# RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**APR1400 Design Certification**

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

**Docket No. 52-046**

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| **RAI No.:** | **419-8517** |
| **SRP Section:** | **04.03 – Nuclear Design** |
| **Application Section:** | **04.03** |
| **Date of RAI Issue:** | **02/23/2016** |

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## Question No. 04.03-8

10 CFR Part 50 Appendix A, General Design Criterion (GDC) 10 requires the reactor core design to include appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation or anticipated operational occurrences (AOOs). GDC 13 requires provision of instrumentation and controls (I&C) to monitor variables and systems that can affect the fission process over anticipated ranges for normal operation, anticipated operational occurrences and accident conditions, and to maintain the variables and systems within prescribed operating ranges. GDC 20 requires automatic initiation of the reactivity control systems to assure that SAFDLs are not exceeded as a result of AOOs and that automatic operation of systems and components important to safety occurs under accident conditions.

To assess compliance with these requirements, Section 4.3 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) states that the reviewed information on core power distribution should demonstrate that "A reasonable probability exists that in normal operation the design limits will not be exceeded, based on consideration of information received from the power distribution monitoring instrumentation; the processing of that information, including calculations involved in the processing; the requirements for periodic check measurements; the accuracy of design calculations used in developing correlations when primary variables are not directly measured; the uncertainty analyses for the information and processing system; and the instrumentation alarms for the limits of normal operation (e.g., offset limits, control bank limits) and for abnormal situations (e.g., tilt alarms for control rod misalignment)."

The DIT/ROCS nuclear design code was first approved by the NRC staff in 1983 and again in 1988 as modified for use in analyzing cores with gadolinium burnable poison. The code documentation indicates that the DIT/ROCS nuclear data libraries for almost all nuclides, including gadolinium, were derived from Evaluated Nuclear Data File Volume B, Version IV (ENDF/B-IV). The staff notes however that ENDF/B-IV includes only elemental data for gadolinium and that data for the individual isotopes of gadolinium first appeared in a later release of ENDF/B-V. It is nevertheless clear that DIT/ROCS employs isotopic cross section data for gadolinium. Accordingly, it is not clear how the gadolinium cross section data used by DIT/ROCS can in fact be based on ENDF/B-IV.

Please clarify the source of the gadolinium isotopic data used by DIT/ROCS. The applicant should either insert the requested information into appropriate sections of the DCD or provide the information in a separate report that is cited in the DCD and included in the list of documents to be incorporated by reference.

## Response

As stated in Section 2.1.2.2 of CENPD-275-P-A, the gadolinium cross section data used by DIT/ROCS is based on the ENDF/B-IV data set. Section 3.3 of CENPD-266-P-A provides details on how the DIT library is obtained from the ENDF/B-IV data set. Two DIT fine energy group cross section libraries are available for design use, an 85 group and a 41 group library. The primary difference between the two libraries is in the condensation of the very high energy regions.

The 85 group library is usually used for DIT design calculations for PWR low enriched uranium cores. The 41 group library has been used for PWR low enriched uranium assemblies that include Gadolinia burnable absorbers. The reduced number of fine energy groups was originally driven by computer storage limitations since significantly finer spatial mesh is needed to properly model Gadolinia fuel pellets in the DIT flux solution and depletion calculations. As indicated in Section 3.3.5 of CENPD-266-P-A, the 41-group library has been tested against the basic 85-group library and has been shown to accurately reproduce reactivity levels, reactivity coefficients, power distributions and reaction rates. In PWR assembly calculations the differences in individual reaction rates are on the order of [ ]TS in magnitude for the two libraries. Infinite multiplication factors differ on the order of [ ]TS over a normal depletion range. Thus either library is acceptable for design use of assemblies with or without Gadolinia burnable absorbers.

The results of benchmarks to measurements described in Section 3.0 of CENPD-275-P-A and Sections 3.5 and 4.0 of CENPD-266-P-A demonstrate that the DIT/ROCS methodology and cross section are adequate for the purpose of nuclear core design analysis. (Note that CENPD-275-P-A and CENPD-266-P-A are References 9 and 5 of Chapter 4.3 of the APR1400 DCD.)

**Impact on DCD**

There is no impact on DCD.

**Impact on PRA**

There is no impact on PRA.

**Impact on Technical Specifications**

There is no impact on Technical Specifications.

**Impact on Technical/Topical/Environmental Report**

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

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## Question No. 04.03-9

10 CFR Part 50 Appendix A, General Design Criterion (GDC) 10 requires the reactor core design to include appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation or anticipated operational occurrences (AOOs). GDC 13 requires provision of instrumentation and controls (I&C) to monitor variables and systems that can affect the fission process over anticipated ranges for normal operation, anticipated operational occurrences and accident conditions, and to maintain the variables and systems within prescribed operating ranges. GDC 20 requires automatic initiation of the reactivity control systems to assure that SAFDLs are not exceeded as a result of AOOs and that automatic operation of systems and components important to safety occurs under accident conditions.

To assess compliance with these requirements, Section 4.3 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (SRP) states that the reviewed information on core power distribution should demonstrate that “A reasonable probability exists that in normal operation the design limits will not be exceeded, based on consideration of information received from the power distribution monitoring instrumentation; the processing of that information, including calculations involved in the processing; the requirements for periodic check measurements; the accuracy of design calculations used in developing correlations when primary variables are not directly measured; the uncertainty analyses for the information and processing system; and the instrumentation alarms for the limits of normal operation (e.g., offset limits, control bank limits) and for abnormal situations (e.g., tilt alarms for control rod misalignment).”

For ex-core monitoring of the core axial power distribution by the Core Protection Calculator System (CPCS), CENPD-170 discusses conversion of the ex-core detector responses to peripheral core power at three core rings and then to a 20-node axial shape using up to eight algorithm constants. These are apparently pre-calculated to represent flat-, saddle-, top-, or bottom-peaked axial shapes at various times over the fuel cycle. It appears that the CPCS uses some degree of pattern recognition on the 3-ring axial power distribution to determine which of the four types of power shapes are present and then uses a cubic spline fit to data.

Please explain how the algorithm constants are developed, whether they are cycle-dependent or bunup-dependent within one cycle, how the 20-node shapes are generated by the DIT/ROCS code and selected to represent a full range of allowed CEA positions and transient axial xenon effects, and how the shapes will be verified against in-core plant data. The explanations should include the addition of clarifying statements to discussion of axial power shapes in DCD Section 4.3 and may include pointers to any supporting details in various sections of the DCD and in referenced technical reports. The applicant should provide the requested information in the DCD itself or in a separate report that is cited in the DCD and included in the list of documents to be incorporated by reference.

## Response

Technical Report APR1400-F-C-NR-14001-P, Rev.0, "CPC Setpoint Analysis Methodology for APR1400" will be revised to incorporate the response to the above question as shown in the attachment. This Technical Report will be referenced in DCD Tier 2, section 7.2 in the response of RAI 328-8422 Q04.04-7.

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**Impact on DCD**

There is no impact on DCD.

**Impact on PRA**

There is no impact on PRA.

**Impact on Technical Specifications**

There is no impact on Technical Specifications.

**Impact on Technical/Topical/Environmental Report**

Technical Report APR1400-F-C-NR-14001-NP, Rev.0, "CPC Setpoint Analysis Methodology for APR1400" will be revised as shown in the attachment.

## APPENDIX C CPCS AXIAL POWER DISTRIBUTION ALGORITHM

The CPCS axial power distribution algorithm uses the measured SAM and BPPCC constants which are determined during the startup test at the site. The SAM and BPPCC constants are determined by using the least square fitting of startup test data, and are used for the whole cycle. Therefore, the SAM and BPPCC constants are cycle-dependent, but they are not burnup-dependent. They are installed using the plant measured data, and thus, further verification is not needed.

The SAM and BPPCC constants are determined by using the power distributions measured during power ascension testing in the BOC. So, the error in axial power distribution may be generated when the axial shapes are different from those of BOC such as flat, saddle, top or bottom peak axial shapes. This error is taken into account in Overall Uncertainty Factor by overall uncertainty analysis.

In overall uncertainty analysis, CPCS calculates the axial power distributions using the SAM and BPPCC constants which are simulated in the BOC for the 4,800 core conditions generated by reactor core simulator. Various power distribution (flat, saddle, top or bottom peak and so on) in table-C1 are used in the determination of the overall uncertainty factors for LPD and DNB-OPM. Therefore, the difference in axial shapes between reactor core simulator and CPCS are considered statistically in Overall Uncertainty Factor.

**Table C-1 Core Conditions of Axial Power Distribution for Overall Uncertainty Analysis** TS