Chapter 12 Discussion Topics List

Chapter 12 Radiation Protection, RAIs 8254, 8275, 8353, 8247, 8098, and 8389
Conference Call with KHNP on APR1400 Design Certification
Wednesday, May 18th, 2016, 7:00 p.m. – 9:00 p.m. (EST)

1. In a clarification call regarding RAI 8254, Question 12.03-17, KHNP indicated that the peaking factor for the peak assembly is 1.2353 and the peaking factor of 1.55 used in Chapter 12 is actually the peaking factor for an individual rod; this would be a conservative assumption for the Chapter 12 shielding calculations, which would be acceptable, for that purpose. However, staff has never agreed on if this approach is acceptable, and Chapter 15 appears to use different peaking factors, e.g., Table 15.4.8-4 uses 1.80. KHNP needs to explain why these different peaking factors are being used and update the DCD to make it clear what the actual normal operation peaking factor is. (KNF)

   (KHNP Response) The peaking factor of 1.80 used in DCD Table 15.4.8-4 and other tables in Chapter 15 (e.g., Tables 15.1.5-12, 15.3.3-3, and 15.7.4-1) is a conservative radial power peaking factor for an individual fuel rod at full power, considering PDIL (Power Dependent CEA Insertion Limit) and power tilt in the core with uncertainty. The peaking factor of 1.55 used in DCD Chapter 12 Section 12.3.2.3 for shielding calculations is the target value of maximum expected radial peaking factor for an individual fuel rod at full power for unrodded condition.
   It is expected that for normal, base load operation of the plant, reactor operation will be with limited CEA insertion so that the actual normal operation peaking factor will not be higher than the nuclear design target of 1.55 during most of the cycle.

2. Regarding RAI 8275, Question 12.03-40, KHNP have agreed to revise the response to correct inaccurate information (stating that the ranges and coverage area of the containment upper and lower monitors are the same). However, KHNP never indicated if they would be revising the technical specifications Table 3.3.11-1 designation for the containment upper operating area monitors from "F" to "E". Radiation Protection and Emergency Planning staff agree that KHNP should either revise the designation or, if the designation is maintained, provide additional information regarding what the alternative means would be, for staff review. (I&C)

   (KHNP Response) The applicant provides the following background and rationale behind designating "F" for the upper operating area monitors and "E" for the lower operating area monitors. This will hopefully explain the reason for not addressing the question about "E" and "F" designation in our second response (RAI 368-8470, Question 14.03.08-14).
   The accident considered in table 3.3.11-1, items 10 and 25 is the fuel handling accident. For this accident, the primary monitors are the lower monitors (item 25) and it is designated as "E" accordingly. For this accident the upper monitors (item 10) performs secondary role of providing the second set of signals into the second one-out-of-two coincidence logic. The upper monitors are, therefore, designated as "F" for this accident. This was explained in the applicant's first response and felt no need to address the subject in the second response.

The above argument changes if one focuses on the DBAs of the NSSS ESFAS and the upper monitors required by RG 1.97 become the primary with designation “E” and the lower ones become the alternatives with designation “F”.
Because of these dual roles, the applicant decides to revised the designation of the upper ones to “E” and add a note for item 25. Please refer to the attached mark-up

3. KHNP has agreed to supplement RAI 8254, Question 12.03-20. However, during a previous call, KHNP stated that they do not need a monitor in the remote shutdown room, and staff said they would look into it. In a follow-up email, staff sent KHNP the following question, which has never been discussed in a follow-up call:
“I’m evaluating to see if they need a monitor in the remote shutdown room as discussed yesterday. When I look at the remote shutdown room (see room 137-A06D on Figure 12.3-6, for normal operation and equivalent accident figures), it is in a non-radiological area (the area is white, labeled as less than 0.0025 mSv/hour, normal operation), however there is a small oblong area of the north west portion of the room that appears to be designated as less than 1.0 mSv/hour (normal operation). Can we ask KHNP to let me know what that area represents (it is probably not a symbol for a radiation zone, but I want them to indicate what it is).” (RP)

   (KHNP Response) The oblong shown in the remote shutdown room on Figure 12.3-6 indicates a “Raised floor”. It means that while the elevation of the concrete floor of the remote showdown room is 137’-6”, the actual floor is raised by one or two more feet so that the electrical cables can be routed beneath this raised floor. For APR1400 the elevation of the actual raised floor is 139’-3”. Since a note for this symbol is missing, KHNP will update Figure 12.3-6 to add a note indicating what this symbol means as a revised response to RAI 8254, Question 12.03-20.

4. The following question was sent in a previous email, but was never discussed. The response to RAI 8254, Question 12.03-13, Part 1, indicates that the temperature and humidity instrumentation are installed at wall mounted piping racks that include valves. For clarification, would a worker need to enter a delay bed room if there was something wrong with the temperature and humidity instruments, or can it be accessed from the piping racks? (RP)

   (KHNP Response) Since instrumentation of temperature and humidity in the charcoal delay bed room is not required, there is no instrumentation installed in that room. Instead, the Compound Building HVAC system provides the room ventilation and the GWMS system is designed to control the temperature and the humidity of the charcoal delay beds. Therefore, access to this room is not required during normal operation.

5. In the call related to questions 1-4, which KHNP sent responses to in “CQ-20160324 & 0325 & 0330”, staff stated they needed to do additional review on item “B) 1.” related to RAI 8353, Question 12.02-21. After doing that follow-up review, staff provided KHNP with additional information needed. Discussion is needed to ensure there is a path forward in order to resolve the issues.

   ** Followings include the previous discussions on CQ-20160324 & 0325 & 0330” related to RAI 8353, Question 12.02-21. The latest NRC questions are written in red and the KHNP latest responses are in blue.*

   (NRC Question) It is unclear why processing 50,000 gallons of concentrate from the holdup tank is appropriate for determining the source term for the concentrate package,

when the capacity of the holdup tank is much greater than 50,000 gallons (we're looking for justification for why 50,000 gallons was chosen) (SD)

(KHNP Response) The design criteria of holdup tank is (1) to allow for holdup of total recycle waste, and (2) to allow for holdup of wastes from specified plant shutdowns.

It was assumed that fuel burnup waste would be processed each time the HT accumulates to 50,000 gal. This assumption is made for the following reasons;

1. The operation must be relatively infrequent. Because of time spent initiating and securing each operation of the boric acid concentrator (BAC), it is more desirable to perform one large batch rather than several small batches.
2. The volume of which the HT is processed must be relatively small to avoid excessive tank sizing. Because of the recycle design, unprocessed liquid in the HT represents boration water from the BAST, and dilution water from the RMWT which cannot be taken credit for when sizing those tanks. Thus to meet the design criteria, the BAST and RMWT sizes vary directly with the quantity of waste allowed to accumulate in the HT. Likewise, the required HT is equal to the maximum surge volume requirement plus the maximum operating volume.
3. Processing level must be at a large enough percent of the total HT height to be easily monitored, and thus not confused by offset in level indicator instrumentation.

Therefore, the maximum operating volume of the HT is determined as 50,000 gals in the relevant operation guideline, because it is approximately 12% of the HT volume, thus, it is easily monitored and small enough to avoid excessive tank sizing. Based on the operation guideline, the volume of 50,000 gals is used for the source term for the concentrate package.

(NRC Notes): While the calculation package states that the BAC package is based on processing 50,000 gallon of CVCS Holdup Tank fluid. The DCD and the response to the following item specifies that the source term is based on a concentration factor of 100. However, the Cs-137 source term is only concentrated by a factor of approximately 35 (the concentration of Cs-137 in the BAC concentrator package is only about 35 times more than the concentration in the CVCS Holdup Tank) and there does not appear to be ion exchanger or other removal mechanism between the CVCS Holdup Tank and the BAC package in DCD Figure 9.3.4-1 (and even if the Boric Acid Condensate IX can somehow be aligned, it's decontamination factor for Cs is 1 anyway, as provided in the response to RAI 12.02-4). Therefore, the concentration factor does not appear to be 100, as KHNP has stated. This also appears to be true for other nuclides. Therefore, KHNP is requested to 1) revise the source terms (0.25% and 1% source terms) for all radionuclides to base the source term on a concentration factor of 100 and update all associated information, including radiation shielding, zoning, and equipment qualification information; or 2) Explain and justify why the values for the total values for the BAC source term are acceptable and ensure that all information in the DCD is accurate. (Please ensure that all other changes to related RAIs that could affect the BAC source term have been considered when making these changes). (SD)

(KHNP Response) The concentration of Cs-137 in the BAC package is 100 times more than the concentration in the Holdup Tank. So the values of BAC source term are acceptable. The liquid activity of holdup tank (HUT) is calculated as follows (Cs-137) ;

$$A_{HUTL} = A_{HUTL1} + A_{HUTL2} + A_{HUTL3} + A_{HUTL4} + A_{HUTL5} = 1.0E + 10\ Bq$$

where

A_{HUTL1} = normal power operation activity,

A_{HUTL2} = additional filling activity from low to high water level,

A_{HUTL3} = shutdown boration waste to HUT activity,

A_{HUTL4} = drain waste to HUT for Reactor Head removal activity, and

A_{HUTL5} = startup dilution waste to HUT activity.

However, to calculate specific activity in holdup tank (a_i^{HUT}), normal power operation activity (A_{HUTL1}) and additional filling activity (A_{HUTL2}) are only considered.

$$a_i^{HUT} = (A_{HUTL1} + A_{HUTL2}) / V_{HUTL1+HUTL2} = 1.9E + 01\ [Bq/cc]$$

where

A_{HUTL1} = normal power operation activity,

A_{HUTL2} = additional filling activity from low to high water level,

$V_{HUTL1+HUTL2}$ = HUTL1 volume + HUTL2 volume

The specific activities for Concentrate Heater of BAC components are calculated as follows (Cs-137) ;

$$a_i^{HTR} = a_i^{BAC} \times CF \quad \text{for particulate} = 1.9E + 03[Bq/cc]$$

where

a_i^{HTR} = influent specific activity to Concentrate Heater [Bq/cc]

a_i^{BAC} = influent specific activity to Boric Acid Concentrator [Bq/cc]

CF = concentration factor = 100

Same specific activity is applied to Concentrate Pump, Concentrate Transfer Pump, Concentrate Cooler, Flash Tank, and Vapor Separator (1.9E+03 [Bq/cc]).
The concentration (1.9E+03 [Bq/cc]) of Cs-137 in the BAC package is 100 times more than the concentration (1.9E+01 [Bq/cc]) in the Holdup Tank.

Other issue

(NRC Notes) In addition, in reviewing DCD Figure 9.3.4-1 to resolve this issue, staff identified another radiation protection issues associated with this figure.

Figure 9.3.4-1 (7 of 7) shows a line from Reactor Makeup Water Tank (RMWT) to the Fire Water Pump Suction, closed with a diaphragm valve (which have been known to have through leakage). The RMWT contains radioactive water and in accordance with 10 CFR 20.1406 would not make sense to use for fire protection water, except if there

were absolutely no other options available. It is unclear why a permanent line connecting the RMWT to the fire protection system is necessary and furthermore, it is unclear why the use of a single diaphragm valve is appropriate to separate the tank from the fire protection system. (RP)

(KHNP Response) The line from RMWT to Fire Water Pump Suction is not used for fire protection water. Instead, the line is designed to use the RMWT water as an emergency water source for the ECSBS (Emergency Containment Spray Backup System), which is a backup system for containment heat removal in case where the CSS (Containment Spray System) is not available during severe accident conditions. The line from the RMWT is run to the ECSBS via a portion of the FPS (Fire Protection System). Therefore, during normal operation, the FPS is not contaminated.
Diaphragm valve was selected to lower the buildup of radioactivity considering the ALARA principle. In addition, since the valve is normally closed, not used during normal operation and the design pressure is low, the leakage due to a failure is not expected to occur at the valve.

6. For the very last item in "CQ-20160324 & 0325 & 0330" related to RAI 8247, Question 12.02-15, discussion is needed to determine if KHNP has any update on the approach going forward and if they understand staff's additional comment.

** Followings include the previous discussions on CQ-20160324 & 0325 & 0330" related to RAI 8247, Question 12.02-15. The latest NRC questions are written in red and the KHNP latest responses are in blue.*

(NRC Question) While the applicant gives source term information for the condensate polishing system in the response to Part 1, there does not appear to any minimum shielding thicknesses provided for the area in DCD Table 12.3-4. It is unclear why this information would not be needed or why it isn't included. In addition, there does not appear to be any shielding calculation provided for the condensate polishing system provided in the shielding calculation package either. It is unclear where they are located or where the resin holdup tanks for the CPS are located in the Chapter 12 figures or if the Chapter 12 figures even show all elevations of the turbine building. Finally, while the response and DCD Section 10.4.6-1 discuss the replacement of resin and the DCD discusses that resin collected in the CPS spent resin holding tanks can be sent to the radioactive waste management system, if necessary. There is no line provided from the resin holding tanks to the radioactive waste treatment system for the transfer of resin in DCD Figure 10.4.6-1.

(KHNP Response) The CPS cation and mixed bed room is located inside turbine building, the shielding calculation for this room was performed in 1-371-N376-001, "Turbine Building Shielding Calculation". According to this calculation, the minimum shield thickness is 4 inches, and KHNP will revise the response to the RAI 141-8098 Question 12.03-8 to include this values in table 12.3-4. Transfer line from spent resin holding tank to radwaste treatment area has not been incorporated currently. The more detailed review including process and transfer method for the spent resin is required to correct this inconsistency.

(NRC Status): KHNP is going to determine if the APR1400 is going to be designed to regenerate CPS resin and determine if the transfer line from the spent resin holding tank to radwaste treatment area is incorporated into the APR1400 design. KHNP will

get back to NRC on this issue and revise the response to Question 12.02-15 and the DCD to provide any appropriate updates. (ME/RW)

(KHNP Response) Since the U.S. practice appears to be different from Korean NPPs, KHNP is still reviewing how to process the spent CPS resin for the APR1400 DC. Therefore, the response to this question needs to be deferred until further information is obtained considering the best current U.S. practices. KHNP will try to provide the response along with the corresponding DCD markups as soon as possible.

(NRC Status) Also, this was not discussed during the call but the CPS demineralizers and spent resin holding tank area do not appear to be identified in the Chapter 12 radiation zone maps for the Turbine Building. The DCD should be updated to identify and zone these areas properly in the DCD Chapter 12 figures. The response to Question 12.02-15 should be updated to include this information. (RP)

(KHNP Response) Radiation zone maps for Turbine Building are provided in DCD Figures 12.3-17 and 12.3-18. As shown below, the radiation zone at EL. 73'-0" is designated as Zone 2 and this area contains CPS demineralizers and the spent resin hold tanks. Shielding is provided around this area. Since no description is provided in Figure 12.3-17, KHNP will update this figure to include that this area contains CPS demineralizers and the spent resin hold tanks as a revised response to RAI 141-8098 Question 12.03-8.

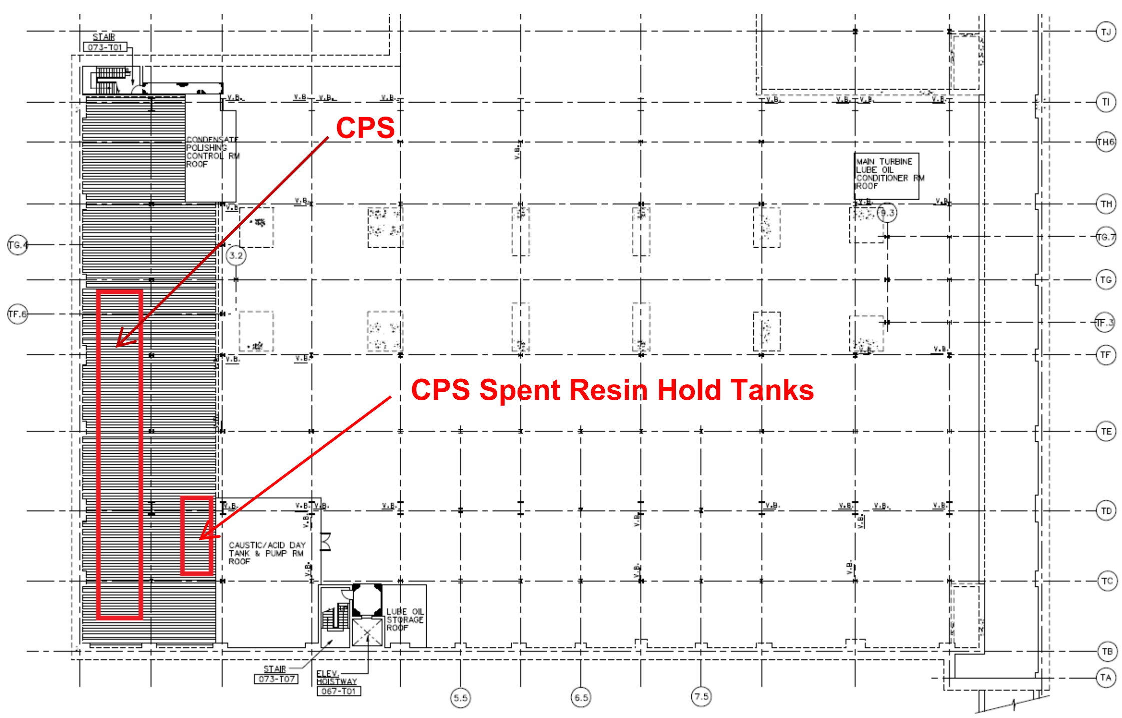


Figure 12.3-17 Radiation Zones (Normal) Turbine Building El. 73'-0"

7. In the most recent CQs discussed ("Response to CQs_04142016 NRC status 4-15-16.docx"), in which staff provided the status, staff needs to discuss the item on RAI

8098, Question 12.03-10, to make sure that KHNP understand staff's comments and are going to include the ceiling thickness(es) for room 077-P01 in the DCD.

** Followings include the previous discussions on CQ-04142016 NRC status 4-15-16.docx" related to RAI 8098, Question 12.02-10. The latest NRC questions are written in red and the KHNP latest responses are in blue.*

Regarding the response to RAI 8098, Question 12.03-10

(NRC Question) Part 4 of the question requested that the applicant update the DCD to include the shielding thicknesses for each wall of room 077-P01 (which is an area potentially greater than 100 rad/hour, probably a VHRA if 100% concentration is considered for resin transfer lines, as discussed above). While the applicant provided Figure 2 in the response, providing all the different minimum wall and floor thicknesses for that room, the applicant did not include any of the information on shielding thicknesses in the DCD. The information needs to be included similar to other radiation shield walls and floors, as specified in the SRP.

(KHNP Response) Since the geometry of Rooms 077-P01 and 068-A07A is too complicated, adding all shield thicknesses for each wall or slab in DCD Table 12.3-4 is difficult as indicated in the staff's comment in the next question. This is why KHNP provided the thickest shield wall or slabs (except for the walls against underground area) for the Hot Pipe Ways in the table as shown below.

"A"

Table 12.3-4 (3 of 7)

Room Number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
Auxiliary Building (cont.)							
068-A06A	Gas Stripper Room	37	35	45	35	30	24
068-A07A	Hot Pipe Way	13	29	31	29	24	24
068-A08B	Hot Pipe Way	33	24	24	20	23	30
068-A09B	Valve Room	19	17	23	19	17	10

KHNP thinks that such detailed information is not balancing with other DC applications. Therefore, since the APR1400 design document (i.e., calculations and drawings) provides the sufficient details of the shielding requirements, KHNP would like to propose to provide this information within the shielding audit or the RAI response processes.

(NRC Status) KHNP agreed to include all minimum shielding thicknesses for room 077-P01 in the DCD. The DCD should be clear what the minimum shielding thickness is for each section of wall, ceiling, and floor. This of course should include any revisions to the source terms, as discussed in item 1 above.

FOLLOW-UP COMMENTS: Staff stated that the applicant could just include the information in Figure 2 of the response to RAI 8098, Question 12.03-10, to resolve this issue, however, staff realized after the call that Figure 2 does not include the ceiling thickness for that room. Therefore, the ceiling thickness should also be included in the DCD. If KHNP wishes to allow for varying ceiling thicknesses, as they did with the

floors, then it should be clear what the values are for different sections of the ceiling. (RP)

(KHNP Response) As discussed during the conference call, the shielding thicknesses for the ceiling of a room are provided as those for the floor of the room located above. The varying thicknesses of a floor will be identified in the updated DCD Tier 1 Table 2.2.1-1 as a revised response to RAI 116-8054 Question 1.

(NRC Status) In addition, regarding the piping ways with complex geometries on the 68' elevation of the Auxiliary Building. As stated in the meeting, the way the DCD is written, DCD Table 12.3-4 provides the minimum required shielding thicknesses for these rooms. The information in the DCD must be accurate. Therefore, the actual physical plant design must have shielding thicknesses as thick or thicker than what is specified in DCD Table 12.3-4 (this statement is applicable to every room). If this is not the case, then DCD Table 12.3-4, needs to be updated to provide complete and accurate information. The actual physical wall thicknesses for radiologically significant walls in these areas with complex geometries in the Auxiliary Building can be verified to be thicker than what is specified in DCD Table 12.3-4 by viewing Tier 1, Table 2.2.1-1, so they are acceptable. However, the actual physical ceiling and floor thicknesses cannot be verified in Tier 1, Table 2.2.1-1, since the specific thicknesses for specific sections of the floor are not specified. Therefore, if the information in DCD Table 12.3-4 does not provide ceiling and floor thicknesses values equivalent to or greater than the actual plant design thicknesses, for all portions of the ceilings and floors, then additional information must be provided, because the DCD must accurately portray the design.

For example, DCD Table 12.3-4 specifies that the minimum required shielding thickness for the ceiling of room 068-A08B in the Auxiliary Building is 30 inches. Therefore, the staff is assuming that the design includes a ceiling that will be 30 inches or more, throughout the entire room, and the SER will be based on this premise. If this is not the case, then the DCD needs to include more specific information regarding the different shielding requirements. (RP)

(KHNP Response) As pointed out by the staff and discussed during the meeting, KHNP will update the DCD Tier 1 Table 2.2.1-1 to provide varying physical concrete thicknesses of the floors. An example for EL. 68'-0" of Auxiliary Building is shown below. The updated Table 2.2.1-1 will be submitted as a revised response to RAI 116-8054 Question 1.

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness	Applicable Radiation Shielding Wall (Yes/No)
Floors	From AJ to AK and From 25 to 26	68'-0"	1'-6"	Yes
Floors	From AI to AJ and From 23 to 26	68'-0"	2'-6"	Yes
Floors	From AF to AI and From 23 to 26	68'-0"	2'-0", 2'-6"	Yes
Floors	From AD to AF and From 23 to 25	68'-0"	2'-0"	Yes
Floors	From AE to AF and From 25 to 26	68'-0"	2'-0", 2'-6"	Yes
Floors	From AB to AD and From 23 to 24	68'-0"	2'-6"	Yes
Floors	From AA to AD and From 24 to 26	68'-0"	2'-0"	Yes

8. Regarding the response to follow-up question 2, associated with RAI 8389, Question 12.03-48, KHNP indicated that they did not want to add the information of the cobalt content of the fuel rods to the DCD, because it is zirconium, instead of steel or nickel. While the URD specifies the cobalt content for steel and nickel based components, EPRI 1003390 provides guidance on the cobalt content of zirconium materials, so information on the cobalt content from these materials should not be excluded from the DCD. Therefore, staff needs KHNP to add the maximum cobalt content of the fuel rods AND the middle grids to DCD Table 12.3-2, as it provides assurance that the overall design limits cobalt within the RCS, to the extent practicable, and ensures that 10 CFR 20.1101(b) and 10 CFR 20.140(b) will be met, which would resolve this issue. KHNP already provided the maximum cobalt content of the fuel rods in the response to the RAI, and Chapter 4 of the DCD indicates that the middle grids are made of ZIRLO, which should also have a cobalt content below 0.005 w/o, which is the maximum value EPRI specifies for Zirconium based material. Therefore, staff expects KHNP to revise the response to RAI 8389, Question 12.03-48, to provide this information. (KNF)

   (KHNP Response) The guidance of EPRI 1003390 for the cobalt content of stainless steels, inconels and zirconium materials is as follows:

   "Personnel should also be made aware of fact that significant levels of impurities are present in materials manufactured using conventional melting practices employed by metal fabricators. Cobalt impurity levels should be less than 500 ppm in stainless steels and less than 200 ppm in Inconels for all nuclear replacement components. <u>There is little cobalt present in the zirconium base Zircaloy alloys used for fuel rod cladding and in some fuel spacer grids, with the cobalt content of Zircaloy typically less than 50 ppm."</u>

   The underlined sentence for the Zirconium material mentions typical cobalt content of zircaloy and doesn't look like it means maximum cobalt content limit. Therefore, we think that it is not necessary to insert the cobalt content of zirconium materials as a limit in the DCD.

9. In the response to RAI 8098, Question 12.03-8, the proposed markup of DCD Table 12.3-4 indicates that the minimum required shielding thickness for the south wall of the volume control tank room is 42 inches. However, Tier 1, Table 2.2.1-1, specifies that this wall is only 3 feet (36 inches) thick. KHNP needs to resolve this discrepancy. (RP)

(KHNP Response) KHNP agrees with the staff's comment. The physical concrete thickness of the south wall of the VCT room will be corrected to be 42 inches to meet the shielding requirement. Table 2.2.1-1 in Tier 1 will be updated as a revised response to RAI 116-8054 Question 1.

10. In the response to RAI 8254, Question 12.03-11, KHNP indicated that the CCW sump drains may also be combined with other potentially radioactive drains, such as the Auxiliary Boiler Blowdown. This statement raises numerous questions and apparent inconsistencies in the DCD.

    1)
        a) First, it appears that monitoring for potentially radioactive fluid in the Auxiliary Steam System occurs prior to the fluid reaching the Auxiliary Boiler Package (apparently this is done by monitor RE-103, according to DCD Table 11.5-2) and if the system contains radioactive fluid, it is routed to the radwaste system prior to entering the Auxiliary Boiler Package. See Figure 10.4.10-1 (Sheet 2 of 3), which shows monitoring of the Auxiliary Steam System after the condensate receiver tank and an optional line to the liquid radwaste system prior to reaching the Auxiliary Boiler Package on Sheet 3 of 3. Therefore, it is unclear if the Auxiliary Boiler Blowdown would be expected to potentially contain radioactivity levels requiring treatment. More importantly, it is also unclear that the description of the blowdown drains in the DCD is accurate and that it will actually connect to the Liquid Waste Management system. While the response to RAI 8321, Question 09.02.02-9, adds a COL item specifying that the COL applicant is to provide the flow diagram of the turbine generator building drain system and the CCW heat exchanger building drain system, DCD Section 9.3.3.2.6 already indicates where the lines run. It states:

           "The discharge from TGB sump pumps are monitored for radiation contamination level. The auxiliary boiler blowdown and drains are routed to the local sump, from which the condensate is routed to the TGB sump for radiation monitoring. When contamination level is detected at or exceeding a predetermined setpoint, the drains are routed to the LWMS for processing and release via the condensate polishing area sump. This approach prevents unintended contamination of the [[WWTF]]."

           The above language implies that the design will include lines from the Auxiliary Boiler Blowdown to the sumps and apparently interconnecting the different sumps in the Turbine Building. Is this information in the DCD accurate? (ME)

           (KHNP Response) The drain system for auxiliary boiler building, turbine generator building and CCW heat exchanger building has design features for minimization of contamination. The description of DCD Section 9.3.3.2.6 is a basic requirement of minimization of contamination for both DC applicant and COL applicant. However, the design for auxiliary boiler building drain system, turbine generator building drain system and CCW heat exchanger building drain system is not included in the DC phase. Therefore, the COL applicant is to provide the detail design including all interconnecting pipes among sumps. The bellow sketch diagram shows the interconnection of the each building drain system for the reference.

The condensate return from the condensate receiver tank is potentially radioactive. The condensate in the condensate return pump suction is monitored for radiation. Upon detection of radioactivity by the radiation monitor, the condensate flow is automatically diverted to the liquid radwaste system by closing and opening of the control valve (for auxiliary boiler package) and control valve (for liquid radwaste system) respectively.

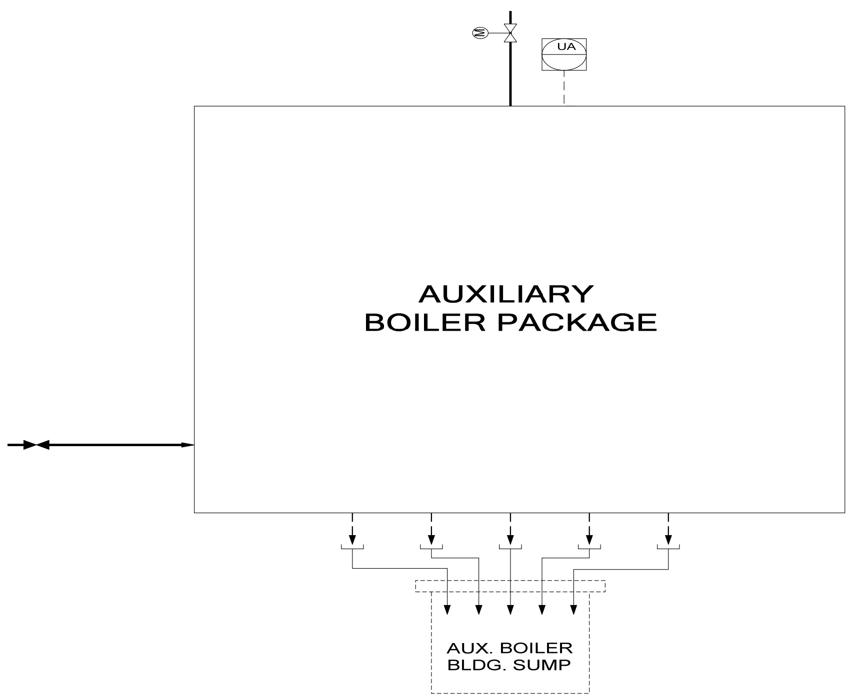
The auxiliary boiler blowdown and drains are routed to the auxiliary boiler building sump and then are routed to the TGB sump for radiation monitoring. When contamination level of the drains is detected exceeding a predetermined setpoint, they are routed to the LWMS for processing and release via the condensate polishing area sump.

Therefore, the auxiliary boiler blowdown would be expected to potentially contain radioactivity levels requiring treatment, and DCD Section 10.4.10.2.3 will be added as shown below and Figure 10.4.10-1 (Sheet 3 of 3) will be revised as shown bellow sketch diagram. KHNP will revise the response to RAI 8326, Question 09.03.03-4.

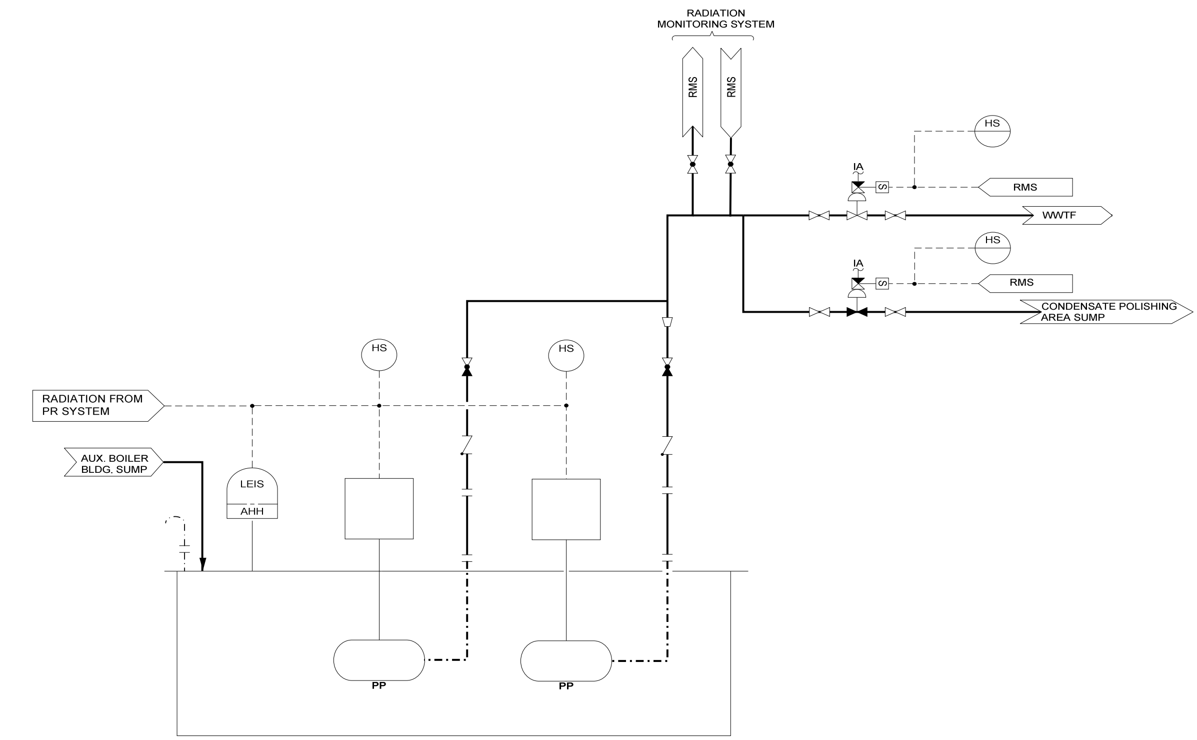
DCD Section 10.4.10.2.3 will be added as shown below:

"The auxiliary boiler blowdown and drains are routed to the Auxiliary Boiler Building Sump and then are routed to the TGB sump for radiation monitoring. When contamination level of the drains is detected exceeding a predetermined setpoint, they are routed to the LWMS for processing and release via the condensate polishing area sump."

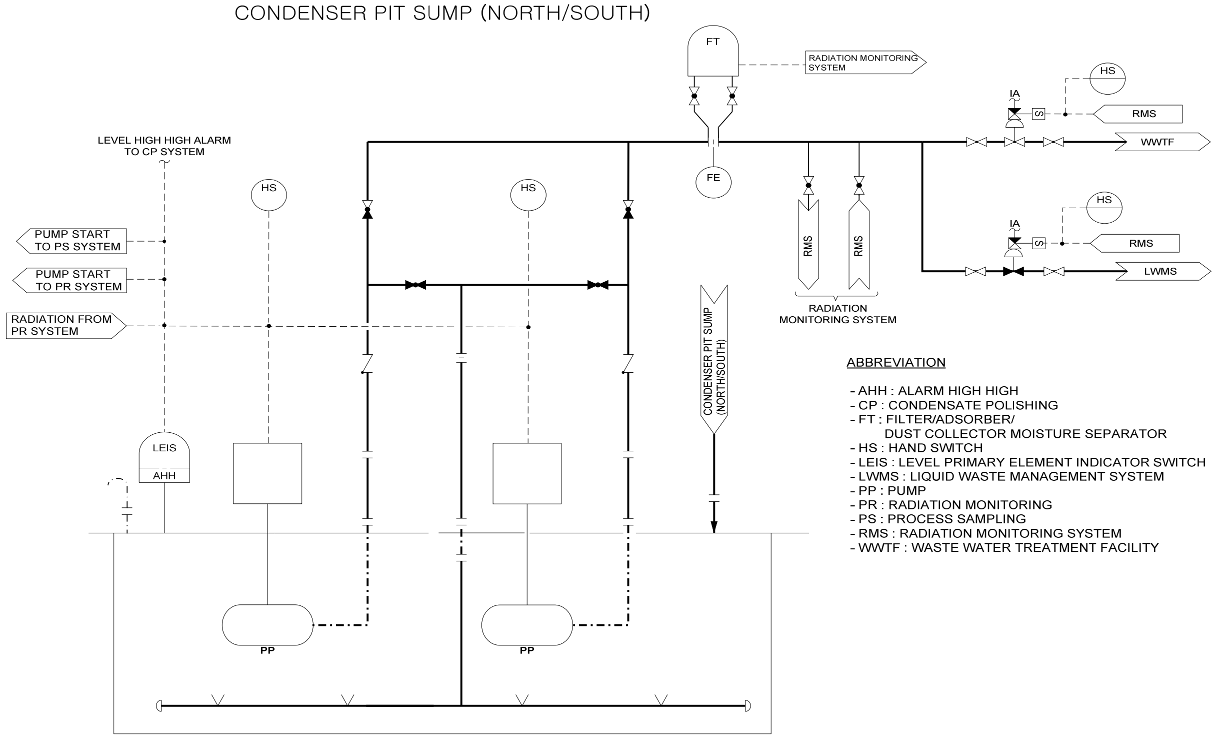
Figure 10.4.10-1 (Sheet 3 of 3) will be revised as shown below:



A reference sketch diagram which shows the interconnection of the each building drain system as follows;



CONDENSER PIT SUMP (NORTH/SOUTH)



CONDENSATE POLISHING AREA SUMP

b) In addition, DCD Section 9.3.3.2.6 also states, “Radiation monitors are provided in the discharge piping from the TGB north and south pit sumps and the condensate polishing area sump to detect contamination levels of the

drains." However, according to DCD Section 11.5, the only sump in the Turbine Building containing a radiation monitor is the Condensate polishing (CPP) area sump (monitor RE-164). KHNP is requested to verify that the information in the responses and the information in the DCD are accurate and make any appropriate corrections.
In addition to the above, there are two other apparent discrepancies associated with radiation monitor RE-103. (I&C)

(KHNP Response) DCD 11.5 will include TGB north and south pit sumps radiation monitor. The response to RAI 244-8326 Q. 09.03.03-4 will be revised to state that the COL applicant is to provide the TGB north and south pit sumps radiation monitor.

2) DCD Section 10.4.10 states: "At the discharge of the condensate return pump, the condensate is monitored continuously for radioactivity. If contaminated, the radiation monitor actuates an alarm in the MCR and automatically diverts the radioactive or potentially radioactive condensate to the liquid radwaste system." However, in viewing Figure 10.4.10-1 (Sheet 2 and 3), the lines for the monitor are located prior to the condensate return pumps. (ME)

   (KHNP Response) It will be revised as follows;
   "At the upstream of the condensate return pump, the condensate is monitored continuously for radioactivity."

3) Also, monitor RE-103 in Figure 11.5-1 (Sheet 1 of 3) appears to be associated with the CCW system, "CC" and not the auxiliary steam system "AS" (this is inconsistent with the description above and what is provided in Chapter 11). (I&C)

   (KHNP Response) The condensate receiver tank belongs to auxiliary steam system. CCW lines shown in the Figure 11.5-1 (Sheet 1 of 3) contain the cooling water to cool down the condensate receiver tank outlet sample monitored by RE-103.

Other issues with the response to RAI 8254, Question 12.03-11:

4) In the response, KHNP proposes replacing paragraph c. on DCD page 9.2-37 with new text. This text includes the statement: "Liquid collected in the local sump of the CCW heat exchangers building can be monitored for radioactive contamination using the radiation monitor."
   The term "can be" implies that it is optional to perform radiation monitoring. Therefore, please explain if it is optional to perform radiation monitoring, or replace the words "can be" with more definitive wording. (NSYS)

   (KHNP Response) It will be revised as follows;
   "Liquid collected in the local sump of the CCW heat exchangers building is monitored for radioactive contamination using the radiation monitor."

5) Finally, the statement quoted in item 4 above and the response to RAI 8321, Question 09.02.02-9, imply that there is a radiation monitor in the CCW sump. However, in reviewing Chapter 11, it would appear that there is no monitor in the sump but instead there are monitors on the CCW supply header. KHNP should

clarify if there is a monitor in the sump or not. If not, the wording associated with these two responses should be modified to clarify that the radiation is monitored by the CCW Supply Header monitors and not by sump monitors. (I&C)

(KHNP Response) DCD 11.5 will include CCW heat exchanger building sump radiation monitor. The response to RAI 8321 Q 09.02.02-9 will be revised to state that the COL applicant is to provide the CCW heat exchanger building sump radiation monitor.