hardness (BHN)**
SA-240 SA-479	304/L/LN Stainless Steel	CS nozzles, Valve Bonnets, Valve Disks	201
SA-182	F304/L/LN Stainless Steel	Valve Bodies, Valve Bonnets, Valve Disks,	201
SA-240 SA-312 SA-376 SA-479	316/L/LN Stainless Steel	Valve Bonnets, Valve Disks, Valve Stems	217
SA-182 SA-336	F316/L/LN Stainless Steel	Valve Bodies, Valve Bonnets, Valve Disks, Main Coolant piping,	217
SA-564	630	Valve Disks, Valve Stems	255
Material	**Grade or Type**	**Pipe**	**Brinell Hardness (BHN)**
SA-358	304 STD	SIS-151	201
SA-312	304(SML)	SIS-1501, SIS-2511, RHS-901R, RHS-2511R, CSS-901, CSS-301	201
SA-312	316(SML)	SIS-2501R, RHS-2501R	217

Table 3.2-3 Debris Source Term

Debris Type	Debris Quantity	Fabricated Density	Material Density	Characteristic Size
RMI	106 ft^3 11,442 ft^2 foil surface area	N/A	490*	0.003 ft
Nukon	46 ft^3	2.4 lbm/ft^3	159 lbm/ft^3	7μm Diameter S_v = 1.742e-5/ft
Epoxy Coatings	18 ft^3	19** lbm/ft^3	94 lbm/ft^3	10μm Diameter S_v = 1.829e-5/ft
Latent Fiber	30 lbm	2.4 lbm/ft^3	159 lbm/ft^3	7μm Diameter S_v = 1.742e-5/ft
Latent Particle	170 lbm	75** lbm/ft^3	168.6 lbm/ft^3	S_v = 1.742e-5/ft

Note: * RMI density is assumed to be that of common stainless steel
** Sludge Density

Table 3.2-4 Debris Concentration Components

TYPE	Quantity	Density	Mass (lbs)
NUKON	23 ft^3	2.4 (lb/ft^3)	55.2
Latent fiber	30 lbm	-	30
Epoxy coatings	18 ft^3	94	1692
Latent particle	170 lbm	-	170
RMI	5.3 ft^3	490	2597
			SUM
			4544.2

Table 3.2-5 Material Wear Rates (Ref. 3-19)

	Wear Rate (inches/year)			
	Coarse Sand		Fine Sand	
Material	7 ft/s	15 ft/s	7 ft/s	15 ft/s
Steel	0.0256	0.0713	0.0016	0.0008
Aluminum	0.0713	0.2945	0.0055	0.0349
Polyethylene	0.0024	0.0181	0	0.0024
ABS	0.0142	0.0815	0.0028	0.0201
Acrylic	0.0390	0.1614	0.0067	0.0559
Geometric Mean of wear ratio	4.618		25.855	

Table 3.2-6 Affected Equipment / Flow Rates

Components	Inner Diameter (inches)	Design Flow Rate (gpm)	Assumed Flow Rate (gpm)	Assumed Flow Rate (ft^3/s)	Assumed Velocity (ft/s)	Reference
Sump Screen						
	0.066					MUAP 08001
Orifice						
Flow instrument orifice on 4" SIS Line 1501	2	1540	2000	4.456	26.685	DCD Fig. 6.3-2
Orifice on 4" SIS Line 2511	2	1540	2000	4.456	26.685	DCD Fig. 6.3-2
Orifice on 4" SIS Line 2501	2	1540	2000	4.456	26.685	DCD Fig. 6.3-2
Flow instr. orifice on 10" RHRS Line 901	7.5	3650	4000	8.913	14.26	DCD Fig. 6.2.2-1
Spray ring orifice on 8" CSS Line 301	6	3650	4000	8.913	17.825	DCD Fig. 6.2.2-1
Spray Nozzle						
Containment Spray Nozzle	0.375	20.98*	22.99	0.051	66.79	DCD 6.2.2.2.4
Piping						
10" SIS Line 151R (SS Sch 160)	8.500	1540	2000	4.456	6.292	Pipe Material Sheet / Crane No. 410
4" SIS Line 1501R (SS Sch 160)	3.438	1540	2000	4.456	15.554	Pipe Material Sheet / Crane No. 410
4" SIS Line 2511R (SS Sch 160)	3.438	1540	2000	4.456	15.554	Pipe Material Sheet / Crane No. 410
4" SIS Line 250R (SS Sch 160)	3.438	1540	2000	4.456	15.554	Pipe Material Sheet / Crane No. 410
16" RHS Line 901R (SS Sch 80)	14.312	3650	4000	8.913	7.978	
10" RHS Line 901R (SS Sch 80)	9.562	3650	4000	8.913	17.873	
16" CSS Line 901 (SS Sch 80)	14.312	3650	4000	8.913	7.978	Pipe Material Sheet / Crane No. 410
14" CSS Line 901 (SS Sch 80)	12.500	3650	4000	8.913	10.459	Pipe Material Sheet / Crane No. 410
10" CSS Line 901 (SS Sch 80)	9.562"	3650	4000	8.913	17.873	Pipe Material Sheet / Crane No. 410
8" CSS Line 901 (SS Sch 80)	7.625	3650	4000	8.913	28.107	Pipe Material Sheet / Crane No. 410
8" CSS Line 301 (SS Sch 40S)	7.981	3650	4000	8.913	25.655	Pipe Material Sheet / Crane No. 410
6" CSS Line 301 (SS Sch 40S)	6.065	1825	4000	8.913	44.426	Pipe Material Sheet / Crane No. 410

*The CS nozzle flow rate is the maximum flow rate of both CS/RHR pumps divided by 348 nozzles.

Table 3.3-1 ECCS and CSS Components Wear vs. Time

	30 DAYS	
Components	**Dia-metrical Wear (inches)**	**Flow Rate Increase (%)**
Orifice		
Flow instrument orifice on 4” SIS Line 1501		
Orifice on 4” SIS Line 2511		
Orifice on 4” SIS Line 2501		
Flow instr. orifice on 10” RHRS Line 901		
Spray ring orifice on 8” CSS Line 301		
Spray Nozzle		
Containment Spray Nozzle		
Piping		
10” SIS Line 151 (SS Sch 160)		
4” SIS Line 1501 (SS Sch 160)		
4” SIS Line 2511 (SS Sch 160)		
4” SIS Line 2501 (SS Sch 160)		
16” RHS Line 901R (SS Sch 80)		
10” RHS Line 901R (SS Sch 80)		
16” CSS Line 901 (SS Sch 80)		
14” CSS Line 901 (SS Sch 80)		
10” CSS Line 901 (SS Sch 80)		
8” CSS Line 901 (SS Sch 80)		
8” CSS Line 301 (SS Sch 40S)		
6” CSS Line 301 (SS Sch 40S)		

Table 3.3-2 ECCS and CSS Settling Velocities

Components	Design Flow Rate (gpm)	Assumed Flow Rate (gpm)	Assumed Velocity (ft/s)	Maximum Settling Velocity (ft/s)	Conclusion
Orifices					
Flow instrument orifice on 4” SIS Line 1501	265	200	6.913	0.37	Debris settling will not occur
Orifice on 4” SIS Line 2511	265	200	6.913	0.37	Debris settling will not occur
Orifice on 4” SIS Line 2501	265	200	6.913	0.37	Debris settling will not occur
Flow instr. orifice on 10” RHRS Line 901	355	300	1.340	0.37	Debris settling will not occur
Spray ring orifice on 8” CSS Line 301	355	300	3.404	0.37	Debris settling will not occur
Spray Nozzle					
Containment Spray Nozzle	1.02	0.862	2.504	0.37	Debris settling will not occur
Piping					
10” SIS Line 151R (SS Sch 160)	265	200	1.131	0.37	Debris settling will not occur
4” SIS Line 1501R (SS Sch 160)	265	200	6.913	0.37	Debris settling will not occur
4” SIS Line 2511R (SS Sch 160)	265	200	6.913	0.37	Debris settling will not occur
4” SIS Line 250R (SS Sch 160)	265	200	6.913	0.37	Debris settling will not occur
16” RHS Line 901R (SS Sch 80)	355	300	0.598	0.37	Debris settling will not occur
10” RHS Line 901R (SS Sch 80)	355	300	1.340	0.37	Debris settling will not occur
16” CSS Line 901 (SS Sch 80)	355	300	0.598	0.37	Debris settling will not occur
14” CSS Line 901 (SS Sch 80)	355	300	0.784	0.37	Debris settling will not occur
10” CSS Line 901 (SS Sch 80)	355	300	1.340	0.37	Debris settling will not occur
8” CSS Line 901 (SS Sch 80)	355	300	2.108	0.37	Debris settling will not occur
8” CSS Line 301 (SS Sch 40S)	355	300	1.924	0.37	Debris settling will not occur
6” CSS Line 301 (SS Sch 40S)	355	300	3.332	0.37	Debris settling will not occur

Table 3.6-1 Ex-Vessel Downstream Effects Regulatory Review

Section 13, Downstream Effects		Response
i.	Review the list of all components and flowpaths considered to determine the scope of the licensee's downstream evaluation (pumps, valves, instruments, and heat exchangers, etc).	Report Sections 3.2.1, 3.2.3 and Figure 3.1-1 and Table 3.2-1 list components and flow-paths.
ii.	Review design and license mission times and system lineups to support mission-critical systems.	Report Section 3.2.2. Design Bases mission time is 30 days for all components.
iii.	Evaluate the vulnerability of the high-pressure safety injection (HPSI) throttle valves to clogging by determining the HPSI system's use?	Report Sections 3.3.3 and 3.3.4. HHSI valves are not vulnerable to clogging due to open design.
iv.	Assess whether the leakage through seals, etc., would increase local dose rates so that credited operator actions, if any, cannot be met.	Report Sections 3.4.1 and 3.4.3. Seals will be specified and procured to meet assumed leakage rates.
v.	Review all LOCA scenarios (i.e., small-break LOCA, medium-break LOCA, and large break LOCA) to assess system operation. For a large-break LOCA or medium-break LOCA, some plants may not need and/or use the HPSI system.	Report Section 3.2.1. LBLOCA is the limiting scenario.
vi.	Review the licensee's evaluation of the extent of air entrainment. Licensee evaluation should include review of plant operating experience. Apart from vortexing, this involves ongoing questions about ECCS and incident report evaluation on the significance of ECCS gas intrusion.	Report Section 3.4.2. System design precludes the formation of gas pockets. Programmatic controls regarding construction and surveillances to assess air ingestion will be submitted as part of the COL.
vii.	Review the characterization and properties of ECCS post-LOCA fluid (abrasiveness, solids content, and debris characterization).	Report Section 3.2.4 and Table 3.2-3, 4 and 5.

viii.	Review the materials of all wetted downstream surfaces (wear rings, pump internals, bearings, throttle valve plug, and seat materials).	Report Sections 3.3.1 and 3.4.1. Wetted materials will be specified and procured to meet wear resistance requirements.
ix.	Review the opening sizes and running clearances in pumps and valves.	Report Sections 3.3.1, 3.3.3 and 3.3.4 for valves. There are no tight clearance valves. Section 3.4.1 addresses pumps. Opening sizes and running clearance will be provided as part of the pump procurement specification.
x.	Review the list of system low points and low-flow areas.	Report Sections 3.3.4 and 3.4.2. Flow velocities in all cases are above the settling velocities of the post-LOCA fluid. System design precludes the formation of gas pockets. Programmatic controls regarding construction and surveillances to assess air ingestion will be submitted as part of the COL.
xi.	Review the range of fluid velocities within piping systems. What is the minimum velocity used to assess settling? What is the maximum velocity used to assess wear?	Report Section 3.2.5 and Table 3.3-2. System flows in excess of pump runout are assumed for wear rate calculations. Less than design flow is assumed for settling evaluations.
xii.	Review the presence and evaluation of equipment strainers, cyclone separators, and other components.	Report Section 3.4.1. MHI does not intend to specify equipment strainers, cyclone separators or other filtering components as part of the pump design or procurement specification. If a pump vendor supplies a pump with such equipment, the manufacturer will be required to confirm that stated that their design is not susceptible to plug or clog during post-LOCA long-term operation and that there will be no negative impact on pump performance or reliability.
xiii.	Review the assessment of changes in system or equipment operation caused by wear (i.e., pump vibration and rotor dynamics). Assess whether the internal bypass flow increases, thereby decreasing performance or accelerating internal wear.	Report Section 3.4.1. A wear assessment and a rotor dynamics study will be part of the pump procurement specification.

xiv. Assess whether the system, piping, or component flow resistance changed, altering flow balances.	Report Section 3.5. There are no significant system, piping, or component resistance changes such that flow balances will be appreciably altered.
xv. Assess whether the system piping vibration response changed for any of the above reasons.	Report Section 3.5. There are no significant system, piping, or component resistance changes such that system piping vibration is appreciably altered.
xvi. Review the listing and evaluation of instrument tubing connections.	Report Section 3.3.5 and Table 3.3-2. All instrument connections, by design, are either at the horizontal or above. Programmatic controls regarding construction will be submitted as part of the COL.
xvii. Review ECCS heat exchanger design to identify those with small (i.e., 3/8" or less) tubes and for which the ECCS is on the tube side. What are the clearances and the potential for fouling?	Report Section 3.3.2. Heat exchanger tubing is ¾" OD. The CS/RHR Heat Exchanger has been specified with a .0005 fouling factor. Fouling is a long term phenomena, CS/RHR heat exchangers are sized considering maximum heat load including fouling.
Section 14. Chemical Effects (partial)	**Response**
ix. Verify the licensee has considered potential downstream effects related to chemical by-product formation.	Report Section 3.4.4. There are no adverse affects on ECCS or CSS pumps, valves, heat exchangers or piping components as a result of chemical precipitants in the post-LOCA fluid.

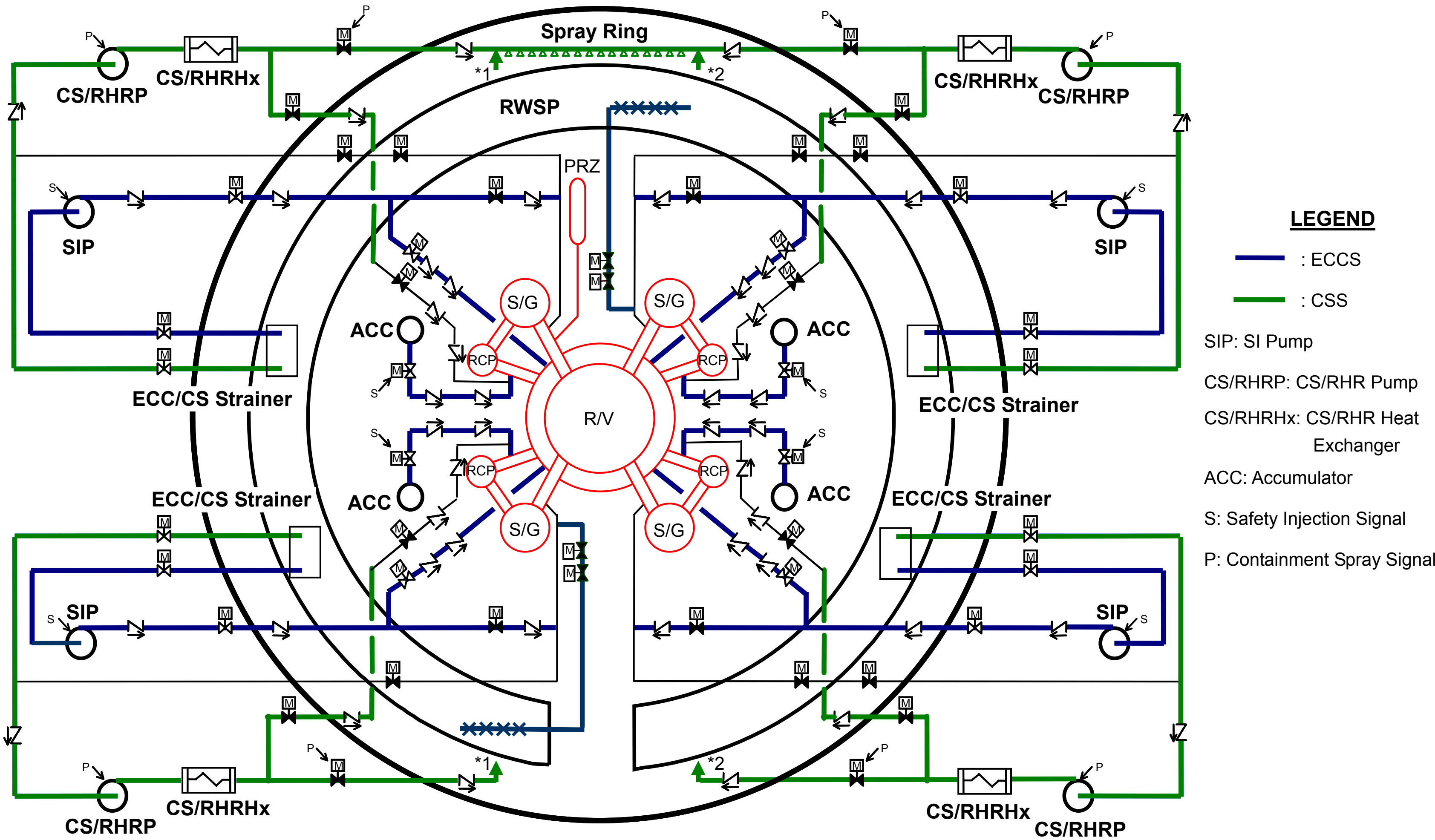

Figure 3.1-1 Schematic flow diagram of ECCS/CSS (Ref. 3-12)

4.0 IN-VESSEL DOWNSTREAM EFFECTS

4.1 Blockage at the Core Inlet

4.1.1 Effect of Blockage of Core Coolant Flow

The following sequence is the US-APWR core cooling path flows in the reactor vessel downstream of the sump strainer. Cooling water will:

1. Come in from ECCS nozzle
2. Pass through the downcomer which is annulus between the reactor vessel and the core barrel
3. Pass through the lower plenum
4. Pass through the flow holes of the lower core support plate
5. Pass through the fuel assemblies
6. Pass through the holes of the upper core plate
7. Flows out from outlet nozzle

The smallest flow hole in the core cooling flow route of the reactor internals is that of the flow holes of the lower core support plate whose size is [ ]. The flow hole of the bottom nozzle in the fuel assembly is [ ]. This is the narrowest gap downstream of the strainer to core inlet, and dictates that the nominal diameter of the strainer holes shall be sufficiently smaller than this gap. [ ] percent margin was considered to limit the debris which may pass through this gap, and no larger than 0.071” (1.8mm) of debris are blocked at the strainer. Finally, the strainer supplier’s standard perforated plate with 0.066” (1.67mm) was selected to the US-APWR strainer specification.

The flow hole of the lower core support plate is over [ ] times the size of the strainer holes. Therefore it is not necessary to consider piling up the downstream debris at any flow paths in the reactor internals. The flow hole of the fuel assembly bottom nozzle is [ ] times the size of the strainer holes. Therefore, it is quite unlikely that the downstream debris may pile up at the fuel assembly bottom nozzle.

4.1.2 Evaluation of Blockage at the Core Inlet

An analysis of the blockage at the core inlet is performed to evaluate the effects of the core inlet blockage due to bypass debris on long-term core cooling (LTCC). The blockage is assumed to deterministically occur at the core inlet. The objective of the analysis is to show that sufficient coolant can enter the core to remove the core decay heat to assure acceptable cladding temperature when the core inlet blockage up to 99.6% occurs.

4.1.2.1 Objective

The objective of this evaluation is to confirm that sufficient long-term core cooling (LTCC) is achieved to satisfy the requirements of 10CFR 50.46. The flow at the core inlet can be suppressed due to the built-up debris penetrating the sump strainer, and to the lower core support plate and the fuel assembly bottom nozzle by ECCS. Therefore WCOBRA/TRAC(M1.0) simulations are run to show LTCC will be maintained in a situation of the blockage at the core inlet by increasing the k-factor to simulate the increment of debris at the core inlet.

4.1.2.2 Approach

In order to enable the increase in k-factor (C_D) in WCOBRA/TRAC(M1.0) analysis, the code are modified and the core inlet blockage due to debris is simulated. In the analysis, it is conservatively evaluated that the blockage at the core inlet will start at the time of 850 seconds after the LOCA (Appendix-E), and that k-factor (C_D) is ramped to 10^9 over the time period of 30 seconds after the blockage starts. The postulated core blockage is modeled in this method. The analysis by WCOBRA/TRAC(M1.0) is carried out for 3000 seconds after the LOCA in order to show that sufficient flow which maintains decay heat removal and coolable core geometry is supplied to the core.

4.1.2.3 Model Description and Assumptions

This section discusses the nodalization and assumptions used in the core inlet blockage calculation by WCOBRA/TRAC(M1.0).

4.1.2.3.1 Nodalization

The nodalization of US-APWR for the core inlet blockage calculation using WCOBRA/TRAC(M1.0) is the same as that described in Reference 4.1-1 and is briefly described as follows.

(1) Vessel Model

(2) Core Model

(3) **Loop Model**

4.1.2.3.2 Assumptions of the Core Inlet Blockage Calculation

The assumptions and conditions in the core inlet blockage calculation by WCOBRA/TRAC(M1.0) are described below.

The conditions and assumptions for the major LOCA parameters used in the core inlet blockage calculation using WCOBRA/TRAC(M1.0) are listed in Table 4.1.2-3. Those shown in Table 4.1.2-3 are the same as US-APWR design certification LBLOCA reference transient case (Ref. 3-10).

<u>The location of blockage</u>
The following two simulations are run with no changes to the standard noding scheme of WCOBRA/TRAC(M1.0) shown in Section 4.1.2.3.1 but with different amount of the core inlet blockage.

The first case (Case-1) modeled 79.8% blockage at the core inlet by ramping the value of k-factor (C_D) up to 10^9 in all core channels except for the lower power (LP) periphery channel [ ]. Since the core bypass flow in the NR region may enter near the top of the core, the blockage of the flow path from the lower plenum to the NR channel ([ ] in Figure 4.1.2-2) is assumed. Figure 4.1.2-9 shows the core channel noding used in Case-1 and those channel [ ] are closed out to represent total blockage at the inlet of the channel. For this modeling approach flow will only enter the core through [ ].

Next, the second case (Case-2) modeled 99.6% blockage at the core inlet by ramping the value of k-factor (C_D) up to 10^9 in all core channels except for the hot assembly (HA) channel [ ]. As well as Case-1, the flow path from the lower plenum to the NR channel is assumed to be blocked. Figure 4.1.2-10 shows the core channel noding used in Case-2 and those channel [ ] are closed out to represent total blockage at the inlet of the channel. For this modeling approach flow will only enter the core through [ ].

<u>The selection of the limiting break</u>

The limiting break will be a double-ended cold leg break which has the minimum driving head contributing to the core flow.

<u>The core radial and axial power distribution</u>

The core channel radial and axial power distribution of US-APWR design certification LBLOCA reference transient case (Ref. 3-10) used for the core inlet blockage calculation are shown in Table 4.1.2-4 and Figure 4.1.2-11. As shown in Table 4.1.2-4, the core radial power distribution is flat other than in the periphery assemblies and the hot assembly. The hot assembly power is conservatively modeled to a high normalized power of [ ]. Moreover the top skewed power shape is limiting in the axial power distribution and this is due to the longer time for the quench front to approach the elevation with the highest power, and its susceptibility to heat up if the core becomes uncovered due to core inlet blockage.

<u>Safety Injection Temperature</u>

The safety injection temperature shown in Figure 4.1.2-12 is used in the core inlet blockage calculation to simulate the rise in RWSP water temperature over a long-term period during LOCA. Temperature change shown in Figure 4.1.2-12 is based on the containment maximum pressure evaluation result at the time of LOCA (Ref. 4.1-2).

<u>The containment back pressures</u>

The containment back pressures used in the core inlet blockage calculation are based on the minimum ones used in US-APWR design certification LBLOCA reference transient case (Ref. 3-10).

4.1.2.4 Calculation Results

This section discusses the results of the core inlet blockage calculation demonstrated by WCOBRA/TRAC(M1.0). The object of this analysis is to establish the behavior of the calculated core inventory and the cladding temperature. It also confirms that the cladding temperature satisfies that of acceptable criteria defined in Appendix-C.

From the calculation results of Case-1 and Case-2, the k-factor (C_D) ramp after 850 seconds blocks all flow into the blocked channel as expected. Consequently, the amount of flow to the unblocked core channel increases.

Next, inventory in the core is examined.
Figure 4.1.2-13 displays the collapsed liquid level of an average assembly core channel ([ ] on Figure 4.1.2-2). As shown in Figure 4.1.2-13, the collapsed liquid level occurring after the core inlet blockage is slightly increasing in both Case-1 and Case-2. Similarly Figure 4.1.2-14 shows the slight increase in the total vessel liquid mass after the core inlet is blocked. The increase in the core liquid level can be attributed to the flow supplied to the core being in excess of the boiloff rate and from liquid inventory from the upper plenum entering the core.

Figure 4.1.2-15 shows that there exists an inventory for the upper plenum global channel. Also Figure 4.1.2-16 and Figure 4.1.2-17 show that the lower power periphery channel allows the liquid from the upper plenum to drain into the core to increase the core inventory.

As shown in Figure 4.1.2-18, the flow supplied to the core is larger than the boiloff rate after the blockage in both cases. The increase in core inventory can therefore be partially attributed to the inlet flow. In addition, the core boiloff rate shown in Figure 4.1.2-18 is calculated by dividing the core power (Q_{DH}; decay heat) by the enthalpy of vaporization [ ] (Ref. 4.1-3) in [ ].

The different inlet flow between Case-1 and Case-2 shown in Figure 4.1.2-18 and the different collapsed liquid level at each downcomer channel shown in Figure 4.1.2-19 - Figure 4.1.2-22 can be explained due to the difference in the resistance at the core inlet. The increase in the downcomer liquid level for Case-2 is calculated to occur due to the large resistance at the core inlet. In Case-1, the downcomer liquid level increases at a slower rate than Case-2.

Although the resistance is ramped in Case-1 to block 79.8% of the core flow area, significant flow is still able to enter the core without the additional build-up of driving head in the downcomer. The higher core inlet flow rate for Case-1 shown in Figure 4.1.2-18 increases the upper plenum liquid inventory and eventually increases the liquid flow rate in the loops. The integral of the hot leg liquid flow rates for each loop are compared in Figure 4.1.2-23 - Figure 4.1.2-26.

Finally, the peak cladding temperature (PCT) of the hot rod is examined.
As shown in Figure 4.1.2-27, the PCT occurs in traditional LOCA analysis space in both cases and after roughly 200 seconds the core is quenched, and no significant heat up occurs thereafter. Since no late heat up occurs, the maximum local and core-wide oxidation calculation for traditional analysis are applicable.

Therefore, it is concluded that sufficient coolant can enter the core after the LOCA to remove core decay heat following up to 99.6% core inlet blockage due to bypass debris penetrating the sump strainer.

4.1.2.5 Summary

The effects of 79.8% and 99.6% blockage of the core inlet flow area were examined using WCOBRA/TRAC(M1.0). A comparison between the calculated core inlet flow rate and the core boiloff flow rate shows ample flow in the core to replace boiloff after the core inlet blockage. Moreover, the PCT plot of the hot rod shows the PCT occurs in traditional LOCA analysis space, and after roughly 200 seconds the core is quenched and no significant heat up occurs thereafter. Since no late heat up occurs, the maximum local and core-wide oxidation calculation for traditional analysis are applicable.
It is concluded that sufficient coolant can enter the core to remove core decay heat with a maximum blockage of 99.6% at the core inlet.

Table 4.1.2-1 Channel Descriptions for US-APWR Vessel Model (1/3)

Table 4.1.2-1 Channel Descriptions for US-APWR Vessel Model (2/3)

Table 4.1.2-1 Channel Descriptions for US-APWR Vessel Model (3/3)

Table 4.1.2-2 Gap Connections for US-APWR Vessel Model (1/2)

Table 4.1.2-2 Gap Connections for US-APWR Vessel Model (2/2)

Table 4.1.2-3   Conditions for the Core Inlet Blockage Calculation

Parameter	Values
Plant physical configuration	
Fraction of SG tube plugged	10% (maximum)
Hot assembly location	Under the open hole
Power-related Parameters	
Core power	4451MWt (100%)
Peaking factor (F_Q)	2.6
Axial power distribution	Top skewed (Figure 4.1.2-11)
Hot rod assembly power ($F_{\Delta H}$)	1.78
Hot assembly burnup	Beginning of life (BOL)
Fuel assembly type	17 X 17 ZIRLO™ cladding
Initial RCS Fluid Condition	
RCS average temperature	583.8°F
Pressurizer pressure	2250 psia
Primary coolant flow	112,000 gpm/loop (thermal design flow)
Accident Boundary Condition	
Break location	Cold leg (in the loop with pressurizer)
Break type	Double-ended guillotine break
Discharge coefficient	1.0
Offsite Power	Not available
Number of SI pumps available	2
Safety Injection flow rate	Minimum
Safety Injection temperature	Figure 4.1.2-12
Safety Injection delay	118 sec

Table 4.1.2-4   Core Channel Radial Power Distribution

Figure 4.1.2-1 US-APWR Vessel (Vertical View)

Figure 4.1.2-2 US-APWR Vessel Noding for Hot Assembly Under Open Hole (Vertical View)

Figure 4.1.2-3 US-APWR Vessel Sections 1 to 2 (Horizontal View)

Figure 4.1.2-4 US-APWR Vessel Sections 3 to 4 (Horizontal View)

Figure 4.1.2-5 US-APWR Vessel Sections 5 to 6 (Horizontal View)

Figure 4.1.2-6 US-APWR Vessel Sections 7 to 8 (Horizontal View)

Figure 4.1.2-7 US-APWR Vessel Sections 9 to 10 (Horizontal View)

Figure 4.1.2-8 US-APWR WCOBRA/TRAC(M1.0) Model Vessel/Loop Layout

Figure 4.1.2-9 Case-1 Core Inlet Blockage Modeling (79.8% Blockage)

Figure 4.1.2-10 Case-2 Core Inlet Blockage Modeling (99.6% Blockage)

Figure 4.1.2-11 Power Shape for the Core Inlet Blockage Calculation

Figure 4.1.2-12 Safety Injection Temperature for the Core Inlet Blockage Calculation

Figure 4.1.2-13 AVG Core Channel Collapsed Liquid Level for Case-1 and Case-2

Figure 4.1.2-14 Total Vessel Liquid Mass for Case-1 and Case-2

Figure 4.1.2-15 Upper Plenum Inner Global Channel Collapsed Liquid Level for Case-1 and Case-2

Figure 4.1.2-16 Total Integrated Liquid Flow at the Top of the Core for Case-1 and Case-2 (HA, AVG, GT channels)

Figure 4.1.2-17 Total Integrated Liquid Flow at the Top of the Core for Case-1 and Case-2 (LP channel)

Figure 4.1.2-18 Integrated Core Inlet Liquid Flow vs. Core Boiloff for Case-1 and Case-2

Figure 4.1.2-19 Broken Loop Downcomer Channel Collapsed Liquid Level for Case-1 and Case-2

Figure 4.1.2-20 Intact Loop-1 Downcomer Channel Collapsed Liquid Level for Case-1 and Case-2

Figure 4.1.2-21 Intact Loop-2 Downcomer Channel Collapsed Liquid Level for Case-1 and Case-2

Figure 4.1.2-22 Intact Loop-3 Downcomer Channel Collapsed Liquid Level for Case-1 and Case-2

Figure 4.1.2-23 Broken Loop Hot Leg Integrated Liquid Flow for Case-1 and Case-2

Figure 4.1.2-24 Intact Loop-1 Hot Leg Integrated Liquid Flow for Case-1 and Case-2

Figure 4.1.2-25 Intact Loop-2 Hot Leg Integrated Liquid Flow for Case-1 and Case-2

Figure 4.1.2-26 Intact Loop-3 Hot Leg Integrated Liquid Flow for Case-1 and Case-2

Figure 4.1.2-27 Case-1 and Case-2 Hot Rod PCT

4.2 Trapping Debris in Fuel Assemblies

4.2.1 Trapping Debris at Grid Spacer

4.2.1.1 Introduction

The bypass debris that would penetrate the sump strainer might be trapped at the grid spacer of the US-APWR fuel assembly. The grid spacer holds the fuel rod by means of two grid spacer springs and four dimples as shown in Figure 4.2.1-1. The maximum diameter of maximum inscribed circle in the grid spacer is about [ ] mm which is larger than that in the bottom nozzle ([ ] mm).

The intermediate grid spacers have mixing vanes on the top of inner straps to increase the mixing of the primary coolant and increase the heat removal efficiency. The mixing flow interfere debris to adhere at the mixing vanes. The top and bottom grid spacers do not have mixing vanes.

The springs and the dimples on the grid strap sheet make the flow hole open instead of the grid cell surrounded by the grid strap sheet. The debris could pass through the flow hole on the grid strap sheet and the coolant as well. Therefore, the flow hole works in favor of the keeping the coolant flow at the position of the grid spacers.

The bypass debris thought the sump strainer will be fine because of the filtering ability of the sump strainers. Therefore, it is not likely that the grid spacer traps the bypass debris coming through the bottom nozzle, but likely that the cooling system maintains well at the grid spacers. This analysis demonstrates that the fuel cladding coolability is acceptable even if the bypass debris clogging at the grid spacers.

4.2.1.2 Methodology

The thermal transfer behavior on the fuel cladding surface is analyzed in the case of the debris clogging at the grid. The analyses take into account the radial heat transfer including the effects of accumulated debris reaching the core at the grid (Ref.4.2-1). In the viewpoint of the consistency, these analyses use the core condition, such as the fuel decay heat and thermal hydraulic condition, based on the results of the WCOBRA/TRAC analysis described at Section 4.1 related to the post-LOCA situation as boundary conditions.

The code ABAQUS™ was used for this analysis using a partial flat plate model focused on a fuel rod with solid elements in the steady heat transfer conditions. Figure 4.2.1-2 shows the schematic drawing of the analysis model. Although the thermal flux effecting heat transfer on the surface decreases with an increase of the surface area contacting with the coolant due to accumulating debris on the cladding, the flat plate model considers the constant heat flux based on the inner cladding diameter conservatively.

The model has one span length with a grid and includes the axial position indicating the maximum heat flux. The grid is located at the center of the model and the cladding has half span length from the grid. The model takes into account the heat flux on the inner cladding surface and the radial thermal transfer between the outer surface of the cladding including the debris and the coolant. Therefore, it is not necessary to consider the pellets and the pellet-cladding gap. It is not assumed that the axial heat transfer exists on both the upper and bottom horizontal surface of the debris, conservatively. The parameters for the analyses are thermal conductivity and the thickness of the debris. This parameter study compares the cladding surface temperature with the acceptable temperature.

It is assumed that the acceptable cladding temperature is up to [ ] °F as described in the Appendix C. This temperature is based on the results of the autoclave corrosion tests which indicate no-acceleration behavior of the cladding corrosion when the temperature on the cladding metal surface is below this acceptable temperature, [ ] °F.

4.2.1.3 Inputs

(1) Power Conditions

The power conditions in the analysis are based on the results of the WCOBRA/TRAC analysis described at Section 4.1. It is estimated that 850 seconds after the LOCA event is the minimum time for debris reaching up to the core in the US-APWR. The heat flux at 850 seconds after LOCA event is the highest heat flux in the analyses because the fuel decay heat is decreasing with the time. Figure 4.2.1-3 shows the heat flux profile on the hot rod at that time and the modeling area including the maximum heat flux at the axial position. The uniform value of heat flux, is [ ] BTU/hr-ft^2 ([ ] W/m^2), which is based on the inner cladding surface and is applied in the analyses, conservatively.

(2) Thermal Hydraulic Conditions

The Thermal Hydraulic conditions in the analysis are based on the results of the WCOBRA/TRAC analysis at 850 seconds after LOCA event as for 4.2.1.3 (1) above. The uniform value of heat transfer coefficient is [ ] BTU/hr-ft^2-°F ([ ] W/m^2-°C) in the analyses, conservatively. This is the minimum value in the modeling area and the lower value in the grid region. The bulk coolant temperature of the coolant is [ ] °F ([ ] °C) as the estimated value in the modeling region.

(3) Geometric Conditions

The outer and inner diameter of the cladding is 0.374 inch (9.50 mm) and 0.329 inch (8.36 mm) as the fabricated value, respectively. These analyses apply the maximum heat flux uniformly based on the inner diameter of the cladding because the cladding diameter affects the value of the heat flux on the cladding surface. The cladding material is ZIRLO™, but there is no impact for the surface temperature of the cladding.

The US-APWR fuel has 11 grids installed the mixing vane except for the top and bottom grid as shown in Figure 4.2.1-1. No.8 grid counted from the bottom is located in the modeling area including the maximum heat flux point as shown in Figure 4.2.1-3. The uniform heat flux is applied on the inner surface of the cladding so that the model considers the maximum value of heat flux at the grid. The grid axial height is the fabricated value, [ ] inch ([ ] mm).

+ ZIRLO™ is a registered trademark of the Westinghouse Electric Corporation.

(4) Assumptions

- ✓ All the debris accumulating on the cladding at the grid distributes and accumulates uniformly. Therefore, the thermal properties of the debris have homogeneity in the analyses.
- ✓ The debris assumed in the analyses is wet and there might be some limited amount of convection inside. The analyses conservatively take into account no convection in the debris.
- ✓ The thermal conductivity of the debris is assumed to be same with CRUD thermal conductivity. It is reported that the thermal conductivity of the CRUD is 0.5 BTU/hr-ft-°F (0.29 W/m-°C) (Ref.4.2-1and 4.2-2). The analyses apply the parameter study about the thermal conductivity, with values varying from 0.1 BTU/hr-ft-°F (0.058 W/m-°C) to 0.9BTU/hr-ft-°F (0.52W/m-°C) at 0.2BTU/hr-ft-°F (0.12W/m-°C) intervals.
- ✓ The debris thickness varies from 0mils (0μm) to 50mils (1270μm) at 10mils (254μm) intervals. The distance between the cladding surface and the grid strap determine the maximum value of the debris thickness. The grid structures, such as spring and dimple, might define the debris thickness because they are located near by the cladding compared with the grid straps.
- ✓ The inlet is the top or bottom of core region for the bypass debris to reach up to the core. Therefore, the top or bottom grid might have probability for clogging debris compared with the intermediate grids. The analyses conservatively assume that the clogging debris occurs at the maximum heat flux position.
- ✓ An Adiabatic assumption is conservatively applied for the axial heat transfer which might exist on both the upper and bottom horizontal surface of the debris at the grid.
- ✓ The material of the springs and dimples supporting the fuel cladding has the same thermal conductivity with the accumulating debris. This assumption is conservative because the metal material generally has higher thermal conductivity compared with

the debris.

- ✓ The analyses have no heat barrier between each material, such as cladding and accumulating debris.

4.2.1.4 Results

Table 4.2.1-1 and Figure 4.2.1-4 show the maximum temperature at the cladding surface as results of the parameter study varying the thermal conductivity and the thickness of the accumulating debris. The maximum temperature is analyzed at the grid central position. The analyses show that the maximum temperature at the cladding surface with accumulating debris is [ ] °F ([ ] °C) in the worst case which uses 0.1 BTU/hr-ft-°F and 50 mils for the minimum thermal conductivity and maximum thickness of the debris, respectively. This temperature bounds the other results of this parameter study. The all analyzed temperatures at the cladding surface meets the acceptable temperature which is [ ] °F as described in Appendix C. Therefore, the cladding cooling is acceptable in the case of debris clogging at the grid straps after a LOCA. The conservatisms used in the inputs and assumptions described in Section 4.2.1.3, especially in the assumption of the maximum heat flux due to the decay heat, enforce this result.

Table 4.2.1-1 Cladding Metal Surface Temp. vs Debris Thickness

	Debris Thermal Conductibity (BTU/hr-ft-°F)				
	0.1	0.3	0.5	0.7	0.9
Debris Thicknness (mils)	deg. °F (deg. °C)	deg. °F (deg. °C)	deg. °F (deg. °C)	deg. °F (deg. °C)	deg. °F (deg. °C)
0					
10					
20					
30					
40					
50					

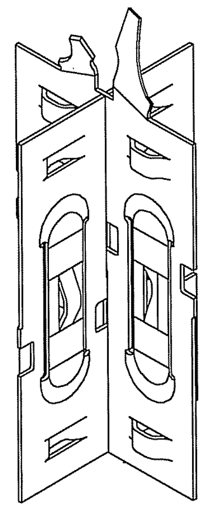

Figure 4.2.1-1 Mitsubishi Grid Spacer (Z3 Type) Schematic View

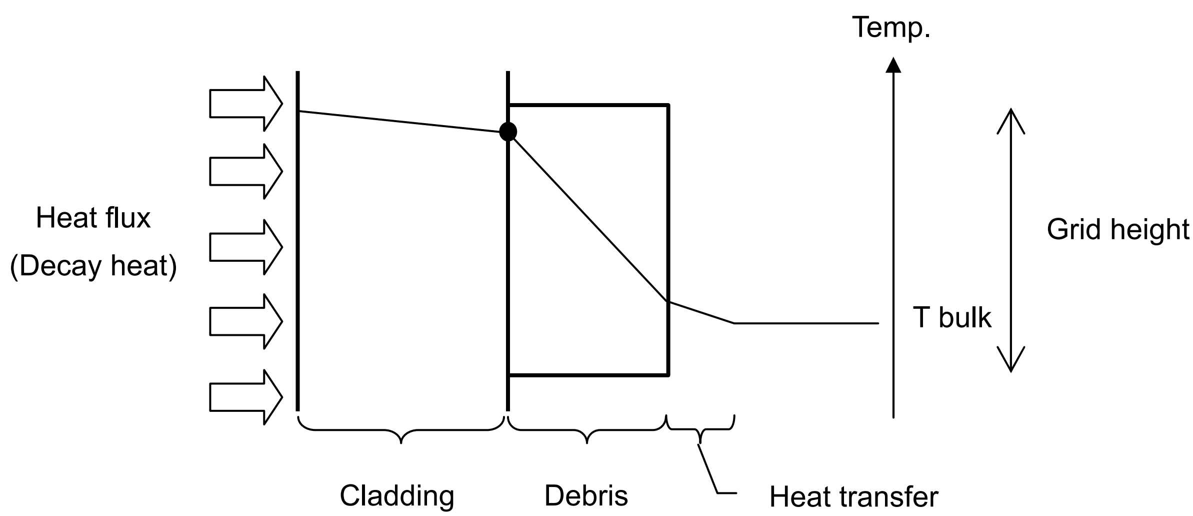

Figure 4.2.1-2 Schematic Drawing of the Analysis Model

Heat Flux (BTU/hr/ft2)

Heat Flux (W/m2)

0 20 40 60 80 100 120 140 160 180

Axial Position from the Bottom of the Active Fuel (inch)

Figure 4.2.1-3 Heat Flux on the Hot Rod at 850 seconds after LOCA Event

Cladding Metal Surface Temp. (°F)

0 10 20 30 40 50 60

Debris Thickness (mil)

Figure 4.2.1-4 Cladding Metal Surface Temp. vs Debris Thickness

4.2.2 Trapping Debris on Cladding Surface

The cladding temperature should meet the acceptable temperature as described in Appendix C. In the case of the debris accumulating on the cladding surface between each grid, the accumulating debris might affect the cladding surface temperature.

The characteristics of bypass debris of the US-APWR are defined in Appendix D, and in Subsection 3.3 of separate technical report (Ref 3-12). As defined, the bypass debris of the US-APWR consists of NUKON fiber, coating particles, and latent particles. In general, it is considered that the particulate will pass through the fuel region unless the fiber debris bed is formed at cladding and capture the fine particles.

Since the bypass debris (NUKON fiber) is defined as very fine because of the filtering ability of the sump strainers, it is very unlikely that it forms debris bed at cladding (Ref.4.2-3). Therefore, the fibrous debris will pass through the cladding surface between the grids, and no fibrous bed will be formed on the cladding surface. The effects of the debris accumulating on the cladding surface between each grid are bounded by the evaluation as described in Section 4.1. It is noted that the effects of the chemical debris is described in Section 4.3.

4.3 Chemical Effects on Fuel Rods

The supply of coolant containing chemical effects products in the recirculation sump water may affect fuel rods in the core after starting post LOCA recirculation. Chemical effects testing were performed to obtain experimental data under simulated plant conditions on the corrosion products that may form in a post-LOCA environment for the US-APWR (Ref. 3-14). The testing provided compositions, characterize properties, and quantify masses of chemical reaction products that may develop in the containment under a representative post-LOCA environment. In regard to coatings it will be unlikely that coatings affect chemically heat removal in the core during a post-LOCA since the standard US-APWR will utilize only DBA epoxy coating systems in containment and the coolant that might contain coatings will not get exposed to such high temperature as likely to have effect on the core.

Epoxy coatings were similarly not considered in the joint USNRC and nuclear industry integrated chemical effects testing – see NUREG-6914, Integrated Chemical Effects Project: Consolidated Data Report (Ref. 4.3-1). Epoxy coatings have been shown to be chemically resistant in both highly acidic and caustic environments. ASTM D-3911 requires the specific conditions anticipated following a loss of coolant accident that would expose the coated surface of the containments to the temperature-pressure environmental parameters described in Reference 4.3-2. Typical DBA testing parameters in that reference shows a temperature condition of [ ], meaning that epoxy coating systems qualified have been shown to be chemically resistant in high temperature such as [ ].

After the time that the RWSP recirculation water affected by bypass debris has reached the core (Appendix E), the cladding temperatures ill be well below the [ ] temperature at which epoxy coatings may be affected by temperature. Figure 4.1.2-27 in section 4.1.2 demonstrates that, even with 99.6% of the fuel entrance blocked, sufficient water is provided to maintain cladding temperatures at around [ ] during long-term core cooling.

Parametric cladding heat-up calculations described in Section 4.2.1 were performed for a blocked grid. These parametric calculations show that for a precipitate with a sufficiently small value for thermal conductivity and a sufficiently large value of deposited thickness, cladding surface temperatures in excess of [ ] may be predicted. However, these same calculations also demonstrate the temperature of the precipitate surface at the boundary of the coolant, where coatings debris might be expected to collect at higher than a temperature, is within about 20°F of the adjacent coolant temperature at the time of estimation. From the fuel rod heat-up calculations described in Section 4.2.1, the surface temperature of the precipitate surface is calculated to be less than [ ] at the time of evaluation with heat transfer considering state of coolant in the core.

Hence epoxy coatings are evaluated to be chemically inert in the post-LOCA chemical environment for the US-APWR and therefore have a negligible effect on post-LOCA precipitant production. Thus, epoxy coatings are evaluated to not present a concern with respect to long-term core cooling.

The following section 4.3.1 discusses predicting the cladding temperature with deposits of chemical impurities after a LOCA.

4.3.1 Chemical Deposition on the Cladding

The supply of coolant containing chemical effects products in the containment vessel may cause precipitation on the cladding after starting post LOCA recirculation. This precipitant on the cladding may reduce the heat transfer from the fuel, thus causing a rise in fuel temperature. In this section, the cladding temperature after LOCA was evaluated using the chemical effect testing data (Ref. 3-14).

4.3.1.1 Introduction

The reactor containment vessel of the US-APWR is designed to seal in radioactive products and to facilitate core cooling after a LOCA. In LOCA scenarios, RWSP collects the water discharged from the break and containment spray water containing chemical impurities and debris. The water is recirculated into the core by the ECCS and into the containment spray by the CSS. Then the chemical impurities and debris resulting from the chemical interaction between coolant and containment materials may move into the core.

The NRC issued specific guidance to the industry for responding to the chemical effects on the core at the GSI-191 Resolution Status Meeting of February, 2007 (Ref. 4.3-4). NRC asked that submittals intended to demonstrate the viability of long-term core cooling should meet the following requirements specific to chemical effects concerns:

1. Chemical concentration effects due to long-term boiling should be assessed.
2. The plate-out of deposits on the fuel rods should be considered.

MHI has carried out the chemical effect testing focusing on the ECCS of US-APWR to determine the chemical concentration effects (Ref. 3-14). The cladding temperature during post-LOCA will be evaluated by the data of chemical impurity concentrations in the chemical effect tests for the US-APWR. MHI chose to evaluate further these issues and their effect on the viability of long-term core cooling for the US-APWR.

4.3.1.2 Objective

The overall methodology deals with calculating the deposition of chemical impurities in the reactor coolant on the cladding surface and then quantifies the impact these deposits for raising the calculated cladding temperature. The purpose of this evaluation is to predict the cladding temperature with deposits of chemical impurities after a LOCA, which are dissolved or suspended in the recirculation sump water during long-term cooling for the US-APWR.

4.3.1.3 Methodology

4.3.1.3.1 Discussion of Major Assumptions

The deposition method makes several assumptions that are conservative, and, as a result, the predictions of deposit thickness and fuel surface temperature should be considered to be bounding rather than a best-estimate.

1. Deposits, once they have been formed, will not be thinned by flow erosion or by dissolution.
2. All deposition takes place on the fuel cladding.
3. Any mist carry-over, which will discharge potential deposits, does not happen in the steam exiting in reactor vessel. Thus, the concentration of such impurities in the core will be maximized, and thicker deposition will be calculated for this elevated concentration of chemical impurities.
4. The boiling point elevation due to the concentration of solutes is not considered. This model will lead to a slight over-estimation of boiling in the core, resulting in a conservative evaluation. In contrast, the boiling point elevation due to head loss caused by the broken loop flow is considered since the less evaporative latent heat is with increasing pressure the more chemical impurities will be deposited.
5. The deposition rate by boiling is equal to the steaming rate times the impurity concentration. When boiling is terminated, decay heat release is conducted by transmission without boiling. The deposition rate by non-boiling is assumed to be proportional to heat flux and is 1/80 of that of boiling deposition at the equal heat flux. This ratio is based on empirical data under boiling and non-boiling conditions (Ref. 4.3-4). For the conservatism starting time of non-boiling is determined based on a time whole the core boiling is terminated.
6. The deposition of impurities on the fuel cladding surface is assumed to be proportional to local heat flux according to the core power distribution. The estimation is conducted at largest heat density position because higher heat flux results in higher temperature and thicker deposition of fuel cladding surface.

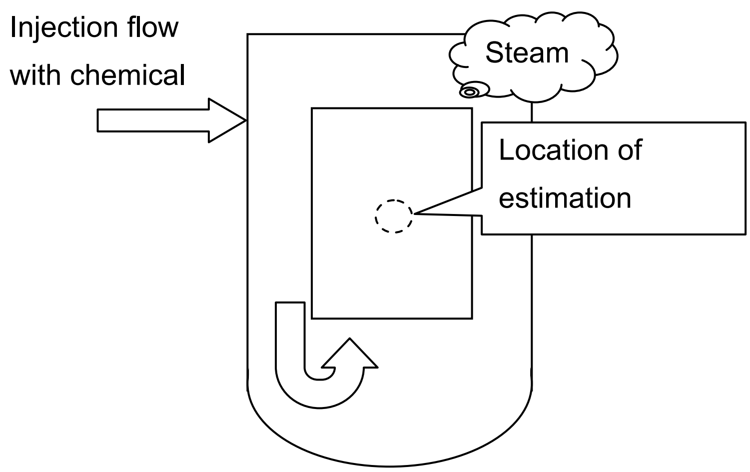

Figure 4.3.1-1 Schematic Model

4.3.1.3.2 Debris Dissolution and Corrosion Rates

The chemical impurities concentrations in the containment pool are estimated from a separate series of chemical effect tests. These concentrations are then used as the input concentrations in the recirculation water flowing into the reactor vessel, which forms the basis for the evaluation of downstream deposition in the core.

The chemical reaction product deposition rate in the core is only affected by the concentrations of the impurities in the inflow, not by the upstream (in containment) dissolution rate. In other words, the same concentrations of chemical impurities in the inflow caused by different dissolution rates in containment result in the same rate of core deposition. If the concentrations are lower, then the rate of the core deposition should be less, irrespective of the corrosion rate.

This is an important consideration because the chemical effects tests are modeled without the core, in which the impurities generated within the containment vessel might be deposited. Thus, these experiments do not simulate the effect of the core, which would be to decrease the impurities concentrations of the coolant water that recirculates back through containment via the break flow and the condensated pure steam vaporized in the core. As a result, the evaluation used sump impurities concentrations that are conservatively high because they neglect this concentration lowering effect cause by core deposition.

It should be noted that when the deposition within the core happens, thereby decreasing the impurities concentrations in the return flow to containment, the subsequent dissolution rates of materials in containment, and hence generation of new chemical impurities, may be accelerated because of lowering concentrations relative to their saturation limits. However, even with this increased rate of dissolution, the resulting concentrations are not expected to exceed those determined from experiments in which deposition is not simulated.

4.3.1.3.3 Modeling of the Core

The estimation is conducted at largest heat density position because higher heat flux results in higher temperature and thicker deposition of fuel clad surface. Decay heat power at the highest heat flux identified peaking factor in the core was used.

4.3.1.3.4 Calculation of Deposition Mass and Fuel Temperature

The following process was used in this to determine the quantity of the deposition of impurities. First, the temperature at the zirconium oxide/deposit interface was calculated using

$$T_{o/d} = q * (xc/kc + xl/kl + 1/h) + Tc \quad \text{.........................(4.3.1-1)}$$

where:

To/d =Temperature at the cladding surface (K)
Tc =temperature of the coolant (K)
q =heat flux at maximum heat load (W/m^2)
xc =thickness of the initial crud layer (m)
xl =thickness of the LOCA scale layer (m)
kc =thermal conductivity of the initial crud layer (W/m/K)
kl =thermal conductivity of the LOCA scale layer (W/m/ K)
h =heat transfer coefficient for thermal resistance of coolant at boundary layer ($W/m^2/K$)

The mass of impurity elements deposited during a time step was calculated simply by multiplying the steaming rate times the concentration of that species.

$$dw = q * dt * C / hfg \quad \text{..........................(4.3.1-2)}$$

where:

dw =deposit mass for time step at unit area (kg/m^2)

dt =time step (s)
hfg =standard enthalpy of vaporization (Joules/kg)
q = heat flux at maximum heat load (W/m^2)
C =concentration of species (kg/kg)

Impurity concentration was treated as a summation of each impurity element.
The thickness added to the LOCA scale was then determined by dividing the mass deposited within the node by the density times the area. Smallest density among assumed deposition elements was used for conservative estimation.

$$dx = dw / D \qquad \text{..........................(4.3.1-3)}$$

where:
dx is the increase in the deposit thickness for the node
D = density (kg/m^3)

If boiling is terminated, growth rate of deposition estimated by (4.3.1-2) is assumed to be 1/80 of that of boiling deposition at the equal heat flux. (Ref. 4.3-4). The core outlet fluid condition is estimated by using minimum safety injection flow rate, core decay heat and core inlet coolant temperature identified with that of recirculation sump water. For the conservatism starting time of non-boiling is determined based on a time whole the core boiling is terminated.

As a result of the chemical equilibrium calculation with OLI StreamAnalyzer^{TM} computer program, the form of the chemical depositions on the cladding are predicted to include: $Al(OH)_3$, $AlOOH$, $NaAlSi_3O_8$, Zn_2SiO_4. The results of the separate effects dissolution experiments will be used to provide quantification of the relative amounts of these species.

The scale layer density which dominates the layer thickness and thermal conductivity are set independently. Thus the lowest density and the lowest thermal conductivity lead to conservative evaluation. The scale deposited on the boiling surface is assumed to be generally from 40 to 58% of porosity with the knowledge described in Ref. 4.3-5, 4.3-6, 4.3-7. Here, the porosity of the deposited scale is assumed to be at 60%.

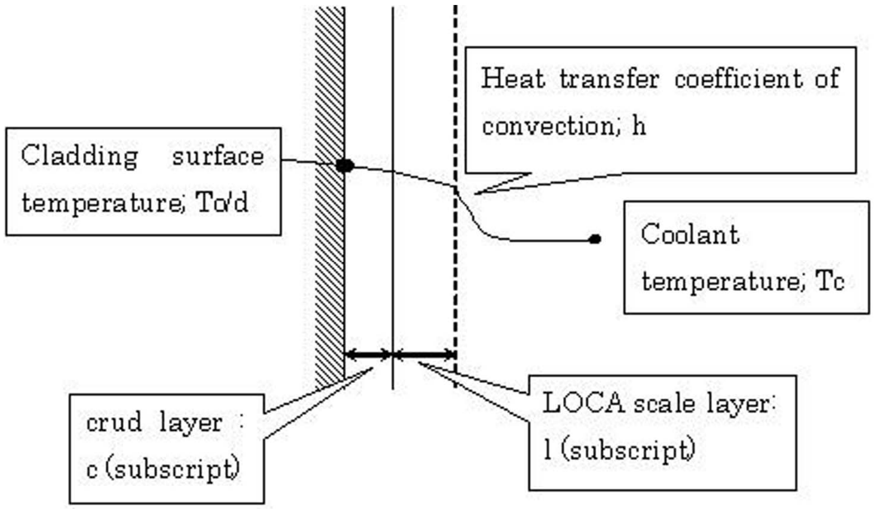

Figure 4.3.1-2 Temperature estimation method

4.3.1.4 Input Condition

The recirculation sump water that interacted with the materials in the containment vessel was assumed to arrive at the reactor vessel at 850 seconds after the beginning of the LOCA (Appendix E). Hence, the evaluation is used the period from 850 seconds to 30 days after LOCA.

4.3.1.4.1 Heat Condition

1. Decay heat
   Decay heat is assumed to be based on ANS-1971 x 1.2 fission product decay curve. Heat flux utilized in this estimation is calculated as maximum value taken into account peaking factor.
2. Boiling termination time
   Boiling was assumed to be terminated at 8 days (=192 hours) after the beginning of a LOCA by using following assumption.

At the cold leg break LOCA, during which boiling continues for a longest time, Safety Injection water supplies from bottom of core through downcommer. Because core is cooled by safety injection water from bottom, boiling shall be terminated from bottom to top sequentially.

Boiling termination time is assumed to be estimated as the time when the energy increase of core inlet coolant calculated by sensible heat, minimum flow rate and core inlet water enthalpy of safety injection becomes beyond above decay heat of whole core.

Because boiling duration time estimated by global heat balance is longer than that estimated by local thermal hydraulic parameter at the place of maximum heat flux, this estimation concludes to have conservatisms.

3. Fluid temperature of core

   Fluid temperature is chosen as 294°F. This temperature is taken into account 10°F margins of maximum temperature in containment vessel in estimated duration. Although fluid temperature decreases as time goes by, fluid temperature assumed to be constant for conservatisms.

4. Initial crud condition

   Initial crud thickness utilized in this estimation is assumed as summations of initial maximum oxidation layer thickness ([ ]) which means allowable maximum cladding oxidation that complies with 10 CFR 50.46 and initial maximum crud layer thickness ([ ]). Smaller thermal conductivity of initial crud is chosen in order to estimate conservatively.

4.3.1.4.2 Chemical Condition

The chemical debris for calculation are defined by the result of the chemical effect tests that represent the recirculated sump chemical concentrations after a LOCA in the US-APWR. The tests temperatures were controlled to be higher than the estimated temperature to ensure conservative production of corroded material.

The water properties in the ECCS and CSS after a LOCA in the US-APWR will transform as follows:

The sum of the dissolved materials and the precipitation after dissolution from those materials (the total impurities) was used as the input condition of the water entering the core for the evaluation of the deposition during boiling and non-boiling.

Figure 4.3.1-3 Impurity concentrations trend at core inlet

The procedure used for predicting trends of dissolution is as follows:

1. Prediction is based on the concentration trend of the recirculation test at 149˚F for 30 days.
2. Remaining concentration balances of the autoclave test result at transient higher temperature after deduction of the one at constant temperature of 149˚F is added to the recirculation test result at 149˚F for each element.
3. When the concentration decreased as time progressed, the concentration is kept constant after the maximum point.

Predicting trends of dissolution is shown in Figure 4.3.1-3.

4.3.1.5 Result of Calculation

The calculated results are shown in Figure 4.3.1-4. 0 second in Figure 4.3.1-4 means the beginning of a LOCA and the figure shows the results after 850 seconds.
It is shown that scale gradually increases during the LOCA. However, after boiling termination, LOCA scale growth rate obviously decreases. LOCA scale thickness becomes about 390 microns at 30 days (720 hours) after the LOCA. It is concluded that the deposited LOCA scale will not block coolant flow path and has no influence upon fuel cladding cooling.
Fuel cladding temperature gradually decreases because the effect of the reducing decay heat is larger than temperature increase effect regarding with thermal resistance of scale formed during the LOCA. Maximum temperature of fuel cladding after recirculation was started was [ ]. Fuel cladding temperature was verified to be maintained lower than [ ] temperature criteria during the entire evaluation period.

Figure.4.3.1-4 LOCA scale thickness and fuel cladding temperature

4.3.1.6 Conclusions

Cladding temperature with deposits of chemical impurities after a LOCA environment for the US-APWR was evaluated.
It is concluded that structural integrity of fuel cladding is retained because scale formed during the LOCA will not have effects on the coolability and fuel cladding temperature will be maintained lower than upper criteria limit.

4.4 In-core Effects

4.4.1 Chemical Effect on Boric Acid Precipitation Evaluation

The US-APWR design uses boron as a core reactor reactivity control method, and there is a procedure that instructs the operators to switch operating DVI lines over to the hot leg injection line (simultaneous reactor vessel and hot leg injection) no sooner than about four (4) hours after the postulated large break LOCA to prevent the core region boric acid concentration from reaching the precipitation point. The switchover time is determined by a method, as described in the DCD Chapter 15, based on assumptions regarding mixing in the reactor vessel (Ref. 3-10).

An analysis method with an appropriate evaluation model is applied to control the boric acid precipitation during long term cooling after LOCA and is similar to that for US representative PWR plants (Ref. 3-10). Generally in the boric acid precipitation evaluation for a PWR plant, only cold-leg break evaluation, which should be the limiting case, is performed,

The limiting scenario for boric acid precipitation is a cold leg break where the core is stagnant with only enough core inlet flow to replace core boil-off. For this scenario, lower plenum flow rate is approximately a factor of 2 to 3 magnitude less than maximum flow rate in the hot leg break case. Furthermore, the spilled SI flow would be repeatedly re-filtered through the sump strainers. Since the volume of settled debris in the lower plenum would be approximately proportional to the flow into the lower plenum, the maximum volume of settled debris in the lower plenum for a cold leg break would be small.

Boric acid precipitation is not an issue for a hot leg break since forced flow into the reactor vessel such that boric acid accumulation will not occur is expected.
Therefore, as for the US-APWR that has similar boric acid concentrating mechanism to that of a representative PWR in the US, the limiting scenario for boric acid precipitation is also a cold leg break.

(1) Effect of Suspended and Settled Sump Debris on Mixing Volume
The suspended debris and settled debris ingested into the core region through the strainers may have some impact on the assumed mixing volume for the evaluation of boric acid concentration during the post-LOCA long term cooling. The debris in the coolant

in the reactor vessel would replace water volume that would otherwise dilute the boric acid in the core region. The amount of debris that bypasses the sump strainer is assumed as some fiber and all of particulate for the US-APWR. The amount of bypass debris that may exist in the mixing volume or that may affect the mixing volume is shown in Appendix–D. The replaced volume of debris shown in Appendix- D would be a small fraction of the liquid mixing volume used for the evaluation of US-APWR boric acid concentration, which is assumed to be that of lower plenum, since the US-APWR has large lower plenum volume of more than one thousand cubic feet.

(2) Effect of Core Inlet Blockage on Mixing Volume

Core inlet blockage due to accumulated sump debris may slightly effect the predicted rate of boric acid concentration accumulation in the core, depending on the specific mixing volume assumptions used in a given boric acid precipitation analysis. However, significant core inlet blockage will not be expected since it is quite unlikely that the downstream debris may pile up at the fuel assembly bottom nozzle described in section 4.1.1. Although it is unlikely that bypass debris accumulates exactly in the core inlet region to predict, only entire or severe core inlet blockage would effectively isolate the lower plenum from the core region. Core inlet blockage that would be expected for cold leg large break scenarios evaluated would not impede flow between the lower plenum and the core region since there will not be sufficient core inlet flow rate and the amount of bypass debris that clog the core inlet in such a break case.

As significant core inlet blockage is not expected, the core region liquid inventory would not be significantly affected, and the core region mixing volume used in the US-APWR analyses would remain effective.

(3) Effect of Blockage on Alternate Core Coolant Flow Paths

Sump debris may accumulate sufficiently to block several alternate coolant flow paths to the core that are expected to dilute the boric acid in the core.

Examples of these flow paths are flow through the neutron reflector region, flow through the hot leg nozzle gaps, and flow to the core from the upper plenum (after switchover to hot leg injection). Only the hot leg nozzle gaps are small enough to capture bypass debris. The US-APWR boric acid precipitation analysis method does not credit the hot leg nozzle gaps as dilution flow path assumptions used in that analysis. There will not be the alternate core flow paths that should be considered to be significant blockage for boric acid precipitation evaluation since the flow areas are enough large as mentioned in section 4.1.1 and are not effective debris traps or filters.

For hot leg breaks, dilution flow is not needed since the core would keep diluted with forced SI flow entering the core through the core inlet. For SI flow to the hot legs after hot leg switch over, the core dilution process would not be impeded unless entire or severe blockage of flow in the upper plenum occurred. Entire or severe blockage between the upper plenum and the core is not expected to occur as described in section 4.4.3.

(4) Effect of Chemical Compounds on Boric Acid Precipitation in the Core Region

Mixing in the core will continue due to convection, diffusion, local turbulence, and bubble mixing phenomena, with little or limited bypass debris accumulating in the core, at the core inlet, or in the lower plenum.

For the flow conditions of the boric acid precipitation scenario, considering chemical compounds of boric acid, there will not be sufficient chemical debris effects on the boric acid to invalidate the licensing basis boric acid precipitation analyses for the US-APWR. Therefore, it is concluded that chemical debris would not significantly affect the boric acid precipitation assessment.

4.4.2 Fuel Swelling and Blockage

Swelling and rupture of the fuel rod cladding during design basis LOCAs is one of the phenomena which licensees are required to evaluate under Appendix K to 10 CFR Part 50. Following a large break LOCA, some of the fuel rods in the core may swell and rupture leaving sharp edges at the rupture locations and a diminished channel flow area. Debris may collect in the restricted channels and at the rough edges of the rupture locations. It is necessary to evaluate the possibility that excessive blockage is produced by the combination of swelling and rupture and debris collection. Such blockage might produce the occurrence of hot spots above the blockage location.

The debris that flows in RCS will pass the bottom nozzle and, based on the expected location of the swelling and rupture, several grids before reaching the rupture location. Therefore, the accumulation of significant debris at the localized rupture location is not generated easily compared with other places where fibers are more likely to gather before the hot leg switchover. Additionally, there would only be a limited number of fuel rod cladding ruptures in the reactor core, and, rupture is most likely to occur in the the highest power fuel rods in the highest power assemblies.

Therefore, there is little possibility that significant blockage will occur due to fuel swelling and fuel rupture in the large break LOCA scenario.

4.4.3 Hot Leg Injection

The US-APWR design uses ECCS hot leg injection no sooner than about four (4) hours after occurrence of the postulated LBLOCA. At this switchover time, the coolant in the RWSP is expected to have been circulating through the ECCS and CSS several times. Therefore particulate and fibrous debris, which is generated by the initial RCS break flow and CS water flow back into the RWSP, is expected to be depleted either by capture on the strainer or by settle-out in low flow rate regions, such as the lower plenum. Thus, the amount of debris injected during the hot leg injection mode is expected to be small enough that the core cooling will not be significantly affected by the debris.

Furthermore, core flow rate would be maintained high enough to remove decay heat since the core power at hot leg switch over (HLSO) decreases to around one third of that at the time core quench is completed.

4.5 Regulatory Summary (In-Vessel)

As described in section 3.6, the US.NRC requested holders of operating reactor licenses to evaluate their ECCS and CSS recirculation functions in light of events regarding the blockage of containment sump strainer (Ref. 3-1, Ref. 3-2). The discussions in this report have included ex-vessel downstream effects and in-vessel downstream effects, which the US.NRC has clarified with the issuance of reference 3-3 and Reference 4.5-1.

These references were intended to address the US.NRC concerns identified in Reference 3-2. There are no other regulatory guidance documents that specify the evaluation of in-vessel downstream effects appropriately. Therefore, addressing the issues identified in the “Review Guidance”(Ref. 3-3) and “Audit Plan” (Ref. 4.5-1) documents should address the concerns identified in Reference 3-1.

Table 4.5-1 shows sections excerpted from “Review Guidance” (Ref. 3-3) and “Audit Plan” (Ref. 4.5-1) and the section in this report where it is addressed.

Table 4.5-1 (a) In-Vessel Downstream Effects Regulatory Review（Ref. 4.5-1）

Downstream Effects on Fuel Checklist（Ref. 4.5-1）	Response
Analyses that should be provided - Potential to clog lower core due to flow induced debris bed.	Report Section 4.1.
- Potential to clog lower core due to filling the lower vessel with a volume of debris.	Report Section 4.1
- Potential for a mid-core blockage (Potential for capture of debris at grid straps or buildup via adhesion (most likely more of a CL LOCA concern)).	Report Section 4.2.
- Potential for heat transfer loss from a chemical film (interaction of high boric acid concentration with debris characterization).	Report Section 4.3.
- Potential for hot leg recirculation to clog upper core – by flow induced debris bed	Report Section 4.4
- Potential for hot leg recirculation to clog upper core – by volume of debris	Report Section 4.4

Table 4.5-1 (b) In-Vessel Downstream Effects Regulatory Review（Ref. 3-3）

xviii. Review the evaluation of downstream effects on reactor fuel and in-vessel components. (Ref. 3-3)	Response
(1) Volume of debris injected into the reactor vessel and core region	Report Section Appendix D.
(2) Debris types and properties	Report Section Appendix D.
(3) Contribution of in-vessel velocity profile to the formation of a debris bed or clog	Report Section Appendix D.
(4) Fluid and metal component temperature impact	Report Section 4.1.2 and 4.2.1
(5) Gravitational and temperature gradients	Report Section 4.1.2 and 4.2.1
(6) Debris and boron precipitation effects	Report Section 4.4.1
(7) ECCS Injection paths	Report Section 4.4.1 and 4.4.3
(8) Core bypass design features	Report Section 4.4.1 and 4.4.3
(9) Radiation and chemical considerations	Report Section 4.3
(10) Debris adhesion to solid surfaces	Report Section 4.2
(11) Thermodynamic properties of coolant	Report Section 4.1.2 and 4.2.1

5.0 CONCLUSIONS

The intent of this technical report is to assess the US-APWR systems and components downstream of the containment sump strainers to ensure that that these systems and components will operate as designed under post Loss-of Coolant Accident (LOCA) Conditions. Downstream systems and components include the Emergency Core Cooling System (ECCS), Containment Spray System (CSS) and the reactor core. This report evaluates the effects of operating with debris-laden, post-LOCA fluid.

This report concludes that the US-APWR Emergency Core Cooling System, Containment Spray System and their components are fully capable of performing their intended functions under post-LOCA operating conditions. I.E. the ECCS and CSS are fully capable of providing adequate core cooling to ensure the reactor core is maintained in a safe, stable condition following a Loss-of-Coolant Accident (LOCA).

The report concludes that debris-laden post-LOCA fluid will not plug or block the reactor core such that cooling flow is reduced below the required flow to maintain long-term core cooling. The report also shows that chemical induced local scale formation on the fuel cladding surface on reactor fuel cladding will not affect the ability to provide adequate decay heat removal. Cladding temperatures are maintained below those required by Section 50.46 of Title 10 of the Code of Federal Regulations (10 CFR).

6.0 REFERENCES

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3-2 U.S. Nuclear Regulatory Commission, Final Safety Evaluation for NEI Guidance Report 04-07, December 6, 2004

3-3 T O Martin, Audit Plan for Verifying the Adequacy of Licensee Responses to Generic Letter 2004-02, USNRC Memorandum, T O Martin to M G Evans, December 5, 2006

3-4 U.S. Nuclear Regulatory Commission, Final Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) WCAP-16406-P, December 20, 2007

3-5 Water Sources for Long-Term Recirculation Following a Loss-of-Coolant-Accident, Regulatory Guide 1.82 Revision 3, November 2003

3-6 U.S. Nuclear Regulatory Commission, Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems, Generic Letter 2008-01, January 2008

3-7 U.S. Nuclear Regulatory Commission, Effects of Insulation Debris on Throttle Valve Flow Performance, NUREG/CR-6902, March 2006

3-8 Design Control Document for the US-APWR Chapter 5, Reactor Coolant and Connecting Systems, MUAP-DC005 Revision 1, August 2008

3-9 Design Control Document for the US-APWR Chapter 6, Engineered Safety Features, MUAP-DC006 Revision 1, August 2008

3-10 Design Control Document for the US-APWR Chapter 15, Transient and Accident Analysis, MUAP-DC015 Revision 1, August 2008

3-11 Design Control Document for the US-APWR Chapter 17, Quality Assurance and Reliability Assurance, MUAP-DC017 Revision 1, August 2008

3-12 US-APWR Sump Strainer Performance, MUAP-08001-NP Revision 2, December 2008

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4.1-2 LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR, MUAP-07012-P, Revision 2, May 2008

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4.3-1. U.S. Nuclear Regulatory Commission, Integrated Chemical Effects Test Project: Consolidated Data Report, Volume 1, NUREG/CR-6914, 2006

4.3-2. U.S. Nuclear Regulatory Commission, Standard Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions, ASTM D-3911-03, 2006

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4.3-4 A. Helaizadeh, H. Muller-Steinhagen, M. Jamialahmadi, “Crystallization Fouling of Mixed Salts During Convective Heat Transfer and Sub-Cooled Flow Boiling Conditions”, ECI Conference on Heat Exchanger Fouling and Cleaning: Fundamentals and Applications, Paper 6, Santa Fe, New Mexico , 2003

4.3-5 Energy and Environment Laboratory CRIEP Report, “Study on Measurement Method of Boiler Scale Porosity (Part 2) –Measurement of the Quantity and the Porosity of Boiler Scale by the Electrolytic Exfoliation Method -, 1978

4.3-6 The Thermal and Nuclear Power, Vol. 29, No. 6, 571-577, 1978

4.3-7 Proc 4th Int. Symp. Environ. Degrad. Mater. Nucl. Power Syst. Water React. 1989, 7.108-7.120, 1990

4.5-1. U.S. Nuclear Regulatory Commission, Draft NRC Staff Review Guidance for Evaluation of Downstream Effects of Debris Ingress into the PWR RCS on Long Term Core Cooling Following a LOCA, November, 2005.

Appendix A

ECC/CS Strainer / Safety Injection Pump Engineering Design Parameters

Table A-1 ECC/CS Strainer and Safety Injection Pump Design Parameters

Description	Specification
ECC/CS Strainer	
Hole diameter of perforated plate	0.066 inch
Equipment Class	2
Seismic Category	I
Safety Injection Pump	
Type	Horizontal multi-stage centrifugal pump
Number	4
Power Requirement	970 kW
Design Flow	1,540 gpm
Design Head	1,640 ft.
Minimum Flow	265 gpm
Design Pressure	2,135 psig
Design Temperature	300°F
Maximum Operating Temperature	Approximately 250°F
Fluid	Boric Acid Water
NPSH Available	21.9 ft. at 1,540 gpm
NPSH Required	15.7 ft.
Material of Construction	Stainless Steel
Equipment Class	2
Seismic Category	I

Note: All data in this Appendix is excerpted from DCD Chapter 6, Section 6.3

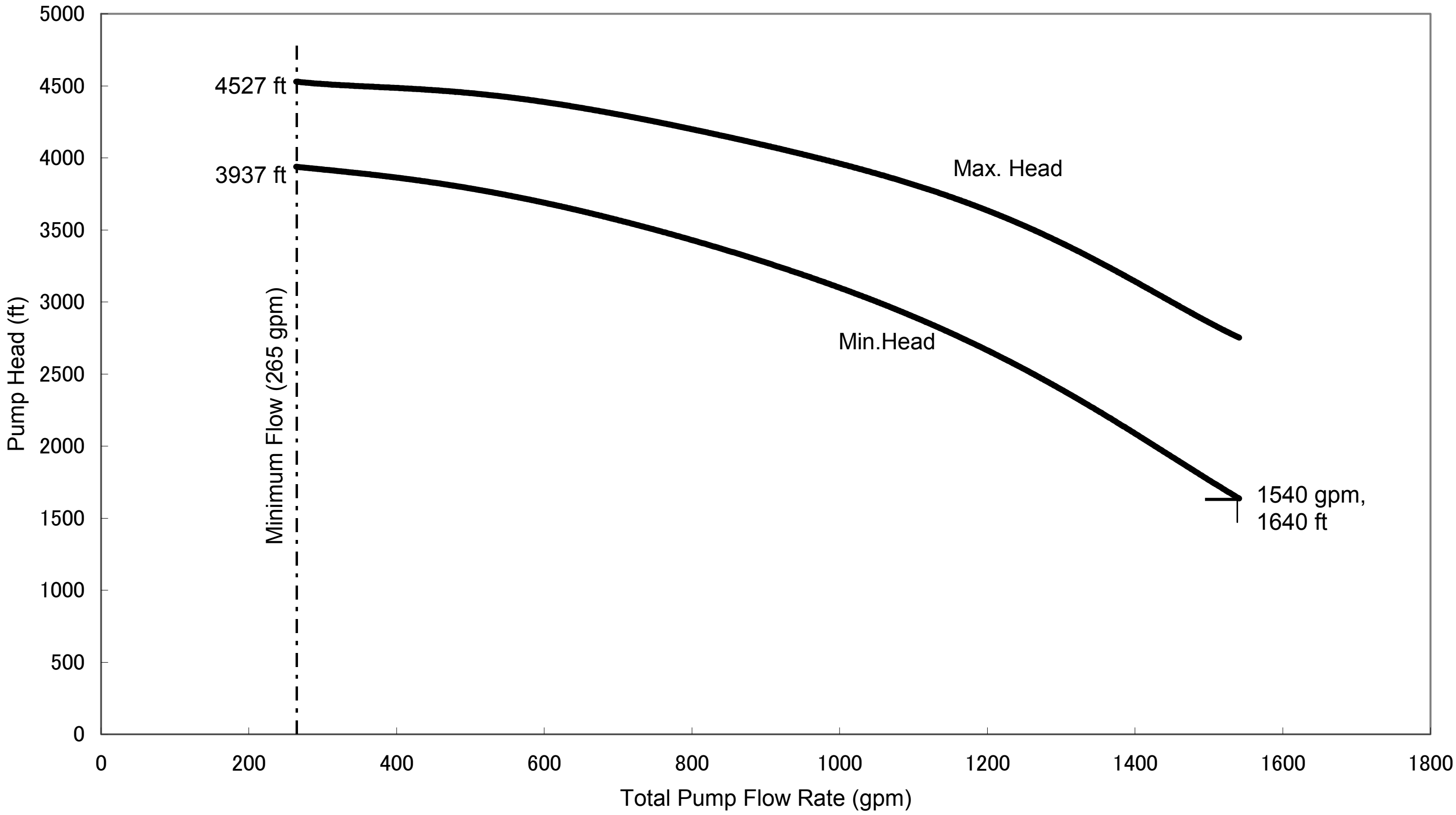

Figure A-1 Safety Injection Pump Performance Flow Requirement

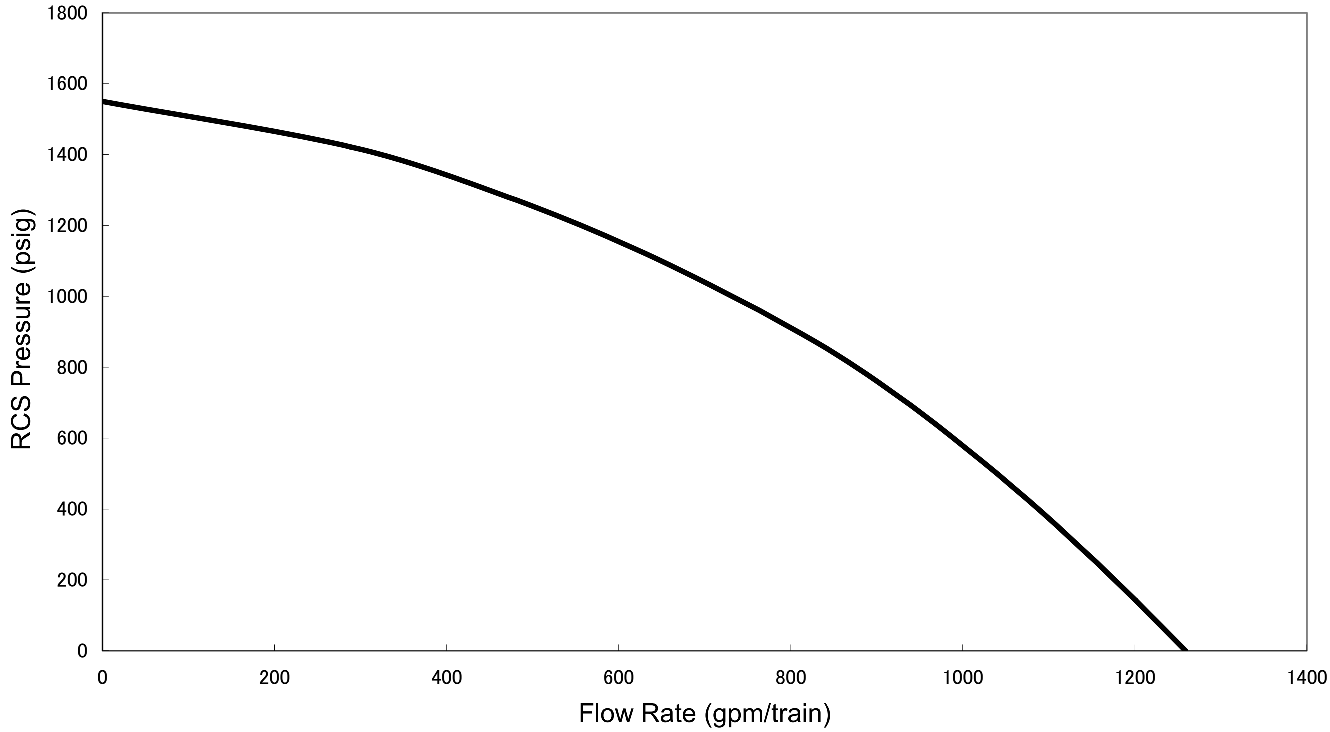

Figure A-2 High Head Safety Injection Flow Characteristic Curve (Minimum Safeguards)

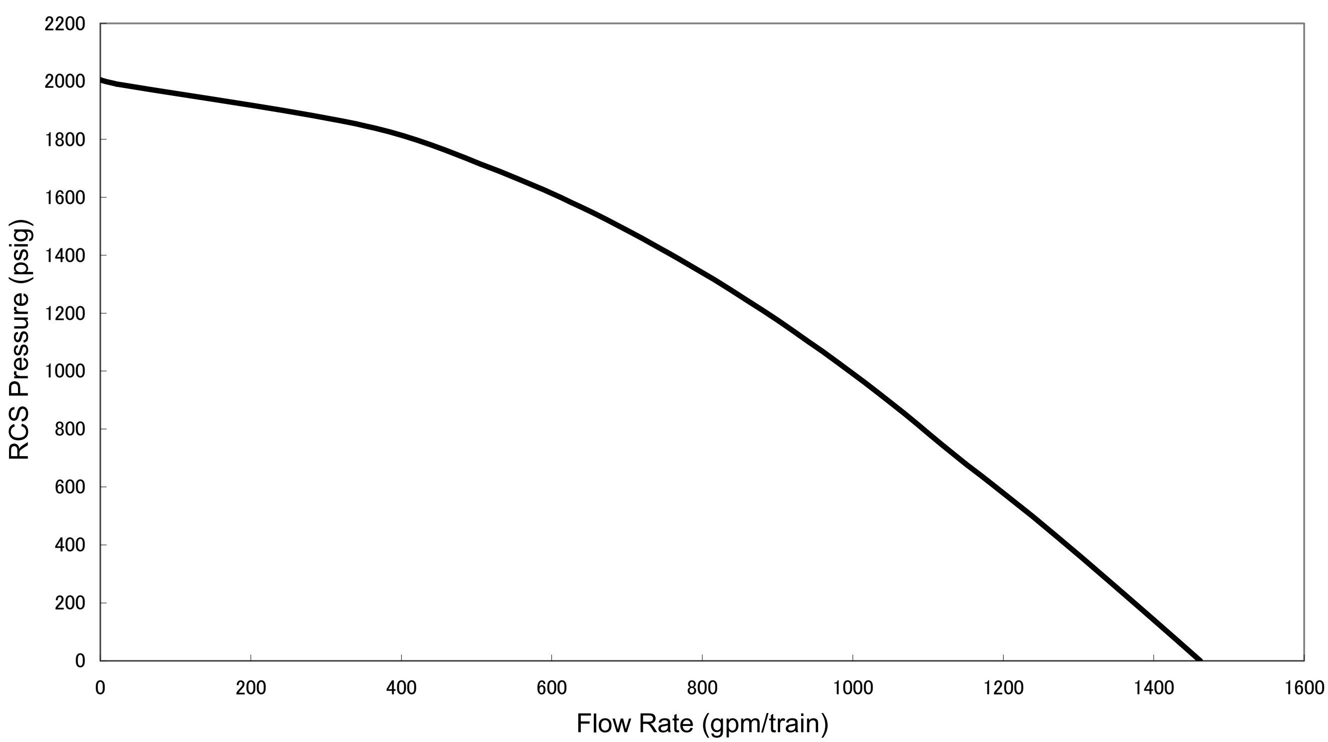

Figure A-3 High Head Safety Injection Flow Characteristic Curve (Maximum Safeguards)

Appendix B

Containment Spray / Residual Heat Removal Pump
Containment Spray / Residual Heat Exchanger
Engineering Design Parameters

Table B-1 Containment Spray/Residual Heat Removal Pump Design Parameters

Containment Spray/Residual Heat Removal Pump	
Number	4
Type	Horizontal, centrifugal type
Power Requirement (kW)	400
Design Flow Rate (gpm)	3,000
Design Head (ft)	410
Minimum Flow Rate (gpm)	355
Maximum Flow Rate (gpm)	3,650
Design Pressure (psig)	900
Design Temperature (° F)	400
Material	Stainless Steel
Normal Operating Temperature (° F)	32 ~ 356
Fluid	Reactor coolant, Boric acid water
Radioactive Concentration (kBq/cm^3)	≥ 37
NPSH Available (gpm)	17.9 ft at 3,650
NPSH Required (gpm)	16.4 ft at 3,650 gpm
Equipment Class	2

Note: All data in this Appendix is excerpted from DCD Chapter 5, Subsection 5.4.7

Table B-2 Containment Spray/Residual Heat Removal Heat Exchanger Design Parameters

Containment Spray / Residual Heat Exchanger		
Number	4	
Type	Horizontal U-tube type	
Heat Transfer Rate (Btu/h)	17.1×10^6	
Overall heat Transfer Coefficient and the effective heat transfer area, UA (Btu/h/° F)	1.852×10^6	
	Tube side	Shell side
Design Pressure (psig)	900	200
Design Temperature (° F)	400	200
Design Flow Rate (lb/h)	1.5×10^6	2.2×10^6
Design Inlet Temperature (° F)	120	99.7
Design Outlet Temperature (° F)	108.7	107.4
Material	Stainless steel	Carbon Steel
Fluid	Reactor coolant, boric acid water	Component cooling water
Radioactive Concentration (kBq/cm_3)	≥ 37	<37
Equipment Class	2	3

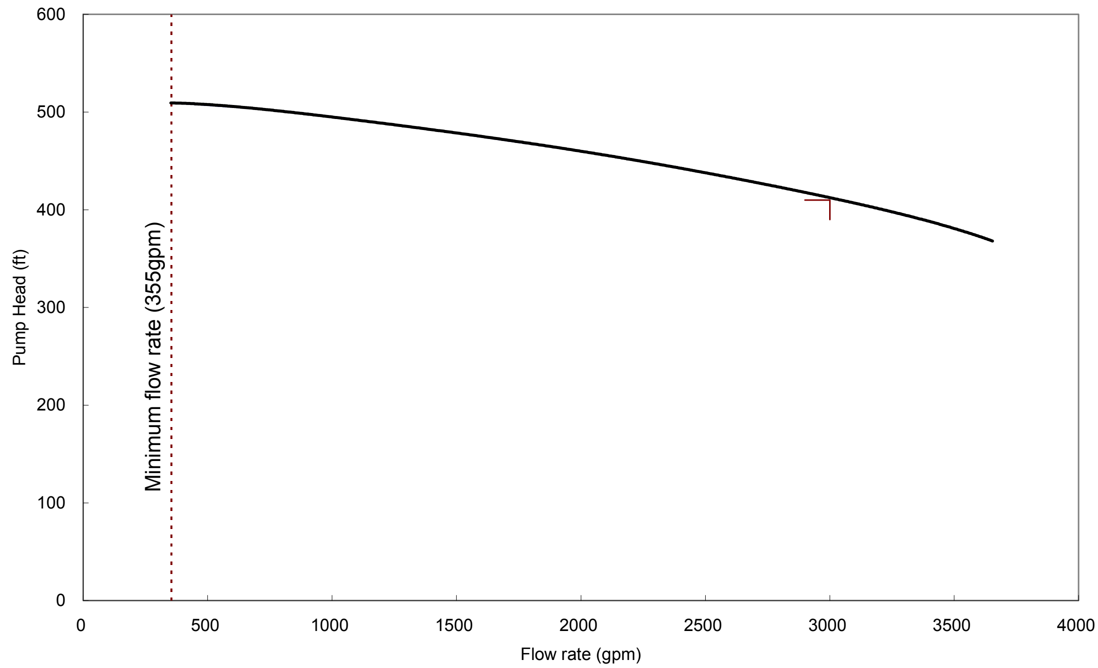

Figure B-1 CS/RHR Pump Characteristic Curve

Appendix C

Long-Term Core Cooling Acceptance Basis For GSI-191

The long term core cooling criteria described in this document are based on the requirements of Title 10 of the Code of Federal Regulations, Part 50.46 (10 CFR 50.46). The criteria are to be used with engineering evaluations that demonstrate acceptable and continuous long-term core cooling successfully after established following the initial recovery of the core subsequent to occurrence of LOCA.

An U.S. industry requested NRC clarify its long-term core cooling requirements under Title 10 of the Code of Federal Regulations Part 50.46 (10 CFR 50.46) to use in developing the GSI-191 debris ingestion evaluation method for reactor fuel (Ref. C-1).
It is requested that the US.NRC provides clarification of the requirements and acceptance criteria for long term core cooling once the core has quenched and reflooded.

The US.NRC responded to the request for clarification by the latter(Ref. C-2) that provides the basis for defining long term core cooling requirements that may be used to address long-term core cooling for GSI-191.

As described in the US.NRC response the long term core cooling acceptance bases defined for GSI-191 are applied after the initial quench of the core and consistent with the long-term core cooling requirements stated in 10 CFR 50.46 (b)(4) and 10 CFR 50.46 (b)(5). These acceptance bases provide for demonstrating that local temperatures in the core are stable or continuously decreasing and that debris entrained in the cooling water supply will not affect decay heat removal.

In order to demonstrate the long term core cooling related to the core and fuels of the US-APWR following acceptance bases are applied for cladding temperature in downstream effects evaluation.

The maximum temperature of the fuel cladding is maintained below [ ] in the situation of the debris reaching up to the core after LOCA. The maximum temperature, [ ], defines acceleration behavior of the cladding oxidation and the effects of the cladding mechanical properties due to hydrogen absorption. The autoclave corrosion tests for [ ] days proved that no-acceleration behavior of the cladding corrosion was observed below [ ] testing temperature. This acceptable temperature is based on the results of zircaloy-4 cladding test, give the conservatism for ZIRLO^{TM} cladding.

REEERENCES

C-1. "Requested NRC Action from Meeting with Westinghouse on April 12, 2006; Acceptance Criteria for Long-Term Core Cooling following Quenching and Reflooding of the Core; PWR Containment Sump Downstream Effects Resolution of GSI-191," , LTR-NRC-06-46, July 14, 2006.

C-2. Nuclear Regulatory Commission, Regarding Pressurized Water Reactor (PWR) Containment Sump Downstream Effects," (Response to Westinghouse Letter LTR-NRC-06-46 Dated July 14, 2006), dated August 16, 2006

Appendix D

Volume of Debris for In-Vessel Downstream Effects Evaluation

D.1 Bypass debris

This Appendix discusses the bypass debris that would pass through the sump strainer perforate plates designed for the US-APWR. The calculation of bypass debris (fiber) was made by engineering judgment, referring the debris penetration test results (Ref. D-1), and comparing the test conditions with design parameters of the US-APWR sump strainer (Ref. D-2). In the calculation, it was conservatively assumed that all of generated particle debris would bypass the strainer.

The result of the bypass debris calculation is summarized in Table D-1.

Table D-1 Bypass debris

Debris	Type	Generation	Bypass	Remarks
Fiber	NUKON	58.5 (ft^3)	16.5 (ft^3)	Includes latent fiber
Particle	Coating	18 (ft^3)	18 (ft^3)	DBA epoxy coating
	Latent particle	1 (ft^3)	1 (ft^3)	

D.2 Volume of debris existed in the reactor vessel

The cold leg break case is discussed here because it is the easiest case to accumulate the bypass debris in the reactor vessel due to the stay of flow.

It is conservatively assumed that all of the fiber bypass debris is accumulated in the lower plenum region in spite of the possibility that it would be suspended. In addition, all of the particle bypass debris is assumed to be accumulated in the lower plenum region. Based on these assumptions, it can be estimated that the total volume of the bypass debris in the reactor vessel is 35.5 ft^3 from Table D-1. However, it is expected that the particle bypass debris (19 ft^3) settles down to the bottom of the lower plenum due to the low flow rate.

D.3 References

D-1. U.S.NRC, Screen Penetration Test report, NUREG/CR-6885

D-2. US-APWR Sump Strainer Performance, MUAP-08001,Revision 2, December 2008

Appendix E

In-Vessel Downstream Effects Evaluation Time

E.1 INTRODUCTION

The debris that potentially impacts downstream components of the US-APWR is generated upstream the sump strainer located in the refueling water storage pit (RWSP). The RWSP is provided as the water source for long term core cooling after a LOCA, and is located at bottom elevation of containment in order to collect containment spray and blowdown water by gravity.

The sump strainer system is designed to filter the debris transported to the RWSP. (Ref. E-1) However, a certain amount of fine debris which may pass through the perforated plate of the strainer system will move downstream in the system and reach the reactor vessel (in-vessel).

The purpose of this document is to demonstrate the time required for the debris to reach the reactor vessel after the accident. Shorter and more conservative time values were used for downstream evaluations provided in subsection 4.1.2, 4.2.1 and 4.3.1.

E.2 FACTS AND ASSUMPTIONS

The following facts and assumptions associated with debris generation and transportation were applied in the evaluation. The detailed discussions were provided in subsection 3.4 “debris transport” and subsection 3.7 “upstream effect” of the technical report. (Ref. E-1)

2.1 The debris due to accident is transported to the RWSP only by return water through drain pipes surrounded by dike and debris interceptor. <Fact> (Ref. E-1)

2.2 Return water consists of containment spray, blowdown water and spilled water, which is supplied from the RWSP during accident. <Fact> (Ref. E-1)

2.3 Return water with the debris travels to the RWSP during recirculation, and fills up ineffective pools before returning to the RWSP through the drain pipe. <Fact> (Ref. E-1)

2.4 Holdup volume consists of “return water on the way to the RWSP” plus “ineffective pools” was calculated, and used for determination of minimum water level of the RWSP during accident. <Assumption> (Ref. E-1)

2.5 The debris entering to the RWSP is transported toward the sump strainer, and filtered by perforated plates. Then, a certain amount of fine debris passes through the perforated plate. Afterward, the debris reaches the reactor vessel . <Fact>

2.6 It was conservatively assumed in the evaluation that when the debris enters to the RWSP, it also reaches in-vessel at the same time. In other words, when the RWSP water is consumed for holdup volumes, and the RWSP reaches its the minimum water level, the debris will reach the reactor vessel. <Assumption>

2.7 In the calculation, it was assumed that all of safety pumps (four trains) would be operated during recirculation in order to estimate earliest time for the purpose. <Assumption>

E.3 CALCUALTIONS

The calculation for debris to reach in-vessel was provided in following table;

Water volumes (m^3)

Return water on the way to the RWSP	: 519.7
Ineffective pools	: 1,124.1
Minimum water level (margin for design basis)	: 80
Total	: 1,723.8

Pump flow rate (four train operation, gpm)

CS/RHR pump	: 2,450 x 4
SI pump	: 1,540 x 4
Total	: 15,960

Required time for debris to reach in-vessel

1723.8(m^3) x 264.178(gal/ m^3) / 15,960(gpm) x 60 (sec) = **1,712 (sec)**

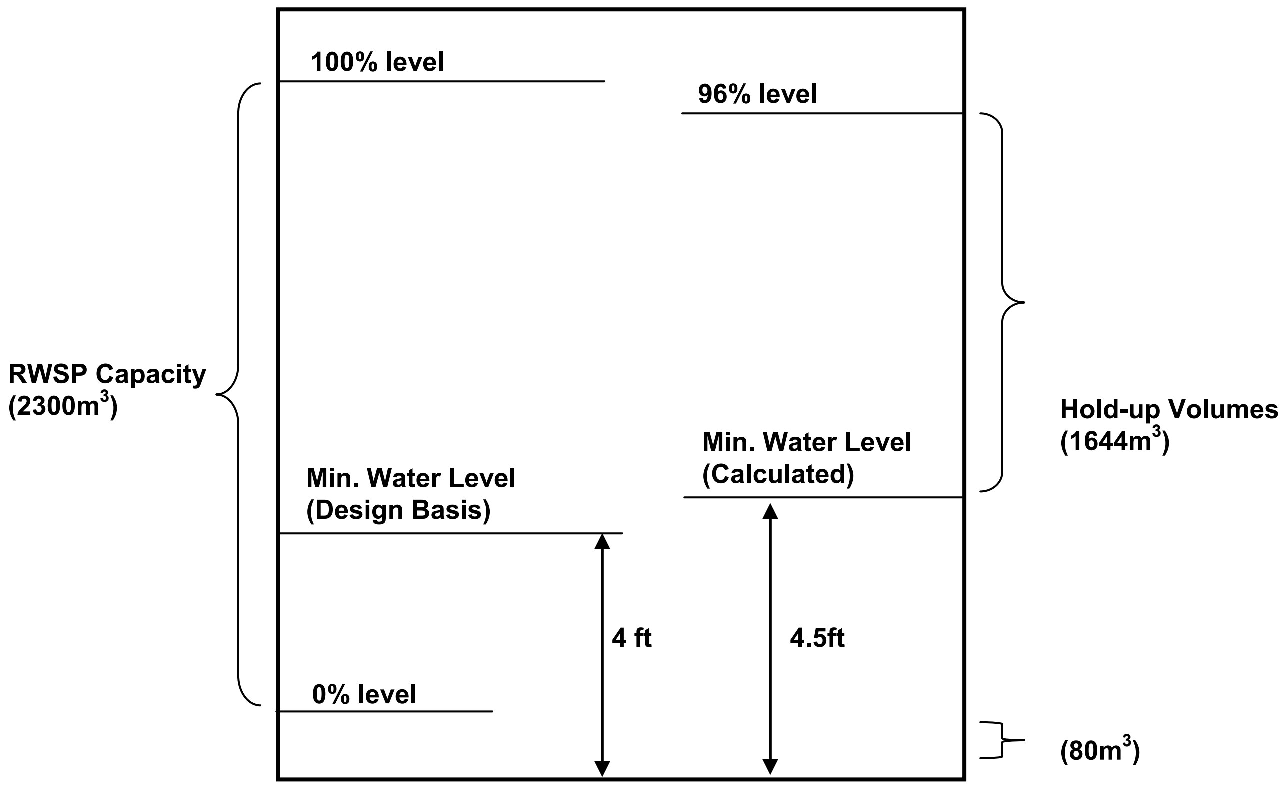

Figure E-1 Minimum water level of the RWSP Note

Note : Original figure was provide in Figure 3-9 of technical report (Ref. E-1)

E.4 RESULT

It was concluded that the time required for debris to reach in-vessel will be approximately 1,700 (sec). Ultimately, a safety margin of 2 is considered, and it was set to be **850 (sec)** for downstream evaluations.

E.5 REFERENCE

E-1 US-APWR Sump Strainer Performance, MUAP-08001(R2), December 2008, Mitsubishi Heavy Industries, Lid.

Appendix F

Confirmation of Calculation of Deposit Process by the Evaluation Tool

F.1 Introduction

To investigate confirmation of the evaluation tool with assumption described in session 4.3.1.3.1, deposition process was simulated by the evaluation tool. Experimental results referred by Brahim et al. (Ref. F-1) are chosen as confirmatory calculation.

F.2 Calculation condition

Calcium sulfate was deposited on an electrically heated tube in a laboratory test reported by Brahim et al. In the test, a calcium sulfate solution near saturation entered a tube at 176°F (80°C) and was heated causing precipitation on the heat transfer surface. The temperature of the heat transfer surface was monitored over time as calcium sulfate precipitant. The fouling resistance was calculated and plotted.

Because concentration of Calcium sulfate and heat flux were kept constant, fouling resistance could be expressed as the following equation by the analytical integration of the equation (4.3.1-2),

$$R = \dot{x} / k = q\ C\ t /(D\ h_{fg}) / k \qquad \text{...(F-1)}$$

where:

R = fouling resistance (m^2K/W),

x = thickness (m),

k = thermal conductivity (W/m/K),

q = heat flux (W/m^2),

C = concentration (kg/kg),

t = time (s),

D = density (kg/m^3),

h_{fg} = latent heat (J/kg).

Thermal conductivity of deposition of Calcium sulfate depends on presentation of water in the pores (Ref. F-2). In this estimation, thermal conductivity of deposition was chosen as largest value without the pore.

F.3 Result

The agreement between the prediction by the evaluation tool and the experimental result by Brahim are shown in Figure F-1.
In this calculation, thermal conductivity of deposition was chosen as largest value in order to confirm conservative prediction by the estimation tool. Even if an assumption of largest thermal conductivity was utilized, fouling resistance became larger than the experimental results.
By conservative evaluation of fouling resistance compared with the experiment, it was concluded that this evaluation tool could conservatively predict chemical deposition processes.

Figure F-1 Comparison of Fouling Resistance for Calcium Sulfate Deposition

F.4 Reference

F-1 Fahmi Brahim, Wolfgang Augustin, Matthias Bohnet, "Numerical simulation of the fouling process", International Journal of Thermal Science, Vol. 42, 2003, 323-334

F-2 S. Krause, "Fouling of heat-transfer surfaces by crystallization and sedimentation", International Chemical Engineering, Vol. 33, 1993, 355-401