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.:	**419-8517**
SRP Section:	**04.03 - Nuclear Design**
Application Section:	**04.03**
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Question No. 04.03-7

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).”

DCD Section 4.3.3.1.1.4 briefly describes how fixed-source adjoint neutron transport calculations are performed with the MCNP (Monte Carlo, N-particle) code to determine the axial shape annealing functions used by the Core Protection Calculator System (CPCS). The staff has not previously reviewed this use of MCNP. The staff needs to evaluate how the MCNP code is used and verified in support of the power distribution monitoring system of the CPCS. During an audit discussion of this topic on January 20, 2016, the applicant displayed an internal technical report that appears to include much or all of the information needed for the staff's review.

Please submit for NRC review a technical report that details and verifies the use of the MCNP code to calculate shape annealing functions in support of the CPCS. The applicant should cite the report in the DCD and either (a) add the report to the list of documents to be incorporated by reference, or (b) insert a detailed summary into an appropriate section of the DCD.

Response

KHNP will submit the technical report for NRC review. The technical report will also be cited in the DCD and a summary will be inserted into Section 4.3.3.1.1.4 of the DCD.

Impact on DCD

Section 4.3.3.1.1.4 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 the Technical/Topical/Environmental Report.

shapes corresponding to the LCO limits on axial power shape from which simulations of core transients are subsequently initiated.

CEA shadowing is the change in ex-core detector response resulting from changing the core configuration from an unrodded condition to a condition with CEAs inserted, while maintaining constant power operation. Although CEA shadowing is a function of the relative azimuthal locations of the higher-power peripheral assemblies and the ex-core detectors, its effect is minimized by placing the ex-core detectors at azimuthal locations where minimum CEA shadowing occurs. CEA shadowing factors can be determined using detailed two- or three-dimensional power distributions calculated by ROCS, representing the cumulative presence of the various CEA banks and the DORT code (Reference 11).

Normalized CEA shadowing factors are relatively constant with burnup and power level changes made without moving CEAs. CEA shadowing factors at the beginning and end of the first cycle are as shown in Table 4.3-12.

Shadowing factors account for the radial power distribution effects and the shape-annealing function acc
Each detecto
from the sec
various regio
of the power
functions, given as fractional response per percent of core height for a three-subchannel system, are shown in Figure 4.3-44.

Axial shape-annealing functions are defined as the fractional contributions of each horizontal slice of core to the ex-core detector responses and they are used to determine shape annealing matrix in CPC for predicting core power distributions. Shape-annealing functions can be determined using particle transport code by calculating ex-core detector responses due to the fission neutrons from each core axial location.

~~Axial~~ The shape-annealing functions are determined utilizing a fixed-source adjoint MCNP calculation. MCNP (Reference 21) is a Monte Carlo N-particle transport code. The three-dimensional geometry model is used for the annealing calculation with representation of the core, ~~vessel~~ reactor internals, vessel ~~internals~~, air gap, ex-core detector, and biological shield. Three adjoint MCNP calculations are performed with an adjoint source in each ex-core detector subchannel. Figure 4.3-45 illustrates the MCNP calculation model used.

The subchannel responses for each axial slice of reactor core can be determined by integrating the fission spectrum weighted adjoint fluxes ascribed to the adjoint source located at one of the subchannels. The annealing curves shown in Figure 4.3-44 are determined using this same technique for other subchannels and normalizing the responses to the total ex-core detector response of all three subchannels.

Because the annealing curve is determined regardless of the axial power shape, the resulting annealing functions, S(z), is multiplied by the peripheral axial power distributions, P(z), to obtain the ex-core detector responses for each subchannel as follows:

$$D_{lower} = \int_0^H P(z)S(z)_{lower}\,dz = \text{Lower detector response} \quad \text{(Eq. 4.3-5)}$$

$$D_{middle} = \int_0^H P(z)S(z)_{middle}\,dz = \text{Middle detector response} \quad \text{(Eq. 4.3-6)}$$

$$D_{upper} = \int_0^H P(z)S(z)_{upper}\,dz = \text{Upper detector response} \quad \text{(Eq. 4.3-7)}$$

The shape-annealing functions are essentially geometric correction factors applied to the peripheral axial power distribution. As such, the effects of time in fuel cycle, transient xenon redistribution and CEA insertion, although affecting the peripheral bundle power shape, do not affect the geometric shape-annealing correction factors.

The ex-core detector temperature decalibration effect is the relative change in detector response as a function of reactor water inlet temperature at constant power. The temperature de code (Reference core detector f detector temperature decalibration effects as a function of inlet temperature normalized to an inlet temperature of 290.6 °C (555 °F).

The application of three-dimensional MCNP code for the shape-annealing function calculations has been tested successfully during startup physics tests of the operating plants. The detailed description for the validation of MCNP application on shape-annealing functions is given in Reference 22.

Final normalization of the CEA shadowing factors, shape-annealing functions, and temperature decalibration constants is accomplished during startup testing.

4.3.3.1.2 Comparisons with Experiments

The nuclear analytical design methods in use for the APR1400 were checked against a variety of critical experiments and operating power reactors. In the first type of analysis, reactivity and power distribution calculations were performed, which produced information concerning the validity of the basic fuel cell calculation. The second type of analysis consisted of a core-follow program in which power distributions, reactivity coefficients,

10. CENPD-188-A, “HERMITE, A Multi-Dimensional Space-Time Kinetics Code for PWR Transients,” Combustion Engineering Inc., July 1976.

11. Oak Ridge National Laboratory, “One-Dimensional, Two-Dimensional and Three-Dimensional Discrete Ordinates Transport Code System,” CCC-650/DOORS3.2, RSICC, 1998.

12. “Methodology Manual – Physics Biases and Uncertainties,” CE-CES-129 Revision 9-P, March 2003.

13 E. G. Taylor, et al., “Combustion Engineering, Inc., Critical Experiments,” WCAP-7102, October 16, 1967.

14. P. H. Gavin and H. G. Joo, “Evaluation of Assembly Peaking Factors by DIT with P1 Scattering and DP2 Cell Coupling,” PHA-88-109, November 2, 1988.

15. CENPD-145, “A Method of Analyzing In-Core Detector Data in Power Reactor,” Combustion Engineering, Inc., April 1975.

16. CENPD-153-P, “INCA/CECOR Power Peaking Uncertainty,” Rev. 1-P-A, Combustion Engineering, Inc., May 1980.

17. C. G. Poncelot, “The Effect of a Finite Time Step Length on Calculated Spatial Xenon Stability Characteristics in Large PWRs,” Trans. ANS, 10, 2, p 571, 1967.

18. A. Gruen, “Meassung Physikalischer Kenngroessen an Leistungsreakoren,” Atomkernenergie, 25, 2, 1975.

19. D. T. Ingersoll, et al., “Production and Testing of the VITAMIN-B6 Fine-Group and the BUGLE-93 Broad Group Neutron/Photon Cross section Libraries Derived from ENDF/B-VI Nuclear Data,” DLC-175, ORNL-6795, January 1995.

20. KNF-TR-ND1-05001, “Evaluation of Fuel Temperature Correlation of PLUS7 for Korean Standard Nuclear Power Plants,” Rev. 0, KEPCO Nuclear Fuel Co., Ltd., March 2005.

21. Los Alamos National Laboratory, “MCNP-A General Monte Carlo N-Particle Transport Code, Version 5-1. 40,” LA-CC-02-083, November 2005.

22. “Validation of MCNP Application on SAF Using Test Results from SKN 1,” KEPCO E&C/ND/TR/16-001, KEPCO Engineering and Construction Company, Inc., March 2016.