

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 1.0 INTRODUCTION AND BACKGROUND

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

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15 2.0 REGULATORY EVALUATION

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

22
23 "The reactor core and associated coolant, control, and protection systems shall be
24 designed with appropriate margin to assure that specified acceptable fuel design limits
25 are not exceeded during any condition of normal operation, including the effects of
26 anticipated operational occurrences."

27
28 The guidance from SRP 4.4 which is applicable to the review of TR ANP-10249P is Acceptance
29 Criterion 1.b, which states: for correlations used to predict critical power, the limiting (minimum)
30 value should be established so that at least 99.9 percent of the fuel rods in the core will not be
31 expected to experience departure from nucleate boiling or boiling transition during normal
32 operation or anticipated operational occurrences.

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3 3.0 TECHNICAL EVALUATION
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5 The critical power for operation of a water cooled reactor is the power below which boiling
6 transition will not occur. Boiling transition is defined as a sudden drop in heat transfer due to
7 the change in boiling mechanisms and is indicated by a temperature excursion of the heated
8 surface. It has been the practice of the NRC staff to associate the occurrence of boiling
9 transition at the surface of the nuclear fuel with failure of the fuel at that location in the core.

10
11 The mechanism for the occurrence of boiling transition is dependent on the conditions within
12 the coolant channel. In the low quality region, the occurrence of boiling transition is associated
13 with a heat flux so large in magnitude that intense boiling occurs causing steam bubbles to be
14 crowded near the surface of the fuel. This bubble crowding prevents additional liquid from
15 reaching the surface so that the fuel surface is blanketed with steam and heat transfer is
16 markedly reduced. This type of boiling transition is generally associated with pressurized water
17 reactors (PWRs).
18

19 The second mechanism for the occurrence of boiling transition occurs in high quality regions
20 and is generally associated with BWRs. At the upper elevations of the core during operation of
21 a BWR, the flow pattern within the coolant channels is expected to be annular with a liquid film
22 at the fuel surface and steam or a mixture of steam and liquid droplets within the interior of the
23 channel. If the heat generation is sufficient to cause dryout of the liquid film or to cause
24 entrainment of the liquid film into the droplet field, a sudden temperature increase associated
25 with boiling transition will occur.
26

27 The ACE/ATRIUM-10 critical power correlation predicts the channel power associated with
28 dryout of the liquid film. The phenomena of entrainment of the liquid film into the droplet field,
29 deposition of droplets into the liquid film, and evaporation from the heated fuel surface are all
30 treated in the model. The solution is obtained by integrating the conservation equations
31 affecting the three fields (liquid film, droplets, and vapor) starting from the core inlet. Thus,
32 axial power distributions are taken into account so that the axial location of dryout, as well as
33 the channel power which will produce dryout can be determined. Previous BWR critical power
34 correlations including SPCB evaluated average channel conditions using the average channel
35 quality. In evaluating average channel conditions, the liquid film at the fuel surface can not be
36 readily distinguished from the liquid in the droplet field. Such correlations are derived for a
37 specific axial power shape and must be modified to predict the critical power for other axial
38 power shapes. The location for dryout of the liquid film is not readily predicted by correlations
39 based on average channel conditions.
40

41 Although the ACE/ATRIUM-10 correlation follows the course of the three fields up a reactor
42 core channel, the formulation remains a correlation since many of the phenomena are
43 determined using empirical constants which are fit to channel dryout data. Phenomena which
44 have been incorporated into ACE/ATRIUM-10 using empirical constants include the effects of
45 non-uniform azimuthal power for the rods of a fuel bundle, part-length and water rods, and
46 turbulent mixing downstream of the fuel element spacer grids.
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1
2 **ACE/TRIUM-10 Database**
3

4 The ACE/TRIUM-10 database for TRIUM-10 fuel contains [] points used to derive the
5 correlation and [] independent points that were used to validate the correlation. In addition,
6 the correlation was compared with [] data points from other but similar fuel designs. All data
7 were taken at the AREVA Karlstein Thermal Hydraulic (KATHY) test facility located in Karlstein,
8 Germany.
9

10 The dryout test assemblies model full size ACE/TRIUM-10 fuel. The heater rods are directly
11 heated by electric current which is passed through the rod surface. The thickness of the heater
12 wall determines the power of the rod relative to other rods in the test assembly. Heater wall
13 thicknesses are varied up the length of the rods so that axial power profiles may be modeled.
14 Thermocouples are located on the highest powered rods at locations below the spacer grids
15 where dryout is expected to occur.
16

17 The database for the TRIUM-10 fuel design contains data for coolant flow rates of [
18] through the test assembly, inlet subcooling of [], and pressures ranging
19 from []. Axial power shapes evaluated were chopped cosine, downskew,
20 and upskew. Part length rods were included and were given the same power shape as full-
21 length rods with the power shape truncated for the part length rods.
22

23 Table 6.2 of TR ANP-10249P lists the physical characteristics of the TRIUM-10 test assembly.
24 The NRC staff requested that AREVA compare the values in the table with those of an
25 TRIUM-10 fuel element and discuss the significance of any differences between the test
26 assemblies and the actual TRIUM-10 fuel. AREVA responded that most of the differences
27 between the test assemblies and the production fuel are within the manufacturing tolerances.
28 The distance between the bundle rod spacers is the same except that an additional spacer is
29 located just above the lower tie plate for the production fuel. Fuel rod dryout would not be
30 expected for that location. The production fuel includes low power blanket regions at top and
31 bottom of each assembly which were not modeled by the test assembly. Fuel rod dryout would
32 not be expected in these low power locations. The part-length rods are shifted slightly upward
33 in the production fuel from the test assembly. The application of ACE/TRIUM-10 accounts for
34 differences in part-length rod elevation. The dryout of part-length rods is evaluated using
35 conservative empirical constants as discussed in the next section. Finally, AREVA noted that
36 the spacer grids used with the test assemblies are of a slightly different design from those of
37 the production fuel elements. An additional dryout test was performed to assess the affect of
38 this change. The effect of the spacer change on the result was not significant. Based on the
39 similarity between the test assemblies and the production TRIUM-10 fuel design, the NRC
40 staff concludes simulation of actual TRIUM-10 fuel by the KATHY test facility is sufficient for
41 development and validation of ACE/TRIUM-10.
42

43 **Determination of Empirical Constants**
44

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

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

4
5 The non-linear constants are used in formations that include grid spacer heat transfer
6 enhancement, onset of annular flow, and entrainment of the liquid film. They were selected to
7 provide the best result in following the trend of the data.

8
9 De-entrainment of droplets while in the annular flow regime are described in an equation using
10 linear constants. These were determined using a linear least square best fit.

11
12 Additive constants are included with the rod assembly local peaking constants (K-factors) so
13 that the predicted critical power will match the experimental critical power. An initial K-factor is
14 determined for each rod from the local rod peaking pattern using the methodology that the NRC
15 staff previously reviewed and approved in Reference 2. The final K-factor which includes the
16 additive constant is used to compute the critical power. The final additive constant for each rod
17 is determined from and averaged over the set of peaking patterns for which that rod is limiting.
18 The iteration is repeated until a convergent solution is obtained.

19
20 Some of the rod locations of the ATRIUM-10 test bundles were not tested for dryout in the
21 AREVA testing program. These rod locations included part-length rods and non-limiting rod
22 positions adjacent to the water rod within the test bundle. The NRC staff questioned how
23 additive constants could be determined for these positions. AREVA responded that a
24 conservative methodology was used by which the part-length rod additive constants were
25 calculated by conservatively assuming that dryout did occur on the part-length rod even when it
26 did not because other locations were limiting. For the full-length rods that were not tested,
27 AREVA determined the additive constants based on experimental data for geometrically similar
28 but slightly more limiting rod positions. The NRC staff agrees that this approach is acceptable
29 and conservative.

30
31 Based on the data from the defining data set, AREVA determined the standard deviation of
32 ACE/ATRIUM-10 in predicting fuel rod dryout. The standard deviation is used in Monte Carlo
33 evaluations to determine the safety limit. An acceptable safety limit is achieved when it is
34 shown that at least 99.9 percent of the fuel rods in the core will not be expected to experience
35 dryout during normal operation or anticipated operational occupancies. This is in accordance
36 with the guidance from SRP 4.4 Acceptance Criterion 1.b.

37 38 **Comparison of ACE/ATRIUM-10 with the Validating Database**

39
40 AREVA separated [] independent data points from the total performed at the KATHY facility.
41 These were used to validate the correlation. In partitioning the data, AREVA placed all the high
42 inlet subcooling data points in the validating data set in order to test the accuracy of
43 ACE/ATRIUM-10 when extrapolated in subcooling. The correlation was shown to still be
44 accurate when extrapolated. The data analysis showed that the critical power is linear with
45 subcooling, therefore, extension of the correlation a few degrees to zero subcooling is justified.

46
47 The correlation statistics were reexamined using the validating data set. The standard
48 deviation of the validating data set is slightly higher than the standard deviation of the defining

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

7 8 **Other Issues Arising During the NRC Staff's Evaluation** 9

10 The range of reactor core conditions for which AREVA proposes to utilize ACE/ATRIUM-10
11 extend slightly outside the range of the tested data. The NRC staff requested justification for
12 the extensions. The extensions involve the upper and lower limit for mass flow rate, the upper
13 and lower limit for subcooling, and maximum rod local peaking limit. For the upper limit on
14 mass flow rate and the upper limit on inlet subcooling, the extension is very small and allows
15 ACE/ATRIUM-10 to be used within the range of data uncertainty. This is acceptable to the
16 NRC staff. AREVA argues that to extend the ACE/ATRIUM-10 to low flow rates is
17 conservative. ACE/ATRIUM-10 was shown to predict a critical power approaching zero for very
18 low flow rates whereas test data shows that the actual critical power is much larger as a result
19 of pool boiling. For the extension of ACE/ATRIUM-10 for minimum subcooling AREVA wishes
20 to extend ACE/ATRIUM-10 about [] for a saturated condition at the core inlet. AREVA notes
21 that in the validation process the correlation was shown to be accurate when extended
22 approximately [] in the direction of greater subcooling. The NRC staff agrees that the
23 accuracy of ACE/ATRIUM-10 has been shown to be relatively insensitive to inlet subcooling so
24 that ACE/ATRIUM-10 may be extended to a saturated inlet condition. AREVA requests to
25 extend the maximum range of local radial power peaking from []. The local radial
26 power peaking of the rods is an input to the ACE/ATRIUM-10 formulation. Inaccuracies would
27 appear as changes in the additive constant. AREVA demonstrated that for a range of power
28 peaks, the changes in the additive constant were small and within the range of the additive
29 constant uncertainty. The NRC staff, therefore, agrees that ACE/ATRIUM-10 may be extended
30 to a maximum local radial rod peaking of [].
31

32 ACE/ATRIUM-10 is designed to predict dryout at the upper elevations of a BWR core where the
33 flow pattern would be annular. The NRC staff notes that under certain conditions, such as a
34 sharply bottom peaked flux shape, boiling crisis might occur in a core below the region where
35 annular flow would begin. Boiling crisis below the annular flow region might not be predicted by
36 ACE/ATRIUM-10. AREVA responded to a NRC staff RAI by comparing the conditions which
37 would be required to produce boiling crisis below the annular region with the range of
38 applicability for ACE/ATRIUM-10. AREVA provided the results from heated simulated fuel
39 bundle tests which included operational conditions for both BWRs and PWRs. Comparison of
40 this data to the range of applicability for ACE/ATRIUM-10 demonstrated that boiling crisis below
41 the annular flow region could not occur within this range. For conditions outside the range of
42 applicability for ACE/ATRIUM-10, AREVA will assume that dryout has occurred. The NRC staff
43 concludes that this approach is acceptable.
44

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

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

14 15 **Application of ACE/ATRIUM-10 in BWR Safety Limit Methodology**

16
17 The currently approved AREVA methodology for demonstrating compliance with GDC-10 of
18 Appendix A to 10 CFR Part 50 is the guidance of SRP 4.4 Acceptance Criterion 1.b. The
19 approved methodology is described in Reference 4. The methodology uses Monte Carlo
20 evaluations to demonstrate that at least 99.9 percent of the fuel rods in the core will not
21 experience dryout during normal operation or anticipated operational occurrences in
22 accordance with the guidance from SRP 4.4 Acceptance Criterion 1.b.

23
24 This methodology will be modified slightly for use with ACE/ATRIUM-10 to take advantage of
25 the channel integration process used with ACE/ATRIUM-10. AREVA provided the explanations
26 and justifications for the modifications in the response to NRC staff RAI 18 (See Reference 3).
27 The key difference between the revised safety limit methodology and the current methodology
28 is that channel bow variation along the length of the fuel bundle is now considered. In the
29 current methodology the effect of channel bow is determined for the limiting plane. The NRC
30 staff reviewed the revised methodology and concludes that the treatment of channel bow
31 remains conservative for the following reasons: the maximum power-to-channel bow sensitivity
32 is used for all rod positions, a conservative bounding core-loading analysis is used, the
33 maximum assembly channel offset is used, no credit is taken for increased burnup of bowed
34 assemblies, and the direction of bow is assumed so as to maximize neutron moderation.

35
36 When the core of an operating reactor is partially loaded with ATRIUM-10 fuel during a
37 refueling outage, the reactor will operate for the next cycle with a mixed core. Approved
38 AREVA methodology for evaluating the critical power for a mixed core is contained in
39 References 5 and 6. Using this methodology, the plant owner performs analyses for the
40 co-resident fuel using a series of input conditions. The critical power of the co-resident fuel is
41 evaluated using the critical power correlation approved for that fuel type. The calculated critical
42 powers are then used to establish the appropriate additive constants and, if necessary, the
43 design specific correlation coefficients are used to provide the best characterization of the
44 critical power performance of the co-resident fuel. This approach is acceptable to the NRC
45 staff.

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1 4.0 LIMITATIONS AND CONDITIONS
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3 The NRC staff concludes that use of ACE/ATRIUM-10, as described in References 1 and 3, is
4 acceptable for plant safety analyses provided that the following conditions are met:
5

- 6 1. The ACE/ATRIUM-10 methodology may only be used to perform evaluations for AREVA
7 ATRIUM-10 fuel. The ACE/ATRIUM-10 correlation may also be used to evaluate the
8 performance of the co-resident fuel in mixed cores as discussed in Section 3.0 of this
9 safety evaluation report.
10
- 11 2. ACE/ATRIUM-10 shall not be used outside its range of applicability defined by the range
12 of the test data from which it was developed and the additional justifications provided by
13 AREVA. This range is listed in Table 2.1 of Reference 1.
14

15 5.0 CONCLUSION
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17 The NRC staff concludes that use of ACE/ATRIUM-10 is acceptable for plant safety analyses
18 as delineated in the TR, and to the extent specified under Section 4.0, Limitations and
19 Conditions, of this Safety Evaluation.
20

21 6.0 REFERENCES
22

- 23 1. ANP-10249P, Revision 0, "ACE/ATRIUM-10 Critical Power Correlation", AREVA NP
24 Inc., April 2006.
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- 26 2. EMF-2209(P)(A), Revision 2, "SPCB Critical Power Correlation", Siemens Power
27 Corporation, September 2003.
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30 Request for Additional Information Regarding ANP-10249P, Revision 0,
31 ACE/ATRIUM-10 Critical Power Correlation," April 9, 2007.
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- 33 4. ANF-524(P)(A), Revision 2, and Supplements 1 and 2, "ANF Critical Power
34 Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation,
35 November 1990.
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- 37 5. EMF-1125(P)(A), Supplement 1, Appendix C, "ANFB Critical Power Correlation
38 Application for Co-Resident Fuel," Siemens Power Corporation, August 1997.
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- 40 6. EMF-2245(P)(A), "Application of Siemens Power Corporation's Critical Power
41 Correlations to Co-Resident Fuel," Siemens Power Corporation, September 2000.
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