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.:	**387-8485**
SRP Section:	**15.00.02 – Review of Transient and Accident Analysis Methods**
Application Section:	**15.00.02**
Date of RAI Issue:	**02/01/2016**

Question No. 15.00.02-1

Transient pressure options in HERMITE 1.6 calculations for DCD Section 15.3 events

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 10 requires the core and associated coolant, control, and protection systems to be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects or anticipated operational occurrences (AOOs). GDC 20 requires, in part, that the protection system be designed to initiate automatically the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of AOOs. GDC 25 requires the protection system to be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

Section 15.0.2 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Subsection III.6, states that the following attributes of the evaluation model should be considered when determining the extent to which the full review process may be reduced for a specific application: (a) Novelty of the revised evaluation model compared to the currently acceptable model, (b) The complexity of the event being analyzed, (c) The degree of conservatism in the evaluation model, and (d) The extent of any plant design or operational changes that would require a reanalysis.

Technical report APR1400-Z-A-NR-14006-P indicates that more accurate methods and additional features were added to HERMITE. The following statement is made in the report (Section 3.7.4): “HERMITE code was initially approved by NRC in 1976 and the CE-Methodology with several improvements was approved in 1992 in the Amendment No. 61 of NPF-41. HERMITE code version in 1992 was 1.5 while current version is 1.6 with added transient pressure option. There is no significant methodology change for this code after the last approval by NRC in 1992, thus this code is applicable to the transient and accident events for the DCD, Tier 2 Chapter 15.” The HERMITE User’s Manual [CE-CES-091-P, Rev. 4] confirms this code change.

Reference VV-FE-0416, Rev. 0, provides the software verification and validation of HERMITE 1.6. Section 3.0 of this document describes the code changes necessary to implement the transient pressure option. Section 5.0 describes eleven test cases used to verify this new feature, and the results indicate that the transient pressure option is working as expected. The results of the independent software review were included in Appendix A. There are two transient pressure options available in HERMITE for inputting pressure versus time tables. The first is an input of pressure ratios relative to the input system pressure, and the second is an input of pressure differences relative to the input system pressure.

However, the applicant has not described how, or if, these new code options are used in the analysis of Section 15.3 events. The staff is therefore concerned that evaluation models for these events have not been adequately described.

Please provide details regarding the use of the transient pressure options in HERMITE 1.6 and their impacts on the analyses of Section 15.3 events. As appropriate, the applicant should ensure that the requested information is either added to the DCD itself or else provided in supporting documents that are docketed and listed for incorporation by reference.

Response

The transient pressure options in HERMITE 1.6 are not used in the analysis of DCD Section 15.3 events because the reactor coolant system (RCS) pressure is continuously increased until a reactor trip occurs and the RCS pressure is much higher than its initial value at the time of minimum DNBR. This assumption is conservative for the minimum DNBR calculation.

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. 15.00.02-2

Calculation of 1-D cross sections for HERMITE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 10 requires the core and associated coolant, control, and protection systems to be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects or anticipated operational occurrences (AOOs). GDC 20 requires, in part, that the protection system be designed to initiate automatically the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of AOOs. GDC 25 requires the protection system to be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

Section 15.0.2 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Subsection III.6, item (A) states that the documentation should include: “An overview of the evaluation model which provides a clear roadmap describing all parts of the evaluation model, the relationships between them, and where they are located in the documentation.”

For the 1-D HERMITE solution described in Section 3.5.2.1 of the technical report APR1400-Z-A-NR-14006-P, cross sections are used that incorporate all reactivity feedback mechanisms including xenon. These cross sections are stated to have been derived from the 3-D ROCS solution at each axial level. However, it is not clear what flux or reaction weightings were used in collapsing the cross sections from 3-D to 1-D. The staff is therefore concerned that the evaluation model has not been adequately described in this regard.
Please provide details on the calculation of 1-D cross sections used for HERMITE, including how these cross sections are flux-weighted and collapsed from the 3-D cross sections obtained from the ROCS solution. As appropriate, the applicant should ensure that the requested information is either added to the DCD itself or else provided in supporting documents that are docketed and listed for incorporation by reference.

Response

The details on the calculation of 1-D cross sections used for HERMITE are as follows;

1. Macroscopic Cross Section

The planar average values for the 2-group macroscopic cross sections and nuclide concentrations are calculated such that these constants, when used in a 1-D or point model, will preserve the total planar reaction rates. Thus, for plane K

$$R_{1D}^{K} = R_{3D}^{K} \qquad 1.1$$

$$\sum_K \phi_K = \sum_{i=1}^{N} \sum_{Ki} \phi_{Ki} \qquad 1.2$$

Where N represents the number of active core nodes in plane K.
Assuming that there is no difference between the planar average fluxes, between the 1D and the 3D, then

$$\phi_K = \frac{\sum_{i=1}^{N} \phi_{Ki} V_{Ki}}{\sum_{i=1}^{N} V_{Ki}} \qquad 1.3$$

So the macroscopic cross sections must be

$$\sum_K = \frac{\sum_{i=1}^{N} \sum_{Ki} \phi_{Ki}}{\phi_K} \qquad 1.4$$

Similarly, a core averaged macroscopic cross section can be defined so as to preserve the total core reaction rate by the following:

$$\bar{\sum} = \frac{\sum_{K=1}^{NP} \sum_K \phi_K V_K}{\sum_{K=1}^{NP} \phi_K V_K} \qquad 1.5$$

2. Representation of Radial Neutron Leakage

The radial leakage of neutrons is represented by calculating the planar average value of DB^2 given as:

$$\overline{DB_K^2} = \frac{\sum_{i=1}^{N} \alpha_i^{XY} \phi_i V_i}{\Delta x \sum_{i=1}^{N} \phi_i V_i} \qquad 2.1$$

Where α_i^{XY} is total radial ROCS type boundary condition (J_{i+}/ϕ_i) for node i,
ϕ_i is the flux in node i, and
Δx is the node pitch in cm.

From the planar average value of DB^2, the planar radial buckling is calculated by

$$B^2 = DB^2/D_K \qquad 2.2$$

The above details are quoted from “User’s Manual for CEKLAPS" (CE-CES-048 Rev. 3-P, August 1994). KHNP will provide the document referred in this information upon request.

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

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Question No. 15.00.02-3

Verification of ROCS-NEM in relation to ROCS-HOD

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 10 requires the core and associated coolant, control, and protection systems to be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects or anticipated operational occurrences (AOOs). GDC 20 requires, in part, that the protection system be designed to initiate automatically the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of AOOs. GDC 25 requires the protection system to be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

Section 15.0.2 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Subsection III.6, indicates that the following attributes of the evaluation model should be considered when determining the extent to which the full review process may be reduced for a specific application: (a) Novelty of the revised evaluation model compared to the currently acceptable model, (b) The complexity of the event being analyzed, (c) The degree of conservatism in the evaluation model, and (d) The extent of any plant design or operational changes that would require a reanalysis.

Report CENPD-266-P-A documents the ROCS/DIT code system. In that report, ROCS was approved for use as a PWR core design and analysis code by the NRC. In addition, DIT and its associated data library were found to be acceptable for the generation of few-group cross sections to predict the physical processes important for PWR design. The regulatory position on the ROCS/DIT system in 1983 is summarized as follows: “We have reviewed the ROCS and DIT computer codes as described in CENPD-266-P and CENPD-266-NP and find them to be acceptable for nuclear core design and safety-related neutronics calculations made by CE in licensing actions for power distributions, control rod worths, depletion, reactivity coefficients and reactivity differentials. We also conclude that the ROCS code, including the fine mesh module MC, is of sufficient accuracy for the generation of coefficient libraries for the in-core instrumentation. The staff, however, recommends that CE perform further verification when the NEM is incorporated into the ROCS code in order to be assured that equivalent calculational biases and uncertainties are obtained with ROCS-NEM as compared to ROCS-HOD."

Table 4.1-4 of the DCD indicates that the Nodal Expansion Method (NEM) was used for APR1400 design and analysis. This is confirmed based on Section 4.3.3.1.1.2 of the DCD, which states the following: "The ROCS program is designed to perform two-dimensional or three-dimensional coarse-mesh reactor core calculations based on a two-group nodal expansion method, with full-core, half-core, or quarter-core symmetric geometries." Based on the regulatory position quoted above, further verification of the ROCS-NEM code is needed.

Please provide verification of the ROCS-NEM solver and discuss the impacts relative to the ROCS-HOD solver. In addition, please provide verification that "equivalent calculational biases and uncertainties are obtained with ROCS-NEM as compared to ROCS-HOD." As appropriate, the applicant should ensure that the requested information is either added to the DCD itself or else provided in supporting documents that are docketed and listed for incorporation by reference.

Response

The Nodal Expansion Method (NEM) was added to the ROCS code as an alternative to the original Higher Order Difference (HOD) formulation. The ROCS code provides reactor power distributions and effective neutron multiplication factors. These data are then used to derive control rod worth, depletion, reactivity coefficients, and reactivity differentials. Use of the NEM achieves significant reduction in computing run times and also improves agreement with measurement data. Although the NEM had not been fully integrated into the ROCS code in 1983, the use of NEM was fully described in CENPD-266-P that was approved by the NRC. Specially, CENPD-266-P explained that NEM had been incorporated into a version of coarse-mesh kinetics code, HERMITE. Furthermore, CENPD-266-P presented numerical comparisons of the NEM and HOD methods for solving the neutron diffusion equations. The results showed that the use of NEM did not have an adverse impact on calculation results and uncertainties.

The DIT/ROCS code's analytical design methods based on NEM were checked against a variety of critical experiments such as KRITZ and operating power reactors as described in subsection 4.3.3.1.2 of the DCD. The comparison of predicted values served not only to verify the code's analytical design methods, but also to provide a set of biases and uncertainties. The bias and uncertainty manual, which is given as Reference 12 (CE-CES-129 Revision 9-P) in DCD Section 4.3 contains a set of biases and uncertainties for the APR1400 core design. And these bias and uncertainty factors were determined with ROCS-NEM by equivalent statistical analysis as compared to that in CENPD-266-P. Also the equivalent biases and uncertainties were obtained with ROCS-NEM as compared to ROCS-HOD. By equivalent, it is understood that the results between the two methods need not be numerically identical, but rather that the biases and uncertainties associated with the ROCS-NEM are not significantly larger than those of ROCS-HOD when such bias and uncertainties are evaluated on the same conservative basis, a 95/95 tolerance limit. Table 1 shows comparisons of biases and uncertainties between ROCS-HOD solver and ROCS-NEM solver.

In conclusion, CENPD-266-P presented numerical comparisons of the NEM and HOD methods and the results showed that the use of NEM did not have an adverse impact on the calculation results and uncertainties. Also, CE-CES-129 contains a set of biases and uncertainties for the APR1400 core design and these bias and uncertainty factors were determined with ROCS-NEM and the equivalent biases and uncertainties were obtained with ROCS-NEM as compared to ROCS-HOD. Thus, CENPD-266-P and CE-CES-129 documents support the verification results of the ROCS-NEM solver and equivalent biases and uncertainties relative to the ROCS-HOD solver.

Table 1 Summary of DIT/ROCS Biases and Uncertainties

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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

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Question No. 15.00.02-4

MCXSEC code for generating macroscopic cross sections

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 10 requires the core and associated coolant, control, and protection systems to be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects or anticipated operational occurrences (AOOs). GDC 20 requires, in part, that the protection system be designed to initiate automatically the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of AOOs. GDC 25 requires the protection system to be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

Section 15.0.2 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Subsection II.1, Item (A) states that the documentation must include “An overview of the evaluation model which provides a clear roadmap describing all parts of the evaluation model, the relationships between them, and where they are located in the documentation.”

In DCD Section 4.3.3.1.1.3, the following statement is made: “The ROCS embedded calculation uses a macroscopic cross-section model based on interpolation of multi-dimensional macroscopic tables. These tables are created by the MCXSEC code, which processes DIT results for all assembly types, and are typically burnup, enrichment, moderator, and fuel temperature dependent for each fine-mesh pin cell type.” However, the applicant has not provided adequate details on the MCXSEC code and its use with the DIT and ROCS codes to allow the staff to complete its review of this portion of the evaluation model.

Please provide details on the MCXSEC code as it used to generate macroscopic cross sections for ROCS based on results from DIT. As appropriate, the applicant should ensure that the requested information is either added to the DCD itself or else provided in supporting documents that are docketed and listed for incorporation by reference.

Response

MCXSEC prepares fine mesh cross section tables for the ROCS code using files saved by the DIT transport code.

The MCXSEC code generates fine mesh macroscopic cross section tablesets for ROCS embedded calculation using files saved by one or more DIT cases. The DIT code performs multi group transport assembly calculations and will produce upon request an output file specifically for MCXSEC containing 2-group mesh region averaged cross sections. The tablesets are multidimensional and can be functionalized on a variety of parameters.

The ROCS code (MC module) uses macroscopic cross sections for all mesh regions. In general, MC uses multidimensional interpolation tables, called tablesets, to obtain these cross sections for a given fine mesh geometry. Multiple cross section tables are allowed. For example, an unshimmed, a shimmed, and a rodded tableset might all be used in a given job to account for all the different types of fuel assemblies in a given core.

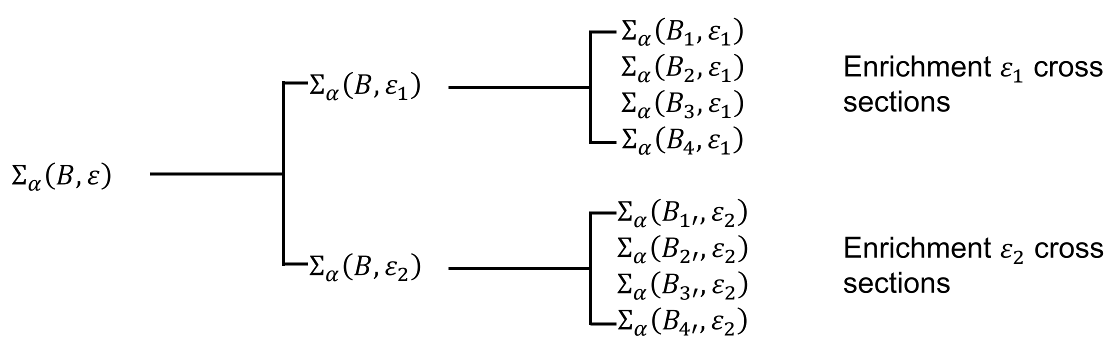
The tablesets will almost always be both burnup and enrichment dependent. The burnup variable will be in units of megawatt-days per metric ton for fueled regions, and integrated absorptions (abs/cm2) for non-fueled mesh regions (e.g., waterholes and shims). The cross section interpolation is performed by multidimensional, piecewise linear interpolation. The following formula illustrates the one-dimensional form of this interpolation for a value Σ(x) between two points x1 and x2:

$$\Sigma_\alpha(x) = \left(\frac{x_2 - x}{x_2 - x_1}\right)\Sigma_\alpha(x_1) + \left(\frac{x_1 - x}{x_1 - x_2}\right)\Sigma_\alpha(x_2)$$

Extrapolation using the same formula outside the range [x1, x2] is employed.

Except for the burnup variable, only two values of the independent variable are used. For the burnup variable, several interpolation points are normally used. Because the DIT depletions are run at two different enrichments, two somewhat different sets of burnup points result. The interpolation scheme must take this into account.

As an example, a table that is both burnup and enrichment dependent would follow the scheme diagrammed in Figure 1 below. Interpolation proceeds from right to left in the figure.



where

ε	enrichment of mesh region
ε_1	first enrichment value in cross section table
ε_2	second enrichment value in cross section table
B	burnup of mesh region
$B_2 \leq B \leq B_3$	the two values of burnup B_2 and B_3 bracket the value B for this mesh region within enrichment ε_1
$B_{1'} \leq B \leq B_{2'}$	the two values of burnup $B_{1'}$ and $B_{2'}$ bracket the value B for this mesh region within enrichment ε_2 (Note that in general, $B_i \neq B_{i'}$)

Figure 1. A Typical Interpolation Scheme

Mathematically the figure implies the following interpolation formula:

$$\Sigma_\alpha(B,\varepsilon) = \left(\frac{\varepsilon_2-\varepsilon}{\varepsilon_2-\varepsilon_1}\right)\left[\left(\frac{B_3-B}{B_3-B_2}\right)\Sigma_\alpha(B_2,\varepsilon_1) + \left(\frac{B_2-B}{B_2-B_3}\right)\Sigma_\alpha(B_3,\varepsilon_1)\right] + \left(\frac{\varepsilon_1-\varepsilon}{\varepsilon_1-\varepsilon_2}\right)\left[\left(\frac{B_{2'}-B}{B_{2'}-B_{1'}}\right)\Sigma_\alpha(B_{1'},\varepsilon_2) + \left(\frac{B_{1'}-B}{B_{1'}-B_{2'}}\right)\Sigma_\alpha(B_{2'},\varepsilon_2)\right]$$

where the values $\Sigma_\alpha(B_2,\varepsilon_1)$, $\Sigma_\alpha(B_3,\varepsilon_1)$, $\Sigma_\alpha(B_{1'},\varepsilon_2)$, $\Sigma_\alpha(B_{2'},\varepsilon_2)$, ε_1, ε_2, B_2, B_3, $B_{1'}$, and $B_{2'}$ are obtained from the table.

MCXSEC obtains its cross section input from DIT cross-section files. In most cases cross section tablesets will be at least enrichment and burnup dependent. This usually means that two DIT "base depletion" calculations will be performed. These calculations will be for identical assembly designs. The only difference will be in the fuel pin enrichments. Each one of these calculations will produce one DIT file for MCXSEC. More complicated tablesets will require DIT restart calculations. For example, if the tableset is to be functionalized on enrichment, control rod insertion, and burnup, several DIT rodded restart calculations will be performed. These calculations are done from several, but not all time points in the base depletion. One variable (e.g. control rod position) is changed at a given time point while all other conditions (fuel temperature, moderator temperature, etc.) are left the same. The cross sections from the restart calculations are usually written to the end of the same file that contains the base depletion cross sections. Thus a DIT file will generally consist of a set of base depletion cross sections and several sets of DIT restart cross sections. MCXSEC will automatically sort the restart cross sections so that the cross sections data is in the order needed for its calculations.

The above details on the MCXSEC code are quoted from “User’s Manual for MCXSEC" (CE-CES-65 Rev. 4-P, January 1997). KHNP will provide the document referred in this information upon request.

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

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