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.:	**55-7940**
SRP Section:	**04.03 – Nuclear Design**
Application Section:	**4.3**
Date of RAI Issued:	**06/29/2015**

Question No. 04.03-3

RAI 4.3-2, Use of ENDF/B-IV cross section for nuclear design

REQUIREMENTS

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 11, "Reactor Inherent Protection," requires that, in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity. GDC 20, "Protection System Functions," 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. In addition, GDC 28, "Reactivity Limits," requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor cause sufficient damage to impair significantly the system’s capability to cool the core.

In accordance with these regulations, a reactor design must include various protection systems and mitigation functions. The regulations also prescribe specific transient analyses to assess the system’s performance under AOOs and design basis accident conditions. The majority of the system protection design and transient analyses depend on accurate nuclear analyses, such as power distribution, the Doppler coefficient, moderator temperature coefficients, and control rod worths.

To achieve these goals, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition – Reactor," Section I, item 2 requires the reviewer to review core power distributions, including normal and extreme cases for steady-state and allowed load-follow transients and covering a full range of reactor conditions of time in cycle, allowed control rod positions, and possible fuel burnup distributions. In addition, the SRP requires the reviewer to examine the applicant's presentation of the core power distributions as axial, radial, and local distributions and peaking factors to be used in the transient and accident analyses. Power distributions within fuel pins is also required as discussed in Regulatory Guide (RG 1.206).

Furthermore, Item 2.F of the SRP requires the reviewer to review measurements in previous reactors and critical experiments and their use in the uncertainty analyses and the measurements to be made on the reactor under review, including startup confirmatory tests and periodically required measurements. And item 2.G requires the reviewer to examine the translation of design limits, uncertainties, operating limits, instrument requirements, and setpoints into technical specifications.

With respect to core nuclear design, the SRP requires the reviewer to examine the correctness and accuracy of the reactivity coefficient calculations and results. Specifically, the SRP indicates that the areas of concern with respect to reactivity coefficients are : "The applicant's presentation of calculated nominal values for the reactivity coefficients, such as the moderator coefficient, which involves primarily effects from density changes and takes the form of temperature, void, or density coefficients; the Doppler coefficient; and power coefficients. The range of reactor states to be covered includes the entire operating range from cold shutdown through full power and the extremes reached in transient and accident analyses."

Regarding control rod design, the SRP requires the reviewer to examine parameters and items such as control rod patterns and reactivity worths throughout the core life, misaligned rods, stuck rods, or rod positions used for spatial power shaping, maximum worths of individual rods or banks as a function of position for power and cycle life conditions appropriate to rod withdrawal transients and rod ejection or drop accidents. The SRP also requires the staff to examine descriptions and graphs of scram reactivity as a function of time after scram initiation and other pertinent parameters and shutdown margin.

ISSUES

All of these parameters, together with many others, are typically calculated in nuclear design using computer codes. The applicant states in Section 4.3 of the APR1400 Design Control Document (DCD) that it has performed reactor nuclear design analyses using the ROCS/DIT codes and nuclear cross section data from the ENDF/B-IV library with some adjustments based on ENDF/B-V data. The staff notes that ENDF/B-IV and ENDF/B-V were developed in 1974 and 1978 respectively and several significant revisions and improvements have been made since then to address the differences identified in the data for many important nuclides, including U-235, Pu-239, Pu-241, and gadolinium.

To assess the impact of different cross section libraries on the core nuclear calculations, the staff first compared the measured values of the total absorption cross section of gadolinium provided in ENDF/B-IV with the data published in the later versions of the ENDF/B libraries. From the comparison, the staff notes large differences between the ENDF/B-IV and the later versions for several important isotopes, particularly for gadolinium.

In order to access the overall impact of the cross section data, excluding gadolinium, the staff also performed a basic criticality benchmark analysis using the SCALE 6.1 computer code for the low enrichment critical experiment, "LEU-COMP-THERM-001-001," from the International Handbook of Evaluated Criticality Safety Benchmark Experiments. The results show that the calculated k_{eff} value using ENDF/B-IV library is 0.00824 (0.824%) lower than the results using ENDF/B-VII continuous energy cross section library. This reactivity difference is about $1.32 in terms of β_{eff}.

In addition, the staff performed a preliminary confirmatory analysis on a single unpoisoned bundle of the APR1400 fuel (PLUS7 fuel). Based on the analyses result, the staff found that the differences between the calculated k_{inf} values using ENDF/B-V and ENDF/B-VII libraries are significant. Since cross sections are the basic values for all nuclear design, inaccuracies in these data will directly affect the calculated power distribution, reactivity, reactivity coefficients, control rod worth and other parameters that are related to reactor safety.

INFORMATION NEEDED

The applicant is requested to demonstrate that using ENDF/B-IV library can adequately predict the APR-1400 reactor physics parameters with sufficient accuracy to ensure that the values used in the safety analyses, transient analyses, and accident analyses in other chapters produce conservative results. The applicant should also evaluate the additional uncertainties of the reactor operating parameters introduced by the deficiencies the ENDF/B-IV cross section library. The parameters should include, but are not limited to:

1. Power distributions and peaking factors;

2. Differential and integral control rod worths;

3. Shutdown CEA reactivity and reactivity shutdown margin;

4. Doppler coefficients; and

5. Moderator temperature and density coefficients.

Given the fact that these parameters are important to the transient and accident analyses, the applicant is requested to evaluate the subsequent impacts on the following transient and accident analyses and update the relevant technical reports and DCD as appropriate:

1. Uncontrolled Control Element Assembly Withdrawal from a Subcritical or Low-Power Startup Condition;

2. Uncontrolled Control Element Assembly Withdrawal at Power;

3. Control Element Assembly Misoperation;

4. The types of AOOs that include one or more CEAs moving or displaced from normal or allowed control bank positions are as follows:

    a. Dropped CEA or CEA subgroup;

    b. Statically misaligned CEA;

    c. Single CEA withdrawal;

5. Identification of Causes and Frequency Classification;

6. Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position; and

7. Anticipated transient without scram (ATWS) as it relates to reactivity insertion.

The staff needs this information to determine the APR1400 nuclear design meets the regulatory requirements of 10 CFR Part 50 and Part 52 and GDC 10, 11, 20, 28 and other design criteria related indirectly to the nuclear design.

Response

The best code system is intended to represent the experimental data as close as possible in terms of total core reactivity, reactivity coefficients, rod worth, power peaking factors, etc. only if the proper data library is used. The best estimated physics parameters are achievable with the proper nuclear data library and associated biases. Also, adequate conservatism is incorporated into the calculated values through the application of proper bias and uncertainty system associated with a specific ENDF data library version. For example, in CE-CES-129 Rev. 9-P (Reference 12 in DCD Section 4.3.6), the bias and uncertainty factors pertinent to ROCS using ENDF/B-VI appear in Appendix G, which are applicable to Calvert Cliffs Units only, while the bias and uncertainty factors for ROCS using ENDF/B-IV appear through the main body of the manual, which are applicable to all plants with some exceptions. That is, the bias and uncertainty factors of each revision are unique for each cross section library. For example, the biases for FTC are 4.1% and 1.3% for ENDF/B-IV and ENDF/B-VI respectively. For the safety analysis, the calculated values of safety parameter are not used directly, but the limiting values of each parameter are used, which are confirmed by applying the bias and uncertainty factors to the calculated values.

Figure 1 shows the differences between measured ITCs[1] and predicted ITC with cross section data based on ENDF/B-IV(of DIT/ROCS) and ENDF/B-VI(of PARAGON/ANC[2]) for OPR1000 plants. Figure 1 also shows the differences between measured ITCs and predicted ITCs using cross section data based on ENDF/B-IV(of DIT/ROCS) for C-E plants. The graph indicates that the predicted ITCs and measured ITCs show good agreement within the test acceptance criteria range regardless of the version of ENDF library.

Since this plot shows that there is sufficient accuracy in predicted ITCs with ENDF/B-IV library as well as ENDF/B-VI library, it was determined that using ENDF/B-IV library can adequately predict the reactor physics parameters of APR1400 with sufficient accuracy. Moreover, the fact that uncertainties from a new code system or cross section library are nearly the same to those of an existing code system or cross section library from the standpoint of statistical methodology or design experience when assuming new biases were properly produced guarantees the same conservatism with an uncertainties from later version of ENDF/B cross-section library.

More examples of ENDF/B-IV validation can be found in the Response to RAI No. 47-7959.

Since the current biases and uncertainties system supports appropriate prediction for nuclear design and safety analysis with sufficient accuracy, and thus, there is no deficiency inherent in ENDF/B-IV for physics analysis, there is no additional uncertainty regarding ENDF version and the evaluation of uncertainty is not necessary.

Note

1. It is an abbreviation of Isothermal Temperature Coefficient.
2. PARAGON/ANC code system has been used for nuclear design of OPR1000 reload cycles since April, 2009.

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Figure 1. ITC Difference (Meas. – Pred.) vs. Boron Concentration

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 Report

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