DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT ANP-10249NP, REVISION 0

"ACE/ATRIUM-10 CRITICAL POWER CORRELATION"

AREVA NP, INC. (AREVA)

PROJECT NO. 728

1.0 INTRODUCTION AND BACKGROUND

Topical Report (TR) ANP-10249P, Revision 0 (Reference 1), describes a new correlation developed by AREVA to predict the critical power for boiling water reactors (BWRs). The new correlation (ACE/ATRIUM-10) will be used to ensure that reactors using AREVA ATRIUM-10 fuel remain within required safety limits during steady-state operation and anticipated operational occurrences. The new correlation will replace the Siemens Power Corporation B (SPCB) correlation (Reference 2), which is currently used to evaluate critical power for BWRs containing ATRIUM-10 fuel. The new correlation provides a more mechanistic treatment for fluid conditions within the reactor fuel bundles and is expected to more accurately predict the critical power. Based on the initial review of TR ANP-10249P, the U.S. Nuclear Regulatory Commission (NRC) staff issued a number of requests for additional information (RAIs). The RAIs and AREVA's response are contained in Reference 3.

2.0 REGULATORY EVALUATION

In its review of TR ANP-10249P, the NRC staff utilized the guidance of Standard Review Plan (SRP) 4.4, "Thermal and Hydraulic Design." SRP 4.4 provides staff guidance for reviewing proposed licensing actions and topical reports against the requirements of General Design Criterion (GDC)-10 which is found in Appendix A of Section 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) to the Commissions regulations. GDC-10 requires the following:

> "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

The guidance from SRP 4.4 which is applicable to the review of TR ANP-10249P is Acceptance Criterion 1.b, which states: for correlations used to predict critical power, the limiting (minimum) value should be established so that at least 99.9 percent of the fuel rods in the core will not be expected to experience departure from nucleate boiling or boiling transition during normal operation or anticipated operational occurrences.

3.0 TECHNICAL EVALUATION

The critical power for operation of a water cooled reactor is the power below which boiling transition will not occur. Boiling transition is defined as a sudden drop in heat transfer due to the change in boiling mechanisms and is indicated by a temperature excursion of the heated surface. It has been the practice of the NRC staff to associate the occurrence of boiling transition at the surface of the nuclear fuel with failure of the fuel at that location in the core.

The mechanism for the occurrence of boiling transition is dependent on the conditions within the coolant channel. In the low quality region, the occurrence of boiling transition is associated with a heat flux so large in magnitude that intense boiling occurs causing steam bubbles to be crowded near the surface of the fuel. This bubble crowding prevents additional liquid from reaching the surface so that the fuel surface is blanketed with steam and heat transfer is markedly reduced. This type of boiling transition is generally associated with pressurized water reactors (PWRs).

The second mechanism for the occurrence of boiling transition occurs in high quality regions and is generally associated with BWRs. At the upper elevations of the core during operation of a BWR, the flow pattern within the coolant channels is expected to be annular with a liquid film at the fuel surface and steam or a mixture of steam and liquid droplets within the interior of the channel. If the heat generation is sufficient to cause dryout of the liquid film or to cause entrainment of the liquid film into the droplet field, a sudden temperature increase associated with boiling transition will occur.

The ACE/ATRIUM-10 critical power correlation predicts the channel power associated with dryout of the liquid film. The phenomena of entrainment of the liquid film into the droplet field, deposition of droplets into the liquid film, and evaporation from the heated fuel surface are all treated in the model. The solution is obtained by integrating the conservation equations affecting the three fields (liquid film, droplets, and vapor) starting from the core inlet. Thus, axial power distributions are taken into account so that the axial location of dryout, as well as the channel power which will produce dryout can be determined. Previous BWR critical power correlations including SPCB evaluated average channel conditions using the average channel quality. In evaluating average channel conditions, the liquid film at the fuel surface can not be readily distinguished from the liquid in the droplet field. Such correlations are derived for a specific axial power shape and must be modified to predict the critical power for other axial power shapes. The location for dryout of the liquid film is not readily predicted by correlations based on average channel conditions.

Although the ACE/ATRIUM-10 correlation follows the course of the three fields up a reactor core channel, the formulation remains a correlation since many of the phenomena are determined using empirical constants which are fit to channel dryout data. Phenomena which have been incorporated into ACE/ATRIUM-10 using empirical constants include the effects of non-uniform azimuthal power for the rods of a fuel bundle, part-length and water rods, and turbulent mixing downstream of the fuel element spacer grids.

ACE/ATRIUM-10 Database

The ACE/ATRIUM-10 database for ATRIUM-10 fuel contains [ ] points used to derive the correlation and [ ] independent points that were used to validate the correlation. In addition, the correlation was compared with [ ] data points from other but similar fuel designs. All data were taken at the AREVA Karlstein Thermal Hydraulic (KATHY) test facility located in Karlstein, Germany.

The dryout test assemblies model full size ACE/ATRIUM-10 fuel. The heater rods are directly heated by electric current which is passed through the rod surface. The thickness of the heater wall determines the power of the rod relative to other rods in the test assembly. Heater wall thicknesses are varied up the length of the rods so that axial power profiles may be modeled. Thermocouples are located on the highest powered rods at locations below the spacer grids where dryout is expected to occur.

The database for the ATRIUM-10 fuel design contains data for coolant flow rates of [ ] through the test assembly, inlet subcooling of [ ], and pressures ranging from [ ]. Axial power shapes evaluated were chopped cosine, downskew, and upskew. Part length rods were included and were given the same power shape as full-length rods with the power shape truncated for the part length rods.

Table 6.2 of TR ANP-10249P lists the physical characteristics of the ATRIUM-10 test assembly. The NRC staff requested that AREVA compare the values in the table with those of an ATRIUM-10 fuel element and discuss the significance of any differences between the test assemblies and the actual ATRIUM-10 fuel. AREVA responded that most of the differences between the test assemblies and the production fuel are within the manufacturing tolerances. The distance between the bundle rod spacers is the same except that an additional spacer is located just above the lower tie plate for the production fuel. Fuel rod dryout would not be expected for that location. The production fuel includes low power blanket regions at top and bottom of each assembly which were not modeled by the test assembly. Fuel rod dryout would not be expected in these low power locations. The part-length rods are shifted slightly upward in the production fuel from the test assembly. The application of ACE/ATRIUM-10 accounts for differences in part-length rod elevation. The dryout of part-length rods is evaluated using conservative empirical constants as discussed in the next section. Finally, AREVA noted that the spacer grids used with the test assemblies are of a slightly different design from those of the production fuel elements. An additional dryout test was performed to assess the affect of this change. The effect of the spacer change on the result was not significant. Based on the similarity between the test assemblies and the production ATRIUM-10 fuel design, the NRC staff concludes simulation of actual ATRIUM-10 fuel by the KATHY test facility is sufficient for development and validation of ACE/ATRIUM-10.

Determination of Empirical Constants

The physical phenomena affecting dryout of the liquid film on a fuel rod surface are described in the ACE/ATRIUM-10 methodology by equations containing a number of empirical constants. ACE/ATRIUM-10 contains three types of constants: non-linear constants, linear constants, and additive constants. The NRC staff requested that AREVA provide additional information

describing the methodology by which the empirical constants were determined (Reference 3). AREVA described the iterative process by which the correlation was fit to the [ ] points of the derivation data base.

The non-linear constants are used in formations that include grid spacer heat transfer enhancement, onset of annular flow, and entrainment of the liquid film. They were selected to provide the best result in following the trend of the data.

De-entrainment of droplets while in the annular flow regime are described in an equation using linear constants. These were determined using a linear least square best fit.

Additive constants are included with the rod assembly local peaking constants (K-factors) so that the predicted critical power will match the experimental critical power. An initial K-factor is determined for each rod from the local rod peaking pattern using the methodology that the NRC staff previously reviewed and approved in Reference 2. The final K-factor which includes the additive constant is used to compute the critical power. The final additive constant for each rod is determined from and averaged over the set of peaking patterns for which that rod is limiting. The iteration is repeated until a convergent solution is obtained.

Some of the rod locations of the ATRIUM-10 test bundles were not tested for dryout in the AREVA testing program. These rod locations included part-length rods and non-limiting rod positions adjacent to the water rod within the test bundle. The NRC staff questioned how additive constants could be determined for these positions. AREVA responded that a conservative methodology was used by which the part-length rod additive constants were calculated by conservatively assuming that dryout did occur on the part-length rod even when it did not because other locations were limiting. For the full-length rods that were not tested, AREVA determined the additive constants based on experimental data for geometrically similar but slightly more limiting rod positions. The NRC staff agrees that this approach is acceptable and conservative.

Based on the data from the defining data set, AREVA determined the standard deviation of ACE/ATRIUM-10 in predicting fuel rod dryout. The standard deviation is used in Monte Carlo evaluations to determine the safety limit. An acceptable safety limit is achieved when it is shown that at least 99.9 percent of the fuel rods in the core will not be expected to experience dryout during normal operation or anticipated operational occupancies. This is in accordance with the guidance from SRP 4.4 Acceptance Criterion 1.b.

Comparison of ACE/ATRIUM-10 with the Validating Database

AREVA separated [ ] independent data points from the total performed at the KATHY facility. These were used to validate the correlation. In partitioning the data, AREVA placed all the high inlet subcooling data points in the validating data set in order to test the accuracy of ACE/ATRIUM-10 when extrapolated in subcooling. The correlation was shown to still be accurate when extrapolated. The data analysis showed that the critical power is linear with subcooling, therefore, extension of the correlation a few degrees to zero subcooling is justified.

The correlation statistics were reexamined using the validating data set. The standard deviation of the validating data set is slightly higher than the standard deviation of the defining

data. Close agreement with the data is still shown. The accuracy of the correlation in predicting dryout elevation is slightly better for the validating data set than for the defining data set. Using both sets of data, a combined standard deviation for ACE/ATRIUM-10 was determined which can be used in the Monte Carlo evaluations to determine the safety limit. The dryout elevation predicted by ACE/ATRIUM-10 is not used except to gain confidence that the correlation is correctly modeling the physical phenomena of fuel rod dryout.

Other Issues Arising During the NRC Staff's Evaluation

The range of reactor core conditions for which AREVA proposes to utilize ACE/ATRIUM-10 extend slightly outside the range of the tested data. The NRC staff requested justification for the extensions. The extensions involve the upper and lower limit for mass flow rate, the upper and lower limit for subcooling, and maximum rod local peaking limit. For the upper limit on mass flow rate and the upper limit on inlet subcooling, the extension is very small and allows ACE/ATRIUM-10 to be used within the range of data uncertainty. This is acceptable to the NRC staff. AREVA argues that to extend the ACE/ATRIUM-10 to low flow rates is conservative. ACE/ATRIUM-10 was shown to predict a critical power approaching zero for very low flow rates whereas test data shows that the actual critical power is much larger as a result of pool boiling. For the extension of ACE/ATRIUM-10 for minimum subcooling AREVA wishes to extend ACE/ATRIUM-10 about [ ] for a saturated condition at the core inlet. AREVA notes that in the validation process the correlation was shown to be accurate when extended approximately [ ] in the direction of greater subcooling. The NRC staff agrees that the accuracy of ACE/ATRIUM-10 has been shown to be relatively insensitive to inlet subcooling so that ACE/ATRIUM-10 may be extended to a saturated inlet condition. AREVA requests to extend the maximum range of local radial power peaking from [ ]. The local radial power peaking of the rods is an input to the ACE/ATRIUM-10 formulation. Inaccuracies would appear as changes in the additive constant. AREVA demonstrated that for a range of power peaks, the changes in the additive constant were small and within the range of the additive constant uncertainty. The NRC staff, therefore, agrees that ACE/ATRIUM-10 may be extended to a maximum local radial rod peaking of [ ].

ACE/ATRIUM-10 is designed to predict dryout at the upper elevations of a BWR core where the flow pattern would be annular. The NRC staff notes that under certain conditions, such as a sharply bottom peaked flux shape, boiling crisis might occur in a core below the region where annular flow would begin. Boiling crisis below the annular flow region might not be predicted by ACE/ATRIUM-10. AREVA responded to a NRC staff RAI by comparing the conditions which would be required to produce boiling crisis below the annular region with the range of applicability for ACE/ATRIUM-10. AREVA provided the results from heated simulated fuel bundle tests which included operational conditions for both BWRs and PWRs. Comparison of this data to the range of applicability for ACE/ATRIUM-10 demonstrated that boiling crisis below the annular flow region could not occur within this range. For conditions outside the range of applicability for ACE/ATRIUM-10, AREVA will assume that dryout has occurred. The NRC staff concludes that this approach is acceptable.

Section 5.7 of TR ANP-10249P provides comparisons of ACE/ATRIUM-10 to data from simulated reactor transients. Load rejection and loss of flow events are used in the comparisons. The correlation was found to predict dryout at or before the time of dryout observed in the tests. The NRC staff questioned the ability of the correlation to predict dryout

times for other types of events. In particular, none of the tests simulated transients involving reactor depressurization. The NRC staff noted that flashing of the liquid film covering the fuel rods during a depressurization transient is not modeled by ACE/ATRIUM-10. The concern was that bubble formation within the liquid film might act to disperse the liquid away from the fuel rods which might lead to earlier liquid film dryout than would be predicted by the correlation. AREVA responded that rod bundle tests for simulated BWR fuel at constant power have been performed and did not show any fuel rod dryout. The test data indicates that fuel element dryout is not of concern during depressurization transients. Depressurization of an operating BWR would not be at constant power, however, since the depressurization would produce additional voiding and cause a reduction in reactor power. The reduction in reactor power would further reduce the occurrence of dryout during a depressurization transient. Therefore, the NRC staff agrees that fuel rod dryout from depressurization is not a concern for BWR applications of ACE/ATRIUM-10.

Application of ACE/ATRIUM-10 in BWR Safety Limit Methodology

The currently approved AREVA methodology for demonstrating compliance with GDC-10 of Appendix A to 10 CFR Part 50 is the guidance of SRP 4.4 Acceptance Criterion 1.b. The approved methodology is described in Reference 4. The methodology uses Monte Carlo evaluations to demonstrate that at least 99.9 percent of the fuel rods in the core will not experience dryout during normal operation or anticipated operational occurrences in accordance with the guidance from SRP 4.4 Acceptance Criterion 1.b.

This methodology will be modified slightly for use with ACE/ATRIUM-10 to take advantage of the channel integration process used with ACE/ATRIUM-10. AREVA provided the explanations and justifications for the modifications in the response to NRC staff RAI 18 (See Reference 3). The key difference between the revised safety limit methodology and the current methodology is that channel bow variation along the length of the fuel bundle is now considered. In the current methodology the effect of channel bow is determined for the limiting plane. The NRC staff reviewed the revised methodology and concludes that the treatment of channel bow remains conservative for the following reasons: the maximum power-to-channel bow sensitivity is used for all rod positions, a conservative bounding core-loading analysis is used, the maximum assembly channel offset is used, no credit is taken for increased burnup of bowed assemblies, and the direction of bow is assumed so as to maximize neutron moderation.

When the core of an operating reactor is partially loaded with ATRIUM-10 fuel during a refueling outage, the reactor will operate for the next cycle with a mixed core. Approved AREVA methodology for evaluating the critical power for a mixed core is contained in References 5 and 6. Using this methodology, the plant owner performs analyses for the co-resident fuel using a series of input conditions. The critical power of the co-resident fuel is evaluated using the critical power correlation approved for that fuel type. The calculated critical powers are then used to establish the appropriate additive constants and, if necessary, the design specific correlation coefficients are used to provide the best characterization of the critical power performance of the co-resident fuel. This approach is acceptable to the NRC staff.

4.0 LIMITATIONS AND CONDITIONS

The NRC staff concludes that use of ACE/ATRIUM-10, as described in References 1 and 3, is acceptable for plant safety analyses provided that the following conditions are met:

1. The ACE/ATRIUM-10 methodology may only be used to perform evaluations for AREVA ATRIUM-10 fuel. The ACE/ATRIUM-10 correlation may also be used to evaluate the performance of the co-resident fuel in mixed cores as discussed in Section 3.0 of this safety evaluation report.

2. ACE/ATRIUM-10 shall not be used outside its range of applicability defined by the range of the test data from which it was developed and the additional justifications provided by AREVA. This range is listed in Table 2.1 of Reference 1.

5.0 CONCLUSION

The NRC staff concludes that use of ACE/ATRIUM-10 is acceptable for plant safety analyses as delineated in the TR, and to the extent specified under Section 4.0, Limitations and Conditions, of this Safety Evaluation.

6.0 REFERENCES

1. ANP-10249P, Revision 0, "ACE/ATRIUM-10 Critical Power Correlation", AREVA NP Inc., April 2006.

2. EMF-2209(P)(A), Revision 2, "SPCB Critical Power Correlation", Siemens Power Corporation, September 2003.

3. R. L. Gardner, AREVA NP Inc., to Document Control Desk, NRC, "Response to a Request for Additional Information Regarding ANP-10249P, Revision 0, ACE/ATRIUM-10 Critical Power Correlation," April 9, 2007.

4. ANF-524(P)(A), Revision 2, and Supplements 1 and 2, "ANF Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation, November 1990.

5. EMF-1125(P)(A), Supplement 1, Appendix C, “ANFB Critical Power Correlation Application for Co-Resident Fuel,” Siemens Power Corporation, August 1997.

6. EMF-2245(P)(A), “Application of Siemens Power Corporation’s Critical Power Correlations to Co-Resident Fuel,” Siemens Power Corporation, September 2000.

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