MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

March 15, 2010

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-10073

Subject: MHI's Response to US-APWR DCD RAI No. 524-4020 Rev. 1

References: 1) "Request for Additional Information No. 524-4020 Revision 1, SRP Section: 12.03-12.04 – Radiation Protection Design Features, Application Section: 12.3," dated January 26, 2010

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Response to Request for Additional Information No.524-4020 Revision 1".

Enclosed are the responses to four RAIs contained within Reference 1.

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

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

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

Sincerely,

Y. Ogata

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

D081
NRO

Enclosure:

1. Affidavit of Yoshiki Ogata

2. Response to Request for Additional Information No.524-4020 Revision 1 (Proprietary version)

3. Response to Request for Additional Information No.524-4020 Revision 1 (Non-proprietary version)

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

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Enclosure 1

Docket No.52-021
MHI Ref: UAP-HF-10073

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

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

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

2. In accordance with my responsibilities, I have reviewed the enclosed "Response to Request for Additional Information No.524-4020 Rev.1" and have determined that the document contains proprietary information that should be withheld from public disclosure.

3. The information in the document identified as proprietary by MHI has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that the information described in the attachment of the response to RAI No.524 question No. 12.03-12.04-34 involves MHI's Know-how, such as surface areas exposed to reactor coolant and actual cobalt content for components.

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

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

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

Executed on this 12th day of March, 2010.

Y. Ogata

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

Enclosure 3

UAP-HF-10073
Docket No. 52-021

Response to Request for Additional Information
No. 524-4020 Revision 1

March, 2010
(Non Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/12/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 524-4020 REVISION 1**
SRP SECTION:	**12.03-12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**1/26/2010**

QUESTION NO.: 12.03-12.04-33

RAI 428-2910, Question 12.03-12.04-22 (a supplemental question derived from RAI 171-1858, Question 12.03-12.04-9 and RAI 172-1864, Question 12.03-12.04-10), requested additional information about the radiation protection features associated with the additional penetrations in shield walls located in some of the piping areas depicted in Figure 12.3-1 included in the response to RAI 172-1864. In their response, the applicant noted that some resin transfer lines ran through areas that were contiguous with areas containing valves. The piping rooms were assigned Radiation Zone IX < 500 rad/h. However, confirmatory calculations performed by the NRC staff using some of the isotopic concentrations provided in FSAR Tier 2 Table 12.2-17 indicate that dose rates could exceed 500 rad/h at one meter from a 2" pipe 90 degree elbow containing resin, even after a year of decay in the mixed bed vessel. The NRC staff requires additional information to be able to confirm the applicant's conclusion that dose rates in the piping areas will remain < 500 rad/h during resin transfers.

The applicant should provide information regarding the analytical code used to evaluate the dose rates in areas traversed by resin lines, the assumptions and input parameters used to determine the area dose rates due to pipes containing resin described in the response to RAI 428-2910.

ANSWER:

The reviewer indicated that the NRC used activity information provided in DCD Tier 2 Table 12.2-17 to perform calculations to confirm the assignment of Radiation Zone IX to the piping rooms containing resin transfer lines. However, MHI did not use the spent resin from the CVCS mixed bed demineralizers as the source term when determining the Radiation Zone for the piping rooms containing resin transfer lines. Instead, the Spent Resin Storage Tank (SRST) activity from DCD Tier 2 Table 12.2-48 is used as the source term for determining the applicable Radiation Zone for the piping area. (NOTE: The spent resin from the mixed bed demineralizer was used as the source to determine the thickness of the shielding wall for the piping area.) The reasons for using the SRST resin instead of the mixed bed demineralizers resin for the dose assessment are that the frequency of spent resin transport from the mixed bed demineralizer is low, the duration of transport is limited; and the piping area has a locked entrance to strictly control access during normal operation and prohibit access during the transport of spent resins. This is consistent with

DCD Subsection 12.3.1.2.1.1 which states that plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy.

The radiation zone in the piping area including the resin transport piping was determined based on a MicroShield dose rate calculation using the SRST resin as the source of the radioactivity of the piping. The calculated dose rate at 1 m from the pipe surface is 185 rad/h, which meets the dose conditions of Radiation Zone IX. The specific geometric and material-related calculation conditions used in MicroShield are provided in the attachment.

As described in MHI's response to RAI 428-2910, Question 12.03-12.04-22 dated September 28, 2009, during normal operation there are no components or valves installed in this piping room that require periodic personnel access for maintenance or operation. Furthermore, there is no equipment that requires monitoring for ESF leakage in this piping room, because all the ESF equipment is installed in the Reactor Building.

Based on the above discussion, MHI concludes that the use of the SRST resin to determine the applicable Radiation Zone for the piping area containing resin transport piping is acceptable with respect to maintaining occupational exposures ALARA without being excessively conservative.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

Attachment

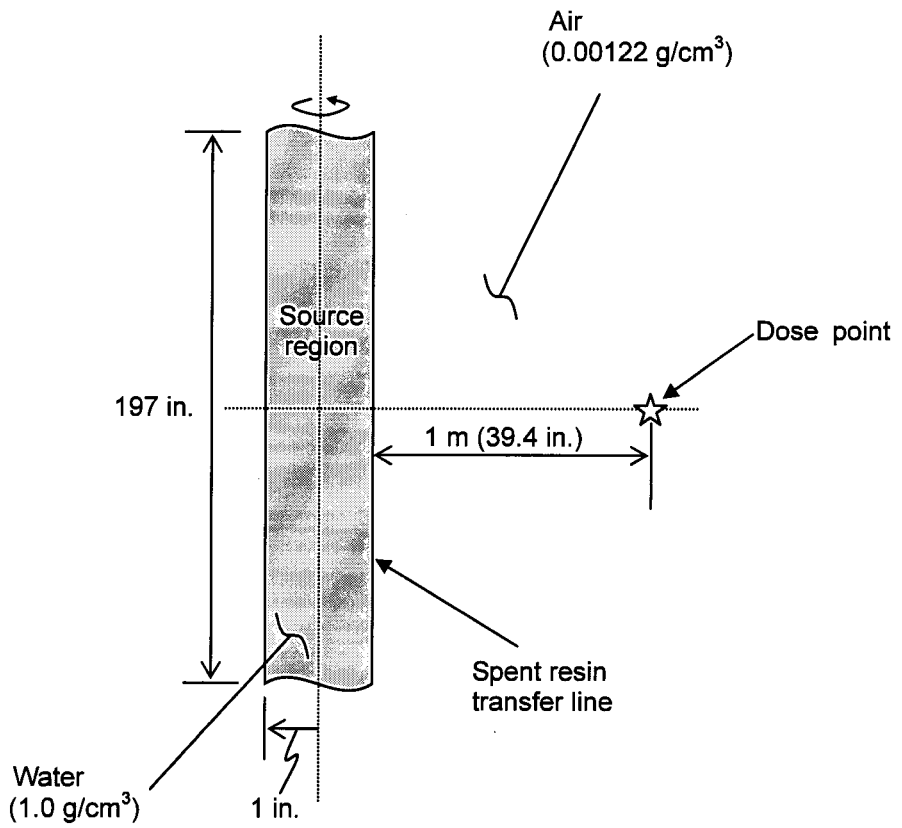

Figure 1 Dose calculation model of spent resin transfer line for MicroShield code
(Source Activity: Values from Table 12.2-48 of the US-APWR DCD Rev.2)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/12/2010

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APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**1/26/2010**

QUESTION NO.: 12.03-12.04-34

RAI 428-2910, Question 12.02-24 (a supplemental question derived from RAI 147-1850, Questions 12.03-12.04-4), requested additional information about specifications for allowable cobalt impurity levels. The applicant's response to this question indicated that the MHI specifications are compliant with the recommendations in industry guidance documents (EPRI TR-1003390 "Radiation Field Control Manual"). However, while EPRI TR-1003390 notes that cobalt impurity levels should be less than 500 ppm in stainless steels and less that 200 ppm in Inconels, Table 12.3-7 contains the following cobalt content limits which exceed these values:

- Inconel and stainless steel components in fuel assemblies 0.05 w% (500 ppm) vs. 200 ppm for Inconels
- Upper core plate, Upper/lower core support plate, lower core barrel 0.10 w% (1000 pp m)
- Main Coolant Piping, casings and internals of Rector Coolant Pumps and Reactor Internals other than listed above 0.20 w% (2000 ppm)

The applicant should revise and update the USAPWR DCD Tier 2 Sections 12.2 to provide cobalt content limits consistent with industry recommendations, or provide the specific alternative approaches used and the associated justification.

ANSWER:

MHI understands that the cobalt content of some components exceed the EPRI requirements, as commented in NRC's question.
However, the surface areas exposed to reactor coolant are only about 0.9% for inconel components and 2.5% for stainless steel components for the total reactor coolant system (RCS) wetted surface areas. MHI has specified a maximum cobalt content of 0.016 mass % for Steam Generator (SG) Inconel tubing, which is lower than the EPRI requirements of maximum 0.02 % (200 ppm), to effectively reduce the total cobalt release contribution since the surface area of SG tubing is about 66.0% of the total RCS wetted surface areas.
The comparison of cobalt release amount between the EPRI requirements and the US-APWR specification shows that the total cobalt release amount of the US-APWR specification is less than that of the EPRI requirement.
Furthermore, additional cobalt reduction can be expected since the actual cobalt content of some domestic and exported components are within the EPRI requirements.(See Attachement-1, Comparison of Cobalt Release Amount from RCS Major Components)
Based on the above, Table 12.3-7 meets ALARA requirement.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

Attachmet-1 Comparison of Cobalt Release Amount from RCS Major Components

(PROP)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/12/2010

US-APWR Design Certification

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RAI NO.:	**NO. 524-4020 REVISION 1**
SRP SECTION:	**12.03-12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**1/26/2010**

QUESTION NO.: 12.03-12.04-35

RAI 262-1972, Question 12.03-12.03-17, Item 8816-Q6 requested additional information about the radiation protection features associated with an inadvertent rapid drain down of the refueling cavity.

In the response to Item 8816-Q6, the applicant indicated that:

- There are a number of fill and drain lines in the refueling cavity administratively locked closed prior to fuel movement. However, a review of US-APWR FSAR Tier 2 Chapters 5, 9 and 16, failed to show any requirement to have fill/drain lines for the refueling cavity locked shut, or any requirement to have two barriers preventing inadvertent drain down through these flow paths.
- A water level alarm would detect a decrease in the refueling cavity water level. However, US-APWR FSAR Tier 2 Chapters 5, 7 and 9 do not mention a Refueling Cavity Level alarm. This implies a reliance on the SFP level alarm. Since there are no physical constraints preventing closing the SFP weir gate, the SFP level alarm may not be able to monitor the Refueling Cavity level under all conditions where fuel is out of the reactor vessel in the refueling cavity. In addition, there are no requirements in Technical Specifications 3.7.12 "Fuel Storage Pit Water Level" or 3.9.7 "Refueling Operations Water Level" to have the SFP Weir Gate open while fuel is out of the reactor vessel in the refueling cavity, or for the SFP Level alarm to be operable.
- If a drain down event were to occur, then water would be added to the RC system to maintain level. However, insufficient information is available to demonstrate that sufficient water would be available for shielding and cooling during a rapid cavity drain down event.

Background

GDC 61 requires that the fuel storage and transfer system be designed to ensure adequate safety and shielding, as well as be designed to prevent the release of radioactive material during normal and accident conditions.

Generic Safety Issue 137 (GSI-137) "Refueling Cavity Seal Failure" was initiated to consider a Reactor Cavity Seal Ring failure as an initiating event for a Spent Fuel Pool accident sequence (see NRC Bulletin 84-03). GSI-137 noted the following possible consequences to a reactor cavity

seal ring failure;

(1) high radiation levels in the containment due to uncovering of spent fuel in transfer;
(2) radioactive material release in the containment building due to rupture of fuel pins (by self-heating after uncovering);
(3) high radiation levels in the spent fuel building due to uncovering of stored spent fuel; and
(4) radioactive material release outside the containment building due to rupture of fuel pins in the storage pool.

While the installation of permanent seal rings between the reactor vessel and the refueling cavity wall, and the installation of cofferdams between the refueling cavity and the spent fuel pool precluded draining of the SFP, it did not specifically address exposure of fuel bundles in the refueling cavity.

In their response to NRC Bulletin 84-03, "Refueling Cavity Water Seal," a number of licensees indicated that if a large seal leak occurred, the refueling cavity could drain rapidly and therefore, the procedures and makeup capability for refilling the refueling cavity would be insufficient to prevent drainage. In anticipation of such an event, the licensees developed an abnormal procedure for safely storing a fuel assembly were one positioned above the vessel flange level during a loss of the refueling cavity water.

Information Notice No. 92-25 "Potential Weakness in Licensee Procedures for a Loss of the Refueling Cavity Water", noted an event where improper movement of the reactor lower internals package resulted in damage to cavity components. The Information Notice noted that under slightly different circumstances, the event could have resulted in a rapid drainage of the Refueling Cavity. Information Notice 84-93 "Potential Loss of Water from the Refueling Cavity" noted that a variety of events might result in loss of water from the refueling cavity. INPO SOER 85-1 "Reactor Cavity Seal Failure" and NUREG 1449 "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States" noted a number of events, besides Refueling Cavity (RC) seal failure that can result in cavity drain down events. Industry experience has shown that valve positioning errors, and problems with freeze seals failures have resulted in significant events. Information Notice 87-13 noted that in the event of a rapid refueling cavity drain down event, high dose rates might result from loss of shielding from around irradiated core components, other than fuel, stored in the refueling cavity.

Therefore, in order to adequately address operating experience considerations and in accordance with the requirement specified by 10 CFR 52.47(a)(22), the following additional information is required:

1. Identify any locations in the Refueling Cavity, where more than one fuel bundle, including any in transit, may be out of the reactor vessel in the Refueling Cavity.
2. Provide information on each location in the refueling cavity, where a licensee could safely store a spent fuel assembly should the reactor cavity undergo inadvertent rapid drain down while one or more fuel bundles were out of the reactor vessel in the refueling cavity.
3. For the identified safe storage areas, where the water level at the lowest point following the drain down event is less than 10 feet above the fuel bundles that are out of the reactor vessel, provide the estimated dose rates, and the associated methods, models and assumptions for determining the dose rates.
4. If the volume of water in the safe storage area is less than the amount to needed to ensure complete coverage of the fuel bundle for the time allotted for ensuring containment closure, then justify the continued use of the Fuel Drop Accident Analysis methodology described in RG 1.183 Appendix B.
5. describe and provide the estimated dose rates, from any other irradiated components, that may be exposed during an inadvertent refueling cavity drain down event

ANSWER:

1. The refueling cavity has two fuel containment racks that are capable of storing three fuel bundles each, meaning that a total of six fuel bundles can be temporarily stored. The locations of the containment racks are shown in Figure 1. These fuel containment racks are installed for the following purposes:
   1) To temporarily store dummy fuel to be used during fuel loading
   2) To temporarily store fuel
      - As a standby for returning the fuel handling system from the fuel handling area; and
      - For use if there is any problem with the fuel handling system when transferring fuel with the refueling machine.

   The fuel handling procedure allows fuel to be loaded and unloaded without using a containment rack.

2. The response to RAI No. 507-3993 Revision 2 Question No.: 09.01.04-16 [item c.1st bullet and item d] confirmed that no paths, other than those described below, exist that are capable of inadvertently draining the refueling cavity.

   - Permanent Cavity Seal (PCS)
     The polar crane and the refueling machine are designed as single failure proof cranes. However, should a load suspended from these cranes, such as a fuel assembly, be dropped, damage to the PCS is prevented by the guard plate installed above the PCS before filling the refueling cavity with water. The PCS is visually inspected before each filling of the refueling cavity. Therefore, the possibility of a leak is minimized. As noted above, the design of the US APWR PCS eliminates the kind of inflatable seal failures experienced in the past. There is the possibility that a very small crack could form in the cavity seal weld and be undetected upon examination. Based on the size of such a crack to escape detection, a small (< 1 gpm) leak is considered.
   - Fuel transfer tube
     The fuel transfer tube is used at relatively low temperatures and pressures. The transfer tube is rinsed thoroughly with demineralized water after the draining of the refueling cavity. There is the possibility that a very small crack could form and be undetected upon examination. Based on the size of such a crack to escape detection, a small (< 1 gpm) leak is considered.
   - Cask pit gate, Fuel inspection pit gate
     The gates are made of stainless steel. Although deterioration of the seal would be detected by periodical inspection and the seal is changed regularly, it is assumed that an undetected aging deterioration of a seal causes leakage of 1 gpm, as the maximum leak rate.
   - Cavity drain valve
     An undetected leak from a cavity drain valve is considered to be at a maximum leak of 1 gpm.

   Thus, if the water level in the refueling cavity decreases due to any of the events described above, a gradual decrease in the water level (1 gpm or less) could potentially occur. Nevertheless, it is possible to maintain the water level above low water level for a certain time period because a refueling water recirculation pump can be used to feed boric acid water from the RWSP or RWSAT into the cavity at a rate of 200 gpm. Since the makeup capability is much greater than the estimated leak rate, the water level in the refueling cavity can easily be maintained. During the period of time of using borated make-up water from the RWSP or RWSAT, it is possible to safely store the fuels that were in transition in either the SFP or to return them to the RV. As described in the response to item 1 above, the containment racks are not utilized during a refueling cavity drain down event. The safe storage areas during a drain down event are the SFP and the RV.

3. As mentioned in Item 2 above, it is possible to maintain the water level above L.W.L for a certain time period by feeding boric acid water from the RWSP or RWSAT if the water level in the refueling cavity begins to decrease. During this period of time, it is possible to safely store the fuels that were in transition in either the SFP or to return them to the RV. Thus, during fuel transfer, the fuel bundles will remain 10 feet or more below the water level.

4. RG 1.183 Appendix B requires the fuel bundles to be kept 23 feet (about 7 m) or more below the water level in the fuel drop accident analysis. As mentioned in Item 2 above, it is possible to maintain the water level above L.W.L for a certain time period by feeding boric acid water from the RWSP or RWSAT in the event that the water level in the refueling cavity begins to decrease. During this period of time, it is possible to safely store the fuels that were in transition in either the SFP or to return them to the RV. Thus, the volume of water in the safe storage area (in the SFP or RV) will not deviate from this requirement (water depth of 23 feet or more) during fuel transfer.

5. As mentioned above, if the water level in the refueling cavity decreases, only a gradual decrease in the water level occurs. Moreover, the water level in the refueling cavity is maintained for a certain time period of time by feeding boric acid water from the RWSP or RWSAT into the cavity. During this period of time, other activated materials (upper internal structures) are transferred to the RV, so that an adequate water level is maintained without causing any safety problems (in terms of shielding). Additionally, a water shield thickness of 10 feet or more is maintained while lifting fuel and safely storing spent fuel in the SFP.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

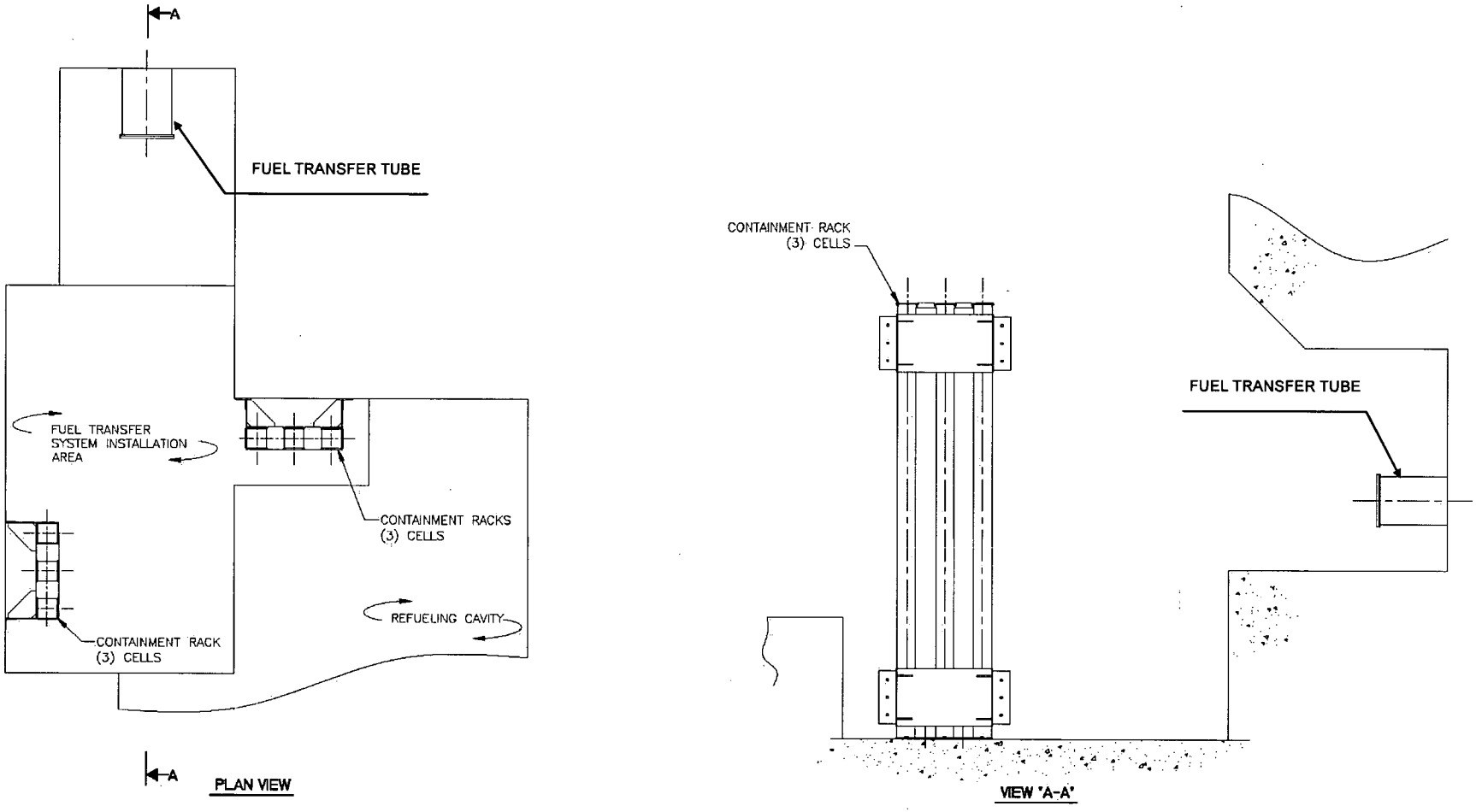

Figure1 Locations of the Containment Racks

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/12/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 524-4020 REVISION 1**
SRP SECTION:	**12.03 – 12.04 Radiation Protection Design Features**
APPLICATION SECTION: 12.3	
DATE OF RAI ISSUE:	**1/26/2010**

QUESTION NO. : 12.03-12.04-36

10 CFR 20.1101(b), 10 CFR 20.1406(b) require licensees to describe design feature to maintain Occupational Radiation Exposure(ORE) ALARA, reduce contamination of the facility, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. 10 CFR 52.47 requires applications to include information describing how operating experience has been incorporated into the design. SRP Section 12.3-4 Acceptance Criteria and Regulatory Guides 8.8 and 4.21, provide guidance for meeting the requirements of 10 CFR 20.1101 and 10 CFR 20.1406 while Regulatory Guide 1.206 sections C.I.5.4 and C.I.9.3 note that applicants should discuss system reliability considerations.

US-APWR Tier 2 FSAR section 12.3.1.1 "Plant Design Features for As Low As Reasonably Achievable" discusses some design features intended to reduce radiation exposure. In addition, a number of these design features can reduce corrosion, and improve component availability so that system and unit reliability is improved. A number of industry standard documents, notably those provided by the Electric Power Research Institute (EPRI) describe changes to previous design practices shown to be erroneous or which represent improvements over previous practices. A number of the design features described in US-APWR FSAR Tier 2 Sections 5, 9.3 and 12.3.1, intended to improve component reliability or reduce radiation exposure, are inconsistent with current operating experience or are incomplete. Examples include:

Some of the specifications provided for valves are not consistent with current industry recommendations for valve reliability or leakage reduction and in some cases may be counter productive.

§ Metallic bellows and diaphragms produce a hermetic seal that results in the lowest achievable leak rates, but they are limited to low-stroke length applications. Industry experience has shown that these bellows will crack and leak when installed on valves subject to frequent movement, such as fine control valves (e.g. Pressurizer spray valves).
§ Fully engineered packing sets, valve stem material and finish specifications, and the use of Live Load packing on critical valves, and valves subject to thermal tapering, are methods that improve valve reliability and reduce leakage.
§ Industry documents note that some important design considerations for check valve reliability are:

- proper sizing to reduce flutter related wear during the most frequent use conditions rather than maximum flow conditions,

- reducing turbulence prior to the check valve,
- the proper identification of the severity of the valve duty (i.e. highpressure drop).
- More frequent inspections for some types of check valves that are subject to severe conditions.

Industry experience has shown that some of the specifications provided for pumps are not consistent with current industry recommendations for pump reliability or leakage reduction and in some cases may be counter productive.

§ Material selection and finishing specifications, such as electropolishing followed by the application of stabilized chrome coating, effectively reduce material deposition and increase pump efficiency.
§ Elimination of cobalt containing Hard Facing alloys in pumps such as Reactor Coolant Pumps have reduced the accumulation of activated corrosion products at some plants.
§ Some plants have improved pump seal reliability by improving pump specifications:

- Specifying the use of seal design features that enhance sealing surface lubrication improves cooling and reduces leaks and failures.
- The use of Seal cartridges, which are pre-assembled mechanical face seal assemblies that contain all of the essential components simplify maintenance and eliminate installation related failures.
- Initial pump specification should include the use of seal cartridge packages, because retrofitting mechanical seal packages into pumps designed to accept conventional packing, limits the fluid circulation around the seal, and results in frequent leaks and seal maintenance.

Some of the specifications provided for piping are not consistent with current industry recommendations for reducing facility contamination and dose rates.
Some plants have additional material specifications related to the use of O-ring type pipe caps and O-ring face seal fittings, which allow easy access by operators and eliminate subsequent leaks. Seals and fittings, that develop a positive seal with minimal effort and are not subject to wear with repeated assembly.

§ A number of plants have installed fitting and connections to facilitate hydrolasing of sections of system piping that are especially prone to radioactive material deposition, like portions of the RHR, CVCS let down, CVCS Spray and Safety Injection.
§ Preconditioning of surfaces by means like electro-polishing and application of stabilized chrome pre-coatings can significantly reduce material deposition in specific sections of piping, such as parts of the PWR spray lines and sections of RCS Letdown lines, that accumulate significant deposits of radioactive material. These design features are particularly cost effective for when used for local applications.

Industry experience has shown that sludge and the liquid inputs to liquid and solids collection tanks contain chemical impurities that can attack tank welds and wall thickness, which may result in tank failure.
§ The use of isolation valves and flushing points allows for the removal of sludge in pipes and lines, thus reducing the risk of component failure.
§ The use of internal spray or sparging components and the availability for recirculation reduces the accumulation of internal sedimentation
§ Industry experience indicates that some specific valves, like Pressurizer Spray Control valves, Letdown Control valves and RCS loop interface check valves can accumulate significant deposits of radioactive material, if not polished

Please revise the US-APWR Tier 2 Sections 5, 9 and 12, to update the description of the design features provided for improving facility reliability, minimizing facility contamination and reducing personnel exposure to be consistent with current industry experience and practices, or provide the specific approaches used and the associated justification.

ANSWER:

US-APWR Tier 2 FSAR Subsection 12.3.1.1 "Plant Design Features for As Low As Reasonably Achievable" discusses some design features intended to reduce radiation exposure. US-APWR Tier 2 FSAR Subsection 12.3.1states that "The design feature recommendations, in accordance with the guidance in RG 8.8, paragraph C.2 are utilized to minimize exposures to personnel". The specific design features to reduce radiation exposure recommended by industry standards, notably those provided by EPRI are already considered in the US-APWR design and where appropriate will be included as part of the purchase specification for various components e.g., valves, pumps and piping. EPRI report TR-016780, "Advanced Light Water Reactor Utility Requirements Document" (ALWR URD) has been used as the guidance for US-APWR detailed design in addition with other industry standards. The Utility Requirements Document (URD) consists of a comprehensive set of design requirements for future light water reactors (LWRs). The requirements are grounded in proven technology of 40 years of commercial U.S. and international LWR experience. US-APWR design considers the EPRI reports requirements. Some of these EPRI reports are also referenced in the ALWR URD document e.g., NP-5479, NP-6617 and NP-3832.

The URD states that , "The NRC has issued their Safety Evaluation Report for the Evolutionary ALWR Utility Requirements Document in NUREG-1242 that endorses URD requirements or provides a definitive NRC position on areas of regulatory concern that differ from URD positions. This indicates that the stable regulatory basis for design of ALWR evolutionary plants has been successfully achieved. The URD requirements together with the SER form the basis for the certification and detailed design of the first unit."

Many of the engineers and supervisors assigned to the US-APWR design have performed similar design work or service work on other nuclear power plants. Through this experience, they have acquired knowledge of the radiation protection aspects which are applied to the US-APWR. Nuclear plant operating experience is incorporated through Nuclear Regulatory Commission (NRC) inspection and enforcement bulletins, information notices, and other documents. Knowledge of radiation protection and ALARA is applied to the US-APWR design. This allows integration of experience and ALARA considerations from plant operators and plant designers and promotes incorporation of recent operating and service experience and lessons learned.

The general mechanical equipment requirements are addressed in URD Volume II, Chapter 1: Overall Requirements, Chapter 12 and are being considered in the design of the US-APWR. The design features examples given in Question 12.03-12.04-36 are summarized in Table 1. Table 1 includes the URD related examples, the impact on the US-APWR DCD and the rationale for addressing the DCD impact.

MHI has submitted the response to the NRC RAI (No. 91-1496 Revision 1) on the design philosophy and the features in compliance with 10 CFR 20.1406, "Minimization of Contamination" and Regulatory Guide (RG) 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning." in RAI No. 91-1496 Revision 1 response.

Impact on DCD

DCD Section 12.3.1.1.1.2.H, 7th paragraph, add last sentence to read:

<u>The design of valves located in the piping carrying radioactive fluids or gases include hermetically sealed (packless) valves, live-loaded packing and graphite packing materials to reduce the potential for fluids or gases leakage</u>.

Revise DCD 12.3.1.1.1.2 E, Tanks, first paragraph, add last sentence to read:

Tanks containing radioactive particulate material shall have smooth welds and mixing, flushing and cleaning capabilities to prevent retention of the radioactive particulate material.

Revise DCD 12.3.1.1.1.2 I, Piping, add last sentence to read:

Provisions are made in radioactive systems for flushing the piping with sufficient water to reduce crud buildup. Welds are made smooth to prevent crud traps from forming.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

TABLE 1

Review of Specific Design Features Proposed By The NRC

RAI 12.03	URD	DCD Impact	Rationale
VALVES			
1. Metallic bellows and diaphragms produce a hermetic seal that results in the lowest achievable leak rates, but they are limited to low-stroke length applications. Industry experience has shown that these bellows will crack and leak when installed on valves subject to frequent movement, such as fine control valves (e.g. Pressurizer spray valves).	Volume 2, Chapter 12, 7.4.4.4.2; Valves used in hazardous gas system supply lines shall be packless and shall use either diaphragm or bellows stem seals whenever possible to prevent or minimize potential gas leakage Volume 3, Chapter 12, 3.3.5; Valves shall be of the packless metal diaphragm (PMD) type, shall have bellows sealed stems or be valves with similar leaktightness characteristics. If packed valves are used, doublestem packing leakoff connections shall be provided. The leakoff connection shall be piped to a sealed collection system or shall be supplied with an external source of clean oil-free gas at a pressure higher than the system pressure.	DCD Section 12.3.1.1.1.2.H, 7th paragraph, add last sentence to read: The design of valves located in the piping carrying radioactive fluids or gases include hermetically sealed (packless) valves, live-loaded packing and graphite packing materials to reduce the potential for fluids or gases leakage.	Per URD rationale, the gaseous radioactive waste system must be leak-tight to minimize escape of radioactive gases and potentially flammable hydrogen. PMD type valves have been successfully used and are necessary to minimize leakage. Diaphragm valves shall not be used where diaphragm leakage can result in gas leakage from the system.
2. Fully engineered packing sets, valve stem material and finish specifications, and the use of Live Load packing on critical valves, and valves subject to thermal tapering, are methods that improve valve reliability and reduce leakage.	Volume 2, Chapter 12, 12.2.2.9; In cases where valve stems are pressurized when the valves are in their normal position (open or closed), and such valves are within the reactor coolant pressure boundary before the second isolation valve or in applications where the leakage would be radioactive or hazardous (e.g. radwaste and chemistry systems), the following requirements apply: a) Valves 2 inches and under in size, except for modulating applications, shall be packless bellows or diaphragm stem seals). Packless	Refer to 1 above	

RAI 12.03	URD	DCD Impact	Rationale
	valves shall have had successful operating experience in similar service conditions; b) Valves greater than 2 inches in diameter or modulating valves shall use live loading of the packing by conical spring washers or equivalent means to maintain a compressive force on the packing. Valves with stem packing in high temperature service shall use graphite packing compatible with system fluid and valve stem materials. Guidelines provided in EPRI Report NP-5697, Valve Stem Packing Improvements, shall be applied for the implementation of graphite stem packing and live loading.		
3. Industry documents note that some important design considerations for check valve reliability are: 1. proper sizing to reduce flutter related wear during the most frequent use conditions rather than maximum flow conditions, 2. reducing turbulence prior to the check valve, 3. the proper identification of the severity of the valve duty (i.e. high pressure drop). 4. More frequent inspections for some types of check valves that are subject to severe conditions.	Volume 2, Chapter 12, 12.2.1.9; Check valves shall be applied in the ALWR in accordance with the guidelines in EPRI Report NP-5479, Application Guidelines for Check Valves in Nuclear Power Plants, as applied to a new plant design. In particular, the Plant Designer shall utilize the guidelines to assure that: a) Check valves are used only where necessary; b) The system requirements, e.g., closure time, leakage and flow rates, for each check valve are completely defined; c) The type and size of valve selected is proven in service and compatible with plant (environment) and system requirements; d) The valve is located and oriented properly in the piping system; e)Valve design features and details are selected to lengthen the service life and improve	None	Design features listed in RAI 12.03 are considered for US-APWR equipment design and procurement specification as applicable.

RAI 12.03	URD	DCD Impact	Rationale
	reliability; f)The manufacturer provides detailed drawings of each valve showing dimensions and weights of all parts, clearances between moving parts, a complete list of materials, torques for all fasteners, lubricants for fasteners before assembly, and details of all locking and retaining devices g) Consideration will be done on parts clearance, disc stability and wear relative to the actual operational flow conditions		
PUMPS			
4. Material selection and finishing specifications, such as electropolishing followed by the application of stabilized chrome coating, effectively reduce material deposition and increase pump efficiency	Volume 3, Chapter 1, 8.3.3; The Plant Designer shall consider the use of electropolished surfaces in those areas of the plant where this treatment will significantly reduce the dose to personnel during maintenance. The costs and dose improvements for at least the following areas shall be specifically evaluated for potential application of electropolishing: • Large diameter reactor system piping; • Steam generator channel heads and divider plates; and • Reactor cavity and transfer canal liners.	None.	MHI will reduce the total cobalt release amount of the US-APWR as low as that of the EPRI requirement for the purpose of ALARA. (See MHI's response to the US-APWR DCD RAI 524-4020, Question 12.03-12.04-34). MHI therefore will not include the finishing methods in this chapter because MHI recognizes that there is no reasonable finishing method to reduce exposure effectively in a new-built plant.
5. Elimination of cobalt containing Hard Facing alloys in pumps such as Reactor Coolant Pumps have reduced the accumulation of activated corrosion products at some plants.		None	US-APWR, DCD 12.3.1.1.1.1E states that, "Equipment specifications for components exposed to high temperature reactor coolant contain limitations on the cobalt content of the base

RAI 12.03	URD	DCD Impact	Rationale
			metal as given in Table 12.3-7. The use of hard facing material with cobalt content such as stellite is limited to applications where its use is necessary for reliability considerations. Nickel-based alloys in the reactor coolant system (Co-58 is produced from activation of Ni-58) are similarly used only where component reliability may be compromised by the use of other materials. The major use of nickel-based alloys in the reactor coolant system is the inconel steam generator tube."
6. Some plants have improved pump seal reliability by improving pump specifications: a. Specifying the use of seal design features that enhance sealing surface lubrication improves cooling and reduces leaks and failures. b. The use of Seal cartridges, which are pre-assembled mechanical face seal assemblies that contain all of the essential components simplify maintenance and eliminate installation related failures. c. Initial pump specification	Volume 2, Chapter 12, 12.4.2.5 Pumps in radiation areas shall be provided with long life bearings and permanent type lubrication where practical. Pump internals should be designed so that they may be readily removed for maintenance; however, if this is impractical, the pump shall be designed to facilitate removal and replacement, e.g., flanged connections and electrical quick disconnects. Pumps in nuclear service shall be provided with drain and flush connections for decontamination.	None	US-APWR DCD 12.3.1.1.1.1 B, RCPs states that; "The RCP design includes the use of an assembled cartridge seal for the No. 2 and No. 3 pump seal that reduces the time required for replacement. The RCP design also includes a spool piece to facilitate the assembly or disassembly of the seal system without the replacement of the motor from the pump." Design features listed in RAI 12.03 are considered for

RAI 12.03	URD	DCD Impact	Rationale
should include the use of seal cartridge packages, because retrofitting mechanical seal packages into pumps designed to accept conventional packing, limits the fluid circulation around the seal, and results in frequent leaks and seal maintenance.			US-APWR equipment specification.
PIPING and FITTINGS			
7. Some plants have additional material specifications related to the use of O-ring type pipe caps and O-ring face seal fittings, which allow easy access by operators and eliminate subsequent leaks. Seals and fittings, that develop a positive seal with minimal effort and are not subject to wear with repeated assembly.		None	Design features listed in RAI 12.03 are considered for US-APWR equipment design and procurement specification as applicable.
8. A number of plants have installed fitting and connections to facilitate hydrolasing of sections of system piping that are especially prone to radioactive material deposition, like portions of the RHR, CVCS let down, CVCS Spray and Safety Injection.		None	US-APWR, DCD 11.4.4.3 states that; "Slurry piping is provided with washing and flushing capability with sufficient water to flush the pipe after each use (e.g., at least two pipe volumes)". Design features listed in RAI 12.03 are considered for US-APWR piping configurations.

RAI 12.03	URD	DCD Impact	Rationale
9. Preconditioning of surfaces by means like electro-polishing and application of stabilized chrome pre-coatings can significantly reduce material deposition in specific sections of piping, such as parts of the PWR spray lines and sections of RCS Letdown lines, that accumulate significant deposits of radioactive material. These design features are particularly cost effective for when used for local applications.	Refer to 4, above	Refer to 4, above	
TANKS			
10. Industry experience has shown that sludge and the liquid inputs to liquid and solids collection tanks contain chemical impurities that can attack tank welds and wall thickness, which may result in tank failure.	Vol 2, Chapter 12, 12.6.1 All tanks containing radioactive fluids shall include provisions for the prevention of unintended retention of particulate material. One or more of the following shall be included: • Purification of radioactive fluids (filtration or ion exchange) prior to entering the tank; • Sloped or cone-shaped tank bottom; • Grinding of internal welds; • Tank flushing capability; • Agitation by recirculation flow capability; • Lancing or chemical cleaning capability.	Revise DCD 12.3.1.1.1.2 E, Tanks, first paragraph, add last sentence to read: Tanks containing radioactive particulate material shall have smooth welds and mixing, flushing and cleaning capabilities to prevent retention of the radioactive particulate material.	DCD 12.3.1.1.1.2 E, Tanks, first paragraph states that, "Whenever practicable, tanks are provided with sloped bottoms and bottom outlet Connections". This design prevents retention of radioactive particulate bottoms.
11. The use of isolation valves and flushing points allows for the removal of sludge in pipes and lines,		Revise DCD 12.3.1.1.1.2 I, Piping, add last sentence to read: Provisions are made in radioactive	US-APWR, DCD 11.4.4.3 states that; "Slurry piping is provided with washing and flushing capability with

RAI 12.03	URD	DCD Impact	Rationale
thus reducing the risk of component failure.		systems for flushing the piping with sufficient water to reduce crud buildup. Welds are made smooth to prevent crud traps from forming.	sufficient water to flush the pipe after each use (e.g., at least two pipe volumes)".
12. The use of internal spray or sparging components and the availability for recirculation reduces the accumulation of internal sedimentation	Vol 2, Chapter 12, 12.6.1 All tanks containing radioactive fluids shall include provisions for the prevention of unintended retention of particulate material. One or more of the following shall be included: • Purification of radioactive fluids (filtration or ion exchange) prior to entering the tank; • Sloped or cone-shaped tank bottom; • Grinding of internal welds; • Tank flushing capability; • Agitation by recirculation flow capability; • Lancing or chemical cleaning capability.	Refer to 10 above	Per URD rationale, where there is a choice in tank location in the system, filtration or ion exchange prior to the flow entering the receiving tanks reduces the potential for crud deposition in the tanks by retaining such material in components designed and located for crud accumulation (i.e., filters and ion exchangers). The use of sloped or cone-shaped tanks provides for more complete drainage, greatly reducing the build-up of settled material. Grinding of interior welds is a recognized method used to avoid the build-up of settled material. Agitation by high recirculation flow insures proper tank drainage and the suspension of radioactive solids. Flushing, lancing or chemical cleaning features permit reduction in maintenance exposure via the removal of at least some crud build-up.
13. Industry experience indicates that some specific		Refer to 4 above	

RAI 12.03	URD	DCD Impact	Rationale
valves, like Pressurizer Spray Control valves, Letdown Control valves and RCS loop interface check valves can accumulate significant deposits of radioactive material, if not polished			