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Your ref: Docket No. 52-006
Our ref: DCP_NRC_002564

July 20, 2009

Subject: AP1000 Response to Request for Additional Information (SRP 12)

Westinghouse is submitting a response to the NRC request for additional information (RAI) on SRP Section 12. This RAI response is submitted in support of the AP1000 Design Certification Amendment Application (Docket No. 52-006). The information included in this response is generic and is expected to apply to all COL applications referencing the AP1000 Design Certification and the AP1000 Design Certification Amendment Application.

Enclosure 1 provides the response for the following RAI(s):

RAI-SRP12.3-CHPB-02 R1

Questions or requests for additional information related to the content and preparation of this response should be directed to Westinghouse. Please send copies of such questions or requests to the prospective applicants for combined licenses referencing the AP1000 Design Certification. A representative for each applicant is included on the cc: list of this letter.

Very truly yours,

D.A. Lindgren / FOR

Robert Sisk, Manager
Licensing and Customer Interface
Regulatory Affairs and Standardization

/Enclosure

1. Response to Request for Additional Information on SRP Section 12

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NRO

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ENCLOSURE 1

Response to Request for Additional Information on SRP Section 12

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP12.3-CHPB-02
Revision: 1

Question:

In AP1000 DCA, Revision 17, Tier 2 DCD Section 12.3.2.2.4, Fuel Handling Area Shielding Design, the applicant decreased the overall minimum allowable water depth above active fuel in the reactor cavity and spent fuel pool from 9.5 feet (Revision 16) to 8.75 feet in Revision 17 during fuel movement. With the reduction in the water level, the applicant did not identify the basis of its parameters included in Section 12.3.2.2.4 or why the change occurred.

The applicant has documented these previous changes in Westinghouse TR-121, "Spent Fuel Pool Water Level and Dose," APP-GW-GLN-121, Revision 0. The applicant also previously responded to requests for additional information related to TR-121 and described assumptions used in its calculations for exposure of workers adjacent to the fuel handling areas.

Provide a complete description of the potential radiological effects and dose estimates associated with the reduction of the minimum water level over active fuel in the refueling and spent fuel pool. Include this information in the DCD and provide a markup in your response.

Additional Question based on 6/4/09 phone call: (Revision 1)

After reviewing APP-GW-N2C-006, Rev. 2, the NRC staff requested further clarifying information related to the calculation.

1. Provide clarification relative to why dose evaluations were performed at two water coverage elevations, 102" and 105" above the active portion of the fuel.
2. Provide clarification relative to why Case D sources (based on once-burned fuel assemblies) are the most conservative source terms; particularly since the total gamma source associated with Case A (core average power per assembly) exceeds that of Case D.
3. Provide more information related to the ALARA controls that are used to lower dose during refueling. This should include a brief discussion on controls used to limit dose for failed fuel.

Westinghouse Response:

1.0 Background

Fuel handling activities in the Auxiliary Building Spent Fuel Pool (SFP) include normal fuel handling operations as performed during refueling outages, i.e., fuel offloaded and reloaded from/to the Reactor Building via the Fuel Handling System (FHS) into and out of the SFP storage racks. The AP1000 FHS component in the Auxiliary Building which lifts and lowers the fuel in the SFP during normal fuel handling operations is the Fuel Handling Machine (FHM).

Reference 4.1, Technical Report 121 (APP-GW-GLN-121, Rev.0) was issued to detail the changes to the water level in the SFP to limit the radiation dose rates to personnel on the FHM to 2.5 mR/hr maximum. At that time, the FHM design was a Sigma style Manipulator Refueling Machine (RM) type and the report was based on the configuration of that machine style.

Since that time, design changes were made to change the FHM design from the (RM) type to a simple gantry style bridge crane. Additional design changes were made to the elevation of the fuel assembly in the SFP water column to provide margin for operator handling of the fuel over the spent fuel racks. Previously, fuel transfer elevation provided a nominal 3 inch bottom gap between the fuel assembly and the top of the fuel racks. Allowing for fuel assembly growth and tolerancing stack-ups, this gap could have been as small as 2 inches. Per our external (NuStart) and internal (Field Services) customers, this gap did not provide sufficient margin for handling fuel. These design changes included detailed dimensional changes to the FHM design and identified the specific elevation range of the water in the SFP and Rx refueling cavity.

As shown in Figure 1.1 below, the current configuration of the SFP, fuel handling elevation, and FHM is illustrated. The minimum water coverage over active fuel during fuel handling operations is 105". The FHM operating deck where the operator stands on is approximately 53 inches above the SFP water surface (based on a minimum SFP water elevation of 134'-0"). The FHM deck thickness is 1.25" thick. This current configuration limits exposure rates to personnel on the FHM to 2.5 mR/hr or less. The configuration of the Containment Building Refueling Machine (RM) is bounded by the FHM since the RM has shielding from the mast structure and more distance from the fuel assembly to the operator.

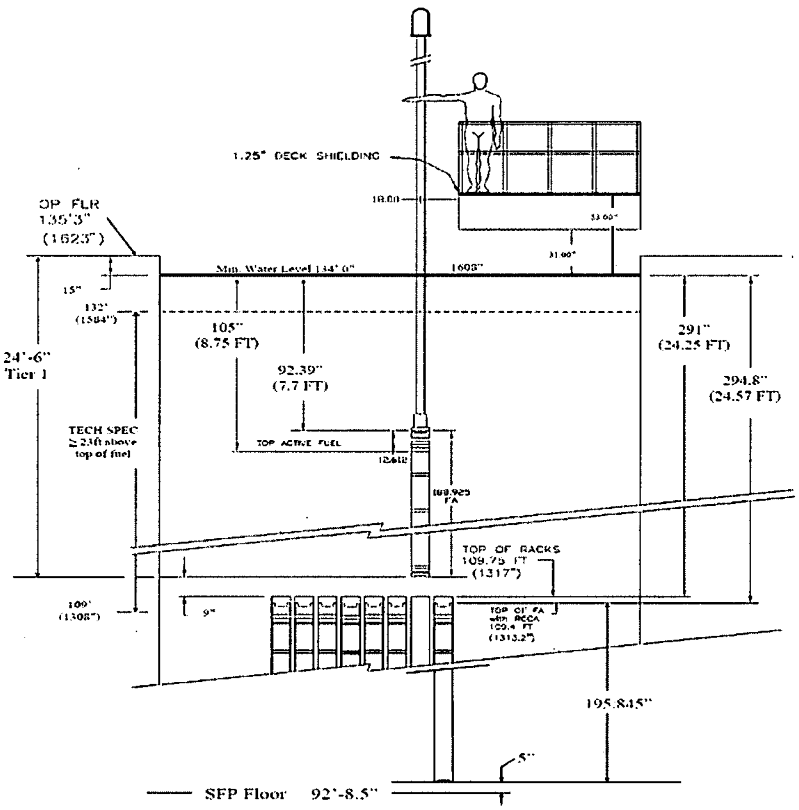

Figure 1.1 - Dimensions Associated with AP1000 Fuel Handling

The following sections extracted from Reference 4.2 (APP-GW-N2C-066, Rev.2, "AP1000 Spent Fuel Shielding Evaluation") provide the background, results, and conclusions of the evaluation used to determine the potential radiological effects and dose estimates associated with the reduction of the minimum water level over active fuel in the Auxiliary Building spent fuel pool and the Containment Building refueling canal. Revision 2 of the evaluation was issued to update it to correspond with the current design of the AP1000 FHS and SFP / Refueling cavity.

2.0 Evaluation Background and Purpose

Previous versions of this evaluation describe analyses of dose rates from spent fuel at the water surface and at location(s) where worker access is required during transport, storage and installation activities. Such activities include the use of the manipulator crane (RM) over the refueling cavity and spent fuel pit. The evaluation also considered the dose rates associated with "manual" handling of spent fuel with long-handled tools (without a mast) using the auxiliary hoist of the manipulator crane.

The purpose of revision 2 of the evaluation is similar in that it is an evaluation of dose rates from spent fuel at location(s) where worker access is required. The major differences are:

a) The water cover is reduced from that previously considered.

b) Previous versions of this evaluation considered a manipulator crane on both the refueling cavity (inside containment) and the spent fuel pit (outside containment). However, the manipulator crane on the spent fuel side has been deleted and replaced by a gantry style crane with a shielded deck that spans the spent fuel pit.

c) The decay gamma source that yields the most conservative dose rates above the water surface is determined by examining spent fuel assemblies with various burnup/power histories. The burnup and power conditions for the spent fuel assemblies that are considered are thrice burned with the core average power (flat power), thrice burned with the corresponding powers associated with each of the 3 cycles, twice burned with the corresponding powers of the 2 cycles, and once burned with the corresponding power of the feed region. Previous analyses considered an assembly with the core averaged power for 3 cycles of operation.

The fuel assembly depletion calculations for the decay gamma source generations are based on the fuel assembly design data using the ORIGEN-S computer code.

The gamma ray transport calculations are based on the spent fuel sources at 100 hours after shutdown using the DORT computer code.

This evaluation was prepared according to Westinghouse document APP-GW-GAP-100.

3.0 Summary of Results and Conclusions

3.1 Results

Various depletion conditions were examined in order to determine the most conservative decay gamma source generation from spent fuel assemblies. The decay gamma sources from a spent fuel assembly with one cycle burnup with the power associated with a feed region was found to yield the most conservative dose rates at or above the water surface.

For normal fuel transfer operations, the AP1000 DCD states that "... the gamma dose rate at the surface of the water is 20 millirem/hour or less." Another design objective is that the radiation dose to an operator during the fuel transfer is less than 2.5 millirem (0.025 mSv) in any one hour. This evaluation demonstrates that these criteria are met.

Directly above the fuel assembly on the assembly center-line and at various elevations above the bridge deck, it is noted that the dose rates can exceed 2.5 millirem/hr (0.025 mSv/hr). However, if more meaningful locations relative to personnel exposure are considered, i.e. between waist level and head level and at least one foot (30.48 cm) from the fuel assembly center-line, the maximum dose rates are generally in the range of 2.0-2.5 millirem/hr (0.02 -0.025 mSv/hr) for 102 inches (259 cm) of water cover and 1.5-2.0 millirem/hr (0.015 - 0.02 mSv/hr) for 105 inches (267 cm) of water cover.

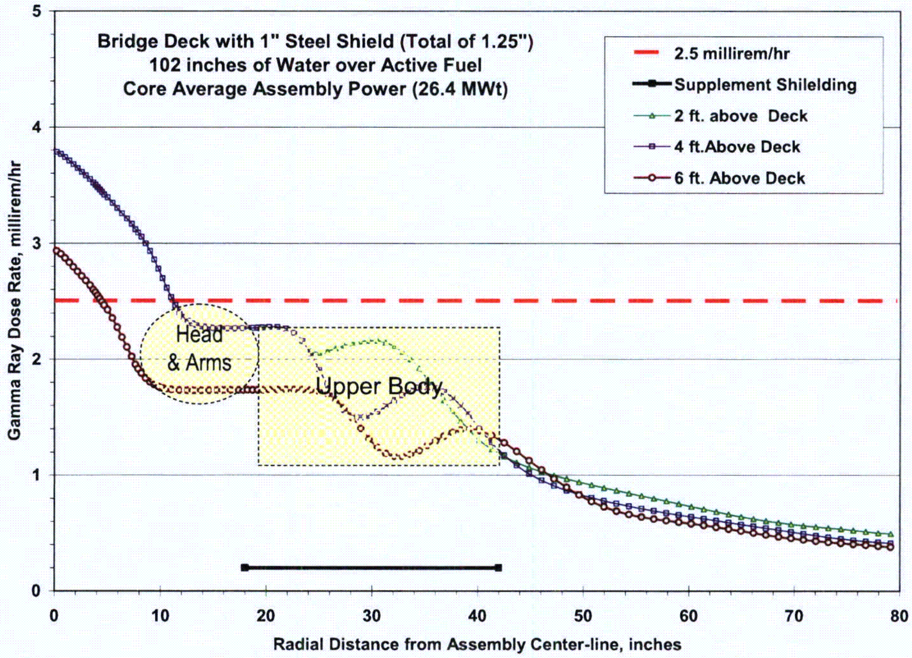

Figure 2-1

Dose Rate on SFP Bridge Deck with 102 inches of Water Cover

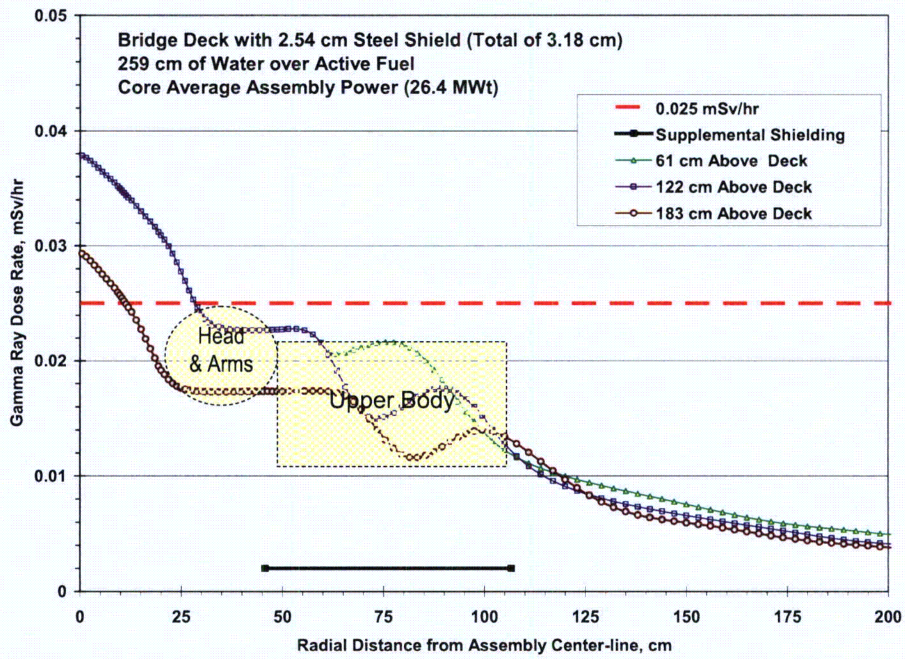

Figure 2-2

Dose Rate on SFB Bridge Deck with 102 inches of Water Cover (SI Units)

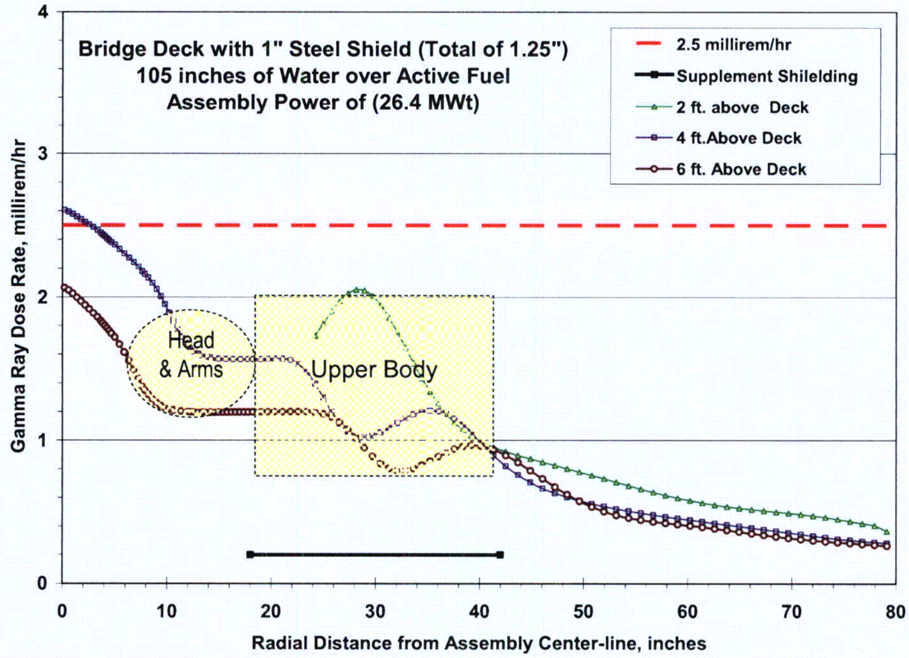

Figure 2-3

Dose Rate on SFP Bridge Deck with 105 inches of Water Cover

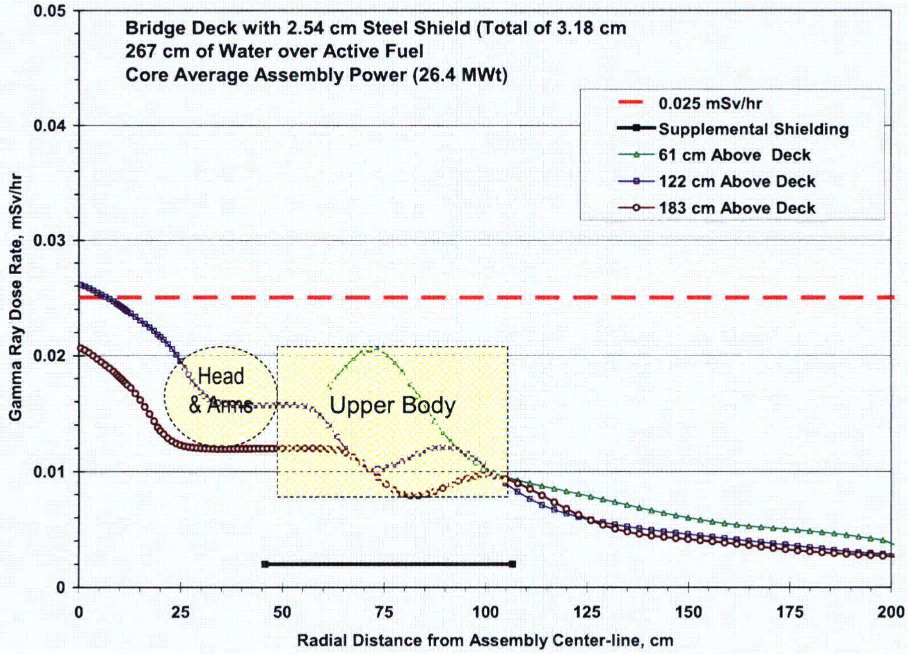

Figure 2-4

Dose Rate on SFP Bridge Deck with 105 inches of Water Cover (SI Units)

3.2 Conclusions / Recommendations

These dose rates are such that it is unlikely that a worker on the bridge deck would accrue a whole body (or deep-dose equivalent) dose of more than 2.5 millirem (0.025 mSv) in an hour. This is illustrated in Figures 2-1 through 2-4, in which the range of dose rates that the head and arms as well as the upper body might realistically receive. In both cases; i.e. with 102 inches (259 cm) and 105 inches (267 cm) of water cover, the dose rates are less than 2.5 millirem/hr (0.025 mSv/hr). Note that Figures 2-1 and 2-2 illustrate the dose rates with 102 inches (259 cm) of water cover where Figure 2-1 is in conventional units and Figure 2-2 is in System Internationale (SI) units. Similarly, Figures 2-3 and 2-4 illustrate the dose rate data for 105 inches (267 cm) of water cover.

It should be noted that these results are highly conservative, since the dose rates are based on a conservative source term that is approximately twice that expected for discharge fuel assemblies.

This RAI response explains why the overall minimum allowable water depth above active fuel was decreased. It provides the basis of the parameters and describes assumptions used with the extract of the calculation. The full calculation APP-GW-N2C-066, Rev.2, "AP1000 Spent Fuel Shielding Evaluation," is available for NRC review on request.

The DCD is not revised to include this information. Rev 17 of DCD Section 12.3.2.2.4,"Fuel Handling Area Shielding Design," discusses the bases for the nuclear radiation shielding and the shielding configurations to a level of detail consistent with the other subsections in 12.3.2.2. Potential radiological effects are not discussed in this section, consistent with the other subsections in 12.3.2.2. Dose estimation methodology is already discussed in DCD section 12.3.2.3, "Shielding Calculational Methods."

3.3 Open Items

There are no open items associated with this evaluation.

4.0 References

4.1 APP-GW-GLN-121, Rev.0, (Technical Report 121, dated 05/22/07), "Spent Fuel Pool Water Level and Dose".

4.2 APP-GW-N2C-006, Rev.2, (CN-REA-05-55), "AP1000 Spent Fuel Shielding Evaluation".

Additional Westinghouse Response based on 6/04/09 phone call:

Response #1

DCD rev 17 defines these two elevations in Tier 1 ITAAC's and Tier 2.

Tier 1, Table 2.1.1-1, Paragraph 5, limits the fuel assembly raise height to 24'- 6" between the bottom nozzle and the operating floor, El 135'-3". This corresponds to 102" of water shielding when the refueling cavity/spent fuel pool water level is at the 134'-0" elevation. This limit is established as a mechanical hard stop for the refueling machine and the fuel handling machine.

Tier 2, throughout Section 9.1 and Section 12 establishes the minimum water coverage as 8.75 feet (105") above the active portion of the fuel. The refueling machine and fuel handling machine will have controls to limit the hoist up travel to satisfy this requirement. These control limits are established to prevent equipment operation at hard mechanical limits.

Response #2

Case A would be the most conservative assumption if the shield medium were air. However, outside of a shield of water and metal, the dose is more sensitive to the energies of the gamma sources. That is, the low energy photons are generally reduced to negligible levels by the shielding media and the higher energy gammas are generally lower in specific source strength. Thus, the most important sources when considering the shielding of fission products tend to be in the ½ to 2 ½ MeV range.

The gamma source strengths considered in the analyses are divided into 20 energy intervals over a range of 0 to 14 MeV/gamma with a total source strength for Case A that is 2.4 percent higher than Case D. However, if the dose rates above the water are considered, rather than the total source strength, the results show that over 90 percent of the dose is associated with 8 energy groups in the range of 0.2 to 3 MeV/gamma

Upper Energy (MeV)	Lower Energy (MeV)	Source, Photons/cc-s		Dose Rate Contribution, %
		Case A	Case D	
3.00	2.00	1.55E+10	1.30E+10	10
2.00	1.50	1.53E+11	1.95E+11	20
1.50	1.00	6.27E+10	3.50E+10	25
1.00	0.80	1.16E+11	1.18E+11	10
0.80	0.70	4.57E+11	5.98E+11	5
0.70	0.60	1.94E+11	1.58E+11	5
0.60	0.40	3.89E+11	3.92E+11	9
0.40	0.20	5.01E+11	4.75E+11	8
Total		**1.89E+12**	**1.98E+12**	**92**

As shown above, the Case D sources in these energy groups are about 5% higher than the Case A sources and the contribution of the 8 energy groups to the total dose rate above the water is greater than 90 percent.

Response #3

As noted in APP-GW-N2C-006, Rev. 2 and the previous response to the RAI, the increase in dose rates associated with the reduction in water cover over a fuel assembly during transfer is offset by the reduction in dose rate provided by the added fuel handling machine bridge deck shielding. The calculations are conservative in that the assembly source strength that is considered in the analysis is roughly twice that associated with discharge fuel assemblies.

Another potential contributor to refueling worker dose is that from activity that is dispersed in the water in the refueling canal and spent fuel pit. This source is typically introduced as particulate activity from the fuel cladding surfaces during fuel handling. It may also be a result of incomplete shutdown crud solubilization evolutions that take place during plant shutdown operations. Again, the addition of shielding to the fuel handling machine bridge deck results in reduced dose rates from this source during refueling operations. It should also be noted that use of the zinc addition system that is included in the AP1000 plant design will tend to minimize this potential source; since zinc addition has been proven to significantly reduce corrosion rates of materials and to reduce plant radiation fields and core crud loadings. Further, improvements in shutdown/startup procedures as recommended by INPO and EPRI should tend to reduce this potential source. The associated reductions in dose rates above the water surface from the activity dispersed in the water results in reduced exposure to refueling personnel that are on the manipulator crane and spent fuel bridge deck as well as those that work on the decks along side the refueling cavity and spent fuel pit.

The impact of the design basis assumption of 0.25% fuel failures is not considered to be a factor in the radiological impact of the change in water cover. Should the plants experience fuel failures, the release of fission products to the primary coolant will not continue after shutdown (except for possible activity "spikes" that may follow the change in fuel temperature at shutdown). The activity in the primary coolant will then be removed by CVS demineralizers and filters to a prescribed end point concentration limit prior to refueling cavity fill and fuel transfer operations.

It should also be noted that radiation levels are monitored by the permanent containment area radiation monitor and by a portable bridge monitor during refueling operations. The containment area radiation monitor is located to best measure the increase in exposure rates for this area. Radiation levels are also monitored by the permanent fuel handling area radiation monitors and by a portable bridge monitor(s) during fuel handling operations. The fuel handling area radiation monitors are located to best measure the increase in exposure rates for this area. The permanent radiation monitors provide an alarm locally and in the main control room. The data provided by the radiation monitors (permanent and portable) when coupled with effective plant health physics and radiation protection programs will ensure that the refueling operations are performed to ALARA standards.

Design Control Document (DCD) Revision:
None

PRA Revision:
None

Technical Report (TR) Revision:
None