RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**225-8254**
SRP Section:	**12.03 – 12.04 – Radiation Protection Design Features**
Application Section:	**12.3 – 12.4**
Date of RAI Issue:	**09/24/2015**

Question No. 12.03-17

10 CFR 20.1101(b) requires that the licensee use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

SRP Section 12.3-12.4 indicates that the acceptability of the facility design features will be based on evidence that the applicant has fulfilled dose limiting requirements and that major exposure accumulating functions (maintenance, refueling, etc) have been considered in the plant design. It also indicates that the evidence should include radiation protection features incorporated into the design, taking into account the state of technology, that will keep potential radiation exposure from these activities ALARA in accordance with 10 CFR 20.1101(b). It states that such features include the ability to reduce source intensity and design measures to reduce the production, distribution, and retention of activated corrosion products.

Plants with high fluid temperatures and high surface heat flux at the fuel clad (high duty cores) have a portion of the total heat transfer to the coolant occur by sub-cooled nucleate boiling. This leads to more severe duty on the fuel and surface boiling which is known to enhance the formation of corrosion product deposits (crud) on the cladding surface. During the pre-application review, staff requested that the applicant identify if the APR1400 has a high duty core, as defined in EPRI Report 1008102, “PWR Axial Offset Anomaly (AOA) Guidelines.” In the pre-acceptance discussions, the applicant indicated that the APR1400 was a medium duty core, however based on the review of information in FSAR Chapter 4, staff calculates the core to be a high duty core.

Therefore, please provide information on design features to reduce crud buildup in the core or to reduce dose rates, monitor radiation levels, or reduce airborne activity levels during refueling. Please update the FSAR, as appropriate, to discuss design features that are not already discussed.

As an alternative, the applicant may justify why the APR1400 design is not susceptible to such crud deposits within the core.

Response

The High duty core can be classified using a calculation for the index of boiling duty according to PWR AOA (Axial Offset Anomaly) Guidelines from EPRI.

From the following equation, the high duty core index can be calculated.

HDCI(High Duty Core Index) = (peak assembly heat flux)*1000/[(assembly flow rate) $*(T_{sat} - \{T_x + T_{out}\})$]

Where,

peak heat flux = (core average heat flux)*(peak assembly power), in BTU/hr-ft^2 or W/m^2

assembly flow rate = (system flow rate at T_{cold})/number of assemblies, in gph or m^3/s

T_{sat} = saturation temperature at system pressure, in ºF or ºC

T_{out} = vessel temperature in the hot legs (ave.), in ºF or ºC

T_x = a temperature correction, 23°F or 12.8ºC

1000 = scaling factor applied to the numerator to present the index as a whole number

HDCI Values to determine high, medium and low duty plants are:

High Duty Plant: $\geq$150 BTU/ft^2-gal-°F

Medium Duty Plant: 120 - 149 BTU/ft^2-gal-°F

Low Duty Plant: $\leq$119 BTU/ft^2-gal-°F

The APR1400 HDCI can be calculated from the following inputs:

Input:

Core power = 3,983 MW_{th} (From DCD Table 4.4-1)

Number of assemblies = 241(From DCD Table 4.1-1)

Number of rods/assembly = 236(From DCD 4.2.2.3)

Active Fuel Height = 12.5 ft (From DCD Table 4.2-1)

Total flow = 446,300 gpm (From DCD Table 4.4-1)

Peak assembly power = 1.2353(From ROCS calculation result)

Vessel T_{out} = 615°F (From DCD Table 4.4-1)

T_{sat} @ 2250 psi = 653°F

Rod Outer Diameter = 0.374 inch (From DCD Table 4.2-1)

Derived Values:

Core power (BTU/hr) = 3,983*3,412,142 =1.359*E+10

Total fuel surface area (ft^2) = 241*236*12.5*0.374*π/12 = 69,611

Average heat flux (BTU/hr-ft^2) = 1.359*E+10/69611 = 195,235

Peak assembly heat flux (BTU/hr-ft^2) = 195235*1.2353 = 241,174

Assembly flow (gpm) = 446,300/241 = 1,852

HDCI = 241174*1000/[1852*60*(653-{23+615})] = **<u>145</u>** BTU/ft^2-gal-°F

The APR1400 has a HDCI value of 145 BTU/ft^2-gal-°F, which is within the range of a medium duty plant. Therefore, APR1400 design is not susceptible to crud deposits within the core that would be seen in a heavy duty plant.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environment Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**225-8254**
SRP Section:	**12.03-12.04 - Radiation Protection Design Features**
Application Section:	**12.3-12.4**
Date of RAI Issue:	**09/24/2015**

Question No. 12.03-22

REQUIREMENTS AND GUIDANCE

10 CFR 52.47(a)(5) requires that the FSAR contain the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR 20.

10 CFR 50, Appendix A, Criterion 61, requires that systems which may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions, with suitable shielding for radiation protection, and with appropriate containment, confinement, and filtering systems.

SRP Section 12.3-12.4 states that the plant should be subdivided into radiation zones with maximum design dose rate zones and the criteria used in selecting maximum dose rates identified. It also indicates that the assumptions and technics used for radiation shielding should be provided and that anticipated operational occurrences should be considered in the determination of plant shielding and zoning.

ISSUE

While FSAR Chapter 12 discusses nitrogen-16 (N-16) within the reactor coolant system and major sources include contribution of N-16, when relevant, the application does not discuss N-16 concentrations in chemical and volume control (CVCS) system piping leaving containment. A review of source term information reveals than N-16 is anticipated to be a significant contributor to purification filter dose during operation and N-16 is still noticeable in the purification ion exchanger source term during operation. Both of these components are located outside containment, in the auxiliary building. While, these components contain significant shielding to limit dose to surrounding areas, it is unclear that N-16 is appropriately considered in piping leaving containment, such as piping running from containment to the purification filter. Therefore, please provide the following

INFORMATION NEEDED

1. Please update the FSAR to provide the estimated time it takes for RCS fluid to travel from the reactor vessel to the nearest containment penetration during normal operation and the estimated N-16 concentration at that point.

2. Please indicate the location of piping running from the CVCS containment penetration to the purification filter and discuss if the shielding for these areas adequately considers N-16 and if there is margin for differences that may occur in piping lengths between the design calculations and actual physical pipe lengths.

3. Discuss if the CVCS piping penetration areas are anticipated to be accessed during normal operation.

Response

1. The N-16 activity in each of the heat exchangers and pipe segments is calculated using the transit time from the RV outlet nozzle to the component. The N-16 activity at the RV outlet nozzle is shown in DCD Table 12.2-5 and Table 12.2-7. The estimated transit time from the RV outlet nozzle to the inside containment isolation valves is conservatively estimated as follows:

   i. Transit time from RV outlet to regenerative heat exchanger (RHX) is assumed to be the same as the transit time from RV outlet to SG (midpoint, 3.85 seconds). Therefore N-16 concentration in inlet of RHX is taken to be 5.68E+06 Bq/g in DCD Table 12.2-7.

   ii. Residence time through RHX: 4.95 seconds

   iii. Transit time from RHX to letdown heat exchanger (LHX): no credit for decay time is taken in the shielding calculation

   iv. Residence time through LHX: 53.63 seconds

   v. Transit time from outlet of LHX to the outmost containment isolation valve before the containment penetration: no credit for decay time is taken in the shielding calculation

   The flow path of the fluid from the LHX outlet to the containment penetration is about 200 feet as shown in Figure 1A. For shielding design purposes, the transit time from the LHX to the containment penetration is not credited for N-16 decay, and the transit time from RV to LHX, 62.43 seconds, was taken for N-16 decay only.

   The estimated N-16 concentration at the exit of the LHX is determined to be 1.94E+04 Bq/g. For conservatism, the N-16 concentration at the containment penetration is assumed to be the same at the exit of LHX, i.e., 1.94E+04 Bq/g for determination of shielding along the paths to the purification filters and ion exchangers. Table 12.2-7 is updated with this concentration.

TS

Figure 1A Letdown pipe layout from the letdown heat exchanger to purification filter

2. The location of piping running (Red-colored pipe) from the containment penetration to the CVCS purification filter is depicted in Figure 2A and 2B below. In the shielding analysis, the letdown piping, in which N-16 is included in the letdown coolant, is included in the analysis for sizing of minimum shield wall thicknesses.

   As discussed in item 1 above, the N-16 concentration at the containment isolation valve for the shielding analysis is assumed to be the same as that at the letdown heat exchanger outlet, and no credit is taken for any further decay along the entire flow path to the purification filters. The shielding calculation also assumes that the contact dose is at a distance of 1-foot from the piping. Hence the shielding analysis basis is conservative and adequately compensates for any differences that may occur in piping lengths between the design calculations and the actual physical pipe lengths.

TS

Figure 2A Location of the letdown line piping running in Auxiliary Building (1)

TS

Figure 2B Location of the letdown line piping running in Auxiliary Building (2)

3. The CVCS letdown piping penetrates the containment at the mechanical penetration room (Room 120-A16B) at building elevation 120'. This room contains a cubicle cooler, which is safety-related. This cooler requires periodic surveillance for its functionality and entry to the mechanical penetration room during normal operation is anticipated. This letdown piping also passes through the mechanical penetration room (Room 100-A13B) at the 100' elevation before it is routed into pipe chases to the purification filters. This room also contains a cubicle cooler that is safety-related. Please refer to Figures 2A and 2B for flow paths of the CVCS piping.

   Both of these two mechanical penetration rooms are equipped with locked doors and entrances that are administratively controlled. Proper radiation signs are to be posted at the entrance. The cubicle coolers are strategically located to be close to the door entrance and as far away as possible from the CVCS piping, in order to minimize stay time for surveillance and for radiation ALARA purposes.

Impact on DCD

DCD Table 12.2-7 will be revised as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environment Report.

Table 12.2-7

N-16 Activity

Location	Activity (Bq/g)
Vessel outlet nozzle	8.22E+06
Vessel outlet line (midpoint)	8.13E+06
Steam generator (midpoint)	5.68E+06
Reactor coolant pump (midpoint)	3.78E+06
Vessel inlet line (midpoint)	3.61E+06

Replace this table with "A" in next page

A

Table 12.2-7

N-16 Activity

Location	Activity (Bq/g)	Accumulated Transit Time (sec)
Vessel outlet nozzle	8.22E+06	0.66
Vessel outlet line (midpoint)	8.13E+06	0.78
Steam generator (midpoint)	5.68E+06	3.85
Reactor coolant pump (midpoint)	3.78E+06	7.56
Vessel inlet line (midpoint)	3.61E+06	8.02
Containment penetration area	1.94E+04	62.43[1)]

[1)] The N-16 concentration at the containment penetration is estimated to be 1.94E+04 Bq/g based on the transit time from RV to SG and the residence times in the regenerative and the letdown heat exchangers before the containment penetration. For shielding analysis, this concentration is assumed to be the same as that at the outlet of the letdown heat exchanger, and no credit is taken for further decay from the letdown heat exchanger outlet to the containment isolation valve.