US-APWR Sump Strainer Performance

Non-Proprietary Version

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Revision History (Sheet 1 of 8)

Revision	Page (Section)	Description
5 (Aug-2011)	1	Section 1.0, 1st paragraph Editorial. The last sentence was changed to present form.
	1	Section 1.0, 2nd paragraph Editorial. The article “the” was added to the first sentence. (Typical two places.) Replaced with the term “sources of potential debris”.
	1	Section 1.0, 2nd paragraph Editorial. The last sentence was divided to two sentences.
	2	Section 2.1, 1st paragraph 1st bullet Editorial. State “strainer systems” in plural form.
	2	Section 2.1, 1st paragraph 5st bullet Editorial. State “a low flow rate’ in singular form.
	2	Section 2.1, 1st paragraph 6st bullet Clarification. Reword “assure core cooling”.
	3	Section 2.1, 5th paragraph, 3rd sentence Editorial. Replaced “for attachment” with “for attaching”
	3	Section 2.1, 5th paragraph Editorial. The article “the” was added to the last sentence. (Typical two places.)
	3	Section 2.1, 6th paragraph, 1st sentence Editorial. Replace preposition “filled by” with “filled with”.
	3	Section 2.1, 7th paragraph, insert a sentence Clarification. Besides nominal strainer surface area, minimal surface area (i.e., 2,730ft^2 per sump) accounting for tolerance is discussed which will be the acceptance criteria of the as-built strainer (ITAAC item).
	3	Section 2.1, 7th paragraph, last sentence Editorial. Reword “A minimum of two out of four safety trains”. Reword “long term core cooling”. Insert “the” before “postulated accident”.
	9	Section 3.1, 13th paragraph Clarification. Replaced with more direct justification for consistency with break criterion 5.
	17	Section 3.4.2, 2nd and 3rd paragraph Clarification. Replaced with the discussion how to determine the amount of fiber bypass debris used for downstream evaluation.
	17	Section 3.5, section title Editorial (terminology). Section title was changed to “debris head loss and vortexing” to include air ingestion issues associated with sump strainer performance.

Revision History (Sheet 2 of 8)

Revision	Page (Section)	Description
5 (Aug-2011)	17-19	Section 3.5 Clarification. Whole section was replaced with discussions regarding the definitions of various head losses and applicability of temperature scaling. Additional discussions regarding potential air ingestion due to excessive head loss caused by two-phase flow were prepared.
	20	Section 3.6.1, 1st paragraph Editorial. The article "the" is added before "ECC".
	20	Section 3.6.2, 1st paragraph Editorial. Insert "to be" before "equal to".
	20	Section 3.6.2.1-a), 1st paragraph Editorial. Reword to "The US-APWR SI pumps and CS/RHR pumps ~~of the US-APWR~~ consist of".
	21	Section 3.6.2.1-a), 2nd paragraph Editorial. Replace "a second one" with "one pump". Replace "remains" with "remaining".
	21	Section 3.6.2.1-b) Clarification. Reworded the terminology "containment overpressure" by "additional containment pressure". Additional clarification for the terminology "additional containment pressure".
	23	Section 3.6.2.1-e), (2) Precipitation of chemical debris, 2nd paragraph Editorial corrections to add articles "the" (typical two places), reword "a PH of" (typical two places), reword "Refer to". Replace "the specific US-APWR" with "the US-APWR specific".
	23	Section 3.6.2.1-e), (2) Precipitation of chemical debris, 3rd paragraph Editorial. Add an article "the" before "temperature".
	24	Section 3.6.2.1-e), (2) Precipitation of chemical debris, 6th paragraph Editorial. Add the article "the" for typical two places.
	24-25	Section 3.6.2.1-e), (2) Precipitation of chemical debris, before the last paragraph, Editorial. The first sentence was changed to present form. The second sentence was reworded by "Al is approximately 135F". Replace with a verbal noun "observing".
	25	Section 3.6.2.1-f) Clarification. Add new subsection f) to define the design basis NPSH requirement used for NPSH calculations.
	27	Section 3.6.2.2, 7th paragraph Editorial. The article "the" is added before "sump water".
	27	Section 3.6.2.2, 9th paragraph Editorial. Replace "water level is used 3.8" with "water level used is 3.8".

Revision History (Sheet 3 of 8)

Revision	Page (Section)	Description
5 (Aug-2011)	27	Section 3.6.2.2, 13th paragraph Clarification. The source of Figure 3-14 was changed to DCD Figure 6.2.1-26 which shows the latest RWSP water temperature profile during accident. The design RWSP peak temperature (i.e., 270F) is clarified.
	28	Section 3.6.2.2, before the last paragraph Clarification. Add Table 3-11 and Table 3-12 to prepare input values for NPSH calculations.
	28	Section 3.6.2.2, last paragraph Clarification. The minimum NPSH margin was updated. (1.2 ft at 204ºF) Editorial changes in the last sentence.
	28	Section 3.6.2.3, section title Clarification. The former section, “Vortexing, Air Injection, and Steam Flashing” was re-titled to “Effects of Air Ingestion on NPSH” The air ingestion assessments were prepared separately for the strainer performance and for the pump performance respectively. The assessment regarding strainer performance was prepared in section 3.5. The assessment regarding pump performance (i.e., effects on NPSH) was prepared in section 3.6.2.3.
	31	Section 3.7.1, 3rd paragraph Editorial. Editorial corrections were made in the second sentence.
	31	Section 3.7.1, 4th paragraph Editorial. The article “The” was added to the beginning of the sentence. Reworded “capable of draining” in the first bullet.
	32	Section 3.7.1, 5th paragraph Editorial. Replace “operated” with “operating”. Reworded “remain” by “the remaining”.
	32	Section 3.7.1, 6th paragraph Editorial. The articles “the” and “a” were added.
	32	Section 3.7.1, 7th paragraph Editorial. Replace “spills out from reactor” with “spills out of the reactor”.
	32	Section 3.7.1, 7th paragraph, 1st bullet Editorial. Reworded “capable ~~to~~ of draining”
	32	Section 3.7.1, 9th paragraph Editorial. Add the articles “the” and “a” in typical four places.
	32	Section 3.7.1, 10th paragraph Editorial. Insert the article “the” to say “all of the debris”. Insert “does” to say, “ and does not contribute”.
	33	Section 3.7.1, 11th paragraph Editorial. Insert “flow” to say, “make-up flow to the RWSP”.

Revision History (Sheet 4 of 8)

Revision	Page (Section)	Description
5 (Aug-2011)	34	Section 3.7.2, 2nd paragraph Editorial. Remove the article "the" before "water evaporation".
	34	Section 3.7.2, 3rd paragraph Editorial. Add the articles "the" in typical two places.
	34	Section 3.7.2, 4th paragraph Editorial corrections were made to whole paragraph.
	38	Section 3.9, 1st paragraph Editorial. Reword to state "reach ~~to~~ the strainers", and "the vital pumps of the US-APWR."
	39	Section 3.9, 2nd paragraph Editorial. The articles "a" and "the" are added in typical two places.
	52	Figure 3-14 Clarification. Figure is replaced with the latest RWSP temperature profile prepared in the DCD Figure 6.2.1-26. The temperature plots was expanded to 800hrs.
	53-55	Figure 3-15, 3-16, 3-17 Clarification. Figures were updated by NPSH calculation based on the design basis strainer head loss.
	57	Table 3-4 Debris Generation Typo. A typographic error regarding fiber debris (0.1875 ft^3) was corrected as 1.875 ft^3.
	59	Table 3-6 Debris Head Loss Clarification. The table was updated to incorporate measured head losses at the test and design basis head loss corresponding to the RWSP temperatures during accident.
	60	Table 3-7 NSPH Calculation Matrix Clarification. The table was updated to incorporate design basis head losses corresponding to the RWSP water temperatures during accident.
	61	Table 3-8 Editorial. Re-titled to "sump fluid flashing calculation cases".
	61	Table 3-9 HL vs. Submergence Clarification. The table was updated to incorporate design basis strainer head losses corresponding to the RWSP temperatures during accident. Initial dry air partial pressure (i.e., 29ft) is updated.
	62-63	Clarification. Table 3-11 and 3-12 were added to prepare inputs for NPSH calculation.
	59	Section 5.0 Editorial. The former statement "consistent with the requirements in RG 1.82" was replaced with "consistent with the guidance in RG 1.82". Other minor editorial changes were made.

Revision History (Sheet 5 of 8)

Revision	Page (Section)	Description
5 (Aug-2011)	B-1	Appendix-B, B.1, 2nd paragraph Editorial. Add the article “the” before “Thin Bed Test”.
	B-1	Appendix-B, B.1, Test-1 Editorial. Relocate the adverb “separately” before the verb “determine”.
	B-2	Appendix-B, B.1 Test-3 Editorial change was made on last sentence of Test-3 “Debris laden test strainer head loss”.
	B-3	Appendix-B, B.2, 3rd paragraph Editorial. Replace “same the configuration” with “the same configuration”.
	B-3	Appendix-B, B.2, 5th paragraph Editorial. Minor editorial changes in typical two places.
	B-3	Appendix-B, B.2, 6th paragraph Editorial. Minor editorial change, “the safety pumps”.
	B-5	Appendix-B, B.4, 2nd paragraph Editorial. Reword to “not contribute to”.
	B-6	Appendix-B, B.5, 1st paragraph, note Editorial. Insert “a” before “computer”.
	B-6	Appendix-B, B.5, 2nd paragraph, 2nd bullet Editorial. The word “elevated” was replaced with “heated”.
	B-7	Appendix-B, B.6, Test-3 Editorial. The article “the” was inserted before “strainer”.
	B-7	Appendix-B, B.7, the last sentence Editorial. Refer new subsection B.8 for tests results discussion.
	B-8	Appendix-B, B-8 Test results Clarification. The section was extended to incorporate the tests results and post-test assessment regarding strainer head loss tests and bypass fiber tests.
	B-12	Appendix-B, B-9 Reference Clarification. A vendor test plan document (B-1) was updated.
	B-12	Appendix-B, B-9 Reference Clarification. A vendor test report (B-12) was added.
	B-16 to 22	Appendix-B, Figures Clarification. Figure B-6 to B-14 (test pictures) were added.

Revision History (Sheet 6 of 8)

Revision	Page (Section)	Description
5 (Aug-2011)	B-24 to 26	Appendix-B, Tables Clarification. Table B-2 to Table B-6 (test results summary) were added.
	C-4	Appendix-C, Table C-1 Editorial. Correct typo regarding unit (micro gram per litter) as per RAI 06.02.02-15.
	C-4	Appendix-C, Table C-1 Clarification. Update recirculation water volume as per RAI 06.02.02-64. Calculated chemical debris were updated accordingly.
6 (May-2012)	3	Section 2.1 Revised description of return water flow path. Updated calculated RWSP hold up volume “93,150 ft^3 (696,800 gallons). Replaced “A minimum of 2,754 (ft^2) per sump accounted for fabrication tolerance will be constructed in the plant” with “Fabrication tolerances shall be specified during strainer procurement to provide the per sump minimum surface area of 2,754 ft^2 for the as-built strainers.”
	6	Figure 2-1. Changed minimum submergence from “12.73” inches to “12.75” inches.
	9	Section 3.1, 13th paragraph Changed maximum fiber debris allocated to one sump from “25.5” lbs to “30.0” lbs, from “10.625” ft^3 to “12.5” ft^3 and from “0.0463” inches to “0.0545” inches.
	11	Section 3.2, 8th paragraph Updated chemical debris amount from “145 lbs of aluminum hydroxide and 160 lbs of sodium aluminum silicate” to “300 lbm of aluminum hydroxide and 330 lbm of sodium aluminum silicate).
	14	Section 3.4.1, 1st paragraph Updated assumptions for debris split from “70%/30%” to “85%/15%”. Section 3.4.1, 5th- 7th paragraphs Replaced 3rd and 4th sentences of 5th paragraph with “The generated debris flows through openings in the SG compartment floors to the buffer areas…the US-APWR disregarded this captured debris in the transport calculation. Deleted 6th & 7th paragraphs.
	16	Section 3.4.1, 8th - 11th paragraphs Updated debris distribution to operable sump strainers. Changed debris split amount from “70%/30%” to “85%/15%”, from “60%/40%” to “67%/33%” and added “operable” and “operable sump” for clarification.
	17	Section 3.4.2, 2nd paragraph Updated the plant fiber debris amount, from “17.4%” to “14.5%”.

Revision History (Sheet 7 of 8)

Revision	Page (Section)	Description
6 (May-2012)	18	Section 3.5, 5^{th} paragraph Editorial change was made. .
	19	Section 3.5, 9^{th} paragraph Changed strainer submergence during long-term recirculation from “1’-0.73’”” to “1’-0.75’””. Replaced “prevents” with “prevent” for editorial change.
	19	Section 3.5, 9^{th} paragraph and 11^{th} paragraph Changed strainer submergence during long-term recirculation from “1’-0.73’”” to “1’-0.75’””. Replaced “prevents” with “prevent” for editorial change. Updated void fraction amount from “0.44%” to “0.43%”. Replaced “preclude” with “limit” for editorial change.
	22	Section 3.6.2.1, (b) Containment Pressure Editorial corrections were made for clarification.
	31	Section 3.7.1, 2^{nd} paragraph Replaced the word “transfer” with “overflow”. Updated “Refueling Cavity Drains”.
	32-34	Section 3.7.1 Updated “Refueling Cavity Drains, and Ineffective pools” subsections.
	34	Section 3.7.2, 1^{st} paragraph Minor editorial change was made in the 1^{st} sentence.
	34-35	Section 3.7.2, 2^{nd}-4^{th} paragraphs Changed borated water volume from “2,300 m^3“to “84,750 ft^3 (634,000 gallons). Changed the design basis minimum water level from “18 m^3“to “1,370 ft^3“, and initial depth water in the RWSP from “18’-4”" to “16’-7””.
	43-44	Section 3.9 Figures and Tables were updated as follows. Figure 3-5 Schematics of Debris Allocation on Operable Sumps (for Bypass Debris) Figure 3-6 Schematics of Debris Allocation on Operable Sumps (for bypass debris)
	47	Figure 3-9 Schematic of Return Water and Hold-up Volumes
	49	Figure 3-11 Minimum Water Level of the RWSP
	57	Table 3-4 Debris Generation Updated chemical debris amount.
	59	Table 3-7 NPSH Calculation Matrix Table was updated by NPSH calculation. Short-term water temperature level changed from “256>T>204” to “270>T>204”

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Revision	Page (Section)	Description
6 (May-2012)	57	Table 3-4 Debris Generation Updated chemical debris amount.
	59	Table 3-7 NPSH Calculation Matrix Table was updated by NPSH calculation. Short-term water temperature level changed from “256>T>204” to “270>T>204”
	60	Table 3-8 Sump Fluid Flashing Calculation Cases Clarification. Table was updated by Sump Fluid Flashing calculation. Design basis temperature changed to “270>T>204” for short-term post-LOCA recirculation.
	60	Replaced “Table 3-9 Head Loss vs. Submergence (Static Head) plus Initial Dry Air Pressure” with “Table 3.9 Static Pressure at Strainer Outlet vs. Vapor Pressure”. New content of Table 3-9 was added.
	61	Updated Table 3-10 Upstream Effect Hold-up Volumes.
	66	Updated the revision number and date for Reference 13.
	67	Updated the revision number and date for Reference 17.
	A-6	Updated Figure A-3 Sump Strainer Arrangement.
	B-7	Typo. Changed from “confucted” to “conducted”.
	C4	Appendix-C Section C.2 Chemical Debris Calculation: Updated Table C-1 Chemical debris of the US-APWR
	C4-C5	Appendix-C, Section C.3 Maximum Recirculating Water Volume Section C.3 was added.
	C5	Appendix-C Section C.4 References Revised section number to “C.4” due to an addition of new section “C.3”.
	D-1	Appendix-D Updated Figure D-1 Containment Plan View (RWSP, EL. 3’-7’’)
	D-2	Appendix-D Updated Figure D-2 Containment Plan View (EL. 25’-3’’)
	D-3	Appendix-D Updated Figure D-3 Section View.
	D-4	Appendix-D Modified Figure D-4 Nozzle and Equipment Detail were added.
	F-3	Appendix-F, Section F.1 Containment Pressure during LOCA Editorial. Replaced “containment” with “steam partial” in the 2nd paragraph.
	F-9	Typo. Replaced “consequencen” with “consequence”.

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Appendices

List of Acronyms

AlOOH	Aluminum Oxy-hydroxide
APWR	Advanced Pressurized Water Reactor
ARL	Alden Research Laboratory
CO/L	Cross Over Leg
CSHL	Clean Strainer Head Loss
CS/RHR	Containment Spray/Residual Heat Removal
CSS	Containment Spray Systems
CVCS	Chemical and Volume Control System
DBA	Design Basis Accident
DCD	Design Control Document
DEGB	Double Ended Guillotine Break
ECCS	Emergency Core Cooling Systems
ECC/CS	Emergency Core Cooling and Containment Spray
GR	Guidance Report
HELB	High Energy Line Break
HVAC	Heating, Ventilation, and Conditioning
ICET	Integrated Chemical Effects Test
LANL	Los Alamos National Laboratory
LOCA	Loss of Coolant Accident
LBLOCA	Large Break LOCA
MBLOCA	Medium Break LOCA
MCP	Main Coolant Pipe
MHI	Mitsubishi Heavy Industries, LTD.
MS/FW	Main Steam and Feed Water
NPSH	Net Positive Suction Head
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PCI	Performance Contracting, Inc.
PWR	Pressurized Water Reactor
PZR	Pressurizer
RCP	Reactor Coolant Pump
RG	Regulatory Guide

RHR	Residual Heat Removal
RMI	Reflective Metal Insulation
RWSP	Refueling Water Storage Pit
RV	Reactor Vessel
SBLOCA	Small Break LOCA
SE	Safety Evaluation
SFS	Sure-Flow Strainer
SG	Steam Generator
SI	Safety Injection
TBE	Thin Bed Effect
TSHL	Total Strainer Head Loss
UNM	University of New Mexico
ZOI	Zone of Influence

1.0 INTRODUCTION

This technical report summarizes the design and evaluation of the standard US-APWR sump strainer, and supports the US-APWR Design Control Document (DCD), Chapter 6, Subsections 6.2 “Containment Systems”, and 6.3 “Emergency Core Cooling Systems (ECCS) (Reference [1]). The design and evaluation described herein is in accordance with the Regulatory Guide 1.82 Rev.3. (Reference [2])

In this report, Section 2 contains a description of the strainer, the type, location, and a summary of the design features relative to the insulation and coating systems which are generally considered sources of potential debris. Section 3 provides the evaluation of sump strainer performance including break selection, debris sources and generation, debris characteristics, debris transport, postulated debris head loss, net positive suction head (NPSH), and upstream effects of the US-APWR. Section 4 summarizes the downstream effect evaluations. Section 5 presents the conclusions of the US-APWR sump strainer performance.

2.0 DESIGN DESCRIPTION

2.1 ECC/CS strainer

The US-APWR emergency core cooling and containment spray (ECC/CS) strainers are designed to be consistent with Regulatory Guide (RG) 1.82 as follows:

- Four independent sets of strainer systems are provided
- The strainers are installed on the bottom floor of the containment to collect the blowdown water during the accident
- The design precludes the drain water impinging directly on the strainers
- The strainers are well isolated from postulated pipe break jets and missiles
- The strainer’s large surface provides a low flow rate on the strainer surface and mitigates debris head loss
- The perforate plates are designed to assure core cooling
- The strainers are constructed of corrosion resistant materials
- The strainers are sized to maintain the performance of the safety-related pumps
- The strainers are designed to meet seismic category I requirements
- The strainers are inspected periodically, during plant shutdown

As shown in the US-APWR DCD Figures 6.2.2-8 and 6.2.2-9, four independent sets of ECC/CS strainers are provided inside the in-containment refueling water storage pit (RWSP). The ECC/CS strainers prevent debris from entering the safety systems that are required to maintain the post-LOCA long-term cooling performance. The RWSP is located at the lowest part of the containment in order to collect containment spray water and blowdown water by gravity. The RWSP is compartmentalized by a concrete structure against the upper containment area, and connecting pipes that drain the collected water from the upper containment are provided in the ceiling of the RWSP. The RWSP protects the ECC/CS strainers from missiles. The ECC/CS strainers are installed on the bottom floor of the RWSP, and are designed to be fully submerged during all postulated events requiring the actuation of the ECCS.

The fully submerged strainers, in combination with the Safety Injection (SI) pump and the Containment Spray/Residual Heat Removal (CS/RHR) pump elevation, provide sufficient NPSH to ensure continuous suction availability without cavitation. The strainer sizing accommodates the estimated amount of debris potentially generated in containment.

There is no debris source (i.e. insulation, paints, concrete wall) in the RWSP. All debris is generated outside the RWSP. The debris will be transported to the RWSP by return water through overflow pipes that are located in the header compartment and reactor cavity. The overflow pipes are positioned so that drain rater does not impinge the strainer system. Vent pipes are also provided to equalize the atmospheric pressure between the RWSP and the upper containment. Detail drawings around the RWSP are provided in Appendix-D.

The standard US-APWR design utilizes a passive disk layer type of strainer systems, “Sure-Flow Strainer (SFS)”, supplied by Performance Contracting Inc. (PCI)“. Figure 2-1 shows a typical plan view of the disk type strainer system of one safety train used in the US-APWR. The strainer is principally constructed of perforated plate with a square flange at the bottom for attaching to the supporting plate, which is covered on the sump pit. A manifold core tube connected to the flange penetrates near the center of the layer disks, and guides the clean water filtered by the layer disks into the sump pit. The joint gap between the components of the strainer is controlled to preclude debris from bypassing the perforate plates. The strainers and supporting plates will be constructed of corrosion-resistant stainless steel. The nominal diameter of holes is designed to be equal or less than 0.066”, consistent with the narrow gap in the downstream systems of the strainer. The downstream narrow gap is discussed in Section 4.0 of this report. The technical description and detailed drawings of the strainer are provided in Appendix-A.

The RWSP is filled with 93,150 ft^3 (696,800 gallons) of borated water during normal operation, and is designed to hold a sufficient water volume during a loss of coolant accident (LOCA). An adequate water level is maintained to submerge the strainer in case of a LBLOCA. The strainers are installed so as to submerge the top of the layer disk 1 foot under the minimum water level. The water balance of the RWSP is summarized in US-APWR DCD Table 6.2.1-3, and its calculation is discussed in Section 3.7 of this report.

Each safety train has a minimum of 2,754 (ft^2) of strainer surface area arranged in 9 modules of 21 stacked disks each. Fabrication tolerances shall be specified during strainer procurement to provide the per sump minimum surface area of 2,754 ft^2 for the as-built strainers. A minimum of two out of four safety trains are required to maintain long term cooling after the postulated accident.

2.2 Insulation

The standard US-APWR design utilizes the zone of influence (ZOI) method for the evaluation of debris generation, as discussed in Section 3.1 and Section 3.2 of this report. The ZOI represents the zone where a given high-energy line break (HELB) will generate debris that may be transported to the strainer. The size of the ZOI is defined in terms of pipe diameters and determined based on the pressure contained by the piping and the destruction pressure of the insulation surrounding the break site. The ZOI for specific insulation types are provided in the approved methodology, the SE of NEI 04-07 (Reference [3])

In the Section 3.3.4.2.1 (Table 3-2) of the SE (Reference [3]), the reflective metal insulation (RMI) is seen to require a largest destruction pressure among the types of insulation made of fibrous and particulate materials. Therefore, the application of RMI for the piping and components subject to jet impingement from a HELB will minimize the generation of problematic insulation debris from the use of fibrous/particulate material insulations.

As a result, the US-APWR design utilizes the RMI, to the greatest extent practicable, for the piping and components subject to jet impingement from a HELB (i.e., reactor coolant system, and main steam and feedwater systems), in order to mitigate the generation of problematic insulation debris.

Following is the design of the insulation applying for the US-APWR equipment and pipe lines:

Equipment

RMI is applied to the reactor vessel (RV), the reactor coolant pumps (RCP), the steam generators (SG), and the pressurizer (PZR) in the areas that have large amount of insulation to be potentially subject to jet impingement from a HELB.

There is no other equipment to be insulated inside containment of the US-APWR. In addition, the heating, ventilation, and air conditioning (HVAC) of the US-APWR requires no insulation, nor even the ventilation filters, which are considered as the potential debris sources caused by HELB.

Piping

RMI is applied to the main coolant pipes (MCP) connecting the RV, the SG, and the RCP, and to MS/FW pipe lines. RMI is applied on the piping located inside the ZOI which are subject to the jet impingement of HELB. The use of fibrous insulation inside containment is limited and excluded from ZOI.

Others

The insulation design excludes the use of fibrous insulation in the ZOI, however the strainer design allows for the use of a limited amount of fibrous insulation inside containment (i.e., 4.5 lbs which is equal to 1.875 ft^3 of NUKON fiber insulation) as an operational margin for future plant modification.

2.3 Coatings

The standard US-APWR utilizes a DBA qualified and acceptable coating system in containment. The coating systems in containment are met with the requirement of Service Level-I coatings categorized in USNRC Regulatory Guide 1.54 Revision 1 (Reference [4]) and relative ASTM requirement described in R.G 1.54. The criteria for those coating systems are contained in ANSI N101.2, “Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities (Reference [5]),” and its successor document, ASTM D 3911, “Standard Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions (Reference [6]).” Only the epoxy type coatings (including primer and top coated) are used. The inorganic zinc coating systems are not used.

The coating design limits the type of coating as DBA epoxy however it allows for the existence of non-qualified or degraded coating (i.e., 200lbs of epoxy coating) inside containment as operational margin for plant life.

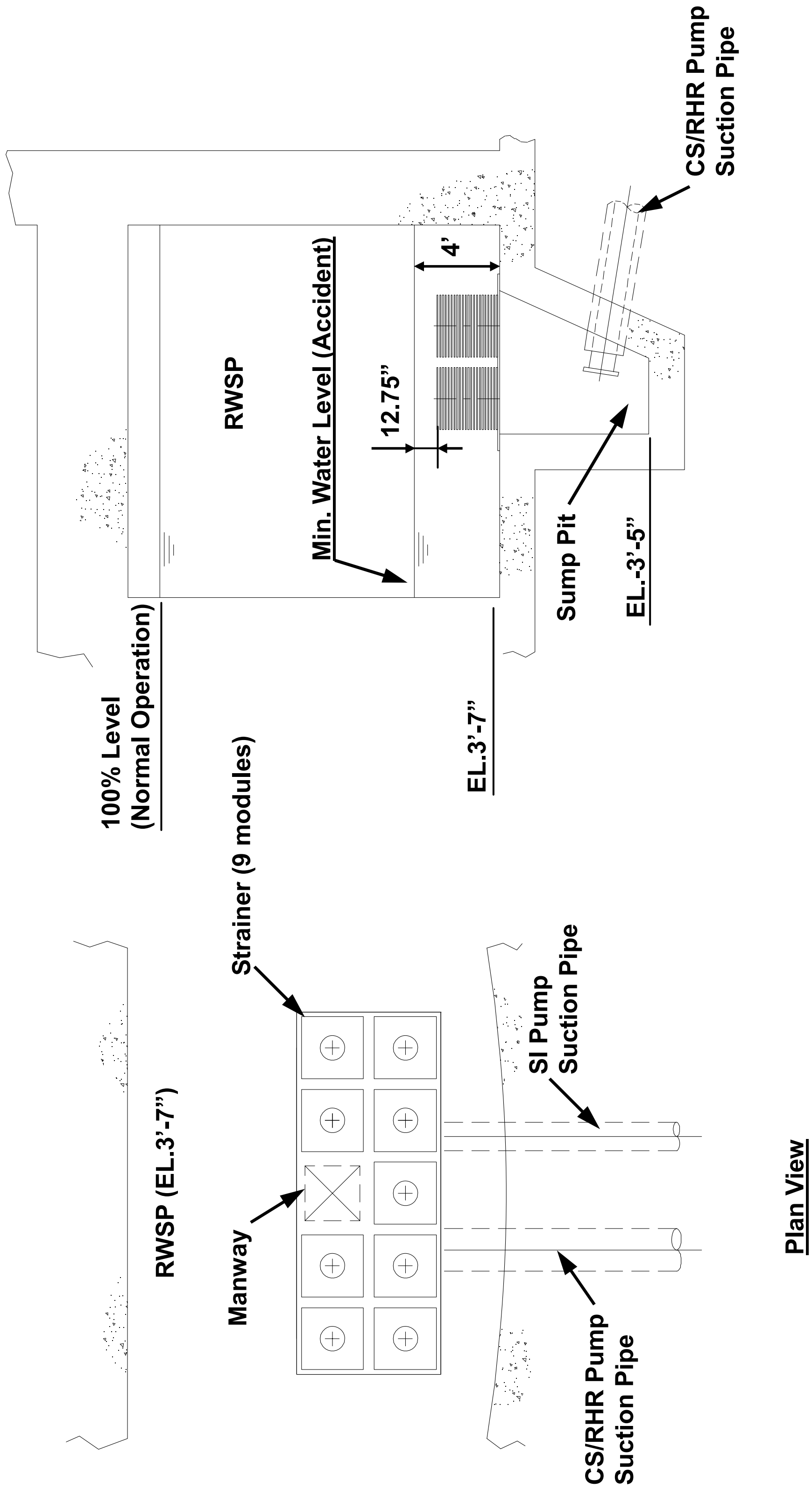


Figure 2-1 ECC/CS Strainer Arrangement

3.0 EVALUATION OF STRAINER PERFORMANCE

3.1 Break Selection

The US-APWR design considers that the pipe breaks in the primary coolant system piping (i.e., RCS piping) have the potential need for reliance on ECCS sump recirculation. A RCS piping large break loss of coolant accident (LBLOCA) and certain RCS piping small break LOCAs (SBLOCAs) would also require ECC/CS sump recirculation. In addition, the secondary side system, i.e. Main Steam and Feed Water (MS/FW), pipe breaks require sump operation.

The break size considered for line breaks (primary and secondary) are double ended guillotine breaks (DEGB). The basis for this break size selection is to provide the largest volume of debris from insulation and other materials that may be within the region affected by the postulated break.

For the break selection, the following break location criteria, which are recommended in the SE (Reference [3]) and comply with RG 1.82 (Reference [2]), are considered:

<u>Break Criteria</u>

1. Pipe break in the RCS or MS/FW with the largest potential for debris
2. Large breaks with two or more different types of debris
3. Breaks with the most direct path to the sump
4. Large breaks with the largest potential particulate debris to insulation ratio by weight
5. Breaks that generate a “thin-bed” - high particulate with 1/8" fiber bed

According to Section 3.3.4.1 of the SE, the breaks are considered for selection only for 2" in diameter and larger HELB pipes. Based on the criteria, the pipe lines considered to be break for sump strainer performance of the US-APWR are listed in Table 3-1.

Section 3.3.5 of the SE describes an approach to the break selection process which includes beginning the evaluation at an initial location along a pipe and stepping along in equal increments (5 foot increments) considering breaks at each sequential location. However, it is not necessary to consider 5-ft increments for the US-APWR evaluation, because of the following conservative considerations:

As discussed in Section 2.2 of this report, the RMI is used for the equipment and pipe lines located in the ZOI to the greatest extent practicable. The design fibrous insulation is excluded from ZOI and constant amount of fiber is allowed for future use regardless of the break location.. Particulate insulations are not used inside the containment. Therefore, only the RMI debris is considered as the potential insulation debris depending on the break location for the US-APWR.

As shown in Table 3-2, the diameter of the ZOI for RMI is defined as 2 inside diameters of the broken pipe. Therefore, most of RMI debris is generated from broken pipe itself. In order to maximum RMI debris generation, the MCP which has a largest inner diameter is selected as the location of the pipe break, which generates largest amount of RMI debris inside containment.

Figure 3-1 and Figure 3-2 show a spherical region within a distance equal to 2 inside diameters of the MCP when the cross over leg (CO/L) nozzle of the SG is broken. As shown, only a small portion of the RMI installed on CO/L and SG is included in the ZOI, so that the generation of RMI debris is relatively limited. Even if the break selection process which includes stepping along in equal increments (5 foot increments) and considering breaks at each sequential location is utilized, the amount RMI debris generated will never exceed all amount of RMI installed on a CO/L. In other words, if all amount of RMI installed on a CO/L is considered as debris, it is not necessary to consider 5-ft increments for break selection. As a result, the standard US-APWR design considers conservatively that all the RMI installed on a CO/L is considered as debris in the evaluation.

For the fibrous insulation (i.e., NUKON), 17 inside diameter of the broken pipe (Table 3-2) is defined as the ZOI in the SE of NEI 04-07 (Reference [3]). Since fibrous insulation is designed to be excluded from the ZOI, design fibrous insulation debris is not calculated using ZOI. Figure 3-3 and Figure 3-4, for reference, show a spherical region within a distance equal to 17 inside diameters of the MCP when the CO/L nozzle of SG is broken.

The maximum amounts of RMI debris is estimated regardless the location of pipe break. This conservative design assumption envelops the break criteria No.1 and No.2.

For Break Criterion 3, it is not necessary for the US-APWR to identify the most direct path to the

RWSP, because of the conservative assumption regarding debris transport ratio, as discussed in Section 3.4 of this report.

Since particulate insulation is excluded from the ZOI of HELB, any particulate debris will be generated from coatings and latent debris. As discussed in Section 3.2 of this report, the coatings debris and latent debris of the US-APWR are conservatively considered constant volumes, regardless of the break location. Therefore, the US-APWR does not require identifying the specific break location which generates maximum volume of particulate debris.

For Break Criterion 5, the US-APWR design limits the use of fiber debris source in containment and the design basis fiber debris forms the debris bed on the strainer disks less than 1/8th inches (0.125 inches) which is the thin bed theoretical thickness. As summarized in Table 3-4, the design basis fiber debris in containment is 34.5 lbs. The maximum fiber debris allocated to one sump is 30.0 lbs (i.e., 12.5 ft^3), which is equivalent to 0.0545 inches of debris bed. Therefore, the US-APWR is deemed to be a thin bed plant which meets the Criterion 5. To evaluate the head loss due to the thin bed effect, prototypical strainer head loss tests were implemented. Further details regarding strainer head loss tests are discussed in Appendix-B.

The following quantitative results demonstrate that a break in the main coolant pipe is the limiting break location (worst case) in terms of debris generation with a comparison between four break locations other than MCP break.

The pipe breaks and locations other than MCP breaks were selected to compare the impact on debris generation as follows:

(1) Pressurizer surge pipe (16in) break at MCP nozzle

As a comparative break location with MCP break inside secondary shield walls, Pressurizer surge line pipe break was selected for the evaluation. The results confirmed that the break location at MCP nozzle generates the maximum amount of fibrous debris in the line.

(2) Main steam pipe (32in) break near the containment penetration

As a representative break location outside secondary shield walls, main steam pipe break was selected and examined the break location generating maximum fibrous debris. The selected break point was located above the floor elevation EL 76'-5" outside the secondary shield walls.

(3) Feedwater pipe (16in) break near the containment penetration

Feedwater pipe breaks outside the secondary shield walls were examined. The selected break point was located above the floor elevation EL 50'-2" outside the secondary shield walls. This is different floor area from the above (2).

(4) Feedwater pipe (16in) break at steam generator nozzle

Secondary line break inside secondary shield walls was also examined. The break point was located at steam generator nozzle that locates upper area inside secondary shield walls.

The ZOI for each insulation type used for debris generation calculations due to MCP breaks is shown in Table 3-2. The energy of secondary lines is lower than that of the MCP (reactor coolant system). Smaller ZOIs due to secondary line break could be applied. However, in the evaluation, the same ZOIs used for MCP were conservatively used for debris generation calculations for secondary line breaks.

The results of the comparable calculation of the four limiting break locations versus MCP break are provided in Table 3-3. It demonstrated that MCP break is the worst case of debris generation for postulated accidents of the US-APWR.

3.2 Debris Generation

The sources of debris at the US-APWR are the insulation debris, coatings debris, latent debris, and miscellaneous debris (i.e. tags, signs, stickers, etc.). For the insulation debris, the US-APWR evaluation concluded that the RMI and fibrous insulation were the potential debris sources following a HELB. Chemical debris generated during post-LOCA long term core cooling and containment spray operation is also defined as a potential debris source of the US-APWR.

The US-APWR design defines a ZOI for the evaluation of insulation and coating debris generation. The damage pressures and corresponding volume-equivalent spherical ZOI of each insulation type are extracted from the guidance of the SE, and are provided in Table 3-2.

In estimating the US-APWR insulation debris generation, a more conservative evaluation rather than ZOI methodology was applied. As discussed in Section 3.1, all of RMI insulation installed on a CO/L was assumed to become debris. In addition, all fibrous insulation inside the ZOI would become fibrous debris, excluding that outside robust barriers. Figure 3-3 and Figure 3-4 show the robust barriers, such as primary and secondary shield walls, which protect components behind them from jet impingement. The estimated amount debris was conservative and enveloped the amount of debris generated at any pipe break locations.

As discussed in Section 2.2, design fibrous insulation is excluded from ZOI. The strainer design allows for the use of a limited amount of fibrous insulation (4.5 lbs which is equal to 1.875 ft^3 of NUKON fiber insulation) inside the ZOI as an operational margin for future plant modification.

As for the coating debris of the US-APWR, the ZOI for qualified coatings is a sphere with a radius four times the MCP inner diameter, which generates largest amount of coating debris. A ZOI of 4D was utilized based on the NRC letter in 2010. (Reference [7]) In the evaluation, the volume of coating debris was calculated by multiplying the surface area of the ZOI sphere by the thickness of the coating film. The thickness of the coating film was defined based on the past experience, and was conservatively assumed to be 650 (μm). As a result, the maximum volume of coating debris was established as 0.082 (m^3), and was rounded to 3.0 (ft^3).

Latent debris is defined as unintended dirt, dust, paint chips, and fibers, which principally consist of fiber and particle debris. The evaluation used a conservative assumption of 200 (lbm) as the upper bound amount of latent debris. The particulate and fiber mix of the latent debris was assumed to be 15% fiber as per the guidance of the SE.

The standard US-APWR does not define specific type of materials for miscellaneous debris, such as tapes, tags or stickers, because these are controlled by foreign material control program established by plant owner. To deal with this uncertainty, a 200 ft^2 penalty of sacrificial strainer surface area per sump is applied as a margin for future detail design and installation of the US-APWR.

Chemical debris of the US-APWR was determined based on the US-APWR chemical effects test as discussed in Section 3.8. Quantification of chemical debris is detailed in Appendix-C. In summary, 300 lbm of aluminum hydroxide and 330 lbm of sodium aluminum silicate were defined as chemical debris used for strainer design.

The amount of design basis debris of the US-APWR is provided in Table 3-4.

3.3 Debris Characteristics

The debris characteristics used in the US-APWR evaluation are presented in Table 3-5.

The US-APWR assumes that all fibrous insulation debris within the ZOI is fine. The term refers to “fines” as discussed in the SE of NEI 04-07. There is no further basis that explicitly defines any fibrous debris size characterization and associated debris size characterization distribution in the guidance.

The SE of NEI 04-07 classified fibrous debris into 4 groups as follows:

1. fines that remain suspended
2. small piece debris that are transported along the floor
3. large piece debris with the insulation exposed to potential erosion
4. large debris with the insulation undamaged and/or still protected by a covering and thereby preventing erosion

Therefore, fines of fiber (insulation) debris are considered as suspended and transportable to the strainer. The Post-LOCA 30-day erosion of fiber insulation debris in the containment is no longer required to be considered because all of the fiber debris was assumed to be fines.

As for RMI debris, the US-APWR applies the NEI GR that 75 percent for small fines and 25 percent for large pieces as the size distribution of RMI debris. These values are based on the size distribution of less than 4 inches as listed in Figure 3-7 of NUREG/CR-6808. The RMI debris is considered as “non-suspended” in the sump pool due to its specific gravity. Effect of erosion during Post-LOCA 30 day operation is not required to consider for RMI debris characterization.

As for coating debris, the SE of NEI 04-07 considers that all coating within the coatings ZOI fails when subjected to DBA conditions. Absent applicable experimental data, a coating debris size value of 100 percent small fines is adopted by the SE of NEI 04-07 for all types of coating material in the ZOI. Therefore, the US-APWR considers that all of coating debris is small fines discussed in the SE of NEI 04-07, and will be suspended in the sump pool in terms of transport. Effect of erosion is not considered for coating debris because it is sufficiently defined in fines.

The latent debris characteristics were based on the SE of NEI 04-07. The latent fiber is comparable to fiberglass (NUKON) insulation, and considered as “fine”. Latent particles are the latent dust and dirt debris mixture, and the size distribution of them is based on the guidance found in NUREG CR6877. Effect of erosion is not required to consider for latent debris.

3.4 Debris Transport

Debris transport is the estimation of the fraction of debris that is transported from debris sources (break location) to the sump strainer. The SE of NEI 04-07 provides the generic transport logic tree to evaluate the fraction of debris for the typical conventional PWR plants.

The SE of NEI 04-07 provides the fraction of debris that is transported to the sump pool. Since the US-APWR has a similar layout feature to the conventional 4 loop PWR plants, the reduction of transported debris can be considered. The US-APWR has a number of gratings, curbs and hold up volumes that can possibly trap the debris during Post-LOCA operation. However, for conservative assumption, the US-APWR assumes that all debris will be transported to the operable sumps. The amount of debris trapped in ineffective pools in containment is not credited in debris transport either.

The US-APWR has four ECCS/CS trains with an independent strainer for each train. The design requires a minimum two trains in operation assuming one train is out of service due to on-line maintenance and another one has a single failure. Therefore, transported debris in the RWSP is assumed to be distributed to two, three or four sumps.

The number of operable sumps during LOCA is a key parameter to determine the debris distribution to each sump. The logic establishes the conditions for subsequent evaluations. For the strainer head loss evaluation, the number of available sumps should maximize the head loss, i.e., assume only two operating sumps. For the bypass debris fraction, the number of available sumps should maximize the amount of bypass debris, i.e., assume four operating sumps. Based on this logic, debris distribution per sump is discussed below.

3.4.1 Debris Split for Head Loss

The US-APWR conservatively assumes that all break-generated insulation and coating debris within the ZOI is transported directly to the closest sump without splitting to another sump, in order to maximize the strainer head loss. Latent debris and chemical debris uniformly distributed in containment apply reasonable assumptions for debris split are applied: 85% of debris is allocated for one sump and 15% for another. The following is the basis of debris splits for latent and chemical debris.

The layout features of the US-APWR associated with debris transportation from debris generation to the RWSP refers to the general arrangements of containment provided in the DCD Chapter 1 Figure 1.2-14 to Figure 1.2-25. Detail drawings around the RWSP are provided in Appendix-D.

The US-APWR is a four main coolant loop plant, and a SG and RCP is located in four independent SG compartments. As demonstrated in Section 3.1, worst case debris generation occurs by a main coolant pipe break. Debris transport senario herein is discussed considering main coolant pipe break that is worst case from view point of debris generation and direct transport toward the RWSP.

Each SG compartment is enclosed horizontally by primary and secondary shield walls. The bottom floor of the compartment is at elevation 25'-3", and supports the SG and RCP legs. Each compartment is isolated by a concrete wall, and access opening at floor level is provided between the loop compartments to allow access. A labyrinth access from outside the secondary shield wall is provided for each SG compartment, and slope is provided in the labyrinth. The top of slope in the labyrinth is at elevation 25'-5", and two inches higher than nominal floor level. In each SG compartment, seven layers of intermediate grating floors are provided for various maintenance purposes during plant shutdown.

When the debris is generated by a pipe break in the compartment, a certain amount of debris might be trapped at layered grating floors. The US-APWR does not take credit for captured debris, and conservatively assumes that all debris falls to the floor. The generated debris flows through openings in the SG compartment floors to the buffer areas. All debris is assumed to transport to the buffer areas. The buffer areas include the reactor cavity, header compartment, and C/V drain pump room. The debris is transported from the buffer areas to the RWSP through 12-inch overflow pipes from both the header compartment (two sets of four overflow pipes) and the reactor cavity (one set of four overflow pipes). The reactor cavity and header compartment are connected to equalize water levels between the two compartments. Containment spray drainage also flows from the refueling cavity through piping to the header compartment. The RWSP overflow piping to the C/V drain pump room installed above the 100 % RWSP water level is not a containment drainage path during a LOCA. Two check valves installed in series in this overflow line prevent water from returning to the RWSP from the C/V drain pump room after a LOCA. The C/V drain pump room is threfore an ineffective volume for containment drainage. Although the buffer areas will capture a portion of the debris, the US-APWR disregards this captured debris in the transport calculation.

Despite the debris transport scenario, the US-APWR assumes that all of break-generated insulation and coating debris will be transported directly to a single sump for conservatism for the NPSH evaluation. No credit is taken for debris settlement on the floor or entrapment in ineffective pool.

Latent debris is defined as fiber and particulates that accumulate on various surfaces throughout the whole area of the containment. The debris is assumed to be uniformly distributed. During an accident, containment spray water and the subsequent water streams on the floor will wash down the latent debris to the containment floor drains. While a significant amount of the latent debris will be led into ineffective pools, it is conservatively assumed that all latent debris will be transported to the RWSP, taking no credit for debris settlement on the floor or entrapment in ineffective pools.

Figure 3-5 shows schematics of debris distribution of RWSP drain pipes, and possible debris allocation patterns for operable sump strainers. It was assumed that the debris allocation for two operable sumps would follow three patterns:

1) 85% debris on one operable sump, and 15% on the remaining operable sump
2) 67% debris on one operable sump, and 33% on the remaining operable sump
3) 50% debris on one operable sump, and 50% on the remaining operable sump

The 85% / 15% debris allocation pattern is used for latent debris as a design assumption and 85% debris allocation for one sump is considered to determine the debris head loss across the strainer during an accident.

RMI debris will not transport to or collect on the strainer surfaces of the US-APWR because of the strainer's lower postulated flow velocity (i.e., 0.0045 fps). Table 4-2 in NEI 04-07 guidance (Reference [3]) provides flow characteristics of the RMI debris, and indicates the terminal settling velocity (i.e., 0.37 fps) and lift over curb velocity (i.e., 0.87 fps) of the RMI fine debris. These flow velocities demonstrate that the RMI debris will settle on the floor and will not rise up to the strainer surface even if transported around the sump.

Because chemical debris which is generated during long term recirculation is considered to be uniformly distributed debris in the containment, the 85% / 15% debris split is applied for strainer head loss evaluation.

3.4.2 Debris Split for Bypass Debris

The bypass debris is defined as the debris which passes through the perforated plate holes of the strainer disks and challenges the downstream component integrity and core coolability (see Section 4.0). The US-APWR assumes that all type of debris accumulated on the strainer will bypass the strainer.

Figure 3-6 shows a schematic of the fiber debris distribution of RWSP drain pipes. The possible debris allocation patterns increase bypass fiber by lowering debris loading per sump. All four sump trains are included in the bypass debris evaluation because this maximizes the bypass fiber. As shown, 14.5 % of plant fiber debris per sump is a possible debris allocation to one sump that would increase the "fraction" of bypass debris. The maximum absolute amount of bypass fiber is determined by multiplying; (1) the absolute amount of debris loading per sump and, (2) the fraction of bypass fiber per sump and, (3) the number of operable sumps. Different amount of debris loading could cause different bypass fraction. The multiple combinations of these parameters can be occurred at the postulated accident, which could result in the maximum amount of bypass debris. It will be difficult to identify limited conditions how much debris fraction corresponding to the debris loading amount could maximize the bypass debris. Therefore, a correlation equation for the calculation of the maximum amount of bypass fiber was developed based on the US-APWR fiber only bypass tests. The tests were implemented to obtain empirical data regarding the bypass fractions corresponding to different debris loadings per sump. The following debris loadings to the strainer were examined to determine the bypass fractions:

Bypass fiber fraction at:

- 12.5% of all fiber debris loading to a strainer
- 25% of all fiber debris loading to a strainer
- 50% of all fiber debris loading to a strainer
- 100% all fiber is loading to a strainer

The summary of the fiber only bypass tests is prepared in Appendix-B. The development of the US-APWR bypass debris equation and design basis bypass fiber used for downstream evaluation are discussed in Appendix-J of MUAP-08013. (Reference [17]).

3.5 Debris Head Loss and Vortexing

The Design Basis Strainer Head Loss (DBSHL) is established to evaluate net positive suction head (NPSH) of the Safety Injection (SI) pumps and the Containment Spray / Residual Heat Removal (CS/RHR) pumps discussed in Section 3.6. The DBSHL has been utilized for air ingestion analyses which potentially adversely affect strainer performance and pump performance.

The DBSHL is a sum of the Debris Laden Strainer Head Loss (DSHL) and the Clean Strainer Head Loss (CSHL). The DSHL is defined with sufficient margin to the Debris Laden Test Strainer Head Loss (DTSHL) measured by the US-APWR sump strainer head loss tests discussed in Appendix-B. The CSHL is the prototype clean strainer head loss calculated separately by a vendor. The strainer head losses are summarized in Table 3-6.

As discussed in Appendix-B, the US-APWR strainer head loss tests are implemented at a targeted water temperature 120ºF. The tests demonstrate that the measured strainer head losses are proportional to the change in flow, indicating that there were no bore holes on the strainer debris bed at the termination of the test. This was also observed visually for each strainer disk after the test tank was drained. Additionally, no vortexing was observed during the tests, and the debris bed was stable during the tests. Therefore, temperature scaling of the head loss test data to the postulated plant sump pool temperatures from the test temperature is applicable, in accordance with the NRC March 2008 guidance. (Reference B-2)

Vortexing, sump fluid flashing, and deaeration are the effects which could adversely affect strainer performance. If excessive air ingestion or vaporization due to these effects occurs at the strainer, an unacceptable increase in strainer head loss will be caused by the increased resistance associated with two-phase flow. Since the US-APWR strainer design precludes unacceptable air ingestion from occurring, the DBSHL utilized for safety evaluation is appropriately supported by the US-APWR strainer head loss tests. The tests were implemented and recorded head losses utilizing single-phase water flow at 120ºF.

For vortexing, the strainer design exceeds the level of vortex prevention provided by minimum submergence alone, due to a combination of low approach velocity, small hole size of the perforated plate, and overall stacked-disc geometry. No vortex formation was observed as a result of testing. (Appendix-B)

For sump fluid flashing, the strainer is designed with sufficient submergence to preclude the occurrence of two-phase flow at the debris bed which can result in an unacceptable increase in strainer head losses. Air ingestion due to sump fluid flashing is not expected to occur, and therefore will not adversely affect pump performance.

The evaluation method to address sump fluid flashing is based on assessing the static head of water associated with the post-LOCA containment minimum water level. The static water head based on the height of the post-LOCA minimum containment water level to various US-APWR sump strainer components is evaluated as follows:

- Strainer top disk (upper surface),
- Strainer discharge (core tube outlet),
- ECC/CS outlet pipe centerline in the sump pit, and
- Centerline of the ECC/CS pump suction inlet

If the static water head height exceeds the calculated head loss across each of the evaluated strainer components/locations, then the static water head will have prevented flashing and/or vaporization.

Table 3-8 provides a summary of the US-APWR sump fluid flashing calculation cases and associated parameters. The strainer top disk and strainer disk discharge were evaluated as representative cases. As discussed in Section 3.7, the minimum post-LOCA RWSP water level at an elevation of 7’-7” (i.e., 4’-0” above the US-APWR RWSP floor elevation of 3’-7”). The minimum RWSP post-LOCA water elevation of 7’-7” results in 1’-0.75” of strainer submergence during long-term recirculation. Sump fluid flashing and/or vaporization are precluded if the static head of water for the component/elevation is greater than the associated head loss. In the evaluation, initial dry air partial pressure (minimum present at the start of the accident, as discussed in Section 3.6.2.2) is considered to contribute to prevent the flashing from occurring when sump water is subcooled.

Table 3-9 summarizes that there is sufficient strainer submergence to preclude flashing at the US-APWR strainer debris bed. Note that the case 1 result is based on the conservative assumptions that the debris head loss of entire strainer disks and all clean strainer components head loss are generated on the top surface of the strainer disk. In addition, initial dry air partial pressure will significantly contribute to prevent flashing once the sump water is subcooled.

For deaeration, air solubility at the strainer and pump elevations were evaluated. Section 3.6.2.3 provides calculations of void fraction due to the design basis strainer head loss at various post-LOCA temperatures. As calculated, the void fraction at sump strainer is approximately 0.43% (volumetric basis). A significant level of void fraction is not expected at the strainer. Therefore, the air ingestion due to deaeration is not expected to adversely affect strainer performance. In conclusion, the US-APWR strainer submergence is adequate to limit vortexing, sump fluid flashing, and deaeration induced by design basis strainer head loss.

3.6 Net Positive Suction Head

3.6.1 System Operation

The US-APWR engineered safety features (ESF) include safety injection (SI) pumps and containment spray/residual heat removal (CS/RHR) pumps. These pumps are normally aligned to the refueling water storage pit (RWSP) inside the containment. Figure 3-9 shows a schematic flow diagram of the ECC and CSS. The SI pumps are automatically initiated by the safety injection signal and the CS/RHR pumps are automatically initiated by the containment spray signal. These pumps take suction directly from the RWSP. Four ECC/CS strainers are installed in the RWSP and each ECC/CS strainer is for one of four trains.

3.6.2 NPSH Available Calculation

The containment pressure is assumed to be equal to the initial containment pressure prior to the start of the accident (for sump fluid temperatures below the saturation temperature corresponding to this containment pressure). This methodology fulfills the requirements of RG 1.1 & RG 1.82 that the NPSH available be evaluated without crediting any increase in pressure resulting from accident conditions (See Appendix E), at low temperatures. This approach ensures that sufficient containment pressure is available under all accident conditions and that defense-in-depth is maintained by preserving the independence of systems designed to prevent accidents and those designed to mitigate the effects of accidents (See Appendix F). It is assumed that the containment pressure remains constant at the pre-accident value consistent with RG 1.82 and RG 1.1, for sump fluid temperatures lower than the corresponding initial saturation vapor pressure. For temperatures higher than this initial saturation pressure, the containment pressure is assumed to be equal to the sump fluid vapor pressure.

3.6.2.1 Assumptions

For the NPSH available calculation, the most limiting conditions were assumed to be applicable to all events.

a) Single Failure

The US-APWR SI pumps and CS/RHR pumps consist of four trains. A single ECC/CS strainer is installed in each train (e.g., one ECC/CS strainer supplies one SI pump and one CS/RHR pump). Therefore, a single failure in any single train does not affect flow rate through any other strainer or ECC/CS train.

The containment spray system has a common spray ring header. Therefore, if the number of operating pumps is smaller, the flow in any one pump is greater. The minimum number of operating CS/RHR pumps is two (one pump is assumed out of service, one pump is assumed to experience a single failure, and the remaining two are operating).

In the NPSH available calculation, the maximum (assumed runout) pump flow rates are conservatively used thus minimizing NPSH available. The calculated NPSH available is therefore greater than would be expected in the case of a single failure.

b) Containment Pressure

For the minimum NPSH available calculation, no additional containment pressure is credited above the initial containment pressure for low sump fluid temperatures (i.e., below approximately 212º F). For higher sump fluid temperatures, the containment pressure is assumed to equal the saturation pressure corresponding to the sump water temperature. The treatment of the containment pressure is discussed in Section 3.6.2.2.

c) Water Level

RWSP water level for NPSH available calculation is the minimum RWSP water level. The details of the calculation of minimum water level are given in Section 3.7. The water level used in NPSH available calculation includes a 5% uncertainty.

d) Head loss

Head loss calculations for NPSH available are prepared based on hydraulic models of the systems aligned to take suction from the RWSP. The system configurations of SI pump suction and CS/RHR pump suction are not changed during an accident. Therefore this system configuration results in the highest sump flow rate, which is used for sizing the ECC/CS strainers. The flow rate for the NPSH available calculation is conservatively based on the maximum pump flow rate. These calculations use Equations 3-5, 3-14 and 3-15 of Crane Technical Paper No. 410, “Flow of Fluids Through Valves, Fittings, and Pipe” (Reference [9]) to determine the head loss due to frictional resistance in the piping and line losses due to other components. The water temperature for head loss calculation (Pipe, fitting, and so on) is conservatively set as 32 deg F to maximize fluid density and the resulting head loss. The head loss used in NPSH calculation includes a 5% uncertainty.

e) Debris head loss

Debris head loss is evaluated by qualified test results conducted specific to the US-APWR plant conditions. The tests plan and results are provided in Appendix-B.

This section describes the treatment of strainer and line head losses. Section (1) describes the methodology for adjusting head loss values measured during testing for temperature effects. Section (2) discusses chemical debris precipitation from solution, which may occur at lower temperatures during long-term recirculation cooling (< 150°F). Therefore, the strainer head loss is divided into two separate regimes:

- Non-Chemical and Chemical Debris at lower temperatures (< 150°F)
- Non-Chemical debris only at higher temperatures (> 150°F)

(1) Head Loss vs. Temperature

The sump strainer testing measured head loss values due to non-chemical and chemical (precipitant) debris, at a single test fluid temperature (120°F). The actual strainer head loss will vary with viscosity (i.e., temperature) of the sump water. The calculated line losses represent a conservative bounding value and do not require temperature adjustment.

Therefore, the total head loss values are given as follows:

$$H_F(T) = H_L + H_{S_{120}} \frac{\mu(T)}{\mu_{120}}$$

Where:

H_F :	Total head loss
H_L :	Line loss
H_S :	Strainer Loss
μ :	Absolute (dynamic) viscosity
Subscript 120:	Property evaluated / measured at 120°F

(2) Precipitation of Chemical Debris

f) NPSH required

Generally, the required net positive suction head (NPSH required) is identified by the pump vendor and will be confirmed through qualification testing of the as-built pump. In the US-APWR design, NPSH required for CS/RHR pump and SI pump are specified to include margin above the nominal NPSHr identified by the vendor in the preliminary design phase. The Design-basis NPSH required for the CS/RHR pump is specified as 19.7 ft, although the nominal value provided by the pump vendor is 16.4 ft at maximum flow rate. Also, the design-basis NPSH required for SI pump is specified as 18.8 ft, although from the nominal value provided by the pump vendor is 15.7 ft. In the NPSH evaluation, the design-basis NPSH required values are used.

3.6.2.2 Calculation Results

NPSH available is the total suction head in feet absolute, determined at the pump suction and corrected to datum, less the vapor pressure of the liquid in feet absolute. This leads to an analysis of energy conditions on the suction side of a pump to determine if the liquid will vaporize at the lowest pressure point in the pump. The typical equation governing the calculation of available NPSH is given as:

$$NPSH_A = H_P + H_{EI} - H_{VP} - H_F$$

Where:

H_P : Absolute containment pressure head at the pump suction (sump surface)

H_{EI} : Elevation head

H_{VP} : Vapor pressure at prevailing water temperature converted to head

H_F : Form and frictional head losses including through the sump screen, entrance losses and piping losses

Crediting the initial dry air partial pressure involves setting the pressure head term (H_P) equal to the minimum partial pressure of air in containment at the start of the event plus the vapor pressure at the maximum initial sump temperature. This initial vapor pressure in the containment is conservatively ignored for additional margin. This method limits the contribution of air pressure to the minimum present at the start of the event and ignores any increase in air or vapor pressure resulting from heatup during the event.

Set $H_P = H_{a\,min} + H_{VP}'$

$$NPSH_A = H_P + H_{EI} - H_{VP} - H_F$$

$$= H_{a\,min} + H_{VP}' + H_{EI} - H_{VP} - H_F$$

$$= H_{a\,min} + H_{EI} - H_{VP} - H_F$$

Where:

H_{VP}' : Initial vapor pressure in the containment converted to head (conservatively neglected)

This equation is limited to use below a saturation temperature corresponding to the minimum containment air pressure ($H_{a\,\min}$). When the temperature of sump water is higher than the saturation temperature corresponding to this pressure, the conventional method of calculating NPSH available is applied, where the containment pressure is assumed to be equal to the sump fluid vapor pressure. Therefore, the following method for calculating NPSH available is applied.

When the temperature of sump water is higher than the saturation temperature corresponding to the minimum containment pressure;

$NPSH_A = H_{EI} - H_F$ (Conventional method) … Eq. (1)

When the temperature of sump water is below the saturation temperature corresponding to the minimum containment pressure;

$$NPSH_A = H_{a\,min} + H_{EI} - H_{VP} - H_F \qquad \text{… Eq. (2)}$$

Based on an assumed initial containment temperature of 120°F (maximum allowed by TS 3.6.5, which corresponds to a vapor pressure of 1.7 psia) and a containment pressure of -0.3 psig (minimum allowed by TS 3.6.4), the minimum dry air partial pressure prior to the event is 12.7 psia.

Conservatively assuming a sump water temperature of 32°F for converting pressure to head, the sump water density would be 62.4 lbm/ft^3. Combined with the minimum partial pressure of air, the minimum hydrostatic head is 29 feet ($H_{a\,\min}$ = 29 ft). The saturation temperature corresponding to this minimum hydrostatic head is 204°F.

Therefore, the following equation for NPSH available calculation with temperature below 204°F is applied:

$$NPSH_A = 29(ft) + H_{EI} - H_F - H_{VP}$$

For static head, the relationship between RWSP minimum water level and pump center elevation is shown in Figure 3-8. In this calculation, the water level used is 3.8’, which includes a minus 5% margin for uncertainty.

The acceptance criteria for reliable operation of the ECCS pumps is that the available NPSH exceeds the required NPSH (i.e., the NPSH margin is greater than zero) for all expected operating temperatures. Therefore, the acceptance criteria for the sump strainer is that the strainer head loss (including non-chemical and chemical debris) is sufficiently low to ensure positive NPSH margin.

The NPSH margin will be evaluated at various Post-LOCA sump water (i.e. RWSP water) transient temperatures. Figure 3-14 is the Post-LOCA sump water temperature profile excerpted from the DCD Chapter 6 Figure 6.2.1-26. As shown in Figure 3-14, the temperature profile has been calculated assuming conservative accident conditions, in order to maximize the peak temperature (i.e., 256°F). The design peak RWSP temperature for NPSH calculations are defined as the bounding temperature (i.e., 270ºF).

As discussed in above, the NPSH evaluation is divided into two regimes based on the saturation pressure corresponding to the initial air partial pressure (204°F). As discussed in Section 3.6.2.1 (Subsection E.2), the low temperature regime is itself divided into two additional regimes to account for precipitation of chemical debris. These three regimes are outlined in Table 3-7. Figure 3-15 shows the available and required NPSH versus temperature, and Table 3-11 and Table 3-12 prepare the NPSH calculation values for CS/RHR pump and SI pump respectively.

Figure 3-16 shows the resulting containment pressure head used in the NPSH evaluation, total head losses, and RWSP water vapor pressure head versus temperature. Figure 3-17 evaluates the NPSH available versus time, based on the RWSP temperature profile shown in Figure 3-14. Note that the step change in the figures is due to the chemical debris precipitation and resulting transition. The actual transition would be smoother and bounded by the chemical debris loss values used in this NPSH methodology.

As illustrated in Figure 3-15 to Figure 3-17, the NPSH available exceeds the NPSH required for all expected sump temperatures (and therefore, at all times throughout the LOCA transient). The minimum NPSH margin calculated with this methodology is approximately 1.2 ft at 204°F (the saturation temperature corresponding to the minimum initial air partial pressure). Therefore, the RWSP strainer and US-APWR design provides sufficient available NPSH to ensure reliable operation of the ECCS system pumps

3.6.2.3 Effects of Air Ingestion on NPSH

Air ingestion into the sump fluid due to vortexing, sump fluid flashing and deaeration may adversely affect pump performance. Section 3.5 discusses the effects of air ingestion on strainer performance and concludes that vortexing and sump fluid flashing are prevented by the US-APWR strainer design. However, deaeration remains a potential source of air ingestion to the pump due to dissolved air in the sump fluid.

It is recognized that a small amount of deaeration will occur due to the difference in the solubility of air in water resulting from the pressure differential across the strainer and debris bed. A conservative assessment was made of the theoretical void fraction (air ingestion rate) which is expected to be minimal. If the gases are released in a manner such that they are ‘captured’ by the strainer discharge fluid flow, they would be entrained in the fluid and could be transported to the ECC/CS pumps.

The solubility of air in water is decreased when the water temperature higher. In other words, the solubility is a maximum at the lowest water temperature of interest. In addition, the solubility is proportional to absolute pressure. Therefore, the amount of deaeration is proportional to the pressure loss across the strainer.

In this evaluation, the solubility difference between strainer inlet and outlet is considered. Air solubility at the water surface at 70 ºF is 0.7992 mol / m^3 (based on Bunsen Absorption Coefficient and partial air pressure above the sump water). Also, the air solubility at the strainer outlet is 0.6541 mol / m^3 (considering the strainer head loss of 7 ft @ 70 ºF and static pressure 1 ft below from water surface). The difference of air solubility is 0.1451 mol / m^3. This quantity of air is assumed to evolve to gas at the strainer outlet. Therefore, void fraction at the strainer outlet, at 70 ºF, is 0.43 %. In addition, the analysis of further gas evolution downstream of the strainer indicates that it is significantly less than the value at the strainer, because the increase in static elevation head compensates for the additional line losses.

This value is considered conservative for the following reasons. The deaeration is calculated assuming all pressure loss occurs at the top elevation of the strainer, where the increased static head is minimum, whereas the pressure loss will distributed over the height of the strainer. The calculation is based on a water temperature of 70 ºF. The solubility of air in water is significantly less at higher temperatures. In addition, the head loss values across the face of the strainer are temperature corrected, and are also significantly less at higher temperatures. The US-APWR Design Basis LBLOCA water temperature is a maximum of approximately 256 ºF, and a realistic minimum long-term temperature would be approximately 120 ºF. Therefore any actual void fraction that could occur at the strainer debris bed is very minimal.

If any void fraction should occur, it would be further reduced before reaching the pump inlet due to the significant static head of water above the ECC/CS pump suction inlets. A void fraction of 0.43 % at the strainer outlet would be reduced to 0.25 % at the pump inlet flange due to the difference in elevation.

As described above, air ingestion due to deaeration is very minimal, but it is not zero. RG 1.82, Rev. 3, provides a conservative method to account for the effects of air ingestion by increasing the NPSH required as follows:

$$NPSH_{required,modified} = NPSH_{required,vendor} \times \beta$$

where $\beta = 1 + 0.50\alpha_p$ and α_p is the air ingestion rate (in percent by volume) at the pump inlet flange. Therefore, the modified NPSH required for the CS/RHR pump is the following:

16.4 (ft) x (1 + 0.5 x 0.25) = 18.45 (ft)

This modified NPSH required for CS/RHR pump is smaller than Design-basis NPSH required of 19.7 ft. Application to the SI pumps results in the same conclusion. This means the effect of air ingestion is within the margin of NPSH required. Moreover, the RG 1.82 NPSH required modification is considered to be a very conservative method to account for air ingestion. Air ingestion can cause a decrease in pump performance and potential shaft vibration, but is much less challenging to long-term erosion and pump performance than the effects of cavitation resulting from insufficient NPSH available.

Therefore, air ingestion due to deaeration is not considered to affect the NPSH evaluation to demonstrate that the US-APWR ECCS pumps can perform their required functions.

3.7 Upstream Effect

3.7.1 Hold-up Volumes

The evaluation of upstream effect is a review of the flow paths leading to the RWSP, identifying those flow paths which could result in blocking the return water that could challenge the RWSP minimum water level evaluation. The evaluation also includes identifying the hold-up volumes, such as recessed areas and enclosed rooms, for which trapped water will not return to the RWSP. The evaluation of the US-APWR was performed, and all of the hold-up volumes were taken account into the minimum water level calculation.
Figure 3-9 shows a schematic of containment spray/blowdown return pathways of the US-APWR, and is provided to supplement the information in the DCD. In the evaluation, two possible choke points were identified; 1) refueling cavity drains, and 2) overflow pipes of the RWSP.

<u>Refueling cavity drains</u>

In the refueling cavity, there are two 8 inches drain pipes which are connected to the header compartment. The header compartment then provides return flow to the RWSP through the 12-inch overflow pipes. Since all of debris are defined as small/fines as discussed in Section 3.3, there is a relatively low possibility for potential blockage to occur in the drains. The US-APWR credits the following gratings (Figure 3-9 and Figure 3-10) for preventing the potential “large debris” from reaching to cavity drains, and from potential blockage, the drain function will be maintained.

- Grating inside secondary shield wall at EL. 55’-1” (Loop-A,B,C, and D)
- Grating inside secondary shield wall at EL. 73’-1” (Loop-A,B,C, and D)
- Grating at upper core internal laydown pit (in refueling cavity)

The following is the technical basis of for the number and size of refueling cavity drains to maintain the return water to the header compartment:

- Two drain pipes are capable of draining water retained on the refueling cavity floor to the header compartment by gravity at maximum containment spray pumps (i.e., four train pumps) operation.

- The drain pipe (8”) is sufficiently sized to pass “small” debris which potentially falls on the refueling cavity without blockage.

Given that four containment spray pumps are operating, the maximum flow rate per drain pipe and associated pressure drop of the pipe were calculated as follows:

Q	: Overflow rate per drain pipe	: 7,860 ft^3/hr (223 m^3/hr) per one drain pipe [Note]
P	: Pressure drop of transfer pipe	: 3.6 ft

Note: Assuming that one drain pipe was blocked and the remaining pipe is operable.

Since the required water head for the drain pipes is 3.6 ft, the water level in this area will be 3.6 ft higher than the water level at the discharge.

Twelve inch diameter overflow pipes provide return flow from the reactor cavity and header compartment. There are two sets of four header compartment overflow pipes to the RWSP, and one set of four reactor cavity overflow pipes to the RWSP. Each overflow pipe discharge is surrounded by a return flow water baffle. The reactor cavity and header compartment receive containment drainage through floor openings in the SG compartments. Mesh debris interceptors are installed above the floor openings and within the header compartment. The debris interceptors are with an 8-in x 8-in mesh, which is smaller than the overflow pipe diameter to prevent clogging of the overflow piping. The debris interceptors are necessary for ECCS operation and are therefore classified as safety-related and seismic category I components. The debris interceptors are sized such that all small debris (i.e., “fines” per NEI 04-07) will transport to the RWSP without blockage from the debris interceptors. The header compartment also receives return flow from the refueling cavity, which is protected from large debris by grating in the upper core internal laydown area.

Given that four trains of safety pumps are operating, the maximum flow rate per overflow pipe and associated pressure drop of the pipe were calculated as follows:

Q: Overflow rate per overflow pipe	14,230 ft^3/hr (403 m^3/hr) per overflow pipe (note)
P: Pressure drop of overflow pipe	1.3 ft

Note: Assuming that 3 overflow pipes are blocked and the remaining pipes are operable.

Since the water head for the overflow pipe can be up to 13.6 ft at minimum RWSP water level, the hydrostatic pressure is sufficient for gravity flow against the overflow pipe pressure loss.

Besides the overflow pipes and refueling cavity drains, no other drains or narrow pathways are assumed to provide make-up flow to the RWSP. Floor drain piping which collects in the Containment sump, such as the SG compartment floor and operating floor, is assumed to become blocked. Containment Spray water is drained to lower containment levels by way of stairway openings, equipment hatch, or compartment access openings. These openings are not considered to be narrow pathways vulnerable to blockage. Since the floor drains are assumed to be blocked, an amount of Containment Spray water is assumed to collect and remain on various Containment levels. The depth of water remaining on the containment floors is assumed to be 1 inch on the EL 76’-5’’, EL 50’-2’’, and most of the refueling cavity. The depth of water in the upper core internals storage area is 75 inches and 2 inches in other recessed areas of the refueling cavity. The 75 inches was conservatively calculated considering the pressure loss of drain piping and the water level in the discharge area of the drain water. The depth of water on the EL 25’-3’’ floor is 6 inches inside the SG compartments and 7 inches outside the SG compartment. This amount of remaining water is factored into the return water hold-up volume discussed in the following calculation:

The US-APWR hold-up volumes are categorized into two groups, “Return water on the way to the RWSP”, and “Ineffective pools”. The calculated values described in the DCD Table 6.2.1-3 *“RWSP Design Features”*. The followings are the definitions of the groups:

<u>Return water on the ways to the RWSP</u>

In a LOCA, the RWSP water returns from containment spray nozzle and broken pipe. The water on the way to the RWSP will decrease the initial RWSP water level. The following are the source of return water to the RWSP.

a. Containment spray water droplets and saturated steam (includes the empty spray header rings and pipes)
b. Condensate water on all of the containment surfaces (includes equipment, walls and ceiling, etc.)
c. Water stream on the containment floors (includes refueling cavity floor)

Ineffective pools

An ineffective pool is defined as a hold-up volume that entraps return water which will not contribute to recovering the RWSP water level. The following are considered as the US-APWR ineffective pools:

a. Reactor cavity
b. Header compartment
c. C/V drain pump room
d. Upper core internal storage area in the refueling cavity
e. NaTB baskets
f. Additional hold-up volume (condensate water between PCCV and RWSP and minor recessed area)

The calculated hold-up volumes of the US-APWR are provided in Table 3-10.

3.7.2 Minimum Water Level

The minimum water level of the RWSP forms the basis for estimating pump water head in the NPSH evaluation, as discussed in Section 3.6. It was conservatively calculated as follows:

During normal operation, the RWSP contains 84,750 ft^3 (634,000 gallons) of borated water (the water volume from 0 (%) to 100 (%) water level), as shown in Figure 3-11. The RWSP allows water evaporation and when the water surface reaches the 96 (%) water level, the makeup operation is activated and continued until 100 (%) water level is recovered. This level is defined as “below nominal water level” of the RWSP, and is used as the initial water level for the postulated accidents. In case of LBLOCA, the water mass in the accumulator tanks can contribute to recover the RWSP, but this source was conservatively disregarded in the calculation.

The design basis minimum water level of the RWSP was set at 4.0 ft above the RWSP floor. The calculated minimum water level was determined by subtracting the hold-up volume from the initial water volume in the RWSP and confirmed that the design basis was higher than the calculated level. Figure 3-11 illustrates the minimum water level of the RWSP. As shown, the design basis minimum water level was determined with a 1,370 ft^3 of water volume margin to the calculated minimum water level.

The US-APWR does not require the RWSP minimum water level for small break LOCA, because it is bounded by the LBLOCA condition. The US-APWR has a different design to operating PWR plants that it does not need "switch over" to continue long term core cooling after the postulated accident. The US-APWR has an in-containment water resource (i.e., the RWSP) for the postulated accident that is normally filled with borated water. Therefore, the strainer is fully submerged during normal operation. The initial water depth in the RWSP is 16’-7”. In case of a LBLOCA, hold-up volumes discussed in Section 3.7.1 will be lost from the reactor coolant system and released into containment. To compensate for the loss of water, the RWSP water is injected into the reactor coolant system, the RWSP water depth decreases, and reaches to the minimum water level (i.e., 4.0ft). In case of a SBLOCA, less water will be lost from the reactor coolant system, therefore, less water from the RWSP will be consumed. This means that the LBLOCA is the worst case for the US-APWR in terms of minimum water level for the strainer.

3.8 Summary of Chemical Effects Tests

The containment system of the US-APWR is designed to accommodate the energy release following a postulated accident. The containment system also permits the recirculation of reactor coolant, the CSS water and the ECCS water to the decay heat removal (DHR) heat exchangers. Water collected in the sump from the reactor coolant system, the safety injection system, and the containment spray system is recirculated through the reactor core to remove residual heat. The recirculation sump contains the strainers to protect the downstream components that are in the reactor coolant, the CSS, and the ECCS flow paths from the effects of debris that could be transported to the RWSP. During operation of the ECCS, debris in the sump fluid would affect long-term core cooling.

The post-LOCA environment creates several challenges to containment materials and debris sources based on temperature, chemical reactions, and effects from sprayed and pooled water. The combination of spray chemicals, insulation, corroding metals, and submerged materials creates a potential condition for the formation of chemical substances that may impede the flow of water through the sump strainers or affect downstream components in the emergency core cooling or reactor coolant systems.

The Integrated Chemical Effects Test (ICET) Project represented a joint effort by the NRC and the nuclear utility industry to simulate the post-LOCA chemical environment present inside a containment structure and to monitor the chemical system for an extended period of time to identify the presence, composition, and physical characteristics of chemical products that may form. The ICET test series was conducted by Los Alamos National Laboratory (LANL) at the University of New Mexico (UNM). The results of the ICET testing are published in NUREG/CR-6914. (Reference [10])

The US-APWR is a low fiber plant that uses sodium tetraborate as a buffer and based on a review of the results presented for ICET Test#5 is expected to have minimal corrosion and reaction products. However, in order to further understand the plant specific interactions between the containment materials and post-LOCA debris with the sump fluid chemistry, MHI elected to perform an ICET experiment for the US-APWR.

The chemical effects tests were conducted in the program attempted to simulate the chemical environment present inside the US-APWR containment recirculation water after a LOCA. Autoclave tests as parts of the testing were performed to simulate the temperature transient of early LOCA phase, and to further understand the sensitivity related to the sump fluid chemistry. The recirculation test was conducted for 30 days at a constant temperature of 149°F. The chemical environment within the tank included boric acid, lithium hydroxide, and sodium tetraborate was added to the test. The autoclave test water temperature was similarly operated with a time-temperature profile representative of the US-APWR post-LOCA operation for the first 100 hours until such time the temperature falls to 149°F.

The objective of the chemical effects tests of the US-APWR is to obtain experimental data under simulated plant conditions on the corrosion products that may form in a post-LOCA environment.

The tests data was used to determine compositions, characterize properties, and quantify masses of chemical reaction products that may develop in the sump water under a representative post-LOCA environment. The tests were performed in Mitsubishi Heavy Industries, Takasago Research and Development Center.

The details of the test plan and its results are provided in separate technical reports respectively. (Reference [12], Reference [13])

The quantification of chemical debris based on the US-APWR specific chemical effects test results (Reference [13]) is evaluated. Appendix-C provides the quantification of chemical debris used for strainer qualification tests discussed in Appendix-B.

3.9 Evaluation Summary

The US-APWR sump strainer performance was evaluated in accordance with the RG 1.82 Rev.3 requirements. The break selection, debris generation, and debris transport were analyzed to identity the potential debris which may reach the strainers in the RWSP assuming a number of conservative considerations. The characteristics of potential debris were set, identified, and referred appropriately, and used in the debris head loss evaluations, as well as the NPSH evaluation of the vital pumps of the US-APWR.

The US-APWR strainer head loss test is performed to determine maximum head loss across the strainer during a postulated accident. The evaluation concluded that postulated debris head loss would satisfy the specified requirements of the standard US-APWR, and have sufficient suction head to operate the plant safely following a post-LOCA event.

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3-1 Plan View of Zone of Influence (RMI, L/D=2.0)

Security-Related Information – Withheld Under 10 CFR 2.390

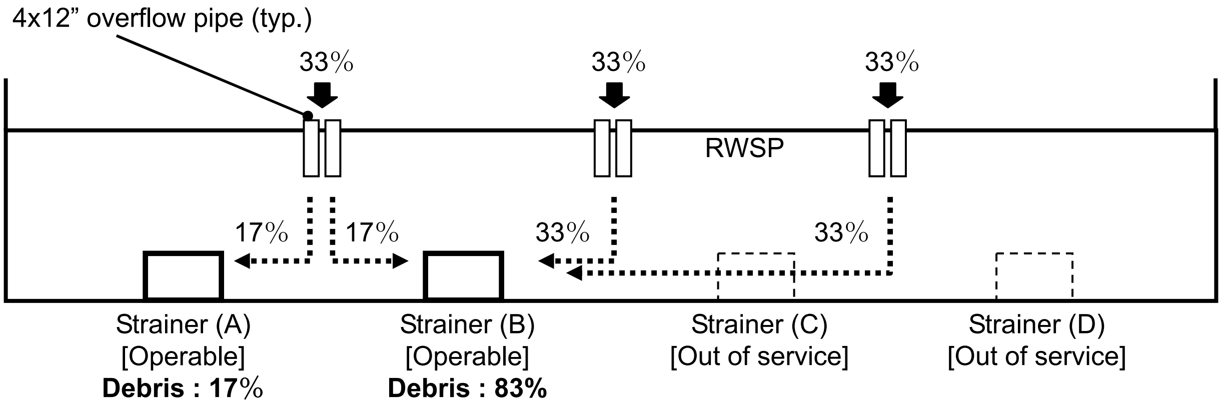
Figure 3-2 Section View of Zone of Influence (RMI, L/D=2.0)

Security-Related Information – Withheld Under 10 CFR 2.390

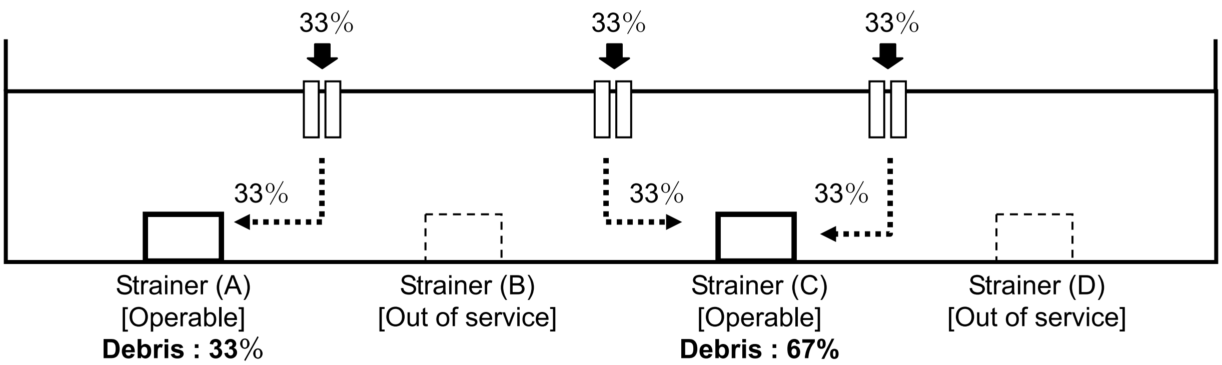
Figure 3-3 Plan View of Zone of Influence (Nukon, L/D=17)

Security-Related Information – Withheld Under 10 CFR 2.390

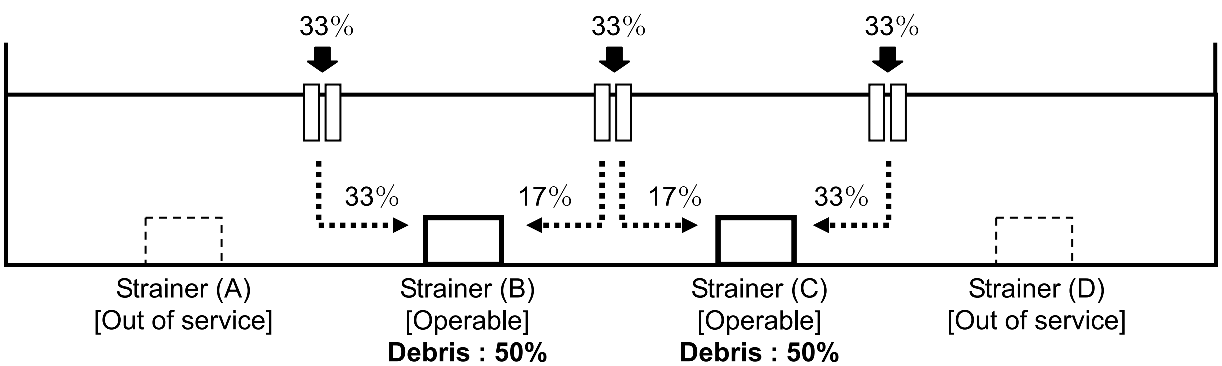
Figure 3-4 Section View of Zone of Influence (Nukon, L/D=17)



Pattern-[1] 83% Debris Allocation on Sump Strainer (Worst case)

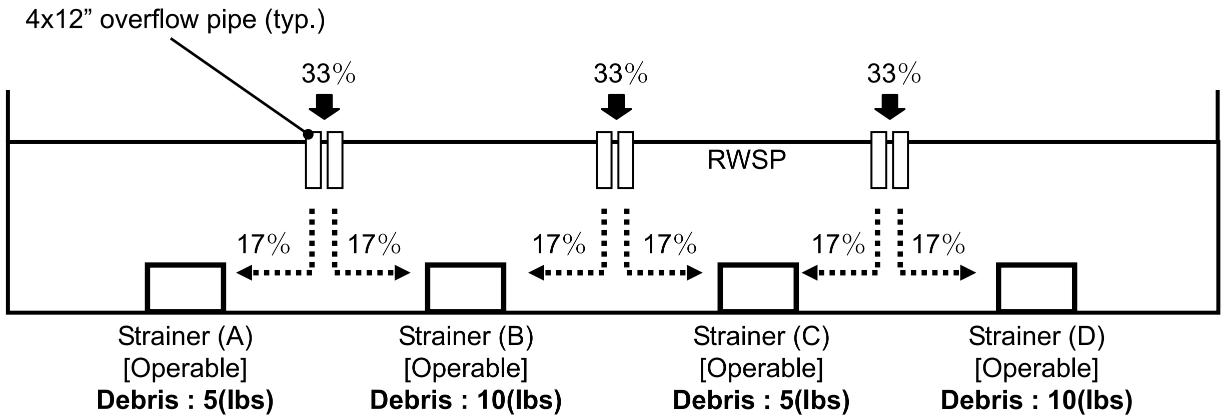


Pattern-[2] 67% Debris Allocation on Sump Strainer

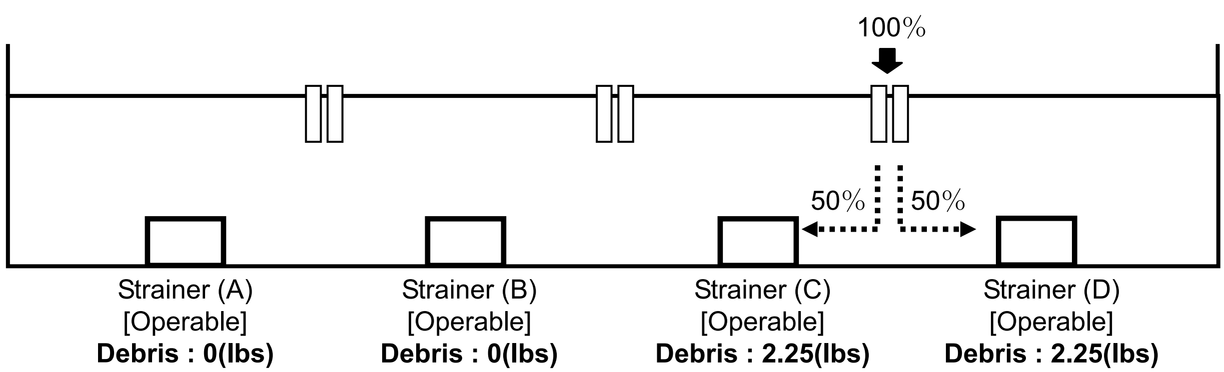


Pattern-[3] 50% Debris Allocation on Sump Strainer

Figure 3-5 Schematics of Debris Allocation on Operable Sumps (for head loss)



Latent Fiber (30 lbs) Allocation



Generated Fiber / Operating Margin Debris (4.5lbs) Allocation

Debris Allocation	Total	Strainer (A)	Strainer (B)	Strainer (C)	Strainer (D)
Amount	34.5 (lbs)	5 (lbs)	10 (lbs)	12.25(lbs)	7.25 (lbs)
Fraction	100 (%)	14.5 (%)	29.0 (%)	35.5 (%)	21.0(%)

Figure 3-6 Schematics of Debris Allocation on Operable Sumps (for bypass debris)

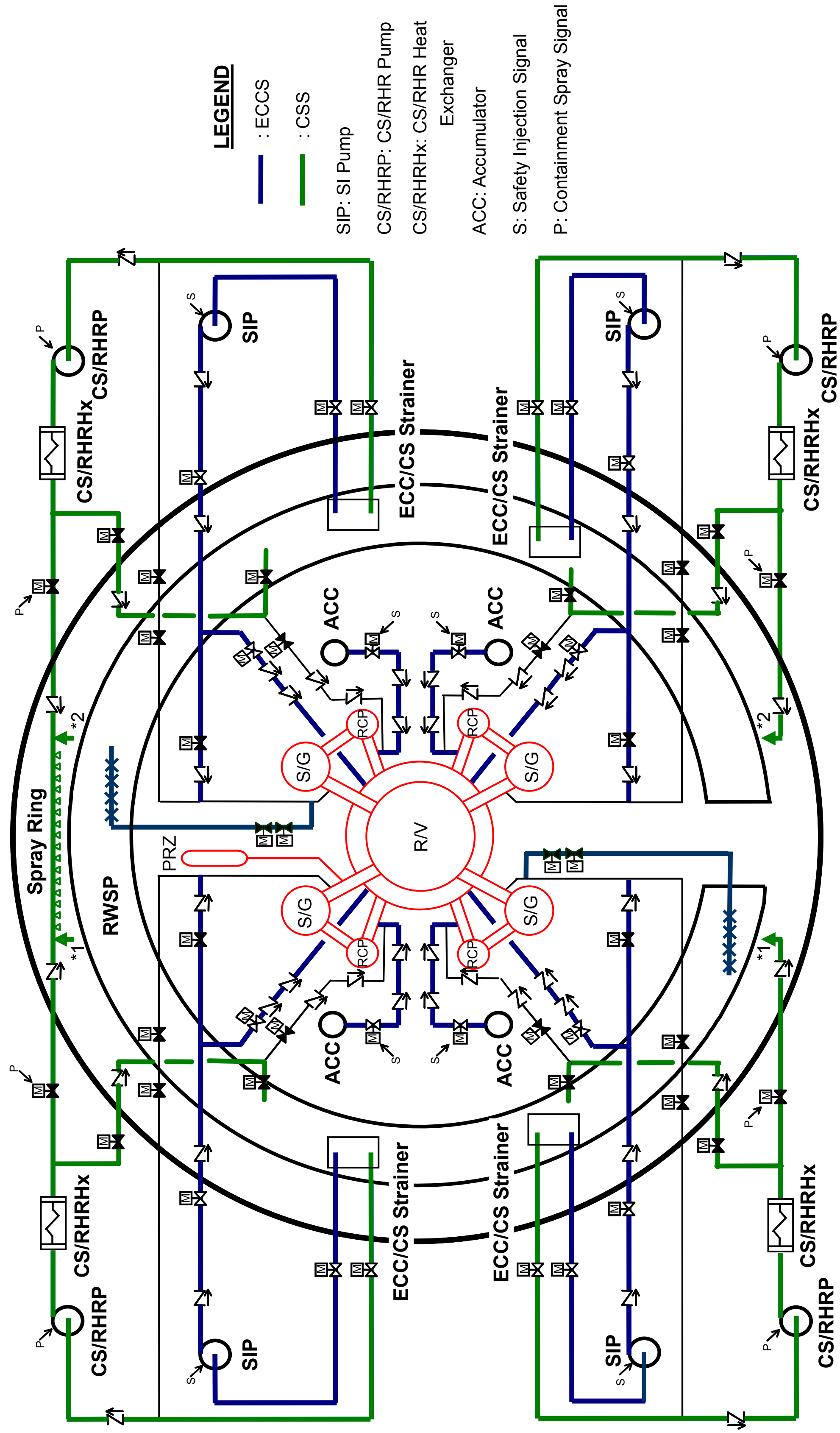


Figure 3-7 Schematic Flow Diagram of ECCS/CSS

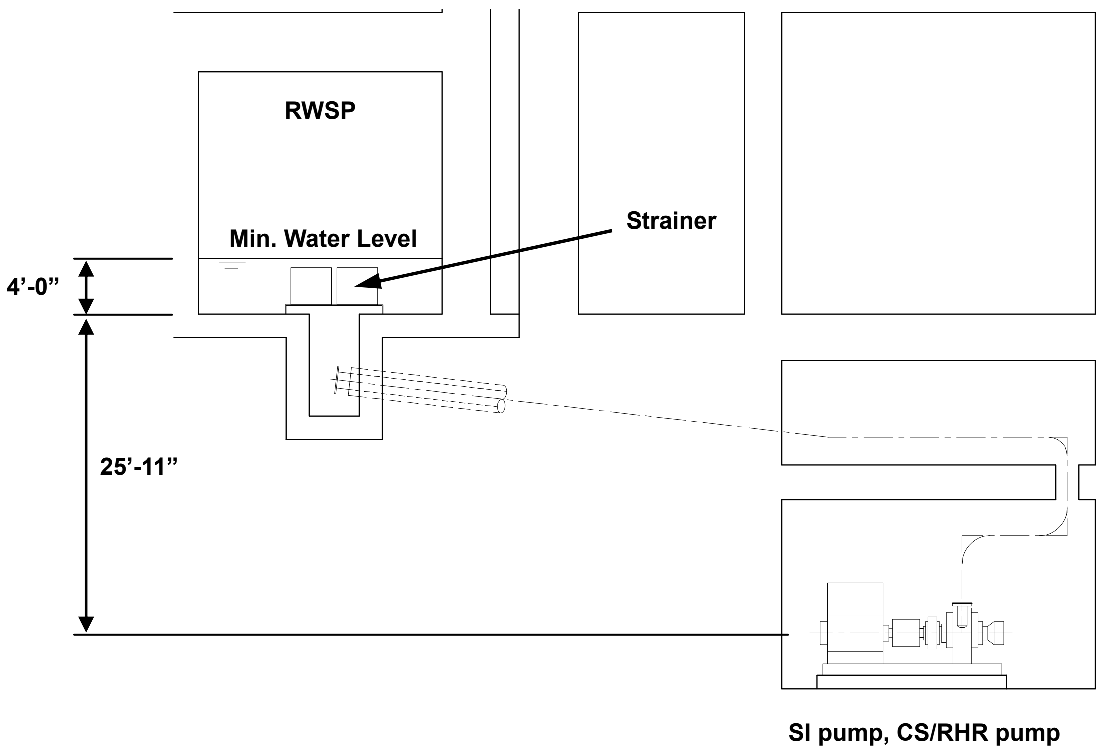


Figure 3-8 Elevation between Minimum Water Level and the Pumps

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3-9 Schematic of Return Water and Hold-up Volumes

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3-10 Gratings Preventing Drains from Blockage

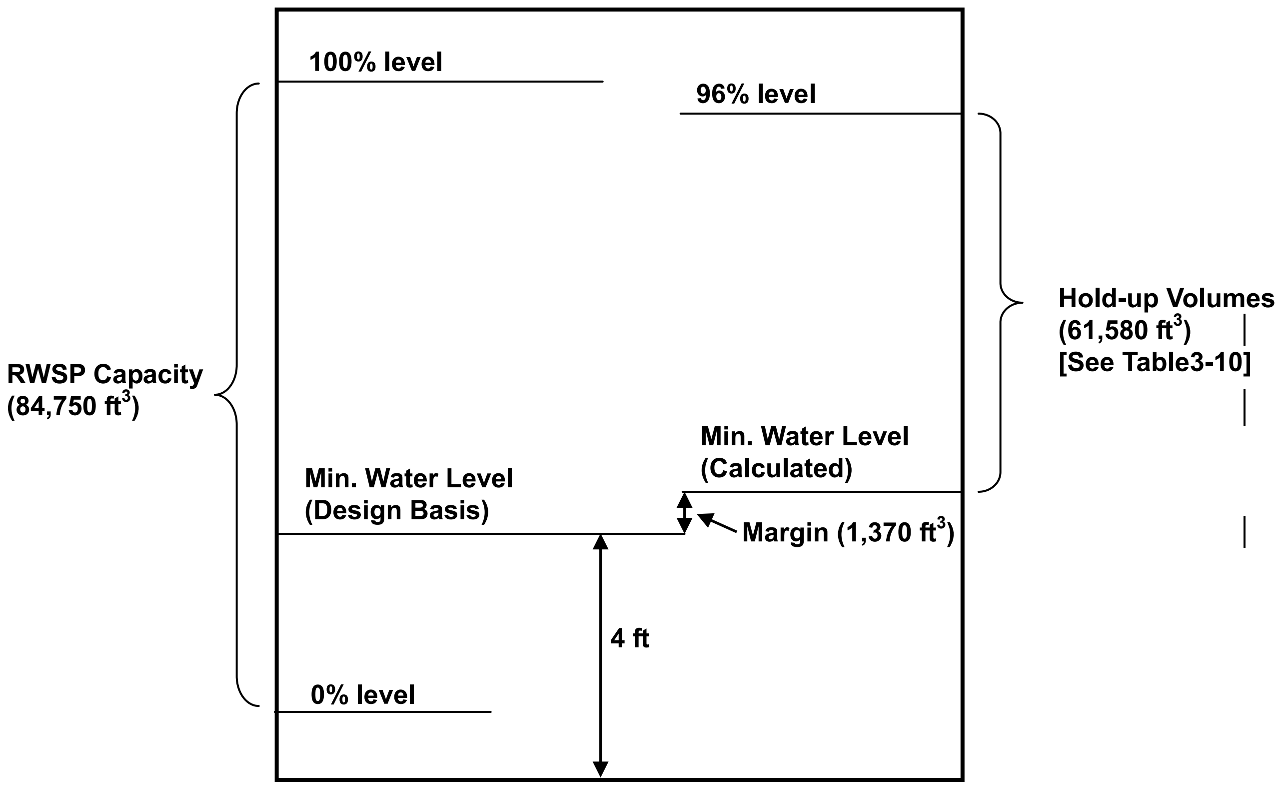


Figure 3-11 Minimum Water Level of the RWSP

Figure 3-12 Total Suspended Solids at pH=7.8 and 150ºF in US-APWR Recirculation Test.
(Figure 4.1-20 from Reference [13])

Figure 3-13 Amorphous $Al(OH)_3$ Solubility at pH=7.8

Figure 3-14 Long-term Recirculation Sump Water Temperature Profile.
(The DCD Figure 6.2.1-26. Format was modified.)

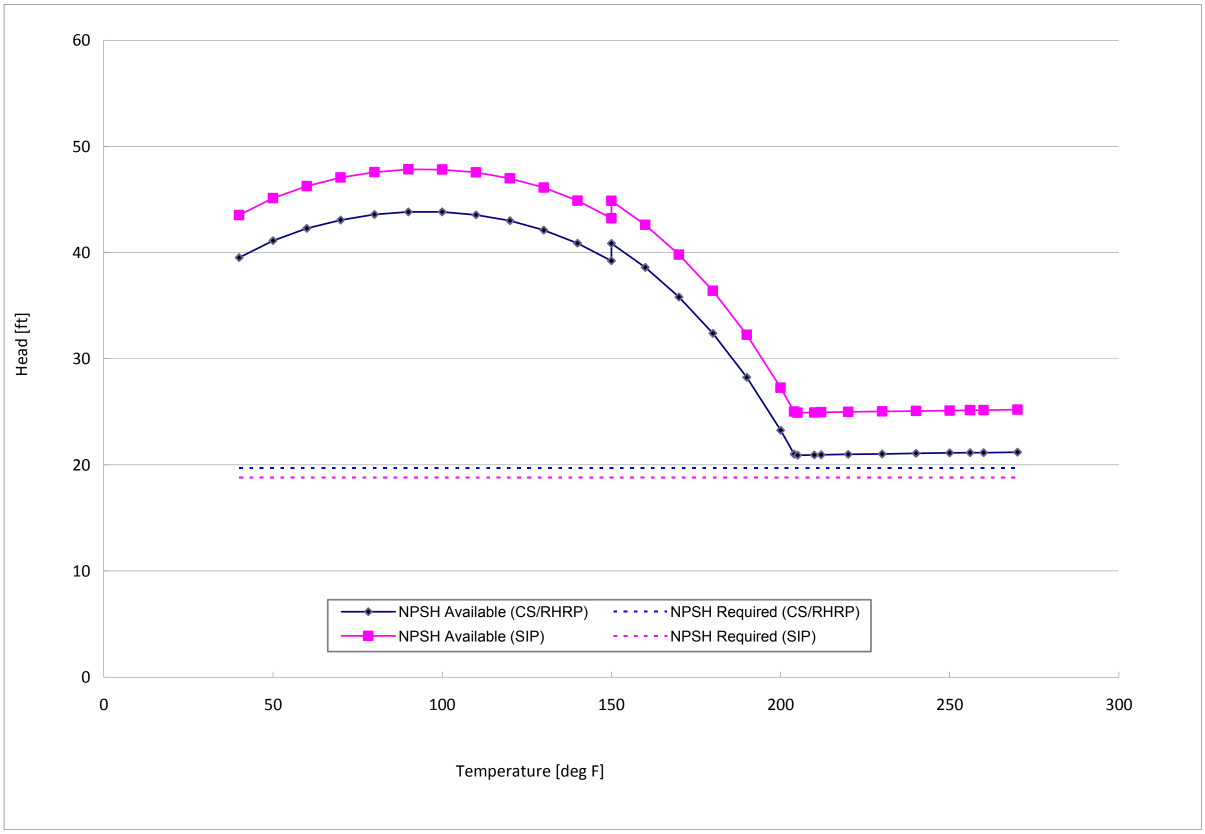


Figure 3-15 NPSH vs. Temperature

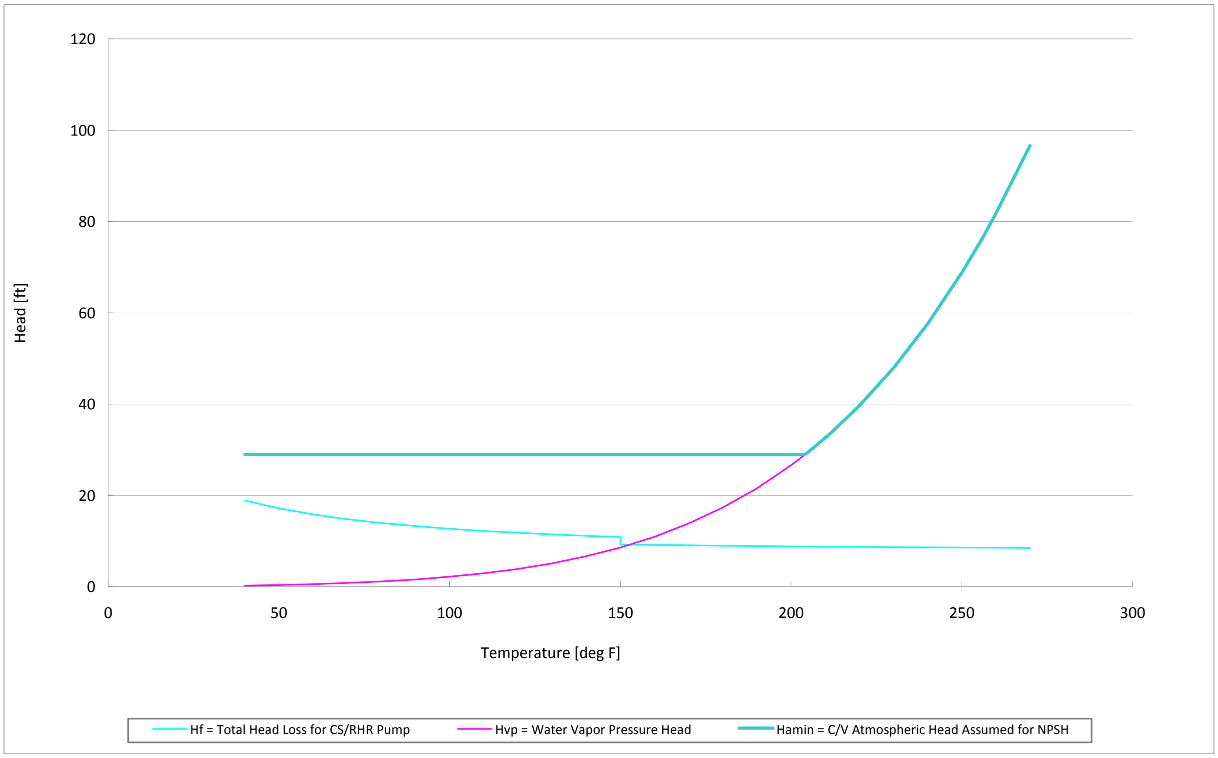


Figure 3-16 Assumed Containment Head, Sump Vapor Pressure Head, and Head Losses

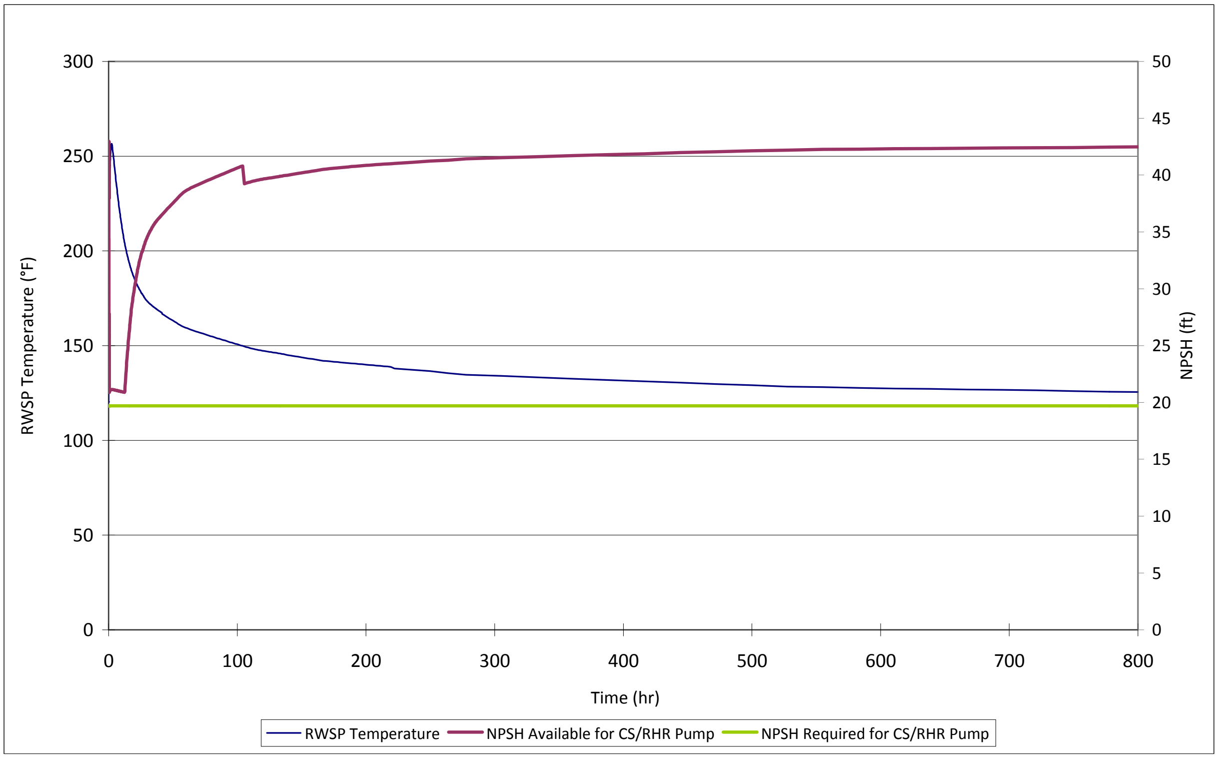


Figure 3-17 NPSH Available vs. Time

Table 3-1 The US-APWR Postulated Break Pipe Lines

Pipe lines	Size	Location					
	Inner Diameter (in)	Inside secondary shield wall					Outside secondary shield wall
		SG compartment				PZR compartment	
		A	B	C	D		
Main coolant pipes	31	X	X	X	X		
PZR surge line	12.81		X				
Accumulator injection lines	11.19	X	X	X	X		
RHR pump inlet lines	8.5	X	X	X	X		
RHR pump outlet lines	6.81	X	X	X	X		
PZR spray line	5.19		X	X		X	
Direct Vessel Injection lines	3.44	X	X	X	X		
Charging line	3.44	X					
Let down line	2.62				X		
Safety injection (SI) lines	3.44	X	X	X	X		
PZR aux. spray line	2.62					X	
PZR safety valve inlet line	5.19					X	
PZR safety depressurizeation lines	6.81					X	
	5.19					X	
	3.44					X	
	2.62					X	
Main steam lines	29.01						X
Feed water lines	14.31	X	X	X	X		X

Table 3-2 Damage Pressure and Corresponding Volume-Equivalent Spherical ZOI Radii

Type	Destruction Pressure (psig)	ZOI Radius / Break Diameter
Transco RMI Darchem DARMET	114	2.0
Unjacketed Nukon, Jacketed Nukon with standard bands	6	17.0

Table 3-3 Debris Generation for Limited Pipe Breaks

Break pipe	Debris type and amount		
	RMI	Fibrous	Coating
Pressurizer surge pipe (16in)	0.21 m^3 [7 ft^3]	0 m^3	0.014 m^3 [0.5ft^3]
MS pipe (31in) (outside secondary shield wall)	0.63 m^3 [22 ft^3]	0 m^3	0.071 m^3 [2.6 ft^3]
FW pipe (16in) (outside secondary shield wall)	0.3 m^3 [11 ft^3]	0 m^3	0.018 m^3 [0.7ft^3]
FW pipe(16 in) (inside secondary shield wall)	0.27 m^3 [10 ft^3]	0 m^3	0.018 m^3 [0.7ft^3]
MCP (design basis)	3.0 m^3 [106 ft^3]	0 m^3	0.082 m^3 [3.0 ft^3]

Table 3-4 Debris Generation

Type		Amount	
		English unit	Metric unit
RMI (Transco)		106 (ft^3) [foil surface area 11,442(ft^2)]	3.0 (m^3) [foil surface area 1063(m^2)]
Fibrous Insulation (Nukon)		0.0 (ft^3) (1)	0.0 (m^3) (1)
Coating (Epoxy)		3.0 (ft^3) (2)	0.082 (m^3) (2)
Latent Debris (200 lbm)	Fiber (15%)	30 (lbm)	-
	Particle (85%)	170 (lbm)	-
Chemical debris	Aluminum Hydroxide	300 (lbm)	-
	Sodium Aluminum Silicate	330 (lbm)	-

Note: Following debris is added as operational margin:

(1) 1.875 (ft^3) of fiber debris

(2) 200 (lbs) of coating debris

Table 3-5 Debris Characteristics

Description	Symbol	Values
RMI debris		
Inter-foil gap thickness	Kt	0.003 (ft)
Fibrous Insulation debris (Nukon)		
Diameter of fiber	D_{f1}	7 (μm)
Fabricate density	C_{f1}	2.4 (lbm/ft^3)
Material density	ρ_{f1}	159 (lbm/ft^3)
Specific surface volume	S_{vf1}	1.742×10^5 (ft^{-1})
Coating		
Diameter of particle	D_{p1}	10 (μm)
Sludge density	C_{p1}	19 (lbm/ft^3)
Material density	ρ_{p1}	94 (lbm/ft^3)
Specific surface volume	S_{vp1}	1.829×10^5 (ft^{-1})
Latent Debris (Fiber) Note		
Fabricate density	C_{f2}	Assumed same as to Nukon
Material density	ρ_{f2}	Assumed same as to Nukon
Specific surface volume	S_{vf2}	Assumed same as to Nukon
Latent Debris (Particulate) Note		
Sludge density	C_{p2}	75 (lbm/ft^3)
Material density	ρ_{p2}	168.6 (lbm/ft^3)
Specific surface volume	S_{vp2}	1.06×10^5 (ft^{-1})

Table 3-6 Debris Head Loss

		Measured	204 ºF	150 ºF	120 ºF
(1)Debris Laden Test Strainer Head Loss (DTSHL) (Test value)[1]	Non-chemical only	1.279 (ft)	0.680 (ft)	0.989 (ft)	1.283 (ft)
	Non-chemical plus chemical	1.538 (ft)	N/A	N/A	1.553 (ft)
(2)Clean strainer head loss of test facility (Test Value)[1]		0.4455 (ft)	0.243	0.347 (ft)	0.446 (ft)
(3)Clean Strainer Head Loss (CSHL, calculated)		0.055 (ft) (120 ºF)	0.051 (ft)	0.053 (ft)	0.055 (ft)
(4)Debris head loss	Non-chemical only	-	0.49 (ft)	0.70	N/A
	Non-chemical plus chemical	-	N/A	N/A	1.17 (ft)
(5) Design Basis Strainer Head Loss	Non-chemical only	-	1.0 (ft)	1.45	N/A
	Non-chemical plus chemical	-	N/A	3.09	4.0 (ft)

Note1: Test values represented at the first strainer head loss test. (Refer to Appendix-B)

Table 3-7 NPSH Calculation Matrix

Post-LOCA	Short term	Interim term	Long term
Water Temp. (F)	270 > T > 204	204 > T > 150	150 > T > 70
Design Basis Strainer Head Loss(DBSHL)[1]	Non-chemical	Non-chemical	Non-chemical plus Chemical
	1.0 ft at 204°F	1.45 ft at 150°F	7.0 ft at 70°F
Applicable $NPSH_A$ Calculation (See Section 3.6.2.2)	Equation (1)	Equation (2)	Equation (2)
Elevation Head[3] (CS/RHR and SI)	29.7 ft		
Line Loss[2, 3] (CS/RHR)	7.8 ft		
NPSH Required[2, 3] (CS/RHR)	19.7 ft		

Notes:

1) Debris (Strainer) Losses are adjusted for viscosity (temperature).

2) Line Losses and NPSH Required for CS/RHR bound those for the SI system. These values represent conservative bounds and are not adjusted for temperature.

3) From US-APWR DCD Table 6.2.2-1.

Table 3-8 Sump Fluid Flashing Calculation Cases

Case	Post-LOCA Recirculation Period	Design Basis Temperature, °F	Debris Load	Minimum Submergence above the RWSP floor
1	Short Term	270>T>204	Non-Chemical	4.0 ft
2	Interim Term	204>T>150	Non-Chemical	4.0 ft
3	Long Term	150>T>70	Non-Chemical and Chemical	4.0 ft

Note: Design Flow Rate 5,200 gpm per sump is considered.

Table 3-9 Static Pressure at Strainer Outlet vs. Vapor Pressure

Case	Design Basis Temperature Range °F	Debris Source for Strainer HL	Design Basis Temperature °F	Static Head(1) (Submergence) ft of water	Assumed Containment Pressure(2) ft of water	Design Basis Strainer HL ft of water	Static Pressure at Strainer Outlet ft of water	Local RWSP Vapor Pressure ft of water
1	270>T>204	Non-chemical	270	1.0	96.6	0.7	96.9	96.6
			204	1.0	28.9	1.0	28.9	28.9
2	204>T>150	Non-chemical	204	1.0	28.9	1.0	28.9	28.9
			150	1.0	28.9	1.5	28.6	8.6
3	150>T>70	Non-chemical plus chemical	150	1.0	28.9	3.1	26.9	8.6
			70	1.0	28.9	7.0	23.0	0.8

Note:

1. Minimum static head at top of strainer.
2. See Section 3.6.2.2 for details of the assumed containment pressures in the NPSH analysis.

Table 3-10 Upstream Effect Hold-up Volumes

[1] Return water on the way to the RWSP	(ft^3)
Containment spray droplets & saturated steam (including the empty spray header rings & pipes)	6,510
Condensate water on the various surfaces	2,300
Water stream on the floor (including refueling cavity)	7,320
Volume of water in piping and floor opening on 2nd floor	400
Subtotal [1]	16,530 (approx. 123,600 gallons)
[2] Ineffective pools	
Reactor cavity	16,660
Header compartment (Including ducts)	14,280
Containment reactor coolant drain pump room (Including containment drain sump)	6,530
Upper core internal laydown pit	810
NaTB Baskets	2,950
Additional hold-up volume	3,820
Subtotal [2]	45,050 (Approx. 337,00 gallons)

Table 3-11 NPSH available Calculation Data (CS/RHR Pump)

NPSHavailable Calculation (CS/RHRP)								
Temp	Hamin	HEL	HF		Hvp	NPSHa	Design Basis	Margin
			Hpipe	HL			NPSHreq	
deg F	ft	ft	ft	ft	ft	ft	ft	ft
40	29	29.7	7.8	11.09	0.28	39.53	19.7	19.83
50	29	29.7	7.8	9.37	0.41	41.11	19.7	21.41
60	29	29.7	7.8	8.05	0.59	42.26	19.7	22.56
70	29	29.7	7.8	7.00	0.84	43.06	19.7	23.36
80	29	29.7	7.8	6.15	1.17	43.58	19.7	23.88
90	29	29.7	7.8	5.46	1.61	43.82	19.7	24.12
100	29	29.7	7.8	4.89	2.19	43.82	19.7	24.12
110	29	29.7	7.8	4.41	2.94	43.55	19.7	23.85
120	29	29.7	7.8	4.00	3.91	42.99	19.7	23.29
130	29	29.7	7.8	3.65	5.13	42.12	19.7	22.42
140	29	29.7	7.8	3.35	6.67	40.88	19.7	21.18
150	29	29.7	7.8	3.09	8.59	39.23	19.7	19.53
150	29	29.7	7.8	1.46	8.59	40.85	19.7	21.15
160	29	29.7	7.8	1.35	10.95	38.60	19.7	18.90
170	29	29.7	7.8	1.26	13.84	35.80	19.7	16.10
180	29	29.7	7.8	1.17	17.35	32.38	19.7	12.68
190	29	29.7	7.8	1.10	21.57	28.24	19.7	8.54
200	29	29.7	7.8	1.03	26.61	23.26	19.7	3.56
204	29	29.7	7.8	1.01	28.89	21.00	19.7	1.30
205	-	29.7	7.8	1.00	-	20.90	19.7	1.20
210	-	29.7	7.8	0.97	-	20.93	19.7	1.23
212	-	29.7	7.8	0.96	-	20.94	19.7	1.24
220	-	29.7	7.8	0.92	-	20.98	19.7	1.28
230	-	29.7	7.8	0.87	-	21.03	19.7	1.33
240	-	29.7	7.8	0.82	-	21.08	19.7	1.38
250	-	29.7	7.8	0.78	-	21.12	19.7	1.42
256	-	29.7	7.8	0.76	-	21.14	19.7	1.44
260	-	29.7	7.8	0.75	-	21.15	19.7	1.45
270	-	29.7	7.8	0.71	-	21.19	19.7	1.49

Table 3-12 NPSH available Calculation Data (SI Pump)

NPSHavailable Calculation (SIP)								
Temp	Hamin	HEL	HF		Hvp	NPSHa	Design Basis	Margin
			Hpipe	HL			NPSHreq	
deg F	ft	ft	ft	ft	ft	ft	ft	ft
40	29	29.7	3.8	11.09	0.28	43.53	18.8	24.73
50	29	29.7	3.8	9.37	0.41	45.11	18.8	26.31
60	29	29.7	3.8	8.05	0.59	46.26	18.8	27.46
70	29	29.7	3.8	7.00	0.84	47.06	18.8	28.26
80	29	29.7	3.8	6.15	1.17	47.58	18.8	28.78
90	29	29.7	3.8	5.46	1.61	47.82	18.8	29.02
100	29	29.7	3.8	4.89	2.19	47.82	18.8	29.02
110	29	29.7	3.8	4.41	2.94	47.55	18.8	28.75
120	29	29.7	3.8	4.00	3.91	46.99	18.8	28.19
130	29	29.7	3.8	3.65	5.13	46.12	18.8	27.32
140	29	29.7	3.8	3.35	6.67	44.88	18.8	26.08
150	29	29.7	3.8	3.09	8.59	43.23	18.8	24.43
150	29	29.7	3.8	1.46	8.59	44.85	18.8	26.05
160	29	29.7	3.8	1.35	10.95	42.60	18.8	23.80
170	29	29.7	3.8	1.26	13.84	39.80	18.8	21.00
180	29	29.7	3.8	1.17	17.35	36.38	18.8	17.58
190	29	29.7	3.8	1.10	21.57	32.24	18.8	13.44
200	29	29.7	3.8	1.03	26.61	27.26	18.8	8.46
204	29	29.7	3.8	1.01	28.89	25.00	18.8	6.20
205	-	29.7	3.8	1.00	-	24.90	18.8	6.10
210	-	29.7	3.8	0.97	-	24.93	18.8	6.13
212	-	29.7	3.8	0.96	-	24.94	18.8	6.14
220	-	29.7	3.8	0.92	-	24.98	18.8	6.18
230	-	29.7	3.8	0.87	-	25.03	18.8	6.23
240	-	29.7	3.8	0.82	-	25.08	18.8	6.28
250	-	29.7	3.8	0.78	-	25.12	18.8	6.32
256	-	29.7	3.8	0.76	-	25.14	18.8	6.34
260	-	29.7	3.8	0.75	-	25.15	18.8	6.35
270	-	29.7	3.8	0.71	-	25.19	18.8	6.39

4.0 DOWNSTREAM EFFECT

The assessment of the US-APWR systems and components downstream of the sump strainer downstream effects caused by post-LOCA debris laden fluid is discussed in separate technical report. (Reference [17])

The report discusses two (2) subject areas: Ex-Vessel and In-Vessel Evaluations, and concludes that the US-APWR ECCS, CSS 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. It 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).

5.0 CONCLUSION

This technical report describes the design and the evaluation of the US-APWR sump strainer. The US-APWR sump strainer design is intended to be consistent with the guidance in RG 1.82 Rev.3. The break selection, debris generation, and debris transport were analyzed to identity the potential debris which may reach the strainers in the RWSP assuming a number of conservative considerations. The characteristics of potential debris were set, identified, and used in the debris head loss evaluations, as well as the NPSH evaluation of the vital pumps of the US-APWR. A plant specific qualified test was conducted to determine the maximum strainer head loss due to the postulated debris accumulation during an accident, and test data was used for NPSH evaluation. The evaluation concluded that the postulated debris head loss would satisfy the specified requirements of the standard US-APWR, and have sufficient suction head to operate the plant safely during post-LOCA conditions.

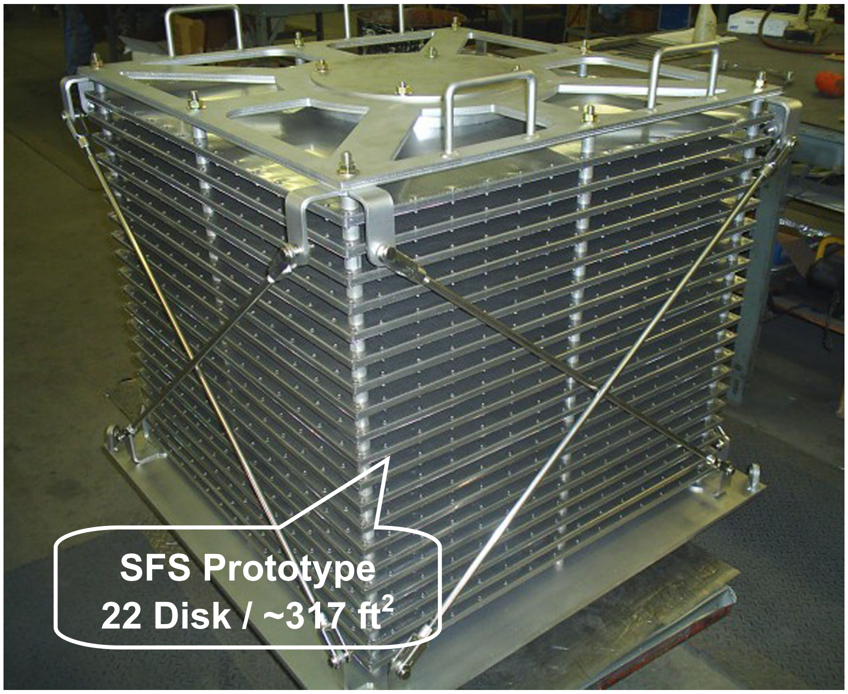
6.0 REFERENCES

1. "Design Control Document for the United States Advanced Pressurized Water Reactor", Mitsubishi Heavy Industries, Ltd., Docket No 52-021.
2. "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 3", Regulatory Guide 1.82 Rev.3, U.S. NRC, November, 2003
3. Safety Evaluation by the NRC related to Generic Letter 2004-02, NEI Guidance Report (Proposed Document Number NEI 04-07), "Pressurized Water Reactor Sump Performance Evaluation Methodology", U.S.NRC, December 2004. (ML043280007, ML043280008)
4. "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants", Regulatory Guide 1.54 Revision 1, U.S. NRC, July, 2000.
5. "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities", ANSI N101.2
6. "Standard Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions", ASTM D 3911
7. NRC letter to NEI, "Revised guidance regarding coatings zone of influence for review of final licensee responses to Generic Letter 2004-02, Potential Impact of Design Basis Accidents at Pressurized Water Reactors", dated April 2010. (ML100960495)
8. "Characterization and Head-Loss Testing of Latent Debris from Pressurized Water Reactor Containment Buildings", NUREG/CR-6877, U.S NRC. (ML052430751)
9. Crane Technical Paper No. 410, "Flow of Fluids Through Valves, Fittings, and Pipe"
10. "Integrated Chemical Effects Test Project: Consolidated Data Report, Volume 1", NUREG/CR-6914, U.S NRC, Dec. 2006. (ML062560105).
11. "Integrated Chemical Effects Test Project: Test #5 Data Report, Volume 6", NUREG/CR-6914, U.S. NRC, Dec. 2006. (ML062560105)
12. "US-APWR Sump Debris Chemical Effects Test Plan", MUAP-08006 Rev.1, Mitsubishi Heavy Industries, Ltd., November 2008. (ML083390119)
13. "US-APWR Sump Debris Chemical Effects Test Results", MUAP-08011 Rev.1, Mitsubishi Heavy Industries, Ltd., May 2012.
14. Argonne National Laboratory, "Aluminum Solubility in Boron Containing Solutions as a Function of pH and Temperature", Sep. 2008. (ML091610696)

15. "Aluminum Chemistry in a Prototypical Post-Loss-of-Coolant-Accident, Pressurized Water Reactor Containment Environment", NUREG/CR-6915 (pg 55, Figure 41), U.S.NRC. (ML070160448)
16. Thermochemical Data of Pure Substances, Ihsan Barin, VCH, 1989.
17. "US-APWR Sump Strainer Downstream Effects", MUAP-08013 Rev.3, Mitsubishi Heavy Industries, Ltd., May 2012.

Appendix-A

PCI Sure Flow Strainer Technical Description and Drawings



The prototype **Sure-Flow® Strainer** (SFS) shown on the left was built in the PCI factory to proof the installation ease of our SFS modular concept.

This is very similar to the 21 disk SFS PCI is proposing to supply herein for the MHI U.S. APWR.

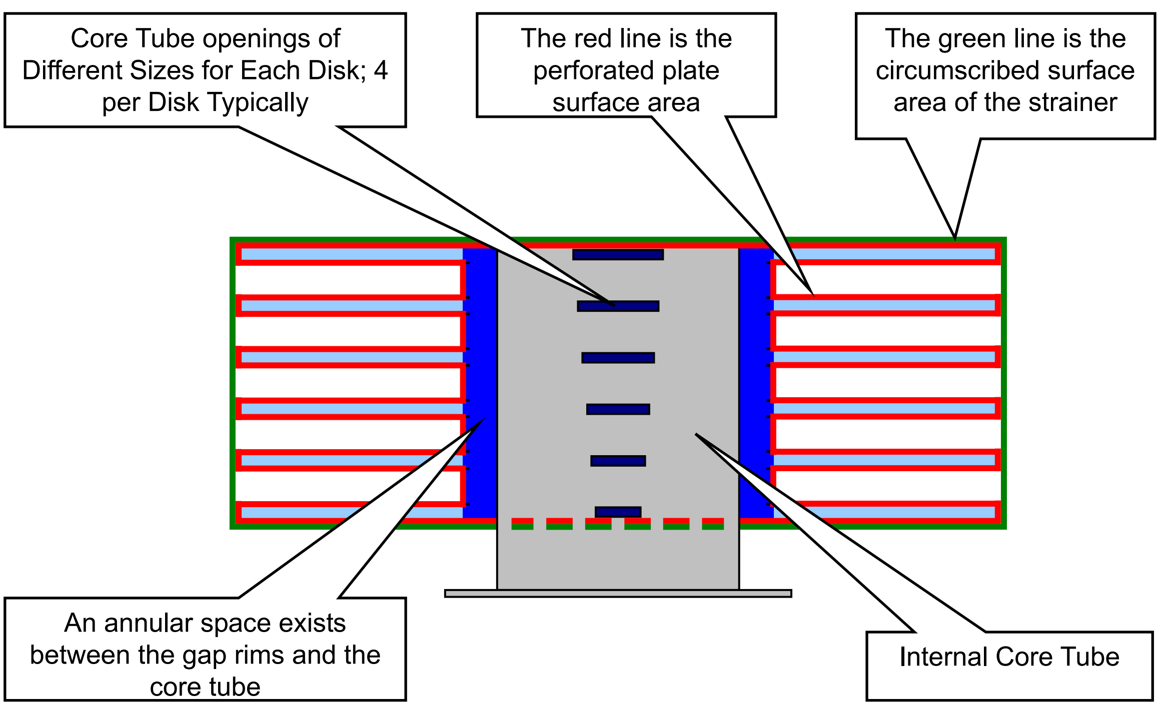
As shown to the left, this 22 disk strainer can be assembled by two men in less than one hour.

The sizes of the disks are approximately 36” x 36”.

<u>Basic Design Components of SFS</u>

The illustration in Figure A-1 below identifies the basic concept of the components and terms applicable to Sure-Flow Strainers.

FIGURE A-1



The following describes and defines the PCI Sure-Flow® Strainer (SFS) major component elements, features and benefits as proposed for the MHI U.S. APWR.

SFS Module Components and Assembly

The following major components are described below and identified in Figure A-2.

1. Base Plate – The SS plate upon which a core tube is attached containing a hole to within the perimeter of the core tube to allow water to communicate with the plenum or sump opening below. Also attaching to this plate are brackets for cross braces and a solid gap rim between the base plate and first perforated disk.

2. Core Tube – A stainless steel cylinder with openings enlarging in locations upstream of the suction source specifically designed to create uniform flow axially along the length of the core tube. This is a patented feature of all Sure-Flow Strainers. The benefits of a uniform suction pressure to each disk and gap are as follows:

    - Results in lower "actual" approach velocities to each disk and gap. A lower approach velocity means the debris bed will form under lower compression than screens without flow control resulting in a lower head loss than same size screens without flow control.
    - Control of the approach velocity justifies the use of reduced scale testing in lieu of testing the entire SFS screen arrangement. A single, full scale module will be used to test the debris bed head loss performance of the entire SFS arrangement.
    - The core tube discourages and prevents vortex formation and air ingestion.
    - The core tube provides design flexibility so that disks and modules can be positioned further from a central suction source to increase the size of screens provided within a given special limitation.

3. Disk – A rectangular chamber fabricated from stainless steel perforated plate to form nominal ½" thick disks. All disks are reinforced to prevent collapse against specified suction pressures using an innovative wire frame design that allows low head loss flow through the disk to its center where water exists into an annulus chamber that exists around the core tubes. The wire frames can be designed to work with light weight sheets; which allows PCI to offer perforated plate designs with holes as small as 0.037" in diameter. The hole size specified by MHI for the US APWR is 0.066" diameter.

4. Gap Rim – A "hoop" made from stainless steel perforated plate to form nominal 1" high rim

with a diameter 3.5” greater than the core tube to provide an annulus chamber between the gap rims and core tube. Gap rims serve to connect the surface area formed by disks together and to provide additional surface area. Gap rims all have solid edge margins so as to not form openings larger than holes at the interface between disks and gap rims. Gap rims are also designed to withstand the specified suction pressures of the system with internal hoops and pipe spacers held in place with tension rods.

5. Tension Rods and Pipe Spacers – There are normally 12 tension rods to a SFS module. Pipe spacers are positioned inside each disk and between disks to provide a continuous tube for the tension rod to pass through the assembly and to assure the components align precisely. The pipe spacers provide a bearing surface when tension is applied to the rods by nuts or couplings. Eight (8) tension rods are positioned along the outside perimeter and four (4) are positioned just inside the gap rims. Tension rods are used to squeeze the disks, gap rims and pipe spacers together into a single assembly without openings that would allow debris larger than the size of holes to enter the internal chamber. They also provide an important structural component to the assembly.

6. Top Frame – At the top of the SFS module assembly is a thick stainless steel plate with flow openings to which tension rods are terminated under torque to complete the assembly. A solid gap rim is usually positioned between the top disk surface and Top Frame to provide additional openings for water to enter the top surface of the top disk.

7. Cross Bracing – To comply with twisting and lateral structural forces cross bracing is added external to the disks between base plates and top frames; and / or connect to intermediate stiffener plates as shown in Figure A-1. When an intermediate stiffener plate with flow openings is used gap rims of ½” height are used instead of 1” high gap rims to increase surface area and protect submergence to the specified water level. The cross braces connect via threaded connections so as to allow a secure and firm fit during assembly.

All components of the SFS module is fabricated from stainless steel to provide worry free service inside nuclear containment environments.

Design Reference History

The following U.S. PWR plants contracted PCI to supply our Sure-Flow Strainers to them for operation. PCI will be applying the same design principles to this design as we have implemented in the following plants.

No.	Plant Name	NSSS Supplier / Loops
1	Callaway	Westinghouse / 4 loops
2	Wolf Creek	Westinghouse / 4 loops
3	Comanche Peak Units 1 & 2	Westinghouse / 4 loops
4	South Texas Project Units 1 & 2	Westinghouse / 4 loops
5	Palisades	CE / 2 loops
6	Point Beach Units 1 & 2	Westinghouse / 2 loops
7	Prairie Island Units 1 & 2	Westinghouse / 2 loops
8	Kewaunee	Westinghouse / 2 loops
9	TVA – Watts Bar 1	Westinghouse / 4 loop Ice Condenser
10	TVA – Sequoyah Units 1 & 2	Westinghouse / 4 loop Ice Condenser
11	FPL - St. Lucie Unit 2	CE / 2 loops
12	FPL – Turkey Point Unit 4	Westinghouse / 3 loops
13	TVA – Watts Bar 2	Westinghouse / 4 loop Ice Condenser

- 17 of the 18 units have been fabricated; shipped and installed as of June 2008.

FIGURE A-2 – TYPICAL SURE-FLOW STRAINER ASSEMBLY

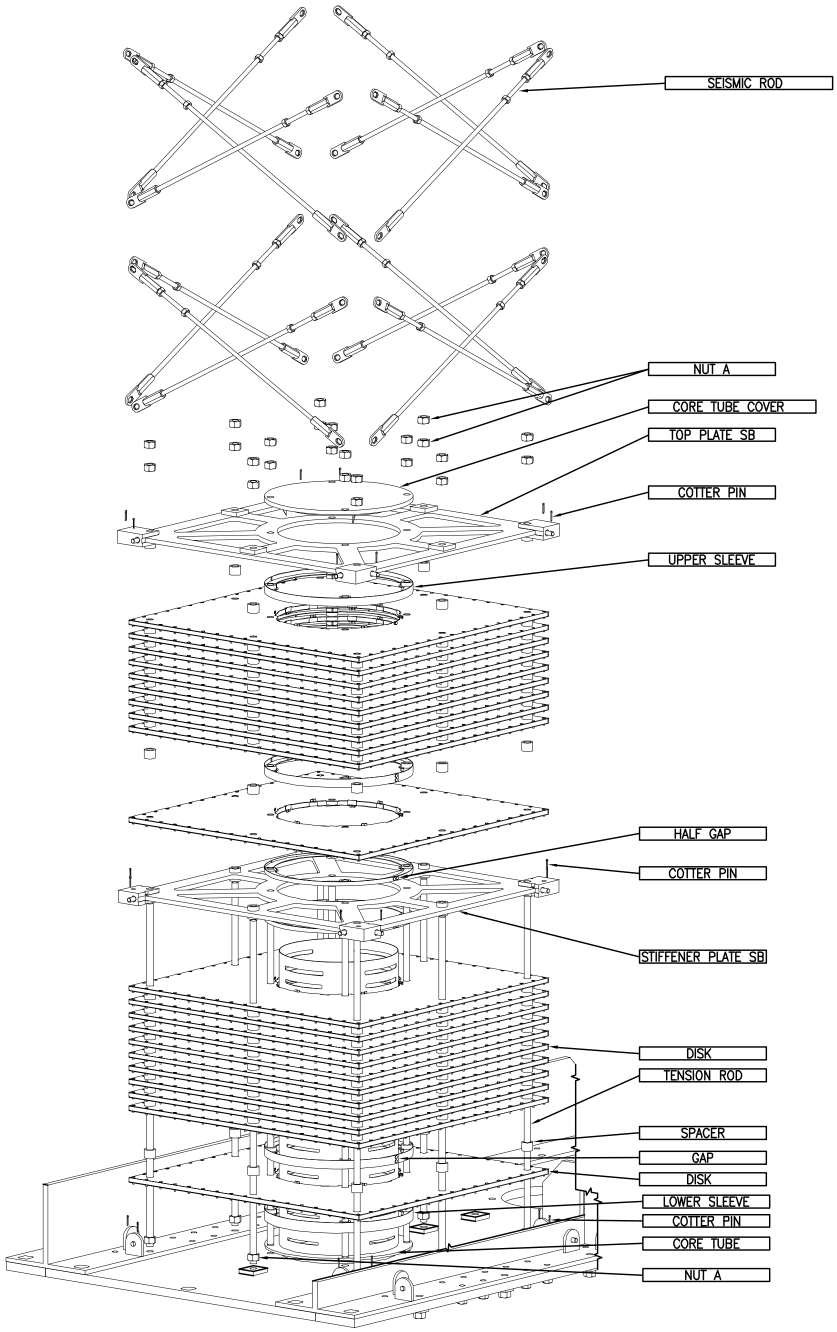


Figure A-3 Sump Strainer Arrangement

Figure A-4 Sump Strainer Module

Figure A-5 Sump Strainer Assemble

Appendix-B

Sump Strainer Head Loss and Fiber Only Bypass Tests

B.1 Introduction

The US-APWR strainer test program (Reference B-1) was conducted to obtain test data for strainer head loss and fiber bypass. The head loss test plan was used to demonstrate acceptable head loss across the sump strainer subject to design basis post-LOCA debris loading. The fiber only bypass test plan was used to demonstrate the maximum amount of fiber that bypasses the sump strainer corresponding to various modes of sump operation and debris loading. The test protocol was developed to reflect the discussions between the staff and MHI; extensive review and evaluation of NRC public meeting documents associated with GL-2004-02, GSI-191, and RG 1.82, Rev. 3; licensee RAI responses; and publicly available NRC trip reports that documented licensee strainer head loss testing. In addition, particular attention shall be paid to the March 2008 guidance document (Reference B-2). The test protocol and test plan also utilized guidance provided in NEI 04-07 Volume 1 and 2 (Reference B-3, B-4).

The following US-APWR test program was implemented:

Test-1 – Fiber Only Bypass Test

Test-2 - Clean Test Strainer Head Loss (CTSHL) Test

Test-3 - Debris Laden Test Strainer Head Loss (DTSHL) Test

Each of these tests is described below. It should be noted that Test-3 is equivalent to the Thin Bed Test due to the limited Design Basis quantity of fibrous debris associated with the US-APWR design.

<u>Test-1 – Fiber Only Bypass Test</u>

The purpose of Test-1 was to establish the maximum amount of fiber to bypass the strainer.

The fiber was added in four batches: 12.5%, 12.5%, 25%, and 50% of the total test fiber quantity. The first batch introduction demonstrated bypass at fiber quantities less than the maximum design basis fiber loading. Fiber introduction in batches two and three represented uniform debris allocation for four and two operable sumps, respectively. The fourth batch introduction demonstrated fiber bypass assuming 100% plant debris accumulation on one sump, with the other operable sumps being clean.

One micron filter bags were utilized to measure the fiber that passed through the strainer. The filter bags were changed before each batch introduction in order to separately determine the fiber bypass for each batch introduction.

Test-2 - Clean Test Strainer Head Loss

The purpose of Test-2 was to establish the clean test strainer head loss (CTSHL) for the US-APWR full-scale test strainer and to establish the system head loss prior to debris loading. Non-chemical and chemical debris was introduced for this test. The test data results were subtracted from Test-3 to determine the “debris loaded” head loss for the strainer.

Test-3 - Debris Laden Test Strainer Head Loss

The purpose of Test-3 was to measure the maximum head loss across the strainer with the design basis debris loading (DTSHL). This test is also deemed as the Thin Bed Test to evaluate the potential for formation of a thin fiber bed that may trap additional particulate debris and increase head loss across the strainer. Floating debris which did not accumulate on the strainer was observed at previous test conducted in June 2010. This issue was resolved by improving debris preparation and introduction procedure discussed in sections B.2 and B.3 below.

The CTSHL and DTSHL are utilized to establish the total strainer head loss (TSHL). The measured CTSHL for Test-2 is subtracted from the DTSHL result for Test-3 and the prototype clean strainer head loss (CSHL, calculated separately) is added to determine the TSHL. Finally, design basis strainer head loss (DBSHL), with a safety margin added to TSHL, is used for the design basis NPSH calculation as discussed in Section 3.6 of this report.

The US-APWR strainer tests were implemented in June, 2011 at Alden Research Laboratory (ARL) located at Holden, MA.

B.2 Test Facility

The US-APWR fiber only bypass test program and strainer head loss test program utilized a mixing tank test protocol and test facility that does not take credit for debris settlement upstream of the strainer. This is intended to conservatively bound plant conditions. The test facility is not intended to be prototypical or similar to the near-field strainer flow dynamics of the plant. The intent of this protocol and facility is to keep debris suspended and prevent settling during the tests. The manner in which debris introduced into the test tank minimizes air entrainment in order to minimize floating debris. Floating debris that was observed during the test was reintroduced.

B.9 References

B-1 AREVA Document 63-9160802-001 "US-APWR Test Plan for ECCS Strainer Performance Testing 2011", PCI EC-PCI-MHIUSA-6032-1010 Rev.1 August, 2011.

B-2 NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Strainer Head Loss and Vortexing, March 2008, USNRC, NRR.

B-3 NEI 04-07 Volume 1, "Pressurized Water Reactor Sump Performance Evaluation Methodology," December 2004.

B-4 NEI 04-07 Volume 2, "Pressurized Water Reactor Sump Performance Evaluation Methodology," December 2004.

B-5 "US-APWR Core Blockage Test", MUAP-11022 Rev.0, Mitsubishi Heavy Industries, Ltd., October, 2011.

B-6 Sure Flow Suction Strainer - Testing Debris Preparation & Surrogates Technical Document No. SFSS-TD-2007-004 Revision 4, January 2009. [ML092430056 & ML092580203]

B-7 NUREG/CR-6877 (LA-UR-04-3970), Characterization and Head Loss Testing of Latent Debris from Pressurized Water Reactor Containment Buildings, July 2005 USNRC.

B-8 Letter from Ho K. Nieh (NRR) to Gordon Bischoff (PWROG), dated December 21, 2007, Final Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191" (TAC No. MD1119) (ML073521072), and Final Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report WCAP-16530-NP "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191" Pressurized Water Reactor Owners Group Project No. 694 (ML073520891

B-9 NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris", Appendix-B, "Transient ECCS Blockage Model", October 1995 USNRC

B-10 Letter from Ervin L Geiger (NRR) to Stuart N Bailey (SIRB), dated October 8, 2010, "Trip Report on Staff Observation of Generic Safety Issue 191 Related Fuel Blockage Testing of AREVA Fuel at Westinghouse Science and Technology Center" [ML102720058]

B-11 AREVA Document 66-9164556-000 "US-APWR ECCS Strainer Performance Test Report 2011", PCI EP-PCI-MHIUSA-6032-1011 Rev.0, August 2011

Figure B-1 US-APWR Test Flume Configuration

Figure B-2 Test Tank (Prototypical Strainer)

Figure B-3 Test Tank (Agitation Area)

Figure B-4 Debris Injection Hopper

Figure B-5 Test Debris (NUKON, Dry)

Figure B-6 Strainer After Tank Water Drain Down at First Fiber Only Bypass Test

Figure B-7 Strainer After Fourth Batch at Secound Fiber Only Bypass Test

Figure B-8 Filter Bags Post First Fiber Only Bypass Test

Figure B-9 Filter Bags Post Second Fiber Only Bypass Test

First Test conducted on June 8, 2011

Second Test conducted on June 15, 2011

Figure B-10 CSHL vs. Flow Rate

First Test conducted in June 8-10, 2011

Second Test conducted in Jun 15-17, 2011

Figure B-11 Flow Rate and Head Loss vs. Time

First Test conducted in June 8-10, 2011

Second Test conducted in Jun 15-17, 2011

Figure B-12 Head Loss Trend Prior to Termination

Figure B-13 Debris Accumulation on The Top Surface of Strainer at Both of Fiber Bypass Only Tests for Each Bach

Figure B-14 DTSHL Pictures for Both Tests

Table B-1 Principle Plant/Test Parameters

Table B-2 Fiber Bypass Weights for Fiber Only Bypass Test

Table B-3 CSHL Test Result

Table B-4 Maximum Measured Head Loss at DTSHL Test

Table B-5 Discrepancy from the Average of two Fiber Bypass Only Test Results

Table B-6 Discrepancy from the Average of two DTSHL Test Results

Appendix-C

Evaluation of Chemical Debris (for head loss)

C.1 Introduction

The chemical effects tests of the US-APWR was planned and conducted in order to obtain experimental data under postulated accident conditions of the plant, and evaluate the corrosion products that may form in a post-LOCA environment. The detail of the test plan and results were provided in the technical reports. (Reference C.3-1, C.3-2) One of the test purposes is to determine the amount of chemical reaction products that may impact on debris head losses of the sump strainer.

The environmental conditions of the chemical effects tests were selected so as to maximize the corrosion products released into the recirculation fluid during test period. The test temperature was evaluated, and it was realized that lower temperatures may result in larger amount of precipitates in the tests. However, it was also realized the difficulty to define how to form and settle the corrosion products in the fluid, because it would be significantly affected by temperature and cooling duration. Therefore, a higher temperature profile was selected for the chemical effects tests, and the sample fluids were examined to identify the concentration of each dissolved / undissolved elements, without filtering or cooling to produce the precipitants. Then concentrations of the dissolved / undissolved elements of the tests were used to quantify the chemical precipitants by commercial analyzer (OLI Stream-Analyzer™).

The purpose of the chemical debris calculation of the US-APWR is to use for debris head loss evaluation and testing discussed in Section 3.7 and Appendix-B of this report. Referring the publicly identified in PWROG TR-WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191", following chemical debris of the US-APWR were quantified by the analyzer:

(1) Aluminum Hydroxide :$AlOOH$
(2) Sodium Aluminum Silicate :$NaAlSi_3O_8$
(3) Calcium Phosphate :$Ca(PO_4)_2$ Note

Note: Calcium phosphate is negligible, because the US-APWR, does not use tri-sodium phosphate for pH control agents.

C.2 Chemical debris calculation

The chemical debris for calculation are defined by the result of the chemical effect tests, representing for the recirculated sump after LOCA in US-APWR. The water properties in ECCS and CSS after LOCA in US-APWR will transform as follows:

The sum of the dissolved and the precipitation after dissolution from those materials, that is the total impurities, is used as the input condition of the chemical debris calculation impacts on the sump strainer head loss.

Procedure for calculation of the concentration trends of dissolutions is follows:

1. The calculation is based on the concentration trend of the recirculation test 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 a concentration decreases as time progresses, the concentration is kept constant after the maximum point.

The transient curve of impurities concentration in the sump water evaluated from the chemical effect testing is shown in Figure C-1.

Figure C-1 Impurities concentration trend of the sump water

As a result of the chemical equilibrium calculation for the sump water containing the impurities using OLI Stream-Analyzer™ computer program, the chemical precipitants of the sump water at 70 °F after 30 days of the recirculation are predicted as shown in Table C-1.

As discussed in section C.1, the calculation purpose is to quantify the aluminum hydroxide and sodium aluminum silicate for bounding evaluation. Although the code outputs the other chemical compositions, they are negligible and excluded from comparison.

Table C-1 Chemical debris of the US-APWR

C.3 Maximum Recirculating Water Volume

It is more conservative to estimate a larger amount of recirculating water volume for calculating the chemical debris amount. Therefore, the recirculating water volume shall be estimated as the maximum amount of recirculating water.

Since the RCS and Accumulators contain large amounts of water in containment, a large-break LOCA shall be considered for determining the recirculating water volume. During a large break LOCA, these volumes shall be released in containment and considered to become recirculating water in the post-LOCA environment. The maximum recirculating water volume is calculated by subtracting the “ineffective pools” volume from the containment water volume. For conservatism, the “ineffective pools” volume shall be minimized and the containment water volume shall be maximized.

The maximum water volume in containment considers the following water volumes:

- 0-100 % water volume in RWSP and below 0% water volume
- Maximum water volume in RCS
- Maximum water volume in Accumulators

Although the "ineffective pools" volume is defined in Section 3.7.2, some areas, such as the reactor cavity and header compartment, become a recirculation flow path. These areas subject to recirculation flow shall be excluded from the "ineffective pools" for determining the maximum recirculating water volume. Only the following areas shall be estimated as "ineffective pools" for determining the maximum recirculating water volume:

- C/V drain pump room
- Condensate water between PCCV and RWSP gap (a part of additional hold-up volume)

Subtracting the minimum "ineffective pools" volume from the maximum water volume in containment, the maximum recirculating water volume becomes 3,200 m^3 (113,000 ft^3).

C.4.References

C.4-1 US-APWR Sump Debris Chemical Effects Test Plan, MUAP-08006 (R1), November 2008

C.4-2 US-APWR Sump Debris Chemical Effects Test Result, MUAP-08011 (R0), December 2008

Appendix-D

Detail Drawings around RWSP

Figure D-1 Containment Plan View (RWSP, EL. 3'-7")

Figure D-2 Containment Plan View (EL. 25’-3”)

Figure D-3 Section View

Figure D-4 Nozzle and Equipment Detail

Appendix-E

Overview of Regulatory Requirements regarding Containment Pressure

This Appendix provides background information on the regulatory requirements for calculating available NPSH evaluations for ECCS pumps.

Safety Guide 1 (RG 1.1) provides the following Regulatory Position:

- Emergency core cooling and containment heat removal systems should be designed so that adequate net positive suction head (NPSH) is provided to system pumps **assuming maximum expected temperatures of pumped fluids and no increase in containment pressure from that present prior to postulated loss of coolant accidents.**
  (Emphasis added)

This position is referenced and expanded upon in Position 1.3.1.1 of RG 1.82 Rev. 3 (referenced by SRP 6.2.2):

- ECC and containment heat removal systems should be designed so that sufficient available NPSH is provided to the system pumps, assuming the maximum expected temperature of pumped fluid and no increase in containment pressure from that present prior to the postulated LOCA. (See Regulatory Position 1.3.1.2.)

For sump pools with temperatures less than 212 °F, it is conservative to assume that the containment pressure equals the vapor pressure of the sump water. This ensures that credit is not taken for the containment pressurization during the transient.

Assuming no increase in containment pressure is appropriate for low temperatures. However, the regulatory guidance does not explicitly address NPSH evaluation at high fluid temperatures (above approximately 212°F) for which the fluid saturation pressure exceeds the initial containment pressure. As indicated in Figure 3-14, the maximum RWSP fluid temperature for the US-APWR is approximately 256°F. This corresponds to a RWSP fluid saturation pressure of 33.1 psia compared to the initial containment pressure of approximately 14.7 psia.

Designing for these conditions is not appropriate, reasonable, or necessary to satisfy the intent of the regulatory guidance. The intent of assuming no increase in containment pressure is discussed in RG 1.1 and reiterated in RG 1.82 (p. 1.82-5):

- Predicted performance of the ECCS and the containment heat removal pumps should be independent of the calculated increases in containment pressure caused by postulated LOCAs in order to ensure reliable operation under a variety of possible accident conditions. For example, if proper operation of the ECCS or the containment heat removal system depends on containment pressure above a specified minimum amount, operation of these systems at a containment pressure less than this amount (resulting, for example, from impaired containment integrity or operation of the containment heat removal systems at too high a rate) could significantly affect the ability of this system to accomplish its safety functions.

The approach typically applied for high sump temperatures (above the saturation temperature corresponding to the initial containment pressure) is to assume that the containment pressure is equal to the fluid vapor pressure, as shown in
Figure **3-16**.

The assumption that the containment atmosphere is saturated at the RWSP fluid temperature represents a conservative minimum bound for all possible accident conditions. In addition, this assumption is independent from calculated increases in containment pressure; instead, the assumed containment pressure is dependent on the RWSP fluid temperature itself.

Further discussion is provided in Appendix F to demonstrate that this assumption represents an appropriate and conservative minimum bounding value. This section describes the physical basis for this assumption and provides a phenomenological basis through comparison of calculated containment pressure versus RWSP fluid saturation pressure for bounding LOCA cases. For robustness, Appendix F also evaluates the contribution to plant risk from this assumption.

Appendix-F

Validity of assumptions regarding Containment Pressure

The NPSH available determination at containment sump temperatures above 204°F is consistent with the intent of the guidance provided by RG 1.82, Rev. 3, Position 1.3.1.1. It is not, however, consistent with the specific wording of the RG. For NPSH available determinations at high containment temperatures (i.e., above 204°F), it is assumed that the containment pressure is equal to the saturated vapor pressure of the sump water since conditions inside containment will be saturated. This recognizes the physical fact that at fluid temperatures above 204°F the containment pressure has to be greater than at -0.3 psig (the initial containment pressure), otherwise the fluid will boil.

The US-APWR meets the intent of RG 1.82 Rev. 3 in the following manner:

- Section 1 – The assumption of saturated conditions inside containment at high sump temperatures is appropriate and reasonable. The assumption of saturated conditions conservatively bound the actual accident containment pressure, i.e. the saturation pressure is at all times less than the containment pressure during all design basis LOCA conditions and is therefore appropriate. This assumption minimizes NPSH available. The assumption of saturated conditions inside containment at high sump temperatures is reasonable as it is consistent with the thermo-dynamic properties of water and takes no credit for any outside influences.

- Section 2 – Design changes required to strictly meet the wording stated in Position 1.3.1.1 of RG 1.82 Rev. 3 for temperatures above 204°F ("assuming the maximum expected temperature of pumped fluid and no increase in containment pressure from that present prior to the postulated LOCA") would be impracticable or otherwise negatively affect overall system health. Design changes would either necessitate providing an additional design source of pressure to keep the containment fluid from boiling or providing vast quantities of additional fluid to account for both surface and sub-surface boiling at 14.7 psia.

- Section 3 – The contribution to plant risk from inadequate containment pressure at high sump fluid temperatures (i.e., rapid depressurization due to impaired containment integrity or operation of containment heat removal systems at too high a rate) is negligible.

F.1 Containment Pressure during LOCA

During LOCA and post-LOCA conditions for the US-APWR, containment pressure always exceeds the saturated vapor pressure at the RWSP liquid temperature, due to the US-APWR design and strong phenomenological correlations for the postulated LOCA described below.

During a LOCA, mass and energy are released from the primary system to both the vapor phase (containment atmosphere) and to the RWSP (liquid phase) inside the containment volume. Steam released from the primary system postulated break maintains the containment atmosphere at saturated conditions during almost all parts of the LOCA transient. Also, fluid condensed by passive heat sinks (such as the containment shell liner, supporting structures and concrete) and the containment sprays is added to the RWSP. The condensed water entering the RWSP will be at the steam partial pressure in the containment atmosphere. Over the long-term, the RWSP liquid temperature is strongly affected by the liquid water condensed from the atmosphere during the containment spray operation. This condensed water is also saturated at the steam partial pressure. Therefore, a higher containment pressure provides higher temperature condensed water, and higher RWSP liquid temperatures. Similarly, a lower containment pressure provides lower temperature condensed water and a lower RWSP liquid temperature. For the purposes of ECCS pump NPSH available determinations, based on these phenomenological correlations, the US-APWR does not consider the RWSP vapor saturation pressure (based on RWSP liquid temperature) to exceed the containment pressure for any postulated DBA.

MHI considers the proposed methodology to be appropriate and conservative under all accident conditions for the following reasons:

- Anticipated Accident Conditions – As mentioned above, the containment RWSP water temperature used for the NPSH available evaluation is dependent on the mass and energy release time-history and mixing with the condensed water flow from containment sprays, which is dominated by the containment atmospheric vapor saturation pressure. As a result, there are no situations where a rapid containment depressurization can induce RWSP water boiling at any given RWSP liquid temperature, even assuming a conservative minimum containment pressure.
- Saturated Conditions – There are no heat sources in the RWSP that would change local temperature or pressure conditions. Furthermore, containment spray injection occurs at post-LOCA. Therefore, it is appropriate to assume that the containment atmospheric conditions are saturated.

The containment response analyses (Table F-2 and Table F-3 and the discussion below) illustrate the LOCA transient responses with assumptions of various conditions and bias settings. In order to estimate the differential pressure between the saturation pressure at the RWSP liquid temperature and the containment pressure, an analytical approach was performed. For the analytical cases, assumptions were intended to cover a range of results including maximum RWSP liquid temperature, minimum containment pressure, and a rapid depressurization with high RWSP liquid temperatures. In each case, bias conditions (as listed in Table F-2 and Table F-3) were assumed to achieve the objectives described above.

The LOCA transients shown in Table F-2 and Table F-3 do not account for the containment vessel leakage rates permitted in the US-APWR technical specifications. However, this leakage rate was evaluated separately and determined to have a negligible effect. In addition, the LOCA transients do not address additional heat removal from the containment air coolers since these are non-safety systems which are isolated during LOCA cases. The non-essential chilled water system provides chilled water to the cooling coils of the HVAC system inside containment during normal operation mode. The containment isolation valves of the non-essential chilled water system are designed to close upon receipt of a containment isolation signal. Therefore, containment depressurization due to heat removal from the containment HVAC system will not occur during LOCA cases.

Figure F-1 through Figure F-4 show the containment pressure and RWSP vapor pressure (saturation pressure at the RWSP liquid temperature) transients for various LOCAs with maximized RWSP liquid temperatures. Under these conditions, containment pressure transients are always above RWSP vapor pressure for all RWSP liquid temperatures.

Figure F-5 shows the containment pressure and RWSP vapor pressure (saturation pressure at the RWSP liquid temperature) transient for a LB-LOCA, with assumptions to minimize containment pressure. RWSP water temperature does not reach 204°F for this case, and thus RWSP saturation pressure is low. Also under these assumptions, containment pressure transients are always above RWSP saturation pressure for all RWSP liquid temperatures.

Figure F-6 and Figure F-7 show the transient of the containment pressure and RWSP vapor pressure (saturation pressure at the RWSP liquid temperature) for LB-LOCAs, assuming rapid depressurization by manual operator action. For this case, an additional 2 containment spray trains are assumed to be activated concurrent with either peak RWSP temperature (Figure F-6) or peak containment pressure (Figure F-7). The RWSP liquid temperature decreases with depressurization of the containment atmosphere. The containment pressure exceeds the RWSP saturation pressure after the additional spray activation.

Figure F-8 shows the distributions of the containment pressure versus RWSP saturated steam pressure at the point of minimum differential pressure between them for the various LOCAs described above. For all of the LOCA biases (high RWSP liquid temperature, low containment pressure, and rapid depressurization with high RWSP liquid temperature), it is shown that containment pressure transients are always above RWSP saturation pressure with a sufficient margin of no less than 10 psid. Also, it indicates a strong correlation between the transient behavior of the containment pressure and RWSP liquid temperature. In cases with low containment pressure, RWSP liquid temperature does not reach 204°F.

The discussions and results presented above with the biased assumptions cover the full LOCA spectrum.

In conclusion, containment pressure is at all times above the saturation pressure at the RWSP liquid temperature. Therefore, the assumption that containment pressure and RWSP vapor pressure are equal when evaluating NPSH available is appropriate and conservative for the US-APWR recirculation system.

F.2 Design Changes to Prevent High Sump Temperatures

To meet the strict wording of RG 1.82, Rev. 3, Position 1.3.1.1, design changes would be required in order either to maintain the RWSP liquid temperature below approximately 212°F or to increase fluid levels in containment post-LOCA. As shown in
Figure **3-16**, changes to the existing system design would need to provide approximately 45 ft of additional static head at high temperatures, which could not be provided by a single component re-design (e.g., strainer loss or pump redesign to change NPSH required). Therefore, significant and impracticable structural and/or system design changes would be required, such as:

- Heat Exchanger Capacity – To maintain sump water temperature below approximately

212 °F in the short-term, heat exchanger capacity would have to be increased significantly. This large increase would be impractical considering that the current systems have sufficient capacities to meet other safety requirements.

- Pump Elevation – The static head could be increased by lowering the pump installation elevation. The resulting increase in overall building height would negatively impact the seismic design.
- Water Volume – The RWSP water volume would need to be increased significantly to provide additional cooling water inventory. An increase in height would negatively affect the overall building height and seismic design.
- Pump NPSH required – A different style of pump may be designed such that it meets the NPSH available flow, pressure and reliability requirements. However this may entail a redesign and reanalysis of the ECCS system, due to the magnitude of the additional NPSH available necessary. Such a redesign may also induce additional risk and is thus not desirable

It is not reasonable or practicable to apply these design changes. Furthermore, the current design (in-containment emergency cooling water source) was chosen to maximize overall system health, which would be negatively impacted by the proposed design changes. In addition, the proposed methodology at high temperatures conservatively bounds the actual containment pressure for all accident scenarios. Therefore, MHI considers the use of the sump vapor pressure for containment pressure in the NPSH evaluation to provide adequate and reasonable assurance that the ECCS system pumps will be able to fulfill their functional requirements with respect to available NPSH.

F.3 Contribution to Plant Risk

Containment accident pressure is the pressure in containment during a postulated accident. In the evaluation of NPSH available, MHI does not credit the containment accident pressure. Instead, MHI assumes that the containment pressure is assumed to be equal to the RWSP fluid vapor temperature pressure for high sump fluid temperatures (greater than 204°F). At these high sump temperatures, this assumption does credit an increase in containment pressure above that originally present prior to a postulated LOCA.

This assumption is shown in Section 1 of this appendix to be a physically appropriate assumption (i.e., necessary to ensure that the RWSP water remains in the liquid state) and to be conservatively bounded by the actual containment accident pressure during all design basis conditions. Section 1 demonstrates that events which could reduce containment accident pressure (e.g., impaired containment integrity or operation of heat removal systems at too high a rate) would also accordingly reduce the peak RWSP fluid temperature, such that the containment accident pressure still bounds the RWSP fluid vapor pressure.

Therefore, there is no increase to plant risk from this methodology, with respect to achieving and/or reaching the peak RWSP temperature. However, a rapid depressurization of containment during this these periods of high RWSP temperature could potentially cause a loss of sufficient NPSH margin (e.g., due to flashing).

This section evaluates the contribution to plant risk from inadequate containment pressure during periods of high sump temperatures during post-LOCA operation. This evaluation was performed to address the five key principles of risk-informed decision making of RG 1.174 Rev. 1.

First, an exhaustive search of initiating events and accident scenarios that can result in high RWSP temperature was performed. Secondly, events that can cause loss of containment pressurization were identified. Finally, a quantitative risk assessment was performed to evaluate the risk from loss of sufficient NPSH margin caused by rapid containment depressurization.

Section 3.1 identifies the potential events which may lead to high RWSP temperature and the maximum time that the RWSP temperature is above 212°F, which is the saturated temperature of water under the containment pressure assuming the most severe post accident containment depressurization. Section 3.2 identifies potential events which may cause a loss of containment pressurization. Section 3.3 evaluates the contribution to plant risk from the most limiting event, a failure of containment isolation. Section 3.4 discusses monitoring programs to ensure the performance of plant equipment used in the risk assessment herein.

It should be noted that the risk assessment is performed for high temperature periods above 212°F, while the NPSH available evaluation uses 204°F to define periods of high temperature. This arises from the use of 212°F as the saturation temperature corresponding to nominal atmospheric pressure (14.7 psia), while 204°F is the saturation temperature corresponding to a conservative minimum US-APWR tech spec initial containment pressure (plus margin). This change is made for simplification and is judged to have a negligible impact within the order of accuracy of the risk assessment.

F.3.1 Identification of Events Leading to High RWSP Liquid Temperature

In the evaluation of NPSH, containment pressure is assumed to be equal to the sump (RWSP) fluid vapor temperature for high sump fluid temperatures. Therefore, containment pressure is important when the RWSP liquid temperature is above 212 °F.

The RWSP contains a minimum of 583,340 gallons of borated water with temperature equal to or below 120°F (US-APWR Technical Specification) during operation at power. Due to the large heat capacity of the RWSP fluid, causes of significant increase in RWSP liquid temperature are limited to events involving high energy release from the RCS or the main steam system inside the containment. Events that may result in RWSP liquid temperature above 212 °F are discussed below.

- **High energy release from the RCS** – LOCA events result in high energy release inside the containment which lead to an increase in RWSP liquid temperature. Table F-1 shows the periods when RWSP liquid temperature exceeds 212 °F during a LOCA, which are categorized into three divisions by break size. These break sizes are consistent with the LOCA break size categorization applied to the PRA. In 2-inch diameter breaks (SB-LOCA), the maximum RWSP liquid temperature was maintained below 212 °F under all conditions. Therefore, SB-LOCA events are excluded from events leading to high RWSP liquid temperature. For MB-LOCA and LB-LOCA events, the RWSP liquid temperature can exceed 212 °F and there is a potential for loss of significant NPSH margin if the event is followed by rapid reduction in containment pressure. The duration for which the RWSP liquid temperature exceeds 212 °F ranges from 1,500 seconds to a maximum of 86,400 seconds (24 hours) for LB-LOCA. Since LB-LOCA events provide bounding results with respect to MB-LOCA events for the durations of high RWSP liquid temperature, these values are also conservatively applied to MB-LOCA events.

  Initiation of feed and bleed operation also involves high energy release in the containment. The safety depressurization valves opened during feed and bleed operation are 4 inches in diameter, and therefore, RWSP liquid temperature increase after feed and bleed would be similar to MB-LOCA events. Feed and bleed is considered in the events leading to an increase in RWSP liquid temperature above 212 °F.

Stuck-open safety valves result in loss of coolant accidents that may or may not cause automatic or manual actuation of safety injection systems and spray systems. According to the NUREG reports (NUREG/CR-5750, NUREG/CR-6820) there has have been two pressurizer safety valve stuck stuck-open events experienced in the US nuclear industry between the period of 1988 and 2002. The leak rates for the two events were 200 gpm and 25 gpm during shutdown. Taking into consideration of the relatively low leak rates caused by the stuck-open pressurizer safety valve stuck open events compared to those of M-LOCA and L-LOCA events, stuck-open safety valve stuck open events are categorized as SB-LOCA events in the PRA. Simultaneous stuck-open safety valve stuck open events involving multiple safety valve failures that result in a consequence energy release equivalent to MB-LOCA or LB-LOCA was were screened out from evaluation. This is because there has been no industrial experience of multiple safety valve stuck -open events. , and eEven if such events were to occur, it is likely that leak rates from each the valves would not be significant, so that the consequence will would not be as severe as MB-LOCA or LB-LOCA.

- **High energy release from the main steam system** – The bounding event of this type in terms of energy release in containment is a main steam line double-ended guillotine break (DEGB). It has been confirmed by thermal hydraulic analysis performed for section 6.2.1.4 of DCD Chapter 6 that the RWSP liquid temperature will not exceed 212 °F if the intact steam lines can be isolated from the faulted line. Should failures in both the main steam isolation valve and main steam check valves occur following a main steam line break, the main steam from multiple SGs will be released from the faulted steam line. In this case, the consequence can be worse than the Section 6.2.1.4 analysis. However, since the main steam isolation valves (MSIVs) and the main steam check valve are independent, the probability of both valves failing is low. Given that the frequency of steam line break in containment is 1.0×10^{-3} /RY, and failure probabilities of MSIV and main steam line check valves respectively 1.2×10^{-3} and 1.0×10^{-4}, the frequency of a steam line break followed by failures of both the main steam line check valve and the MSIV of the faulted SG is 1.2×10^{-10} /RY. The frequency of main steam release from multiple SGs would be lower than this value since the closure of MSIVs of the intact main steam lines can also prevent main steam release from intact SGs. Thus, the frequency of a main steam line break event resulting in energy release from multiple SGs in the containment is estimated to be more than three orders of magnitudes lower than the core damage frequency of the plant, and the contribution to plant risk is negligible.

Within the potential sources for high RWSP liquid, MB-LOCA event, LB-LOCA event and initiation of feed and bleed are considered to be the limiting events that may result in the loss of sufficient NPSH, when followed by rapid containment depressurization.

F.3.2 Sources for Reduction in Containment Pressure

Potential sources for a reduction in containment pressure are listed below, along with the bounding event sequence from the existing PRA and / or other justification for screening out the source from additional consideration. Sources which require further evaluation are detailed in Section 3.3.

- CV Structural Failure – Even for the worst case large LOCA event, the peak containment pressure is 59.5 psig according to the design-basis accident analysis (See DCD Table 6.2.1-1). The US-APWR design applies a pre-stressed concrete containment vessel (PCCV) and its ultimate pressure is 201 psig (DCD Section 19.2.4). This provides sufficient margin against pressure increase after LOCA events. Therefore, a CV structural failure that will result in rapid depressurization is unlikely to occur, and the contribution to plant risk is negligible.
- Operator Error (Isolation) – There is no operational procedures that allow operators to open isolation valves that will result in containment bypass during an accident. The only possibility that the operator will allow containment bypass is by operator error. In a condition where a LOCA has occurred, the containment isolation signal is initiated and the alarm annunciated. A feature of the US-APWR design to prevent human errors that could lead to inadvertent opening of containment isolation valves is that the operators cannot manually control the containment isolation valves without resetting the containment isolation signal. It is unlikely that the operator will inadvertently open any containment isolation valves in such a situation. Therefore, an operator error that opens an isolation valve and results in significant depressurization of the containment is unlikely to occur, and the contribution to plant risk is negligible.

- Operator Error (Heat Removal) – There is a possibility for the containment to experience depressurization through the actuation of additional containment spray trains or the HVAC system inside containment. Actuation of additional containment spray trains is analyzed explicitly in Section 1. As shown in Figure F-6 and Figure F-7, containment pressure is maintained above the RWSP saturation pressure after actuation of additional containment spray trains, which implies that this event will not lead to loss of sufficient NPSH available. The event of operators inadvertently restarting HVAC systems in the containment is considered unlikely to occur, since the HVAC system inside containment is isolated by the containment isolation signal. The operator would not be able to initiate containment cooling by the HVAC system unless containment isolation signal is reset. Therefore, the contribution to plant risk is negligible.
- CV Isolation Valve Failure – Failure of a containment isolation valve was considered to be the most likely event which may result in a rapid depressurization due to the number of valves and penetration lines. This event is discussed in detail in Section 3.3 below.

Within the potential sources for a reduction in containment pressure, failure of a containment isolation valve is considered to be the most likely event that may result in the loss of sufficient NPSH available. The contribution of this event to plant risk is discussed in Section 3.3.

Pre-existing containment isolation failures are excluded since such failure will not cause rapid containment depressurization. Such failures will relax containment pressurization throughout the accident rather than causing rapid depressurization after reaching an elevated RWSP temperature and, therefore, will not result in significant loss in NPSH available margin.

F.3.3 Result of Probabilistic Assessment for the Loss of the Containment Isolation

Probabilistic analysis of significant depressurization due to loss of containment isolation during the period of high RWSP liquid was performed. The frequency of the RWSP liquid temperature being above 212 °F was estimated from frequencies of M-LOCA events, L-LOCA events, and accident sequences involving feed and bleed operation. LOCA events and feed and bleed operation resulting from fire and flood accident scenarios have been considered as well as internal events. In the evaluation of fire and flood scenarios, fire and flood that potentially impact the availability of containment isolation valves have been assessed. Quantification of risk from inadequate containment pressure was performed by first estimating the frequency of accident sequences that leads to high RWSP temperature using the PRA results of for internal and external events PRA results, and then multiplying the probability of having failure in the penetration line or isolation valve that leads to containment bypass. Assumptions applied in the risk quantification are the following:

- The probability of an occurrence of rapid containment depressurization was evaluated assuming the only likely depressurization event from Section 3.2, the failure of a containment isolation valve.
- One hundred (100) lines were conservatively assumed for the number of penetrations, and 24 hours was conservatively assumed as the period of high RWSP liquid temperature. Although failures of isolation valves (depending on the size) do not all result in rapid containment depressurization, it was conservatively assumed that failures in any of the penetration lines during the 24 hour period will cause a loss of sufficient NPSH margin.
- For fire scenarios involving leading to feed and bleed operation, the PRA conservatively assumed that one train of the isolation valves are inoperable, since there is a high probability that one train of the isolation valve is affected by the fire event. Similarly, for flood events accompanying LOCA, one train of the isolation valve is conservatively assumed to be unavailable since there is a high chance the isolation valve is affected by the flood.

With these conservative assumptions, the total frequency of the sequences was determined to be two orders of magnitude lower than the core damage frequency described in Chapter 19 of the DCD. Dominant accident scenarios that lead to loss of sufficient NPSH are the followings:

- Fire events followed by failures in the emergency feed water system (EFWS), and containment isolation:

  The plant is tripped after a fire event. Random and fire induced failures occur in the EFWS or main steam system, resulting degradation in the heat removal function from SGs. Operators initiate feed and bleed. Containment pressure and temperature increases due to the energy released from the RCS as a result of feed and bleed. One train of the containment isolation valve is inoperable due to the fire that has affected the cables. Mechanical failure occurs in the isolation valve of that intact train or any containment penetration line piping during the period where the RWSP is above 212 °F. Containment experiences a rapid depressurization and safety injection pumps and containment spray pumps become inoperable due to loss in NPSH margin. Feed and bleed is no longer operable and the eventually the core is damaged.

- MB-LOCA events followed by failure of containment isolation:

  A medium break LOCA event occurs and the plant is tripped. Random and fire induced failures occur in the EFWS, resulting in degradation in of the heat removal function from SGs. Operators initiate feed and bleed. Containment pressure and temperature increases due to the energy released from the RCS via the breach and the RWSP temperature exceeds 212 °F. Mechanical failure occurs in the isolation valves or any containment penetration line piping and the containment experiences a rapid depressurization. Safety injection pumps and containment spray pumps become inoperable due to loss in NPSH margin. Safety injection functions are no longer operable and the plant fails to mitigate the MB-LOCA event.

The evaluated core damage risk from inadequate containment pressure is two orders of magnitude lower than the core damage frequency described in Chapter 19 of the DCD, even with the conservative assumptions as mentioned above. The contribution to plant risk from inadequate containment pressure is therefore considered negligible.

F.3.4 Monitoring Programs

DCD, Chapter 16, Technical Specification, Section 5.5.8 Inservice Testing Program commits the US-APWR to the use of the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and to its applicable Addenda for the testing of ASME Code Class 1, 2, and 3 Components.

The ECCS pumps are ASME Code Class pumps and will therefore be tested in accord with the ASME OM Code. ASME OM ISTP ISTB requires that Group A pumps be pre-service tested to confirm design basis capability. In accordance with the OM Code, once in-service they will be tested quarterly to detect degradation and bi-annually (Comprehensive Pump Test) to confirm the ability to meet design basis requirements. The use of the ASME OM Code is required as per 10CFR50.55a.

The containment isolation valves, including interlocks, are ASME Code Class valves and will therefore be tested in accord with the ASME OM Code Section ISTC. Containment isolation valves are considered Category A valves and will be tested and monitored in accord with program requirements. The US-APWR is committed to 10CFR50, Appendix J. and the COL Applicant is committed to a monitoring and testing program as per DCD Chapter 6, COL 6.2(8). The use of the ASME OM Code and Appendix J is required as per 10CFR50.55a. Additional details regarding the containment isolation valve monitoring and testing can be found in DCD Chapter 6 Engineered Safety Features, Sections 6.2.4 and 6.2.6; Chapter 14 Verification Programs; and Chapter 16 Technical Specifications, Section 5.5.16.

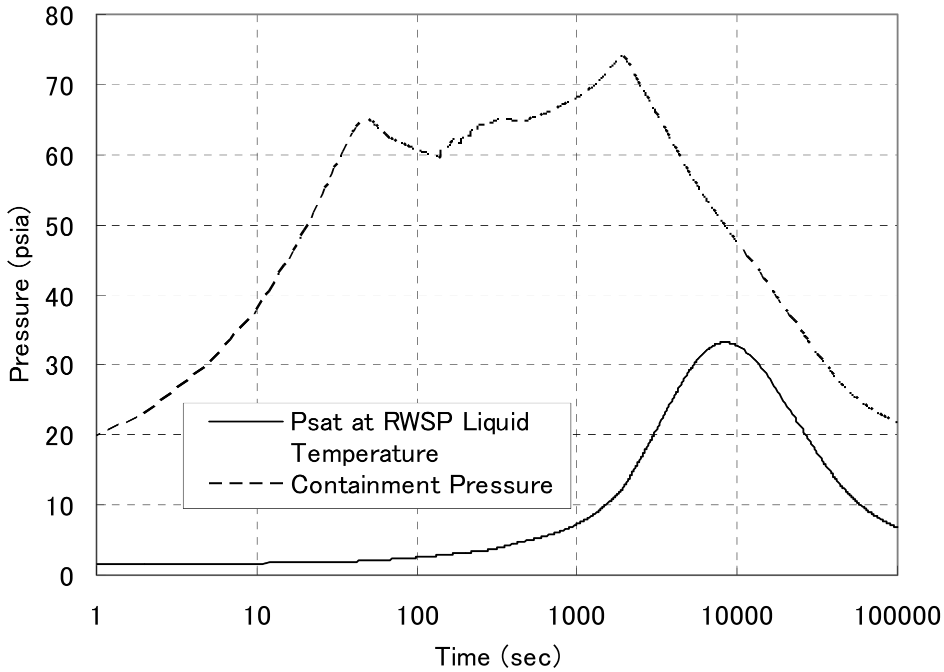


Figure F-1 LB-LOCA – Pump Suction (Case 1)

Figure F-2 LB-LOCA – Hot Leg (Case 2)

Figure F-3 MB-LOCA – 1ft2 Break (Case 3)

Figure F-4 MB-LOCA – Small Break (Case 4)

Figure F-5 LB-LOCA – Cold Leg (Case 5)

Figure F-6 LB-LOCA – Pump Suction (Case 6)

Figure F-7 LB-LOCA – Pump Suction (Case 7)

Figure F-8 Correlation between Containment Pressure and RWSP Saturated Steam Pressure (At Minimum Differential Pressure)

Table F-1 Duration of RWSP Temperature > 212 °F after LOCA

LOCA Type	Break Diameter	Duration of RWSP Liquid Temp. above 212 °F after LOCA
LB-LOCA	Over 8 inch	1500 sec to 1 day (86400 sec)
MB-LOCA	2 inch to 8 inch	1500 sec to 1 day (86400 sec)
SB-LOCA	Under 2 inch	N/A

Table F-2 LOCA Cases and Assumptions for Figure F-1 through Figure F-4

Case		1	2	3	4
Figure		Figure F-1	Figure F-2	Figure F-3	Figure F-4
Break Type		LB-LOCA	LB-LOCA	MB-LOCA	MB-LOCA
Break Condition		Pump Suction Split Break $3ft^2$	Hot Leg Double Ended Break Cd=1.0	Cold Leg $1ft^2$	Cold Leg 4inch Diameter
Offsite Power					
Available ESFs					
Containment Spray Initiation					
Containment Spray Flowrate					
Heat Removal System Capability					
Intended Biases					
Mass and Energy Calculation Code					
Assumptions for Conservatism in Mass and Energy Calculation					
Containment Response Calculation Code					
Containment Volume Modeling					
Assumptions for Conservatism On Containment Response Calculation (Objectives)	Containment Free Volume				
	Estimated Heat Sink Amount				
	Heat Transfer Coefficient for Heat Sink				
	Initial RWSP Liquid Volume				
Peak RWSP Liquid Temperature, ^0F					
Containment Pressure at Peak RWSP Liquid Temperature, psia (psig)					

Table F-3 LOCA Cases and Assumptions for Figure F-5 through Figure F-7

Case (Base Case)		5	6 (1)	7 (1)
Figure		Figure F-5	Figure F-6	Figure F-7
Break Type		LB-LOCA	LB-LOCA	LB-LOCA
Break Condition		Cold Leg Reference Case of WCOBRA/TRAC	Pump Suction Split Break $3ft^2$	Pump Suction Split Break $3ft^2$
Offsite Power				
Available ESFs				
Containment Spray Initiation				
Containment Spray Flowrate				
Heat Removal System Capability				
Intended Biases				
Mass and Energy Calculation Code				
Assumptions for Conservatism in Mass and Energy Calculation				
Containment Response Calculation Code				
Containment Volume Modeling				
Assumptions for Conservatism on Containment Response Calculation (Objectives)	Containment Free Volume			
	Estimated Heat Sink Amount			
	Heat Transfer Coefficient for Heat Sink			
	Initial RWSP Liquid Volume			
Peak RWSP Liquid Temperature, ^0F				
Containment Pressure at Peak RWSP Liquid Temperature, psia (psig)				