AREVA

ACE/ATRIUM-10 Critical Power Correlation

ANP-10249NP-A
Revision 2

Topical Report

March 2014

AREVA Inc.

RECEIVED APRIL 1, 2014
T. N. WILLS
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 21, 2014

Mr. Pedro Salas, Manager
Site Operations and Regulatory Affairs
AREVA NP Inc.
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION TOPICAL REPORT ANP-10249PA, REVISION 1, SUPPLEMENT 1P, REVISION 0, "IMPROVED K-FACTOR MODEL FOR ACE/ATRIUM-10 CRITICAL POWER CORRELATION" (TAC NO. ME7964)

Dear Mr. Salas:

By letter dated December 21, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11363A039), AREVA NP Inc. (AREVA) submitted Topical Report (TR) ANP-10249PA, Revision 1, Supplement 1P, Revision 0, "Improved K-factor Model for ACE/ATRIUM-10 Critical Power Correlation," to the U.S. Nuclear Regulatory Commission (NRC) staff for review. By letter dated January 29, 2014 (ADAMS Accession No. ML14009A444), an NRC draft safety evaluation (SE) regarding our approval of TR ANP-10249PA, Revision 1, Supplement 1P, Revision 0, was provided for your review and comment. By letter dated February 19, 2014 (ADAMS Accession No. ML14052A263), AREVA provided comments on the draft SE. The NRC staff's disposition of the AREVA comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that TR ANP-10249PA, Revision 1, Supplement 1P, Revision 0, is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

NOTICE: Enclosure 2 transmitted herewith contains Proprietary Information. When separated from Enclosure 2, this transmittal document is decontrolled.

In accordance with the guidance provided on the NRC website, we request that AREVA publish approved proprietary and non-proprietary versions of TR ANP-10249PA, Revision 1, Supplement 1P, Revision 0, within three months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The approved versions shall include an "-A" (designating approved) following the TR identification symbol.

The NRC staff acknowledges that AREVA intends to incorporate ANP-10249PA, Revision 1, Supplement 1P, Revision 0, information into the previously accepted TR ANP-10249PA, Revision 1, "ACE/ATRIUM-10 Critical Power Correlation," dated September 2009 to create Revision 2 of the accepted TR. Therefore, Revision 2 of TR ANP-10249PA can be submitted as the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, AREVA will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Sher Bahadur

Sher Bahadur, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 728

Enclosures:
1. Final SE (Non-Proprietary)
2. Final SE (Proprietary)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT ANP-10249PA, REVISION 1, SUPPLEMENT 1P, REVISION 0, "IMPROVED K-FACTOR MODEL FOR ACE/ATRIUM-10 CRITICAL POWER CORRELATION"

AREVA NP INC.

TAC NO. ME7964

1.0 INTRODUCTION AND BACKGROUND

By letter dated December 21, 2011, AREVA NP Inc. (AREVA) submitted to U.S. Nuclear Regulatory Commission (NRC) staff for review and approval for referencing in licensing action Topical Report (TR) ANP-10249PA, Revision 1, Supplement 1P, Revision 0, "Improved K-factor Model for ACE/ATRIUM-10 Critical Power Correlation" (References 1 and 2). This TR documents a revision to the Critical Power Ratio (CPR) correlation for ATRIUM-10 fuel for boiling water reactors (BWRs). The significant change in the Supplement 1P document, relative to ANP-10249PA, Revision 0 (Reference 3), is a revision in the K-factor methodology used to determine the additive constants for the CPR correlation. The K-factor methodology was revised in response to deficiencies in the axial averaging process.

The K-factor is a modeling parameter that characterizes the effect on CPR of radial fuel rod peaking distribution within an assembly. The critical power varies inversely with the K-factor (i.e., as K-factor increases in value the critical power decreases in value). The K-factor has two parts; the first part depends solely on the rod peaking factors of the specific rod and its neighbors and the second part named an additive constant accounts for other effects such as spacing and geometry determined from experimental data. This CPR correlation is applicable to steady-state design and analysis, core monitoring, anticipated operational occurrences (AOOs), accidents, loss-of-coolant accidents (LOCAs), and instability analysis for the ATRIUM-10 fuel design.

Pacific Northwest National Laboratory (PNNL) has acted as a consultant to the NRC in this review. PNNL evaluated this TR using the criteria and guidance provided in NUREG-0800, "Standard Review Plan," (SRP), Sections 4.4, "Thermal and Hydraulic Design" and SRP Chapter 15, "Transients and Accident Analyses." Supplement 1 to the TR was evaluated.

ENCLOSURE 1

2.0 REGULATORY EVALUATION

The NRC staff used the guidance of SRP, NUREG-0800, Section 4.4, "Thermal and Hydraulic Design" for the review of ANP-10249PA Supplement 1P, Revision 0. SRP Section 4.4 acceptance criteria are based on meeting the requirements of General Design Criteria (GDC) 10 and GDC 12 of Appendix A of Title 10 of the *Code of Federal Regulations* Part 50.

GDC 10 states: *The reactor core and associated coolant, control, and protection systems shall be designed with the 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.*

GDC 12 states: *The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.*

GDC 10 requires proper thermal-hydraulic design of the reactor core and associated systems is necessary to assure that sufficient margin exists with regard to maintaining adequate heat transfer from the fuel to the reactor cooling system (RCS). Failure to maintain sufficient margin can result in a transition from nucleate boiling to film boiling on the fuel cladding surface which decreases the heat transfer coefficient at the clad surface and the surface temperature rises significantly, eventually leading to fuel failure and the release of fission products to the RCS.

GDC 12 requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations that result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed. Compliance with GDC 12 provides assurance that the thermal-hydraulic design of the reactor core and associated systems protect the reactor from the consequences of power oscillations that could challenge the integrity of the fuel and result in the release of fission products.

The K-factor methodology in the original ACE correlation for ATRIUM-10 fuel design (Reference 3) was revised to rectify the deficiencies in the axial averaging process. The additive constants were updated during the K-factor methodology update. Evaluations performed by AREVA have confirmed that the original ACE/ATRIUM-10 Critical Power Correlation (CPC) form did not require revision as a result of the K-factor update. The range of applicability of the CPC remains unchanged.

The NRC staff has found that the ACE/ATRIUM correlation as improved by Reference 2 is acceptable to predict critical power and its 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 Boiling Transition (BT) during normal operation or AOOs.

3.0 TECHNICAL EVALUATION

In Supplement 1P, Revision 0, the K-factor methodology is revised to address deficiencies identified in an AREVA Condition Report (Reference 4) regarding the calculation of the K-factor within the ACE/ATRIUM-10 and ATRIUM 10XM CPR correlations (References 2 and 7). These deficiencies were shown to have influenced the predicted results in a non-conservative manner for this CPR correlation, for fuel assemblies with downskew axial power shape. Initial evaluations performed on existing fuel cycle designs that use the ACE correlation concluded that this deficiency has no significant impact for ATRIUM 10XM fuel designs.

The technical issues of this review exactly paralleled those of the review of Supplement 1P to ANP-10298PA, "Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation" (Reference 7). A plant specific application of the improved K-factor model for ATRIUM 10XM fuel design was previously approved for Brunswick Steam Electric Plant (BSEP), Units 1 and 2 (References 5, 8, and 9).

3.1 Critical Power Correlation Coefficient Changes

In order to maintain equivalent CPC behavior consistent with the change in K-factor, it was necessary to adjust the coefficients related to the [boundary condition] (Reference 3). The changed [ ] that are related to [ ] are listed in Section 3.1 of Reference 2. These changes in the coefficients impacted the [ ] and were [ ]. The revised coefficients are listed in Section 3.1 of Reference 2.

3.2 Improved K-factor Model

As documented in the submitted TR, some of the ACE/ATRIUM-10 CPR correlation coefficients are changed from the formulation in ANP-10249PA, Revision 1 as described in Section 3.1 of the safety evaluation (SE). Additional change in the correlation is in the definition of the radial peaking function used to calculate the assembly K-factor. The main purpose of the radial peaking function is to capture the effect on critical power of the radial power distribution. It is also sensitive to effects of lattice geometry and spacer grid design, due to their effect on the two-phase flow distribution in the assembly subchannels. This function includes the additive constants associated with each rod location in the lattice and captures the local thermal-hydraulic behavior within the fuel assembly by means of statistical fitting to the experimental data constituting the correlation's database.

The assembly K-factor is determined from local k-factors calculated for each rod in the assembly based on local rod power. In the original formulation (see ANP-10249PA, Revision 1), the local rod K-factor was calculated as an axial average by integrating over the heated length of the rod. This approach was intended as a means of capturing the total effect of individual rod power on critical heat flux behavior in the assembly, including the effect of axial variation in individual rod power distribution. However, the integration introduces a subtle bias into the local rod k-factor values with the implicit non-physical assumption that downstream conditions as well as upstream conditions affect local dryout behavior. In the actual application, however, it was shown that this effect was a concern only for conditions with downskew power distribution (see AREVA Condition Report, 2011-7210 FA Engineering Condition (Reference 4)).

The correction for this deficiency, as documented in Supplement 1 (Reference 2) was to re-formulate the dryout power model to calculate the local radial peaking term as a function of axial location, rather than integrating to obtain an axial average value. Equation 3.1 of Reference 2 is solved for the margin to dryout at each location and the power is adjusted until the node with minimum margin is at dryout. This means that the margin at each axial node is based on the integration of the equation up to the node, i.e., the solution at the limiting node is independent of the conditions in the node above the limiting node (dryout node). Use of the local conditions rather than the axial average results in a definition of the local rod K-factor that more accurately captures the effect of axial power distribution on dryout behavior. However, the effect is significant only with downskew power shapes. For cosine and upskew shapes, the integration does not introduce a discernible difference in the predicted dryout behavior, since the averaged value is quite close to the local value for the typical dryout location with these power shapes. For downskew power shapes, the dryout location is typically much lower down, and the axial averaged value can differ significantly from the local value.

3.3 Calculation of Additive Constants

The critical power behavior of the individual fuel rods within the fuel bundle is influenced by the spacers and the bundle geometry. Additive constants are factors that distinguish the critical power performance of each rod and they are position dependent. They are considered as a flow/enthalpy redistribution characteristic for a given bundle and spacer design. All other components of the model were unchanged, but the axial resolution of the model was increased to more accurately capture the shape of the axial power distribution for each rod in the assembly. With these revisions implemented, the additive constants were re-derived, using the same procedure documented originally in ANP-10249PA, Revision 1. Some of the empirical coefficients of the ACE/ATRIUM-10 CPR correlation were changed as per Section 3.1 of Reference 2. Improved K-factor model generated a new set of additive constants for use with this correlation in applications to reactor analyses. Estimation of initial additive constants for full length and part-length rods as well as the iteration scheme for the determination of final additive constants are discussed in detail in Section 3.4 of Reference 2. The observed changes in additive constants are generally small (Figure 3-5 of Reference 2) and consistent with the improved K-factor methodology implementation.

Additive constant uncertainty was recalculated per the procedure discussed in Section 3.5 of Reference 2. The improved K-factor methodology has caused a decrease to the overall additive constant uncertainty by 1.6 percent from Reference 3.

Additive constant uncertainty for high peaking rods was calculated as an additional incremental uncertainty from a statistical distribution of the respective additive constant uncertainties. A summary of the process for the additional incremental uncertainty is described in Section 3.4 of the SE.

Overall, the effect of the new additive constants on the correlation predictions is extremely small. The mean Experimental Critical Power Ratio (ECPR) which is ratio of calculated to the measured critical power, for the correlation with this correction is [ ], with a standard deviation of [ ], compared to an ECPR of [ ] and standard deviation of [ ], with the axially averaged K-factor. The fit of the correlation to its database is essentially the same, due to using the same statistical fitting methodology and criteria as was used originally. The NRC staff finds the methodology for calculating the additive constants acceptable.

3.4 Additive Constants for Controlled Assemblies with Higher Peaking Factor

Additive constant uncertainty for high peaking rods was calculated as an additional incremental uncertainty from a statistical distribution of the respective additive constant uncertainties. Table 3-1 of Reference 2 lists the incremental uncertainty for high peaked rods and the additive constant uncertainty for high peaked rods to be used in the safety limit analysis.

The basis for the additional uncertainty to be applied for assemblies when local peaking exceeds [ ] had been agreed upon and accepted by NRC staff (References 12 and 13). Safety Evaluation Report for Reference 12 provides the clarification,

> *Although local peaking factors may be exceeded in controlled bundles, these bundles by definition are not limiting bundles; consequently, they do not factor in the calculation of the minimum critical power ratio (MCPR) safety limit. If, however, in the process of calculating the MCPR safety limit, the local peaking factor of [ ] is exceeded, an additional additive constant uncertainty is applied on a rod-by-rod basis in accordance with Table 3.15 of Reference 12.*

This condition has been extended to the treatment of additive constants for ACE/ATRIUM-10 correlation and the additional uncertainty is listed in Table 3-1 of Reference 2 for high peaked rods. This is acceptable to the NRC staff.

3.5 Transient Benchmarking

Transient dryout tests were performed to confirm the fact that steady-state dryout correlations are conservative for use in BWR transient methodology for the ATRIUM-10 fuel design when using the ACE/ATRIUM-10 CPC. The limiting transient test for benchmarking was the simulated load rejection without bypass events that consist of power and pressure ramps and flow decay and the simulated loss of flow events that consist of flow decay and power decay. Transient critical power tests were repeated using the revised K-factor methodology.

The results obtained with the improved formulation in benchmarking to transient CPR data, as documented by AREVA, show essentially identical behavior to that reported for the original formulation of the correlation. In addition, the evaluations for the Brunswick plant show no significant changes in operational limits or Technical Specifications.

3.6 Application of ACE/ATRIUM-10 to Co-resident Fuel

ANP-10249PA (Reference 2, Section 3.14)) states that the ACE/ATRIUM-10 CPC may also applicable to other vendor fuel designs (Reference 10). However, the application of ACE/ATRIUM-10 correlation for other vendor fuel or future AREVA fuel designs with different critical power performance requires assessment, determination of uncertainties, and the determination of boundaries. References 10 and 11 list the necessary steps for the application of approved AREVA CPCs to various co-resident fuel designs that were previously exposed in reactors. The process and methodology for evaluating the critical power performance of both resident and co-resident fuel consist of either an indirect or direct evaluation process. The indirect process is used when the co-resident fuel critical power correlation is available but the experimental critical power data for the co-resident fuel is not. The direct process is used when the experimental critical power data for the co-resident fuel are available.

The indirect and direct processes for the application of AREVA CPCs to various co-resident fuel designs are detailed in Section 3.0 of Reference 10.

4.0 CONCLUSION

The deficiencies in the ACE/ATRIUM-10 CPR correlation that are corrected with the changes introduced in the formulation for the K-factor, and corresponding minor adjustments in the procedure for the derivation of the additive constants, represent a small but locally discernible improvement in the correlation, particularly for application to conditions with downskew power distributions. The overall effect of the changes documented in the TR submitted for review is to introduce more physically realistic modeling of the local subchannel hydrodynamics that influence dryout behavior in a fuel rod array.

The proposed corrections are acceptable improvements in the dryout modeling approach used in the ACE/ATRIUM-10 CPR correlation.

The NRC staff acknowledges that AREVA intends to incorporate ANP-10249PA, Revision 1, Supplement 1P, Revision 0, information into the previously accepted TR ANP-10249PA, Revision 1, "ACE/ATRIUM-10 Critical Power Correlation," dated September 2009 to create Revision 2 of the accepted TR. Therefore, Revision 2 of TR ANP-10249PA can be submitted as the accepted version of the TR.

5.0 LIMITATIONS AND CONDITIONS

1. The ACE/ATRIUM-10 methodology may only be used to perform evaluations of AREVA ATRIUM-10 fuel design. 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 3.6 of this SE.
2. ACE/ATRIUM-10 correlation shall not be used outside the range of applicability defined by the range of the test data prescribed in Table 2.1 of Reference 2.

6.0 REFERENCES

1. Letter NRC:11:121 from Pedro Salas (AREVA NP Inc.) to US NRC, "Request for Review and Approval of ANP-10249PA, Revision 1, Supplement 1P, Revision 0, "Improved K-factor Model for ACE/ATRIUM-10 Critical Power Correlation," AREVA NP Inc.,

December 21, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML11363A039).

2. ANP-10249PA, Revision 1, Supplement 1P, Revision 0, "Improved K-factor Model for ACE/ATRIUM-10 Critical Power Correlation," AREVA NP Inc., December 2011 (ADAMS Accession Number ML11363A041).

3. ANP-10249PA, Revision 1, "ACE/ATRIUM-10 Critical Power Correlation," AREVA NP Inc., September 2009 (ADAMS Accession Number ML0936314161).

4. Letter BSEP 11-0040 from Michael J. Annacone (BSEP) to US NRC, "Additional Information Supporting License Amendment Request to Add Analytical Methodology ANP-10298PA to Technical Specification 5.6.5, Core Operating Limits Report (COLR)," Progress Energy, April 6, 2011 (Enclosures: AREVA Operability Assessments CR 2011-2274 and CR 2010-7210) (ADAMS Accession Number ML111020442).

5. ANP-3086(P), Revision 0, Brunswick Unit 1 and Unit 2 SLMCPR Operability Assessment Critical Power Correlation for ATRIUM-10XM Fuel – Improved K-factor Model, February 2012 (ADAMS Accession Number ML120760256).

6. Letter BSEP 12-0104 from Michael J. Annacone (BSEP) to US NRC, "Response to Request for Additional Information Regarding License Amendment Request for Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," and Revision to Technical Specification 2.1.1.2 Minimum Critical Power Ratio Safety Limit," Duke Energy, September 21, 2012 (ADAMS Accession Number ML122770484).

7. ANP-10298PA, Revision 0, Supplement 1P, Revision 0, "Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation," AREVA NP, Inc., December 2011 (ADAMS Accession Number ML11363A123).

8. Letter BSEP 12-0031 from Michael J. Annacone (BSEP) to US NRC, "Request for License Amendments - Addition of Analytical Methodology Request for License Amendments - Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)" and Revision to Technical Specification 2.1.1.2 Minimum Critical Power Ratio Safety Limit," Progress Energy, March 6, 2012 (ADAMS Accession Number ML120760256).

9. Letter from US NRC to Michael J. Annacone (BSEP), "Brunswick Steam Electric Plant, Units 1 and 2, Draft Safety Evaluation for Amendments Regarding Addition of Analytical Methodology Topical Reports to Technical Specification 5.6.5 and Revision to Minimum Critical Power Ratio Safety Limit (TAC Nos. ME8135 and ME8136)," US NRC, February 11, 2013.

10. EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to the Co-Resident Fuel," Siemens Power Corporation, August 2000 (ADAMS Accession Number ML031290413).

11. EMF-1125(P)(A) Supplement 1, Appendix C, "ANFB Critical Power Correlation Application for Co-resident Fuel," Siemens Power Corporation, May 1997.

12. EMF-2209(P)(A), Revision 1, "SPCB Critical Power Correlation," Siemens Power Corporation, July 2000 (ADAMS Accession Number ML031290413).

13. Letter from J. F. Mallay (Siemens) to US NRC, "SER Conditions for EMF-2209(P), Revision 1, SPCB Critical Power Correlation," April 20, 2000 (ADAMS Accession Number ML003708323).

Attachment 1: Resolution of Comments (Non-Proprietary)

Principal Contributors: Mathew M. Panicker, NRC
Pacific Northwest National Laboratory staff

Date: March 21, 2014

NRC-IC-09-033
received 9/29/09

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 23, 2009

Mr. Ronnie L. Gardner, Manager
Site Operations and Regulatory Affairs
AREVA NP Inc.
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: FINAL SAFETY EVALUATION FOR AREVA NP, INC. (AREVA) TOPICAL REPORTS (TR) EMF-2209(P), REVISION 2, ADDENDUM 1, "SPCB ADDITIVE CONSTANTS FOR ATRIUM-10 FUEL" AND ANP-10249 (P), REVISION 0, SUPPLEMENT 1, "ACE ADDITIVE CONSTANTS FOR ATRIUM-10 FUEL" (TAC NOS. MD8754 AND ME0162)

Dear Mr. Gardner:

By letters dated May 1, 2008, and July 31, 2008, AREVA submitted TRs EMF-2209(P), Addendum 1, "SPCB [Siemens Power Corporation B] Additive Constants for the ATRIUM-10 Fuel," and ANP-10249(P), Revision 0, Supplement 1, "ACE Additive Constants for ATRIUM-10 Fuel," to the U.S. Nuclear Regulatory Commission (NRC) staff for review. By letter dated July 6, 2009, an NRC draft safety evaluation (SE) regarding our approval of TRs EMF-2209(P), Addendum 1, and ANP-10249(P), Revision 0, Supplement 1, was provided for your review and comments. By letter dated July 24, 2009, AREVA commented on the draft SE. The NRC staff's disposition of AREVA comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that TRs EMF-2209(P), Addendum 1, and ANP-10249(P), Revision 0, Supplement 1, are acceptable for referencing in licensing applications for boiling water reactors to the extent specified in the TRs and enclosed final SE. The final SE defines the basis for acceptance of the TRs.

Our acceptance applies only to material provided in the subject TRs. We do not intend to repeat our review of the acceptable material described in the TRs. When the TRs appears as references in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from the TRs will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that AREVA publish accepted proprietary and non-proprietary versions of the TRs within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of these TRs, AREVA and/or licensees referencing them will be expected to revise the TRs appropriately, or justify their continued applicability for subsequent referencing.

Sincerely,

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 728

Enclosure 1: Non-Proprietary Final SE
Enclosure 2: Proprietary Final SE

Document transmitted herewith contains sensitive unclassified information. When separated from enclosures, this document is decontrolled.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NON-PROPRIETARY FINAL SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AREVA NP, INC. (AREVA) TOPICAL REPORTS

EMF-2209(P), REVISION 2, ADDENDUM 1

"SPCB ADDITIVE CONSTANTS FOR ATRIUM-10 FUEL," AND

ANP-10249 (P), REVISION 0, SUPPLEMENT 1

"ACE ADDITIVE CONSTANTS FOR ATRIUM-10 FUEL"

AREVA NP, INC.

PROJECT NO. 728

1.0 INTRODUCTION AND BACKGROUND

AREVA NP, INC. (AREVA) submitted, by letters dated, May 1, 2008, and July 31, 2008, the following topical reports (TRs): EMF-2209(P), Revision 2, Addendum 1, "SPCB Additive Constants for ATRIUM-10 Fuel" and ANP-10249 (P), Revision 0, Supplement 1, "ACE Additive Constants for ATRIUM-10 Fuel," for U.S. Nuclear Regulatory Commission (NRC) staff review and approval. These submittals are in response to the Title.10 of the *Code of Federal Regulations* (10 CFR) Part 21 notification, dated October 8, 2007.

The above stated AREVA submittals document revisions made to the ACE and SPCB critical power correlations additive constants for ATRIUM-10 fuel for boiling water reactors (BWRs). The additive constants were revised in response to an error discovered in the evaluation of the laboratory data when accounting for the power distribution and the power contained in the part-length fuel rods. Evaluations have confirmed that the SPCB critical power correlation coefficients do not require revision as a result of the error.

The SPCB correlation was developed for two fuel types, the ATRIUM-10 and the ATRIUM-9 fuel designs. However, application of the SPCB correlation to ATRIUM-9 fuel does not require revision as this fuel design does not contain part-length fuel rods. AREVA also noted that the error discussed in these reports is restricted to critical heat flux (CHF) testing of the ATRIUM-10 fuel. Application of the ACE and SPCB additive constant correlation to co-resident BWR fuel containing part-length fuel rods using the NRC approved method described in References 1 and 2, do not require revision.

ENCLOSURE 1

2.0 REGULATORY EVALUATION

In its review of EMF-2209 (P), Addendum 1, and ANP-10249 (P) Revision 0, Supplement 1, the NRC staff utilized the guidance of Standard Review Plan (SRP) 4.4 "Thermal and Hydraulic Design." SRP 4.4 implements the requirements of General Design Criterion (GDC) 10 which is found in Appendix A to 10 CFR 50 to the Commissions regulations. GDC-10 states the following:

> The reactor core and associated coolant, control, and protection systems shall be designed with the 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 EMF-2209 (P), Addendum 1, and ANP-10249 (P) Revision 0, Supplement 1, is Acceptance Criterion 1.b, which states that for correlations used to predict critical power, the limiting (minimum) value should be established so that at least 99.9% 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 occupations.

3.0 TECHNICAL EVALUATION

3.1 Test Data Modifications

The AREVA test facility uses electrically heated rods to simulate the behavior of the fuel bundle in the reactor core. The electrical power generated in the individual rods is readily calculated by knowing the voltage, current, and/or the resistance of the various components. The surface of the simulated rods serves as the electrical conductor for the full length rods. The part-length rods carry the current on the surface of the rod in one direction and then through an inner copper conductor in the other direction. Consequently, the power for the part-length rods should account for the power associated with current at the surface of the rod and in the portion of the inner copper conductor that is contained within the heated length. The initial method for determining the power distribution within the bundle did not properly account for the power of the inner copper conductor of the part-length rods in the test bundle. The test data power distributions and the total power generated in a bundle were modified to properly account for the power present in the inner copper conductor in the part-length rods.

3.2 Power Distributions

AREVA assessed the impact of the modified additive constants on all the pertinent power distributions. AREVA recalculated lattice peaking powers and noted that, when the power carried in the inner copper conductor of the part-length rods is included, the relative power delivered by the part-length rods in the lower end of the lattice (in the fully rodded region below the end of the part-length rods) of the bundle, increased compared to the previously reported powers. Consequently, on a normalized relative power basis, the radial peaking factors of the part-length rods increase, and the radial peaking factors of the full-length rods decrease in the fully rodded region of the bundle. See Figures 3.1 and 3.2 of References 1 and 2.

The inclusion of the power associated with the inner copper conductor of the part-length heater rods impacts the axial power shape of the part-length rods, and consequently impacts the bundle average axial power. However, because the power associated with the inner copper conductor is such a small fraction of the overall bundle power (much less than 1 percent), the impact is small.

The development of the ACE and SPCB correlations was based on selected axial power shapes. The adjustment to the additive constants included the axial power shapes from measurements of the individual rod axial shapes for both, full-length rods and part-length rods. The part-length heater rods accounted for the incorporation of the inner copper conductor. An example comparing the bundle average axial power shape for the bundle STS 17.1 is shown in Figure 3.3 of References 1 and 2. The calculations show that the impact is small, and that the impact on the bundle axial power shape was included in the revised additive constant calculations.

3.3 Additive Constants

Having corrected the respective power distributions, both the lattice power and the bundle power, AREVA performed calculations to determine the boiling transition values of f-effective (SPCB), and the K-factors (ACE), respectively, for each test in the data base. The boiling transition values of f-effective are those values that result in a critical power ratio of 1.0 at the measured operating condition. [

] A detailed description of the determination of the new additive constants is provided in responses to requests for additional information (RAIs) in Reference 3. The newly derived additive constants supersede the additive constants that were presented in References 4 and 5.

3.4 Evaluation of Transient Critical Power Data

AREVA re-analyzed the transient critical power tests presented in References 4 and 5 using the revised initial bundle powers, axial power shapes, f-effective and K-factors values. The repeated analysis was performed consistent with References 4 and 5. The calculated time of boiling transition of each test for the repeat analysis are presented in Table 6.1 of References 1 and 2, and Table 7.1 in Reference 2.

Table 7.1 of Reference 2 indicates that two of the tests listed in 7.1 are slightly non-conservative. The explanation for the minor non-conservatisms provided by AREVA is that in one of the tests (Test STS-17.8-u6.2), simulating a flow decay event along with a correspondent power decay, the power decay was delayed by nearly a full second after the initiation of the flow decay. Typically, an event of this kind experiences an instantaneous power decrease during a flow decay transient. Consequently, the test is considered "atypical," and thus is not a true representation of a realistic plant event. The other test that indicates a minor non-conservatism is Test STS-29.5-H100.1. For this test, AREVA pointed out that Test STS-29.5-H100.4 had very similar initial boundary conditions, but that Test STS-29.5-H100.4 had a lower bundle power, and is representative of how the transient calculation is performed in a licensing procedure. But, in Test STS-29.5-H100.1, the initial bundle power was too high and thus not representative of realistic licensing event. Also, the higher power case would not be analyzed because boiling transition is to happen at a lower bundle power.

The analysis conducted by AREVA in support of this issue indicated that the changes to initial bundle powers, axial power shapes, f-effective and K-factors values, did not impact conclusions in References 4 and 5. The repeated analysis for each of these parameters demonstrated that the ACE and SPCB steady-state "Dry-out" correlations continue to be appropriate for use in evaluating transient events.

4.0 CONCLUSION

The NRC staff finds that the revisions AREVA provided in the submittal regarding the uncertainties associated with the additive constants are acceptable. The revised additive constants will supersede the additive constants for the ATRIUM-10 that is presented in References 4 and 5.

The additive constants were revised in response to an error discovered in the evaluation of the laboratory data when accounting for the power distribution and the power contained in the part-length fuel rods.

Application of SPCB to ATRIUM-9 fuel does not require revision, as this fuel design does not contain part-length fuel rods. Since the error discussed in this report is restricted to CHF testing of the ATRIUM-10 fuel, applications of ACE and SPCB to co-resident BWR fuel containing part-length fuel rods using the NRC approved method described in Reference 1 do not require revision.

The NRC staff acknowledges that AREVA will combine this safety evaluation with the previously approved TRs, to issue Revision 3 of TR EMF-2209, and Revision 1 of TR ANP-10249 . All parts of the latest revisions have been approved by the NRC staff. Therefore, Revision 3 of TR EMF-2209, and Revision 1 of TR ANP-10249, can be submitted as the approved versions of the TRs. This will allow use of current plant technical specification (TS) references without modifications to the standard TSs.

5.0 REFERENCES

1. Letter, Ronnie L. Gardner, Manager, Site operations and Corporate Regulatory Affairs, AREVA, to the U.S. Nuclear Regulatory Commission, requesting review and approval of EMF-2209 (P), Addendum 1, "SPVB Additive Constants for Atrium-10 Fuel," dated May 1, 2008.

2. Letter, Ronnie L. Gardner, Manager, Site operations and Corporate Regulatory Affairs, AERVA, to the U.S. Nuclear Regulatory Commission, requesting review and approval of ANP-10249 (P), Revision 0, Supplement 1, Revision 0, ACE Additive Constants for ATRIUM-10 Fuel," dated July 31, 2008.

3. Responses to Request for Additional Information Regarding adjustments to Additive Constants for ATRIUM-10 fuel design, EMF-2209 (P), Revision 2, Addendum 1, Revision 0, dated October 2008.

4. EMF-2209 (P)(A), Revision 2, "SPC Critical Power Correlations," September 2003.

5. ANP-10249 (P)(A), Revision 0, "ACE/ATRIUM-10 Critical Power Correlation," August 2007.

Principle Contributor: A. Attard NRR/DSS

Date: September 23, 2009

RESOLUTION OF AREVA NP, INC. (AREVA)

COMMENTS ON DRAFT SAFETY EVALUATION FOR TOPICAL REPORTS (TRs)

EMF-2209(P), REVISION 2, ADDENDUM 1

"SPCB ADDITIVE CONSTANTS FOR ATRIUM-10 FUEL," AND

ANP-10249 (P), REVISION 0, SUPPLEMENT 1

"ACE ADDITIVE CONSTANTS FOR ATRIUM-10 FUEL"

By letter dated July 24, 2009 (ADAMS Accession No. ML092120299), AREVA provided two (2) corrections to references listed in the draft safety evaluation (SE) for TRs: EMF-2209(P), Revision 2, Addendum 1, "SPCB Additive Constants for ATRIUM-10 Fuel" and ANP-10249 (P), Revision 0, Supplement 1, "ACE Additive Constants for ATRIUM-10 Fuel." The following are the NRC staff's resolution of these corrections:

Draft SE comments for EMF-2209(P), Revision 2, Addendum 1, and ANP-10249 (P), Revision 0, Supplement 1:

1. Page 4, Line 36: date of the reference revised to May 1, 2008.

   NRC Resolution for Comment 1 on Draft SE:

   The staff reviewed the AREVA recommendation and found it acceptable.

2. Page 4, Line 41: date of the reference revised to July 31, 2008.

   NRC Resolution for Comment 2 on Draft SE:

   The staff reviewed the AREVA recommendation and found it acceptable.

ATTACHMENT

nrc-ic-07-050

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 20, 2007

Mr. Ronnie L. Gardner
Site Operations and Regulatory Affairs Manager
AREVA NP Inc.
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: FINAL SAFETY EVALUATION FOR AREVA NP, INC. (AREVA) TOPICAL REPORT ANP-10249P, REVISION 0, "ACE/ATRIUM-10 CRITICAL POWER CORRELATION" (TAC NO. MD2186)

Dear Mr. Gardner:

By letter dated May 2, 2006 (Agencywide Documents Access and Management System Accession No. ML061250309), AREVA submitted Topical Report (TR) ANP-10249P, Revision 0, "ACE/ATRIUM-10 Critical Power Correlation," to the U.S. Nuclear Regulatory Commission (NRC) staff. By letter dated July 18, 2007, an NRC draft safety evaluation (SE) regarding our approval of TR ANP-10249P, Revision 0, was provided for your review and comments. By e-mail dated July 20, 2007, Jerald S. Holm, Product Licensing Manager, AREVA, stated to NRC staff that AREVA did not have any comments on the draft SE and therefore, there will be no changes to the text of the SE.

The NRC staff has found that TR ANP-10249P, Revision 0, is acceptable for referencing in licensing applications for boiling water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for acceptance of the TR. Enclosed with this letter are the non-proprietary (Enclosure 1) and proprietary (Enclosure 2) versions of the final SE.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that AREVA publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, AREVA and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Ho K. Nieh, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 728

Enclosures: 1. Non-Proprietary Final SE
2. Proprietary Final SE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT ANP-10249P, 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.

Principle Contributors: A. Attard
W. Jensen

Date: August 20, 2007

AREVA

May 2, 2006
NRC:06:019

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Request for Review and Approval of ANP-10249P, "ACE/ATRIUM-10 Critical Power Correlation" April 2006

AREVA NP Inc. requests the NRC's review and approval of the topical report ANP-10249P, "ACE/ATRIUM-10 Critical Power Correlation." This report presents a new critical power correlation and its application to the ATRIUM™-10[1] BWR fuel assembly design. It is requested that the NRC complete its review of the topical report by May 2007. Proprietary and non-proprietary versions of the report are provided on the enclosed CDs.

AREVA NP considers some of the material contained in the enclosed documents to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure.

Sincerely,

Ronnie L. Gardner

Ronnie L. Gardner, Manager
Site Operations and Regulatory Affairs
Framatome ANP, Inc.

Enclosures

cc: G. S. Shukla
F. M. Akstulewicz
Project 728

[1] ATRIUM is a trademark of AREVA NP Inc., an AREVA and Siemens company registered in the United States and various other countries.

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Tel.: 434 832 3000 · Fax: 434 832 3840 · www.areva.com

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Jerald S. Holm. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the topical report ANP-10249P, "ACE/ATRIUM-10 Critical Power Correlation", dated April 2006, and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure.

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Jerald L Holm

SUBSCRIBED before me this 1

day of May, 2006.

Susan K McCoy

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/10/2008

SUSAN K. McCOY
COMMISSION EXPIRES
NOTARY
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STATE OF WASHINGTON

April 9, 2007
NRC:07:014

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Washington, D.C. 20555-0001

Response to a Request for Additional Information Regarding ANP-10249P Revision 0, "ACE/ATRIUM-10 Critical Power Correlation"

Ref. 1: Letter, Ronnie L. Gardner (AREVA NP) to Document Control Desk (NRC), "Request for Review and Approval of ANP-10249P, 'ACE/ATRIUM-10 Critical Power Correlation'," NRC:06:019, May 2, 2006.

Ref. 2: ANP-10249Q1P, "Response to Request for Additional Information – ANP-10249P," April 2007.

AREVA NP Inc. (AREVA NP) requested the NRC's review and approval of the topical report ANP-10249P, "ACE/ATRIUM-10 Critical Power Correlation" in Reference 1. This report presents a new critical power correlation and its application to the ATRIUM-10[1] BWR fuel assembly design. A response to a request for additional information regarding this topical report is provided in Reference 2.

A proprietary and a non-proprietary version of Reference 2 are enclosed on CDs.

AREVA NP considers some of the material contained in Reference 2 to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure.

Sincerely,

Ronnie L. Gardner

Ronnie L. Gardner, Manager
Site Operations and Regulatory Affairs
AREVA NP Inc.

Enclosures

cc: H. D. Cruz
J. H. Thompson
Project 728

[1] ATRIUM is a trademark of AREVA NP Inc., an AREVA and Siemens company registered in the United States and various other countries.

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AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Jerald S. Holm. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the report ANP-10249Q1P Revision 0, *Response to Request for Additional Information - ANP-10249P,* dated April 2007, and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information".

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 6th day of April, 2007.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/10/2008

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STATE OF WASHINGTON

ANP-10249Q1NP
Revision 0

Response to Request for Additional Information -

ANP-10249P

April 2007

AREVA NP Inc.

Table of Contents

This document contains a total of 103 pages.

Question 1. Table 2.1 gives the range of applicability for ACE/ATRIUM-10 in terms of mass flow rate, pressure, inlet subcooling, maximum design local peaking, maximum tested local peaking.

a. Table 2.1 lists the flow range of the correlation as [ ]. Section 6.3.3 describes the tested flow rate range to be between [ ]. Please provide justification that ACE/ATRIUM-10 can adequately predict critical power beyond the tested range of flow rate.

b. The inlet subcooling range is stated to be [ ]. Section 6.3.3 describes the data base for the ACE/ATRIUM-10 correlation to be]. Please provide justification that ACE/ATRIUM-10 can adequately predict critical power beyond the tested inlet enthalpy range. Discuss limitations of the correlation with regard to two-phase saturated conditions at the core inlet.

c. The value for maximum design local peaking is greater than the value for maximum tested local peaking. Please provide justification that ACE/ATRIUM-10 can adequately predict critical power beyond the tested values of local peaking.

Response 1:

Response 1a (Upper Mass Flow Rate Limit):

The limit on maximum mass flow rate is set to [ ]. The maximum mass flow rate of the ACE/ATRIUM[1]-10 data base is [ ]. The upper correlation limit was determined by [ ]. The KATHY test loop uncertainties are provided in Reference 1, Table 6.1.

The need to [ ] was a conclusion reached during code benchmarking to the ACE/ATRIUM-10 data base. Several points in the data base were reported to be out of the range of applicability, thus preventing complete benchmarking. As an

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example, consider the case of a critical power calculation where the maximum mass flow rate is set to [ ] . Now, begin a critical power calculation with the mass flow rate set to [ ] . This mass flow rate must first be converted to kg/s as required for use in the ACE/ATRIUM-10 correlation. Exact conversion of this value (using a mass conversion factor of 0.45359 kg/lbm) results in a mass flow rate value of [ ], which lies below the upper limit value of [ ]. If however, an approximate rounded value of 0.454 kg/lbm is used for the mass conversion factor, the resulting mass flow rate becomes [ ] which lies above the specified upper bound, and is thus reported as being out of the range of applicability. In effect, a small difference that resulted from use of limited precision numbers (rounding) caused an acceptable input mass flow rate to be unacceptable by the correlation.

AREVA NP Inc. considers it very important that production codes implementing the ACE/ATRIUM-10 correlation be able to benchmark to the ACE/ATRIUM-10 data base, providing the same results as that which is reviewed and approved. The benchmarking should be performed with the computer codes using the same switches and options that are used in production calculations. Therefore, the upper mass flow rate limit must be increased slightly to make sure that benchmark calculations with valid input are properly performed.

[

]

Table 1.1 Critical Power Sensitivity to High Mass Flow

[

] The prescribed upper flow limit is justified.

If the mass flow rate exceeds the upper mass flow limit of the ACE/ATRIUM-10 critical power correlation described above, [ ].

Response 1a (Lower Mass Flow Rate Limit):

[ ] is prescribed in ANP-10249P (Reference 1). The ACE/ATRIUM-10 correlation is [ ]. It was derived and fit with the implicit assumption that the [ ].

The low flow critical power performance of BWR assemblies and the behavior of ACE/ATRIUM-10 are described on page 3-39 of ANP-10249P. The data and correlation predictions are shown in Figure 3.24 on page 3-40 of ANP-10249P. [ ]

[

]

The lowest binned mass flow rate in the defining data set was [ ]. The standard deviation for the data in this bin was [ ]. The data from STS-17.8 at [ ] have a standard deviation of [ ]. The statistical uncertainty of the ACE/ATRIUM-10 critical power correlation will not be adversely affected [
].

Therefore, it is justified to use the ACE/ATRIUM-10 critical power correlation [
].

Response 1b (Upper Inlet Subcooling Limit):

The upper limit of inlet subcooling in the ACE/ATRIUM-10 data base is [

] . For reasons stated in Response 1a above, the upper limit on inlet subcooling is

]

Table 1.2 Critical power Sensitivity to Inlet Subcooling

] does not adversely affect the statistics or uncertainty of the ACE/ATRIUM-10 critical power correlation. The prescribed upper inlet subcooling limit is justified.

It is generally accepted (Reference 2, Figure 8.3, page 332; Reference 3, Figure 4-39 on page 146) that increasing the inlet subcooling increases the critical power. [

]

[ ] This will lead to a conservative estimate of the critical power.

Response 1b (Lower Inlet Subcooling Limit):

A model assumption requires that the [ ].

When the ACE/ATRIUM-10 data base was partitioned for correlation development, the very high inlet subcooling data was all placed in the validating data set. The highest inlet subcooling in the defining data set was [ ]. The highest inlet subcooling in the validating data set was [ ]. This was done deliberately with the expectation that a physically based model should be able to predict the critical power with variation in inlet subcooling. The correlation coefficients and initial qualification were developed with the defining data set only. During the validation of the correlation, the correlation was extrapolated in inlet subcooling from [ ]. The ECPR of the high inlet subcooling data points is [ ] and the standard deviation is [ ]. The ECPR shows that the ACE/ATRIUM-10 critical power correlation can be extrapolated in inlet subcooling. The standard deviation shows that the statistics and uncertainty of the ACE/ATRIUM-10 critical power correlation will not be adversely affected by moderate extrapolations in inlet subcooling. Note that the high inlet subcooling data [ ].

The lowest measured inlet subcooling associated with the ACE/ATRIUM-10 data base was [ ]. Extrapolation of the ACE/ATRIUM-10 critical power correlation [ ]

Response 1c (Maximum Rod Local Peaking Limit):

Figure 3.19 in Reference 1 shows that the critical power with respect to rod peaking function is well-behaved. Thus, extending the design local peaking [ ] is appropriate. The justification for using a design local peaking of [ ] is described below.

First, consider the behavior of the additive constants and additive constant uncertainty. In the ATRIUM-10 critical power data base, there are four rod positions where dryout was observed under conditions of different local peaking. These rod positions, tests, and local peaking factors are shown in Table 1.3. The values of local peaking vary from [ ]. Within each test, a measured additive constant for those runs where dryout was observed on the rod is computed (column 4) for the number of data points where dryout was observed (column 6). For a given rod, the change in additive constant between tests is shown (column 5). These differences are small. The mean ECPR and standard deviation of ECPR are provided in columns 7 and 8. Finally, in column 9, the additive constant uncertainty associated with combining the observations in the data sets from different tests is computed. There is no apparent trend in additive constant uncertainty with peaking factor. All of the additive constant uncertainties shown are less than the additive constant uncertainty that is to be used with the ACE/ATRIUM-10 correlation shown on page 3-30 of ANP-10249P. The additive constant uncertainty does not appear to be increasing as the peaking factor increases.

Next, consider the ACE/ATRIUM-10 predicted critical power. The ACE/ATRIUM-10 predicted change in CPR when the peaking is increased from [ ] was checked [

]

Because ACE/ATRIUM-10 becomes slightly conservative as the peaking factor is increased to [ ], and the uncertainty in additive constant does not appear to have a dependence on the local peaking factor, setting the design limit on peaking factor to [ ] is justified.

Table 1.3. Additive Constant Uncertainties for Rods of Different Local Peaking

Question 2. **Page 3-3 states that ACE/ATRIUM-10 is a critical power correlation, not a critical heat flux correlation. Nevertheless, for a locally highly peaked heat flux which might result from an off normal reactor condition, local film boiling might result. Discuss how the occurrence of local film boiling will be predicted if it were to occur as a result of an anticipated operational occurrence at a facility using ACE/ATRIUM-10.**

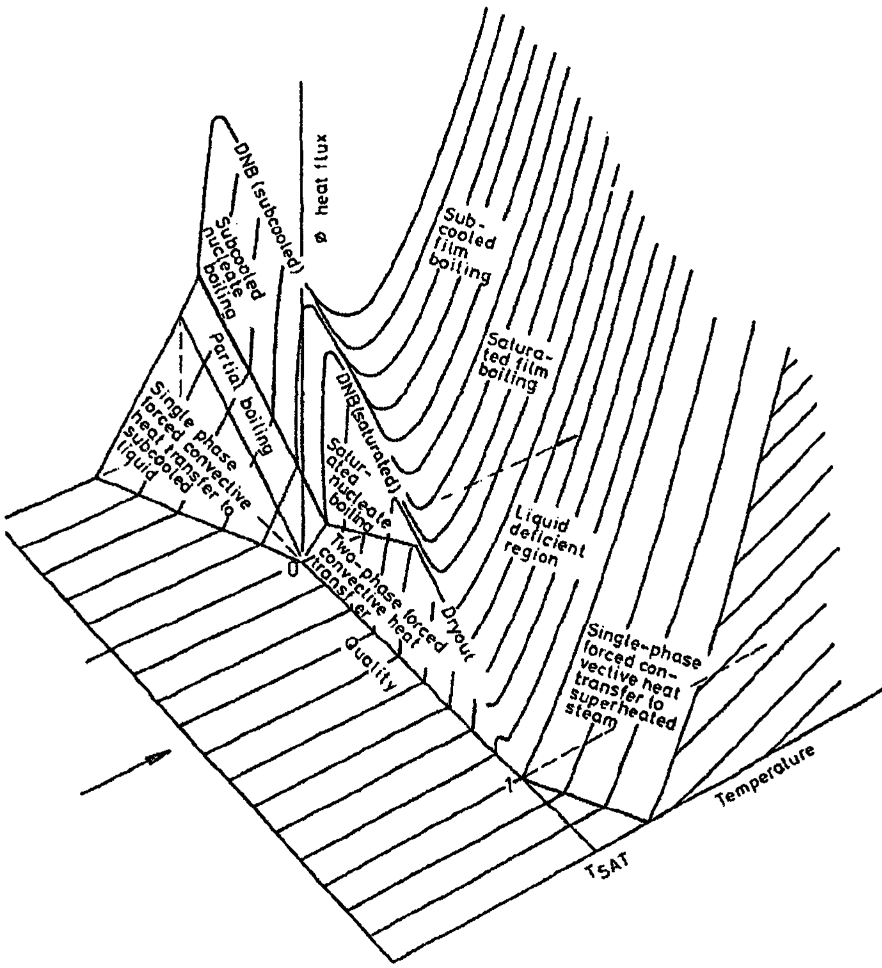
Response 2:

The [ ] provides a complete and adequate representation of BWR dryout. The justification for this conclusion is provided below.

The mechanism for critical heat flux implied in the question is known as DNB (Departure from Nucleate Boiling), a phenomenon known to occur in PWR's (Pressurized Water Reactors). In BWR's (Boiling Water Reactors), the mechanism for critical heat flux is dryout or disappearance of the annular liquid film. The request by the staff can be answered by addressing the following question: "What conditions are required to achieve DNB?"

[ ] Although helpful, this information does not provide conclusive evidence to answer the question "What conditions are required to achieve DNB?"

Next, consider the 3D boiling map associated with convective boiling in a vertical heated channel Figure 2.1. This figure portrays the heat flux in terms of the quality and surface temperature. The surface is divided into regions according to the mechanism of heat transfer. Critical heat flux corresponds to the peak in the three dimensional surface at qualities less than 1.0 and temperatures slightly above the saturation temperature. The critical heat flux peak is divided into three parts – critical heat flux associated with subcooled DNB, critical heat flux associated with saturated DNB, and critical heat flux associated with annular flow dryout. It is observed that dryout occurs at higher qualities and DNB occurs at lower qualities and with subcooled boiling. Thus, the question "What conditions are required to achieve DNB?" can be answered by [
]



Reference 2, page 174

Figure 2.1. 3D Boiling Map

[

]

The magnitude of these effects on dryout quality may be quantified with the critical power data for the ATRIUM-10 design (the data used for correlation development and validation).

[

]

[

]

[

]

Thus, examination of the ATRIUM-10 critical power data does not lead to the conclusion that DNB can be reached. While helpful, this finding is not sufficient to answer the question, "What conditions are required to achieve DNB?"

In order to answer this question, a set of data covering a broader set of conditions is required. Consider the AREVA NP Inc. [ ] collected at the Columbia University test facility. This test covered the following conditions:

These conditions cover [ ] a large part of the BWR conditions where critical power is evaluated.

[

]

Based on the foregoing analysis, it is concluded that it is unlikely that DNB can occur in a BWR assembly. While the foregoing analysis is very helpful in determining if DNB can occur in a BWR under the range of operating conditions, it is still not quite sufficient.

The final factor that determines if DNB can occur is the following. Consider the range of applicability associated with the ACE/ATRIUM-10 correlation [

]

Whenever application of the ACE/ATRIUM-10 critical power correlation falls outside the range of applicability, as prescribed and applied in ANP-10249P (Reference 1), critical heat flux is assumed to have occurred.

The conclusion is that it is not possible to achieve DNB in a BWR under the conditions that the ACE/ATRIUM-10 critical power correlation will be applied. [

]

Figure 2.2. Flow Regime Map

Figure 2.3. Flow Regime of [ ]

Question 3. Section 3.1 provides values for the [ ] and the [ ] for the ATRIUM-10 design. Please provide a reference source for equation 3.6 and describe how these values were determined.

Response 3:

Equation 3.6 was developed by AREVA NP Inc. During the initial correlation development activity, [ ] was treated as a constant (and it is a constant for the ACE/ATRIUM-10 correlation). The applicability of the underlying model to other fuel designs was considered an important part of qualifying the design. Many different designs were evaluated, and concept correlations were developed for 4x4, 7x7, 8x8, 9x9, and 10x10 designs. Those designs that did not preserve [ ] were not adequately fit by the preliminary model [ ]. To make the model applicable [ ] is required in Equation 3.6.

The values used in Equation 3.6 are purely empirical and were determined using the defining data set [ ]. The values are assembly design dependent. An overview of the method for determining the correlation coefficients is provided in Appendix A. Different values were examined. The values used in the ACE/ATRIUM-10 correlation provided the best result.

Question 4. **Section 3.2 provides values for the [ ] and the "weighting factor." It is stated that [ ] found good results using a value of [ ] much smaller than that selected for ACE/ATRIUM-10. Please provide the technical basis (qualitatively and quantitatively), by which each of these constants was determined.**

<u>Response 4:</u>

The work of [ ] (and others) was used to provide a range of physically reasonable values to consider for [ ] (see Appendix A for description of how correlation parameters were determined). Different values were examined and the best overall correlation for ACE/ATRIUM-10 was achieved with the values shown in Equations 3.7 and 3.8 of Reference 1.

Question 5. Equation 3.10 is described as the equation for [ ]. Please give the source of this equation and justify its basis. Discuss how the four input constants were determined for ATRIUM-10 fuel.

<u>Response 5:</u>

Equation 3.10 was developed by AREVA NP Inc. [

]

Question 6. Equation 3.15 is described as the equation for [ ]. Please give the source of this equation and justify its basis. Discuss how the [ ] were determined for ATRIUM-10 fuel.

Response 6:

Equations 3.15 was developed by AREVA NP Inc. Equation 3.15 forms [

].

An overview of the process for determining correlation coefficients is provided in Appendix A.

Question 7. Equation 3.20 is described as the equation for [ ]. Please give the source of this equation and justify its basis. Discuss how the four input constants were determined for ATRIUM-10 fuel.

<u>**Response 7:**</u>

Equations 3.20 was developed by AREVA NP Inc. [

]. These coefficients, in combination with the other coefficients, provided the best result.

Question 8. **Equation 3.22 is described as the equation for the [ ]. The source of the equation is stated to be [ ]. Please describe how the form of the equation for ATRIUM-10 fuel was determined. Values of [ ] input constants are given on page 3-11. It is stated however that there are [ ] for ATRIUM-10 at pressures, flow rates, and [ ] that can be used to correlate the coefficients. Please discuss how the coefficients were determined for ATRIUM-10 fuel.**

Response 8:

The derivation of the form of the equation for [

] The coefficients are purely empirical. They are determined, using the defining data set [ ], according to the process described in Appendix A. The coefficients for ATRIUM-10 are those that provide the best result.

Question 9. Equation 3.29 provides a constant value of the [ ] scaling factor. The factor is said to be set to a value which eliminates critical power correlation bias with [ ]. Discuss the process by which the bias was eliminated and the process by which the constant was selected.

<u>Response 9:</u>

[

]

Question 10. **The [ ] for the purpose of [ ] (equation 3.31) appears to have a constant value. Justify that this value is a constant regardless of [ ] or variances in [ ]. Explain the significance of this quantity and describe how it is used.**

Response 10:

[

]

This value is shown to be appropriate when the ACE/ATRIUM-10 correlation is applied to the ATRIUM-10 data base and the ECPR is plotted as a function of mass flow rate – Figures 4.2 and 5.2 in the topical report. There are no significant trends as a function of mass flow rate.

] The values chosen were those that provided the best behavior for ATRIUM-10. [

] An overview of the process for determining correlation coefficients is provided in Appendix A. [

]

The inlet flow rate is determined using methods previously approved by the U.S. NRC.

Question 11. **Equation 3.37 sets the [ ] to a constant value. On page A-44 other values for this constant are given. Please provide and justify your basis for selection of the value for the [ ] in equation 3.37.**

<u>Response 11:</u>

[

]. Several values were considered and the critical power correlation ECPR and standard deviation are not sensitive to the value chosen. The value chosen was within the range of reasonable values.

Question 12. Equation 3.38 gives the value for [ ] . Please describe how this value was determined. Describe how this parameter is used in the ACE/ATRIUM-10 correlation and discuss its significance.

Response 12:

AREVA NP, Inc. has measured the [ ] to confirm the values [ ] that are used. The measured value of [ ] is well within experimental uncertainty of the value used.

[ ]

Table 12.1 Critical Power Benchmark of Defining Data Set [ ]

Question 13. Page 3-18 states that the assembly [ ] is set to the most limiting rod in the assembly. Page 3-20 shows how an average assembly [ ] is determined. Which is used to determine the critical heat flux in equation 3.1?

Response 13:

For safety analysis, the [ ] from Equation 3.46 is chosen for use in Equation 3.1, as shown. A [ ] is calculated in Equation 3.48. [ ]

Question 14. Section 3.10.3 states that as the [ ]. If this is so why is it conservative to use [ ] values of the additive constant "ℓ"?

Response 14:

[

]

Question 15. Section 3.10.4 describes rod positions for ATRIUM-10 fuel which are not tested. These rods are at the corners of the water channel and include part length rods. Please provide justification that a valid critical power correlation can be obtained for ATRIUM-10 fuel if all positions are not tested.

Response 15:

This question will be addressed in two parts. First, the additive constants for inner water channel corner positions will be justified. Second, additive constants for part length rod positions will be justified.

Rod Positions 34 and 78

The rod positions are shown in Figure 15.1. The inner water channel corner positions (34, 38, 74, and 78) are highlighted.

Figure 15.1. Rods on Outside Corners of Inner Water Canister

Rods at the corners of the inner water canister all have [ ]. In the ATRIUM-10 design, rod position 74 is strictly symmetric to rod position 38. Rod position 38 was peaked in test STS-29.4. [ ].

[ ]

Figure 15.2. Peaking Pattern for STS-29.4 (Total)

[

]

Figure 15.3. Peaking Pattern With Rod Position 38 Limiting

[

]

Figure 15.4. Peaking Pattern With Rod Position 34 Limiting

[

]

[ ]

Figure 15.5. Peaking Pattern With Rod Position 78 Limiting

[ ]

Table 15.1. Assessment of Additive Constants in Positions 34, 38, and 78

[

]

[

]

Question 16. Section 5.7 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. Please comment on the ability of the correlation to predict dryout times for other types of events which might be evaluated using ACE/ATRIUM-10. In particular none of the tests simulated transients involving reactor depressurization. The staff notes that [

]

Response 16:

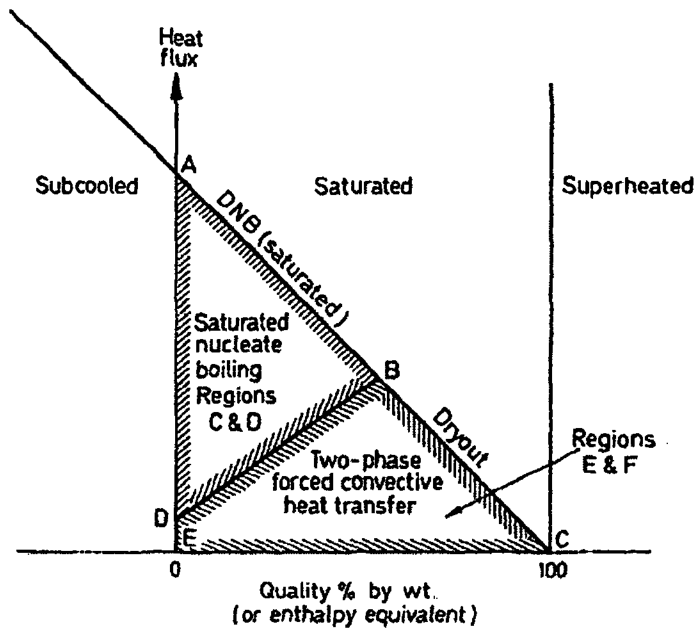
The load rejection and loss of flow events are the most challenging CPR events in a BWR and these events are therefore the principal CPR limiting events simulated in testing. The character of depressurization events will be discussed from the perspective of [

]

Consider first the boiling heat transfer and flow regime in relation to critical heat flux, Figure 16.1. This figure provides a map of boiling regime as a function of quality and shows the critical heat flux boundary. For a given state, a decrease in pressure increases the quality. Thus, for a constant heat flux, the state point moves to the right in Figure 16.1. Increasing the quality and moving to the right in Figure 16.1 leads away from nucleate boiling[1] and towards two phase forced convection heat transfer (annular flow regime). The heat transfer is so efficient that it is not possible to achieve the superheat at the rod surface necessary for nucleation.

The behavior in a boiling water reactor is fundamentally different than the behavior in a pressurized water reactor. In a pressurized water reactor, the starting point is single phase subcooled liquid in forced convection heat transfer at high heat fluxes. A decrease in pressure

1 It was already ascertained in the response to QUESTION 2 that, within the range of applicability of the ACE/ATRIUM-10 correlation, it is not possible to achieve DNB. Thus, critical heat flux will occur to the right of point B in Figure 16.1.



Reference 2, Page 251

Figure 16.1. Boiling Regime Map

(increase in quality) leads to movement toward the right in Figure 16.1. This movement is towards nucleate boiling and towards DNB.

[

]

Figure 16.2. Decreasing Pressure in a PWR

Figure 16.3. Decreasing Pressure in a BWR

Consider now measured critical power data from BWR and PWR. [

]

Depressurization in a BWR increases the void fraction. More voids result in less moderation and with less moderation, the nuclear power decreases. Thus, depressurization transients are of much less concern with respect to CPR in a BWR.

The principal depressurization event in BWR's is pressure regulator failure in the open position. The depressurization rates and pressures at scram were collected from licensing basis analyses of this transient for [

]

Figure 16.4. Depressurization Rate for Pressure Regulator Failure Open Transient

Figure 16.5. System Response to Pressure Regulator Failure Open Transient

The depressurization rate for one case is shown in Figure 16.4. The depressurization rate in this case is [ ]. The system response to this depressurization is shown in Figure 16.5. The core power level declines due to voiding but the core flow rate remains relatively constant.

[

]

As expected, throughout the transient, the MCPR increases. The behavior of the ACE/ATRIUM-10 correlation is consistent in its ability to predict the CPR with the approved SPCB correlation (Reference 8). This confirms that [ ] is acceptable and appropriate for modeling depressurization events.

Figure 16.6. Comparison of ACE to SPCB

[

]

Just as previous critical heat flux correlations are applicable to calculate CPR for the depressurization event, ACE/ATRIUM-10 correlation is also applicable, and appropriate for use.

Question 17. Table 6.2 lists the physical characteristics of the ATRIUM-10 test assembly. Please compare the values in the table with those of an ATRIUM-10 fuel element. Include part length rods and number and location of spacer grids. Compare the design of spacer grids used in the test assembly with those of ATRIUM-10 fuel. Discuss the significance of any differences between the test assemblies and the actual fuel that will be evaluated by the correlation.

Response 17:

The principal physical attributes of the production ATRIUM-10 design compared to the CPR test assembly is provided in Table 17.1. Nearly all of the parameters are identical. [

]

Table 17.2 compares axial features of the ATRIUM-10 design compared to the CPR test assembly. [

]

[

]

The next comparison is between the test spacers and the production spacers. This comparison is provided in Table 17.3. [

]

[

]

Table 17.1. Physical Attributes of ATRIUM-10 Compared to CPR Test

Table 17.2. Comparison of Axial Features of ATRIUM-10 to CPR Test

Table 17.3. Comparison of KATHY Test Spacer to Production Spacer

Question 18. It is the staff's understanding that ACE/ATRIUM-10 provides essentially a best estimate fit to data for which a standard deviation about the mean has been determined. The staff further understands that AREVA plans to demonstrate compliance with GDC 10 of Appendix A to 10CFR50 of the Commissions Regulations by showing that Acceptance Criterion 1.b of SRP 4.4 is met. Acceptance Criterion 1.b requires that at least 99.9% of the fuel rods in the core not experience dryout during normal operation or anticipated transients. Please describe how ACE/ATRIUM-10 will be used to demonstrate that this requirement is met and indicate when this methodology will be submitted for staff review.

Response 18:

The currently approved AREVA NP Inc. methodology for demonstrating compliance with GDC 10 of Appendix A to 10CFR50 (acceptance criterion 1.b of SRP 4.4) is Reference 9. This methodology has been generically approved by the U.S. NRC.

A comparison of the interface to this methodology between the ACE/ATRIUM-10 critical power correlation and the SPCB critical power correlation (Reference 8) is provided in Table 18.1. The key difference between the SPCB and ACE/ATRIUM-10 correlations that must be accounted for in the safety limit methodology is [

] The channel bow magnitude is not uniform along the axial length of the assembly and the in-channel void fraction affects the impact of the channel bow on the local peaking factors.

The following steps are performed in the safety limit methodology to address the impact of channel bow on the MCPR Safety Limit for the SPCB correlation: [

1.

]

[

]

[

]

To illustrate the safety limit methodology treatment of channel bow with the ACE correlation, a sample calculation was performed for a limiting ATRIUM-10 assembly that is operating at the safety limit MCPR power. The purpose of this sample calculation is to demonstrate the conservatism of the proposed approach relative to a more detailed approach [

]

[

] . Although the methodology for determination of safety limit does not use interpolation or extrapolation of local peaking on void fraction, the objective in this discussion is to examine what the "actual" assembly might be like for purposes of comparison with the methodology. Thus in establishing the "actual" bundle a linear extrapolation from the local peaking factors from 40% void and 80% void to higher void fractions is used when nodal voids exceed 80%. The detailed approach can then be compared to the proposed approach in order to demonstrate that the proposed approach is conservative.

[

]

These comparisons show that the proposed conservative approach can be used to bound a detailed approach for the determination of the impact of channel bow on [ ].

There are a number of conservatisms in the currently approved safety limit methodology for the determination of channel bow that are present for both the SPCB and ACE correlations. [

]

These specific attributes of the safety limit methodology provide additional assurance that the safety limit determined with the ACE correlation implemented into the methodology as described above is conservative.

The approach described above demonstrates that the ACE correlation can be used with the current generically approved safety limit methodology in a conservative manner. AREVA NP Inc. intends to submit a revision to the safety limit methodology for review at a later time (estimated submittal date is fourth quarter of 2007). The purpose of this future submittal is to [ ].

Table 18.1. Comparison of SPCB and ACE Interface to Safety Limit Methodology

Table 18.2. Typical local peaking factor sensitivity to channel bow

Figure 18.1. Sensitivity of planer maximum changes in local peaking factor as a function of lattice void fraction

Figure 18.2. Axial bow fraction as a function of elevation

Figure 18.3. K-Factor bowed minus K-Factor unbowed

Question 19. On page A-5, at the top of the page are listed 3 assumptions used in development of ACE/ATRIUM-10. Provide the qualitative and quantitative basis supporting these assumptions. Can these assumptions be classified as conservative?

Response 19:

The first stated assumption is [ ]. Quantitatively, this was described in detail in Question 1 and shown in Figure 3.24 in ANP-10249P, page 3-40.

The second stated assumption is that [ ]. Figure 5.5 in Reference 1 confirms this conservatism for ACE/ATRIUM-10.

The third stated assumption is [ ].

Question 20. Starting on the same page, 7 additional assumptions associated with writing the [ ] are presented. Please provide in detail the basis of these assumptions.

Response 20:

[

]

[

]

Question 21. On page A-9, equation A-18 references [ ] Do the referenced papers contain data in support of this premise?

Response 21:

[

]

Figure 21.1. [ ]

Question 22. On page A-12, the last sentence alludes to the inclusion of [ ]. Why would one assume that the inlet subcooled coolant would not be solid liquid?

<u>Response 22:</u>

[

]

Question 23. On page A-13, the first sentence of the first paragraph states that the right hand side of equation A-40 "commonly appears" in the definitions of fluid flows. Please provide a reference to this effect.

Response 23:

See Reference 3, Equation 4.91, on page 157. It is equilibrium quality.

Question 24. On page A-16, bottom of the page, the parameter [ ] is referenced. Where is it applied?

Response 24:

[

]

Question 25. It is not clear to the staff, starting with equation A.55, if the equation and the remainder of the discussion on this page and the next, pertains to rod critical power, sub-channel power, or assembly power. Please discuss in detail. Also, should not [

] ?

Response 25:

The power referred to in Equation A.55 on pages A-16 and A-17 is the assembly active channel power per unit length that appears as heat at the rod surface according to the definition on page A.5. [

]

Question 26. Section A.4 describes the axial differencing scheme for solution to the ACE/ATRIUM-10 correlation by which a reactor channel is divided into axial segments. Discuss how adequate noding is determined. Compare the noding used to correlate test data to that needed to evaluate reactor operation and transients. Discuss requirements for minimum noding structure for application of the correlation within the thermal/hydraulic computer programs in which it will be applied.

<u>Response 26:</u>

[

]

A nodal sensitivity study was performed to determine the largest size node that should be used. The nodalization study was performed with the defining data set, and the effect of node size on ECPR (Figure 26.1), correlation standard deviation (Figure 26.2), [

] was determined. The highlighted points in each of these figures corresponds to the node size used in developing the correlation [

].

The end user of the critical power correlation does not have the ability to change the nodalization. ACE/ATRIUM-10 is implemented in a code library and the nodalization is performed automatically within the library. The maximum node size is specified within the library [

].

Figure 26.1. ECPR Sensitivity to Number of Nodal Volumes

Figure 26.2. Standard Deviation of ECPR Sensitivity to Number of Volumes

Figure 26.3. [ ]

Question 27. Describe the procedures which will be in place for implementation of ACE/ATRIUM-10 which will ensure that the correlation is being properly applied within the range of physical conditions for which it has been validated. Discuss the training requirements for personnel utilizing the correlation.

Response 27:

ACE/ATRIUM-10 is implemented with a code library. Every computer code that uses the ACE/ATRIUM-10 correlation for calculation of critical power is linked to this code library. For every calculation with the critical power correlation, the limits of applicability are checked within the code library. [

]

The training for application of ACE/ATRIUM-10 is expected to include the following as a minimum:

- Overview of correlation
- Description of correlation inputs
- Description of correlation output
- Description of correlation interface files
- Limits of applicability
- Review of SER and any restrictions on use

Question 28. On page A-24, [ ] is assumed to be constant over the interval j-1 to j. This is only true if the interval is small. Please provide justification for this assumption.

Response 28:

A nodal sensitivity study was performed to address the assumption of linear variation in [ ]. The results of this sensitivity study are provided in Figure 26.1, Figure 26.2, and Figure 26.3. [

]. As described in Question 26, the maximum allowable node size is set in the correlation code library to insure that the node interval does not become too large.

Question 29. On page A.41, the second paragraph dealing with the [ ] needs to be supported with additional explanation.

Response 29:

[

]

Question 30. General topic: In this section, (Chapter 4), the statistical performance of the ACE correlation relative to the raw data is presented. However, no comparison(s) are made to those results of the SPCB-10 correlation statistics regarding overall correlation uncertainty, K-factor and additive constant values and uncertainties. I.e., please provide a table (graph) displaying statistical uncertainties associated with the ACE correlation and the SPCB-10 correlation, K-factor and additive constant uncertainties.

Response 30:

Table 30.1 provides a comparison of the key statistical performance attributes between ACE/ATRIUM-10 and SPCB for ATRIUM-10. [

]

Table 30.1. Comparison of ACE/ATRIUM-10 Statistical Performance to SPCB

Question 31. The first paragraph on page 4-6, the last sentence alludes to "optimizing" the correlation. Please clarify the statement.

Response 31:

[

] These unknowns are empirically fit to design specific experimental critical power data. The process of determining the coefficients is outlined in Appendix A. The coefficients are "optimized" as prescribed there to provide the best overall fit to the data.

[

]

Question 32. Please provide additional explanation regarding the last paragraph on page 4-6.

Response 32:

[

]

Question 33. Please provide additional explanation regarding Figure 4.8 on page 4-8.

Response 33:

[

]

Question 34. On page 5-7, Section 5.3, the importance of the ACE correlation in being able to predict [ ] is discussed. It was demonstrated in chapters 4 and 5 that in a few cases the ACE correlation under and over predicts [ ]. Discuss the impact of the under and over prediction on the overall correlation uncertainty.

Response 34:

[

]

[

]

Question 35. On page 5-16, the ACE correlation was applied to a different assembly. The [ ] had to be optimized to enhance the performance of the ACE correlation. Please provide additional discussion regarding the optimization process.

Response 35:

The optimization process is described in greater detail in Appendix A. A fuel assembly with a different design will have different performance and behavior, depending on how large the differences are between the design and ACE/ATRIUM-10. The ATRIUM-10P design has [

]. The conclusion from the study was that the correlation model that is the basis for ACE/ATRIUM-10 model can be successfully applied to other fuel assembly designs.

Question 36. On page 5-17, the exclusion of high inlet sub-coolant data is alluded to. Please discuss the reason as to why the defining data base did not include high inlet sub-cooled data.

Response 36:

[

] uncertainty that is used in the safety limit methodology is based on inclusion of the high inlet subcooling data. [

]

Question 37. On page 5-19, the second paragraph from the bottom requires additional explanation. Please provide additional clarification.

Response 37:

The ACE/ATRIUM-10 critical power correlation is a best-estimate correlation to steady-state critical power data. This means that the correlation is not biased, does not exhibit any significant trends in key experimental variables, and is well-behaved. It is appropriate for use in the safety limit methodology.

In addition to being used to determine the safety limit, the critical power correlation is also used in transient analysis to establish the licensing basis ΔCPR. The application of a steady-state critical power correlation to transients is accepted to be conservative. The conservatism is confirmed by comparing the measured time to dryout to the calculated time to dryout with test data simulating the load rejection transient and the flow runback transient. If the correlation predicted time to dryout is less than the measured time to dryout, the correlation is conservative in predicting the transient CPR.

Experimentally, several simulations of the transients are performed. Some of the simulations do not lead to dryout, and the correlation also does not predict dryout. These results were not provided because they do not contribute to understanding the performance of the correlation in prediction of dryout in simulated transients (they are only qualitative because dryout is not predicted by the correlation and dryout is not measured in the test).

In a few cases, the ACE/ATRIUM-10 correlation predicts dryout when dryout was not measured in the test. This occurred in three of the transients. In none of the cases where dryout was measured was the time to dryout less than the calculated time to dryout. These indicate that the ACE/ATRIUM-10 correlation is conservative when applied to transients.

Question 38. Is the data base for ACE correlation the same as that for SPCB correlation?

Response 38:

No. As with SPCB, the ACE/ATRIUM-10 correlation development was performed according to the [ ]. The process of evaluating the data for use in correlation was repeated. Only data that could be shown to be unreliable were excluded. The justification for exclusion of these data points is described below.

Some data points were added. The steady-state critical power data points of STS-48.1 were not included in the development of SPCB because the correlation development had been completed prior to these data becoming available. These data points were added to the critical power data base used for ACE/ATRIUM-10 development. [

]

[

]

[

]

[

]

Figure 38.1. STS-17.4 Critical Power vs. Inlet Subcooling and Flow Rate

Figure 38.2. STS-17.4 Dryout Rod at Highest Flow Rate

Figure 38.3. Critical Power vs. Flow Peaking Rods 23, 24, 25, 33, 43 (ΔH=46 kJ/kg)

References

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

2. J. G. Collier and J. R. Thome. Convective Boiling and Condensation, 3rd Edition Paperback. Clarendon Press, Oxford. 1996.

3. R. T. Lahey, Jr. and F. J. Moody. The Thermal-hydraulics of a Boiling Water Reactor, Second Edition. American Nuclear Society, 1993.

4. [ ]

5. R. B. Bird, W. E. Stewart, and E. N. Lightfoot. "Transport Phenomena, 2nd. Ed." John Wiley and Sons, 2002.

6. H. Schlichting. "Boundary Layer Theory, 6th Ed." McGraw-Hill, 1968.

7. [ ]

8. EMF-2209(P)(A) Revision 2. "SPCB Critical Power Correlation." Framatome ANP, Inc., September 2003.

9. 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.

10. EMF-1997(P)(A) Revision 0. "ANFB-10 Critical Power Correlation: High Thermal Peaking Results." Siemens Power Corporation, July 1998.

11. [ ]

12. [ ]

Appendix A Overview of Determination of Correlation Coefficients

]

[

]

[

]

Figure A.1 Correlation Fitting Process

Appendix B [ ]

AREVA

July 31, 2008
NRC:08:054

Document Control Desk
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Washington, D.C. 20555-0001

Request for Review and Approval of ANP-10249P, Revision 0, Supplement 1, Revision 0, "ACE Additive Constants for ATRIUM-10 Fuel"

Ref. 1: Letter, Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), "Publication of ANP-10249PA, Revision 0, "ACE/ATRIUM-10 Critical Power Correlation," NRC:07:048, September 26, 2007.

AREVA NP Inc. (AREVA NP) requests the NRC's review and approval of the enclosure, ANP-10249P, Revision 0, Supplement 1, Revision 0, "ACE Additive Constants for ATRIUM-10 Fuel." This report presents revised ACE critical power correlation additive constants for ATRIUM-10 fuel for BWRs. The constants were revised in response to an error discovered in the evaluation of the laboratory data when accounting for the power contained in the part length fuel rods.

Proprietary and non-proprietary versions of the topical report supplement are enclosed.

AREVA NP considers some of the material contained in the enclosed document to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure.

If you have any questions related to this submittal, please contact Ms. Gayle F. Elliott, Product Licensing Manager. She may be reached by telephone at 434-832-3347 or by e-mail at gayle.elliott@areva.com.

Sincerely,

Ronnie L. Gardner, Manager
Corporate Regulatory Affairs
AREVA NP Inc.

Enclosures

cc: H. D. Cruz
Project 728

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the information contained in topical report ANP-10249P, Revision 0, Supplement 1, Revision 0, "ACE Additive Constants for ATRIUM-10 Fuel," dated July 2008 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

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(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 31st

day of July, 2008.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
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My Commission Expires Oct 31, 2010

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ANP-10249NP
Revision 0
Supplement 1
Revision 0

ACE Additive Constants for ATRIUM-10 Fuel

July 2008

AREVA

ANP-10249NP
Revision 0
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Revision 0

ACE Additive Constants for ATRIUM-10 Fuel

Nature of Changes

Item	Page	Description and Justification
1.	All	This is a new document.

Contents

Tables

Figures

This document contains a total of *21* pages.

Nomenclature

Acronym	Definition
ACE	AREVA Critical Power Evaluator
BT	Boiling Transition
BWR	Boiling Water Reactor
CPR	Critical Power Ratio
ECPR	Experimental Critical Power Ratio (Measured / Calculated Critical Power)
K_{exp}	Experimental K-factor
PLR	Part Length Rod

1.0 Introduction and Summary

This document presents revised ACE critical power correlation additive constants for ATRIUM™*-10 fuel for boiling water reactors (BWR). The additive constants were revised in response to an error discovered in the evaluation of the laboratory data when accounting for the power contained in the part length fuel rods. Evaluations have confirmed that the ACE critical power correlation constants do not require revision as a result of the error.

Since the error discussed in this report is restricted to CHF testing of the ATRIUM-10 fuel, applications of ACE to co-resident BWR fuel containing part length fuel rods using the NRC approved method described in Reference 1 do not require revision.

This document presents the revised additive constants for ACE and some of the associated results. Included also is a discussion of the impact of the error on the test data and the modifications made to correct it. These revised additive constants supersede the additive constants for ATRIUM-10 provided in Reference 2 (Table 3.3, pg. 3-27).

The impact of the revision on the transient behavior is also addressed. Repeating the evaluation exactly as was done in Reference 2 shows the revision remains conservative. However, an inconsistency was found in the earlier work with respect to the definition of the time of boiling transition for the calculated results. Upon resolving this inconsistency, two of the transient cases no longer show a conservative result. These two cases are considered not prototypic of actual licensing analyses. The transient evaluations are discussed in more detail in Sections 6 and 7.

* ATRIUM is a trademark of AREVA NP Inc.

2.0 Revised Additive Constants

The revised additive constants for use with the ACE critical power correlation for the ATRIUM-10 fuel are shown in Figure 2.1. For comparison purposes, the ACE additive constants from the original submittal, Reference 2, are shown in Figure 2.2. The comparison between the predicted and measured critical power data with ACE for the ATRIUM-10 database is shown in Figure 2.3.

Figure 2.1 Revised ATRIUM-10 Additive Constants for ACE

Figure 2.2 Original ATRIUM-10 Additive Constants for ACE

Figure 2.3 Comparison of Predicted and Measured Critical Power Data with ACE for ATRIUM-10 Data

3.0 Test Data Modifications

The test facility uses electrically heated rods to simulate the behavior of the fuel bundle in the reactor core. The thermal power generated in the individual rods is readily calculated by using the known voltage, current, and/or the resistance of the various components. The surfaces of the simulated rods serve as the electrical conductors for the full length rods. The part length rods carry the current (DC) on the surface of the rod in one direction and then through an inner copper conductor in the opposite direction. Consequently, the power for the part length rods should account for the power associated with the current at the surface of the rod and in the portion of the inner copper conductor that is contained within the heated length. The initial method for determining the power distribution within the bundle did not properly account for the power of the inner copper conductor of the part length rods in the test bundle. As the result, the power distributions, both axial and radial, and the total power generated in a bundle, required modification to properly account for the thermal power generated in the inner copper conductor in the part length rods.

3.1 *Lattice Peaking Distribution*

The power distribution can be visualized by lattice. When the power carried in the inner copper conductor of the part length rods are included, the relative power delivered by the part length rods in the lower lattice (fully rodded region below the upper end of the part length rods) of the bundle increases compared to the previously reported values of power. Therefore, on a normalized relative power basis, the radial peaking factors of the part length rods increase and the radial peaking factors of the full length rods decrease in the fully rodded region of the bundle. Figure 3.1 shows the original, uncorrected power distributions of the lower lattice region and the upper lattice region (region above the end of the part length rods) for test STS 17.1 as presented in Reference 2 (pg. 6-36). The corrected power distribution accounting for the inner copper conductor is shown in Figure 3.2. The upper lattice radial power distribution is not impacted because the upper lattice does not have inner copper conductors.

The ACE critical power correlation is impacted because the lower lattice power distribution is used in the process of establishing the additive constants applied to each rod position in the fuel. The changes in the relative power distribution in the lower lattice are accounted for in the correlation through changes in the additive constants. Changes in additive constants are brought about because the base K-factor values change. The base K-factor values are the values that include only the effect of pin local peaking and do not include local effects from spacers and geometry which are accounted for in the additive constants.

3.2 *Bundle Power*

The previously reported bundle operating power for each data point is slightly larger than it should be. The treatment of the power associated with the inner copper conductor was not correctly calculated in the previous interpretation of the data. The previous methodology included the entire length of the inner copper conductor of the part length rods in the determination of the bundle power. This was incorrect since the inner copper conductor extends beyond the heated length of the bundle. The new method includes only that part of the inner copper conductor that lies within the heated length of the bundle in the calculation of the bundle power. The correction consisted of determining the fraction of the previously reported power that was produced in that part of the inner copper conductor which was not within the heated length of the test bundle. This fraction is a constant for a particular test and the test power is corrected by reducing the previously reported power by this fraction. For example, in test STS 17.1, the fractional power change is [ ]. Reference 2 identifies the measured power for a test (run 325.1) as [ ]. The adjusted power for this condition, as used in calculations becomes:

[ ]

3.3 *Axial Power Distribution*

The inclusion of the power associated with the inner copper conductor of the part length heater rods impacts the axial power shape of the part length rods, and consequently impacts the bundle average axial power. However, because the power associated with the inner copper conductor is such a small fraction of the overall bundle power (much less than 1%) the impact is small. This revision developed the required axial power shape from measurements of the individual rod axial shapes and, for the part length heater rods, the incorporation of the inner copper conductor. An example of comparing the bundle average axial power shape for the bundle STS 17.1 is shown in Figure 3.3. Although the impact is small, the impact on the bundle axial power shape was included in the revised additive constant calculations.

Figure 3.1 Original Peaking Pattern STS 17.1
Dryout detections indicated by heavy outline

Figure 3.2 Revised Peaking Pattern STS 17.1
Dryout detections indicated by heavy outline

Figure 3.3 Bundle Axial for STS 17.1

4.0 Impact on Additive Constants

Using the corrected axial and radial power distributions and the corrected bundle power, calculations were performed to determine the boiling transition values of K-factor for each test in the data base. The experimental K-factors are those values that result in a critical power ratio of 1.0 at the measured operating condition. [

] The resulting set of additive constants is shown in Figure 2.1. The analyses show that the resulting additive constant uncertainty becomes [ ]. The additive constant uncertainty from Reference 2 was [ ].

The uncertainty in the additive constants associated with high local peaking was also determined. [

] The resulting ATRIUM-10 additive constant uncertainty for high local peaking is shown in Table 4.1.

Table 4.1 Additional Additive Constant Uncertainty for High Local Peaking

5.0 ACE Conservatisms

The revised additive constants result in the set of K-factor values shown in Table 5.1 for the tests comprising the ACE ATRIUM-10 database. This set of values provides an average [ ]. The average ECPR and standard deviation values from Reference 2 were [ ], respectively. Comparing these K-factor values with the K_{exp} values that place the ECPR of the bundle at 1.0, the number of rods calculated to be in boiling transition for each data point of each test bundle can be determined. This allows for the calculation of the ratio of the number of rods calculated to be in boiling transition to the number of rods observed to be in boiling transition. This value is determined to be [ ] for the revised additive constants.

Table 5.1 K-Factor Values for ECPR Evaluation of ATRIUM-10

The predicted values of critical power compared to the measured values are presented in Figure 2.3 for the ATRIUM-10 data base.

6.0 Evaluation of Transient Critical Power Data

Analysis of the transient critical power tests presented in Reference 2 were repeated using the revised initial bundle powers, axial power shapes, and K-factor values. The repeat analysis was performed consistent with Reference 2, assuming that all of the power delivered by the electrically heated rods, including the power associated with the inner cooper conductors, was deposited with no time delay to the fluid. The calculated time of boiling transition of each test for the repeat analysis are presented in Table 6.1 along with the measured time to boiling transition and the calculated time presented in Reference 2. As illustrated in Table 6.1, the changes to initial bundle powers, axial power shapes, and K-factor values did not impact the conclusions in Reference 2. The repeat analysis demonstrated the ACE steady-state dryout correlation continues to be appropriate for use in evaluating transient events.

A second analysis was performed to bound the impact of the thermal time constant associated with conducting the heat from the inner copper conductors of the 8 part length rods, through the electrical insulation, and into the fluid. The analysis accounted for the thermal time-constant effect by neglecting all power generated in the inner copper conductors, including contributions to the axial power shape and bundle power as a function of time. This analysis demonstrated very little sensitivity to the inner copper conductors.

The second bounding analysis was performed because explicit modeling of the inner copper conductors and the associated thermal time constant response could not be directly modeled in the XCOBRA-T code. Explicitly modeling full length rods *without* inner copper conductors and part length fuel rods *with* inner copper conductors would have required the use of multiple rod models.

Table 6.1 XCOBRA-T Results Using Nominal K-Factor

7.0 Redefinition of Calculated Time of Boiling Transition

During the preparation of the transient analyses discussed in Section 6.0 the definition for the calculated time of boiling transition from XCOBRA-T was found to be inconsistent between earlier work, such as with SPCB in Reference 3, and that used in Reference 2. In the earlier work, the definition of calculated time of boiling transition was taken as the first time step in which critical power is calculated to be at or below 1.0. However, the work presented in Reference 2 assumed the calculated time of boiling transition to occur at the last time step before critical power is calculated to be below 1.0. The difference arising from these two definitions is that the historical definition indicates more conservatism by one calculational time step when considering its application to demonstrating the applicability of a steady-state CPR correlation to transients.

So as to avoid mixing issues, the re-analysis presented in Section 6.0 and in Table 6.1 does not address the definition for redefinition of the calculated time of boiling transition. Table 6.1 shows only the impact of the error discovered in the evaluation of laboratory data when correctly accounting for the overall bundle power and the power contained in the part length fuel rods.

The transient analyses using XCOBRA-T presented in Reference 2 were performed with a computational time step size of 0.05 seconds which is consistent with the experimental data acquisition rate which was also 0.05 seconds. As a consequence, the calculated time of boiling transition presented for all tests in Table 6.1 will indicate that boiling transition occurred 0.05 seconds later and the difference in measured and calculated times will be 0.05 seconds less. The XCOBRA-T results using the historical definition of calculated time of boiling transition are presented in Table 7.1.

With the exception of two test points, Table 7.1 shows the calculated time of boiling transition is less than or equal to the time of boiling transition measured in the transient tests. An inspection of these two tests and the three others which have a calculated time equal to the measured time, was undertaken. The particular test points considered are STS_17.8_H1.4, STS_17.8_H2.2, STS_17.8_U2.2, STS_17.8_U6.2, and STS_29.5_H100.1. The inspection concluded the following;

a) For test point STS_17.8_U6.2, inspection of the boundary conditions show the test was, in fact, very extreme. While the test was intended to simulate a flow decay event along with a corresponding power decay, the power decay was delayed by nearly a full second after the initiation of the flow decay. Actual flow decay events in operating power plants experience an instantaneous power decrease during a flow decay transient as a consequence of neutronic feedback. This test is viewed as an atypical extreme condition.

b) For test point STS_29.5_H100.1, it was observed that the boundary conditions had an initial power that was too high. Test point STS_29.5_H100.4 had very similar initial conditions, used the same protocol for the transient boundary conditions, but had a lower initial bundle power. The calculated time of boiling transition was conservatively predicted in the lower power case. It is important to note that this lower power case is representative of how XCOBRA-T is applied in licensing; the higher power case would not be analyzed because boiling transition was already predicted to occur at the lower power. Therefore, test point STS_29.5_H100.1 is viewed as an atypical extreme condition.

c) The algorithm used in the Reference 2 work to extract the "measured time of boiling transition" from experimental data has been determined to represent the strictest and most conservative definition. The algorithm is based on evaluating the derivatives of each thermocouple trace to determine the first moment where the trace exhibits a defined upward trend. An alternate conservative definition may also be applied that is based on determining the location where a change of 2 °F measured temperature (or more) occurs, and then the measured time of boiling transition is taken to be one experimental data acquisition time step size (0.05 seconds) earlier. Using this definition of the measured time of boiling transition, the predicted boiling transition time would be less than or equal to the measured time for all experimental data indicated in Table 7.1 except for STS_29.5_H100.1.

For test points STS_17.8_U6.2 and STS_29.5_H100.1, it would be appropriate to remove them from consideration as they represent atypical extreme test conditions. However, since both test points were provided in Reference 2, they are also presented herein for consistency with the original analysis. In addition, it was elected to maintain the conservative definition of "measured time of boiling transition" criteria utilized in the Reference 2 work to maintain consistency.

Inspection of Table 7.1 shows that the predictions of the remaining test points are conservative and the ACE correlation continues to be appropriate for use in evaluating transient events.

Table 7.1 XCOBRA-T Results Using Historical Definition of Calculated Time of BT

8.0 References

1. EMF-2245(P)(A), Revision 0, Applications of SPC Critical Power Correlations to Co-resident Fuel, August 2000.

2. ANP-10249PA, Revision 0, ACE/ATRIUM-10 Critical Power Correlation, August 2007.

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

December 21, 2011
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Request for Review and Approval of ANP-10249PA Revision 1, Supplement 1P, Revision 0, "Improved K-factor Model for ACE/ATRIUM 10 Critical Power Correlation"

AREVA NP Inc. (AREVA) requests the NRC's review and approval for referencing in licensing action ANP-10249PA, Revision 1, Supplement 1P. Revision 0, "Improved K-factor Model for ACE/ATRIUM 10 Critical Power Correlation."

Upon approval, AREVA intends to incorporate the enclosed Supplement 1 information into the previously approved Topical Report ANP-10249PA, Revision 1 "ACE/ATRIUM-10 Critical Power Correlation," dated September 2009 to create Revision 2 of the approved Topical Report. Please provide recognition of this in the final Safety Evaluation for the enclosed Supplement 1.

Proprietary and non-proprietary versions of the topical report supplement are enclosed.

AREVA considers some of the material contained in the enclosed document to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure.

In support of the Office of Nuclear Reactor Regulation's prioritization efforts, the prioritization scheme matrix is attached.

If you have any questions related to this submittal, please contact Mr. Alan B. Meginnis, Product Licensing Manager at 509-375-8266 or by e-mail at alan.meginnis@areva.com

Sincerely,

Pedro Salas, Manager
Corporate Regulatory Affairs
AREVA NP Inc.

cc: H. D. Cruz
Project 728

ANP-10249PA, Revision 1, Supplement 1P, Revision 0
Improved K-factor Model for ACE/ATRIUM 10
Critical Power Correlation

TR Prioritization Scheme Matrix
*Industry input on shaded areas was not requested.

Factors	Select the Criteria that the TR Satisfies	Points Assigned for Each Criteria	Total Points (if points are cumulative, total them for each factor in this column)
TR Classification (Points are cumulative)	Generic Safety Issue	6	1
	Emergent Technical Issue	3	
	Standard TR	1	
Applicability (Points are not cumulative)	Industry-Wide Implementation	3	2 (TR is applicable to all BWRs)
	Applicable to entire groups of licensees (BWROG, PWROG, BWRVIP, etc.)	2	
	Applicable only to partial groups of licensees	1	
Specialized Resource Availability (Points are cumulative)	NRC staff expertise is readily available (The NRC staff will evaluate this criteria)	1.5	
	Technical data is available/readily accessible (The NRC staff will evaluate this criteria)	1	
	TR approval is needed by a certain date to support a licensing activity. Explain when and why.	0.5	
Total Points (Add the total points from each factor and total here):			3

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the report ANP-10249PA, Revision 1, Supplement 1P, Revision 0, "Improved K-factor Model for ACE/ATRIUM 10 Critical Power Correlation," dated December 2011 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 20th day of December, 2011.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/10/12

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
01-10-12
STATE OF WASHINGTON

ANP-10249NPA
Revision 1
Supplement 1NP
Revision 0

Improved K-factor Model for ACE/ATRIUM-10 Critical Power Correlation

December 2011

AREVA NP Inc.

AREVA

AREVA NP Inc.

ANP-10249NPA
Revision 1
Supplement 1NP
Revision 0

Improved K-factor Model for ACE/ATRIUM-10 Critical Power Correlation

AREVA NP Inc.

ANP-10249NPA
Revision 1
Supplement 1NP
Revision 0

Improved K-factor Model for ACE/ATRIUM-10 Critical Power Correlation

Nature of Changes

Item	Page	Description and Justification
1.	All	New document.

Contents

Tables

Figures

Nomenclature

Acronym	Definition
ACE	AREVA Critical power Evaluator
AOO	Anticipated Operational Occurrence
BT	Boiling Transition
BWR	Boiling Water Reactor
CPR	Critical Power Ratio
ECPR	Experimental Critical Power Ratio; the ratio of calculated to the measured critical power
LOCA	Loss Of Coolant Accident
MCPR	Minimum Critical Power Ratio
PLR	Part Length Rod

1.0 Introduction and Summary

This document presents a revision to the ACE critical power correlation for ATRIUM™[1] 10 fuel for boiling water reactors (BWR). The K-factor methodology was revised in response to deficiencies found in the axial averaging process. In addition, the additive constants were updated as a result of the change to the K-factor model. Evaluations were performed that confirmed that some of the ACE/ATRIUM-10 critical power correlation coefficients require revision as a result of these changes.

Reference 1 provides a description of the rod local peaking function (called K-factor). The revised K-factor method for use with the ACE/ATRIUM-10 critical power correlation is described in this document. This revised method supersedes the one described in Reference 1. This document also describes the minor changes in the method for determining additive constants that became necessary due to the changes in the K-factor methodology.

The comparison between measured and predicted critical power data is shown in Figure 1-1. The correlation experimental critical power ratio (ECPR) mean with the revised K-factor methodology and updated additive constants is [ ] and the ECPR standard deviation is [ ]. The ECPR mean and standard deviation from Reference 1 were [ ] and [ ] respectively.

The range of applicability of the critical power correlation is unchanged from Reference 1. The revised correlation is applicable to steady-state design and analysis, core monitoring, anticipated operational occurrences (AOO's), accidents, LOCA, and instability analysis for the ATRIUM-10 fuel design. This correlation can also be used to model non-AREVA co-resident fuel consistent with the approved methodology described in Reference 2.

1 ATRIUM is a trademark of AREVA NP Inc.

[ ]

Figure 1-1: Comparison of Calculated to Measured Critical Power

2.0 Standard Review Plan Requirements

There are no critical power correlation specific requirements in the standard review plan.

3.0 Revised Correlation

All modern critical power correlations contain a function that accounts for rod peaking. This function is called K-factor in the ACE correlations. The model equation for the ACE correlation is given in Equation 3.1 of Reference 1 (including symbol definitions). The revision is in the [ ] term:

[ ] (3.1)

The K-factor, [ ]

This assumption was found to be inappropriate because (1) it allows downstream conditions above the location of dryout to non-physically influence the critical power, and (2) it provides equal weighting to all axial locations (low power regions as well as regions far from the location of dryout). Both of these problems were found to be capable of influencing the predicted results in a non-conservative manner.

3.1 *Critical Power Correlation Coefficient Changes*

The change in K-factor in the critical power correlation [ ] can introduce small changes to the [ ]. To maintain equivalent critical power correlation behavior, it was necessary to adjust the coefficients related to the [ ]. These coefficients are described in Section 3.6 of Reference 1. The changed [ ] are the following:

[ ]

[

] The following values are obtained:

It is observed that the changes to the [ ] are small, as expected.

3.2 *Rod Peaking Function*

The K-factor characterizes the rod peaking effect on the bundle critical power. The critical power varies inversely with K-factor. That is, as K-factor increases in value, the critical power decreases in value. [

]

[

]

This description of the local rod peaking function is unchanged from the description in Reference 1.

3.3 *Applying Rod Peaking Function in the Critical Power Correlation*

[

] The maximum of the averaged

K-factors over all the rods was then chosen for use in the critical power correlation according to Equation 3.46 in Reference 1. This averaging of the axial K-factor distribution for each rod was found to be inappropriate for the reasons discussed in Section 3.0 and is therefore excluded in the revised K-factor method.

[

] Thus this solution explicitly addresses both problems noted in Section 3.0.

In the revised method, [

]

3.4 *Method for Calculating Additive Constants*

The spacers and bundle geometry characteristics influence the critical power behavior of the individual rods within the fuel bundle. Therefore, a factor is needed to distinguish the critical power performance of each rod. These position dependent factors are termed additive constants. Additive constants can be considered as a flow/enthalpy redistribution characteristic for a given bundle and spacer design.

In critical power testing, [

]

In accordance with the [ ] the CHF database was randomly divided into a defining data set and a validating data set. Approximately [ ] was set aside as the validating set of data. The remaining [ ] form the defining data set and were used to develop the critical power correlation. The additive constants for all the rod positions were determined from the defining data set. The calculation of additive constants uses the same partition of data as was used during the critical power correlation development. [ ]

The defining and validating data sets used for correlation development in Reference 1 are unchanged. The additive constants are determined [ ]

[ ]

[ ]

3.4.5 Additive Constants for ACE/ATRIUM-10 Correlation

The revised ATRIUM-10 additive constants are shown in Figure 3-4. For comparison purposes, the original ACE/ATRIUM-10 additive constants from Reference 1 are compared in Figure 3-5. The observed changes in additive constant are generally small and [ ]

Figure 3-1: Adjacent Rod Identification for K-factor Calculation

Figure 3-2: Rods Observed to Dryout in Testing

Figure 3-3: Peaked Symmetric Rods Not Observed to Dryout in Testing

[ ]

Figure 3-4: ACE/ATRIUM-10 Additive Constants

Figure 3-5: Additive Constant Comparison

3.5 *Additive Constant Uncertainty*

The overall uncertainty in additive constants is determined [ ]. The following steps are applied.

[ ]

[ ]

The resulting overall additive constant uncertainty for ACE/ATRIUM-10 is [ ]. The additive constant uncertainty from Reference 1 is [ ].

An additional high peaking uncertainty is imposed in the MCPR safety limit methodology for those rods whose local peaking exceeds [

] Table 3-1 shows the results of these calculations.

Table 3-1: Additive Constant Uncertainty for High Local Peaking

3.6 *Critical Power Correlation Conservatisms*

With the improved K-factor model, the ACE/ATRIUM-10 correlation has an average ECPR of [ ] with a standard deviation of [ ]. From Reference 1, the average ECPR was [ ] with a standard deviation of [ ]. The correlation was used to assess each rod in each of the tests. The associated critical powers of each rod were then compared to the measured critical power and a count made of the number of rods which were predicted to be in boiling transition (BT) and this was compared to the number of rods actually observed to be in boiling transition in the experimental data. With the revised K-factor methodology and additive constants, this ratio of predicted to measured rods in boiling transition is [ ]. This compares with a value of [ ] in Reference 1.

4.0 Transient Benchmarking

An industry accepted standard in BWR transient methodology is that steady-state dryout correlations are conservative for use in transient methodology. Transient dryout tests [ ] were performed to reconfirm this for the ATRIUM-10 fuel design when using the ACE/ATRIUM-10 critical power correlation.

The limiting transient tests of interest are the simulated load rejection without bypass (LRNB) events that consist of power and pressure ramps and flow decay and the simulated loss of flow events that consist of flow decay and power decay. The power, pressure, and flow were all controlled by a function generator. The forcing functions were programmed to produce the transient rod surface heat flux typical of the various events.

A total of [ ] ATRIUM-10 LRNB and loss of flow transients were run which were either measured or predicted to have dryout. Of these [ ] transient critical power tests, [ ] The initial conditions for these tests are given in Table 5.7 of Reference 1.

Evaluations of the transient critical power tests were repeated using the revised K-factor methodology. [ ] The AREVA NP transient thermal hydraulic code XCOBRA-T (References 4 and 5), was used to predict the transient test results using the ACE/ATRIUM-10 critical power correlation. The test power forcing function provides the boundary condition of power, which is modeled in XCOBRA-T [ ]

The results [ ] are summarized in Table 4-1.

[

]

The transient benchmark results with the revised correlation are consistent with those presented in Reference 1.

Table 4-1: XCOBRA-T Transient Dryout Results, [ ]

Table 4-2: XCOBRA-T [ ] Transient Dryout Results for [ ]

[ ]

Table 4-3: XCOBRA-T [ ] Transient Dryout Results [ ]

[ ]

5.0 [ ] K-factor Method

With the revised K-factor method, the critical power correlation is used to [

]

6.0 Implementation of Revised K-factor Methodology

The improved K-factor methodology will be implemented into MICROBURN-B2 (Reference 8), POWERPLEX-III (core monitoring), SAFLIM2 (Reference 6), SAFLIM3D (Reference 7), XCOBRA (Reference 11), XCOBRA-T (References 4 and 5), RELAX (Reference 9), RAMONA5-FA (Reference 10), and AURORA-B (References 13 and 14). It will be used in core design and analysis, core monitoring, MCPR safety limit methodology, AOO's, LOCA, and other codes and methods that use the critical power correlations. It will also be in the non-AREVA co-resident fuel methodology (Reference 2).

The MCPR safety limit methodology performs a rod-by-rod evaluation to estimate the number of rods in BT associated with a particular safety limit. [

]

7.0 References

1. ANP-10249PA Revision 1, "ACE/ATRIUM-10 Critical Power Correlation," AREVA NP Inc., September 2009.

2. EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-resident Fuel," Siemens Power Corporation, August 2000.

3. C. Bennett and N. L. Franklin. "Statistical Analysis in Chemistry and the Chemical Industry," Marbern House, October 1987.

4. XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Analysis," Exxon Nuclear Company, February 1987.

5. XN-NF-84-105(P)(A) Volume 1 Supplement 4, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis Void Fraction Model Comparison to Experimental Data," Advanced Nuclear Fuels Corporation, June 1988.

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

7. ANP-10307PA, Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, Inc., June 2011.

8. EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 / MICROBURN-B2," Siemens Power Corporation, October 1999.

9. EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP Richland, Inc., May 2001.

10. BAW-10255PA, Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," AREVA NP, Inc., May 2008.

11. XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, Inc., January 1987.

12. [ ]

13. ANP-10300P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios," AREVA NP Inc., December 2009.

14. ANP-10300P, Supplement 1, Revision 0, "Qualification of AREVA NP Neutronic Methodology for Extended Power Uprate Conditions and Extended Operation Domains," AREVA NP, Inc., June 2011.

AREVA

ACE/ATRIUM-10 Critical Power Correlation

ANP-10249NP
Revision 2

Topical Report

March 2014

AREVA Inc.

Abstract

A new critical power correlation for the ATRIUM-10 fuel design is described. This new correlation is based on a [ ]. Previous correlations were based on the Macbeth formulation of critical heat flux. In addition to predicting the critical power, [ ]

Revision 2 Nature of Changes

Item	Page	Description and Justification
1.	All	Company name changed
2.	All	Page header/footer format and content changed
3.	2-1	Correlation ECPR mean and standard deviation updated
4.	2-3	Figure 2.1 updated
5.	3-1, 3-2	Equation 3.1 updated with description of K-Factor
6.	3-6	Updated [ ]
7.	3-15	Updated correlation coefficients [ ]
8.	3-16	Add symbols for K-Factor and additive constant and [ ]
9.	3-18	Deleted Equations 3.45 and 3.46. Delete text stating [ ]
10.	3-19 to 3-28	Extensively revised to describe updated additive constant calculation methodology, additive constants, and uncertainties
11.	3-40	Added discussion of critical power correlation conservatism according to Supplement 1P
12.	4-38	Updated reference describing how experimental additive constant is determined.
13.	A-41	Updated K-Factor [ ]. Deleted footnote.
14.	A-42	Updated equation A.181 and A.182 [ ]
15.	A-42	Updated equation A.184 [ ]
16.	B-10	Deleted K-Factor example 7 because it is no longer part of the K-Factor methodology

Revision 2 replaces the items identified above. The information from Supplement 1P to Revision 1 is the basis for the modifications. Supplement 1P represents the information that was reviewed and approved by the NRC as stated in the SER. As demonstrated in Supplement 1P, the balance of sample analysis results presented in the Revision 2 document remain representative of those that would be obtained with the Supplement 1P modifications. Therefore, the remaining Revision 1 results have not been updated for issuance of Revision 2.

NOTE that the Supplement 1P modifications listed above attain approved status through the SER for Supplement 1P. The remainder of the document retains approved status associated with the Revision 0 SER and the Revision 1 SER.

Revision 1 Nature of Changes

Item	Page	Description and Justification
1.	2-3	Figure 2.1 Updated with Supplement 1 Figure
2.	3-27	Table 3.3 Updated with Supplement 1 values
3.	3-28	Table 3.4 Updated with Supplement 1 values
4.	3-30	Additive constant uncertainty updated with Supplement 1 value
5.	5-22	Table 5.8 Updated with Supplement 1 values

Revision 1 replaces the items identified above. The information from Supplement 1 to Revision 0 is the basis for the modifications. Supplement 1 represents the information that was reviewed and approved by the NRC as stated in the SER. As demonstrated in Supplement 1, the balance of sample analysis results presented in the Revision 1 document remain representative of those that would be obtained with the Supplement 1 modifications. Therefore, the remaining Revision 0 results have not been updated for issuance of Revision 1.

NOTE that the Supplement 1 modifications listed above attain approved status through the SER for Supplement 1. The remainder of the document retains approved status associated with the Revision 0 SER.

Contents

Tables

Figures

This document contains a total of 258 pages.

Nomenclature

Acronym	Definition
AOO	Anticipated Operational Occurrence
BOHL	Beginning of Heat Length
BWR	Boiling Water Reactor
CHF	Critical Heat Flux
CPR	Critical Power Ratio, defined to be the critical power divided by the power of the assembly.
DC	Direct Current
ECPR	Experimental Critical Power Ratio, defined to be the ACE/ATRIUM-10 calculated critical power divided by the experimentally measured critical power
EOHL	End of Heated Length
LOCA	Loss of Coolant Accident
LRNB	Load Reject with No Bypass
MCPR	Minimum Critical Power Ratio
PLHR	Part Length Heater Rod

1.0 Introduction

This document describes ACE/ATRIUM™-10*, AREVA Inc.'s critical power correlation for boiling water reactors (BWR). This correlation is designed for application to steady-state design and core monitoring, transient anticipated operational occurrences (AOO's), transient accidents, LOCA, and instabilities for ATRIUM-10 fuel designs.

The starting point for a correlation is the theoretical model that describes the point of maximal heat transfer in boiling, sometimes termed critical heat flux, boiling transition, or dryout. For ACE/ATRIUM-10, a theoretical model is constructed that uses [

].

Although the theoretical model provides greater [

]. The theoretical model for the correlation is robust and is expected to be easily applied to future fuel assembly designs.

* ATRIUM is a trademark of AREVA Inc. a company registered in the United States and various other countries.

2.0 Summary

The ACE/ATRIUM-10 correlation can be used to accurately predict assembly critical power [ ] for the ATRIUM-10 fuel design. The correlation provides accurate prediction of the limiting rod within an assembly. The impact of local spacer effects and assembly geometry on critical power is accounted for by two different sets of parameters. The first is a set of constants, one constant for each rod in the assembly. These constants are called additive constants and are presented in [ ] for the ATRIUM-10 design. The second set of parameters provides [ ].

For comparison of correlation predictions to experimental data, an experimental critical power ratio (ECPR) is defined to be the ratio of the calculated critical power to the measured critical power. The ECPR distribution associated with ACE/ATRIUM-10 is adequately represented with a normal curve using an overall mean of [ ].

2.1 *ACE/ATRIUM-10 Database*

The ACE/ATRIUM-10 database is comprised of [ ] steady-state data points taken on [ ] different test assemblies. The rod axial power shapes of the tests were [ ] peak-to-average chopped cosine and [ ] peak-to-average upskew and downskew. The database was compiled from tests performed exclusively at the AREVA thermal hydraulic test facility located in Karlstein, Germany.

During the correlation development, the database was randomly divided into a defining data set and a validation data set. In accordance with the criteria set forth in [ ], approximately [ ] was set aside as the validating set of data. The remaining [ ] form the defining data set and were used to develop the critical power correlation. [ ]. In addition, another [ ] validation points taken from steady-state critical power tests of fuel design variants not included in the database were analyzed. In addition to the steady-state data, transient tests were performed on an ATRIUM-10 test assembly with both a cosine and upskew axial power distribution as part of the correlation validation.

The dryout tests were designed to represent the range of local conditions present in an operating BWR fuel assembly. The database and correlation address the effects due to operating pressure, mass flow rate, inlet subcooling, axial power profile, and local peaking distribution. Table 2.1 provides a range of applicability based on parameters tested, design experience, and correlation capability.

2.2 *ACE/ATRIUM-10 Comparison to the Database*

The ACE/ATRIUM-10 critical power correlation has been used to predict the critical power for each data point in the database. The ECPR has been determined for each test point and is used along with the standard deviation of the ECPR [ ] as the basis to determine the ability of the correlation to predict the onset of dryout. Comparison of the predicted to the measured critical power for ATRIUM-10 is shown in Figure 2.1.

Table 2.1. ACE/ATRIUM-10 Range of Applicability

Figure 2.1 Comparison of Predicted to Measured Critical Power Data

3.0 ACE/ATRIUM-10 Correlation

The single phase subcooled flow at the inlet of a BWR fuel assembly rapidly transitions through bubbly flow to annular flow. In the MCPR (Minimum Critical Power Ratio) limiting fuel assemblies, much of the active length of the fuel assembly is in annular flow. A liquid film on the rod and a steam-water mixture in the center region characterizes the annular flow regime. As the flow progresses upward, the water film changes because of evaporation, entrainment, and the deposition of water droplets onto the liquid film. A rapid temperature excursion occurs when the cooling effectiveness of the liquid film is lost. The loss of this liquid film is variously termed dryout, boiling transition, and critical heat flux (CHF).

The ACE/ATRIUM-10 correlation is a new correlation based on [

]. A detailed derivation of the ACE/ATRIUM-10 critical power correlation form is provided in Appendix A. The correlation is based on [

]

* [

]

The resulting overall additive constant uncertainty for ACE/ATRIUM-10 is [

]

An additional high peaking uncertainty is imposed in the MCPR safety limit methodology for those rods whose local peaking exceeds [

]

[ ] Table 3.4 shows the results of these calculations.

Table 3.4. Additive Constant Uncertainty for High Local Peaking

3.15 ***Critical Power Correlation Conservatisms***

The ACE/ATRIUM-10 correlation has an average ECPR of [ ] with a standard deviation of [ ]. The correlation was used to assess each rod in each of the tests. The associated critical powers of each rod were then compared to the measured critical power and a count made of the number of rods which were predicted to be in boiling transition (BT) and this was compared to the number of rods actually observed to be in boiling transition in the experimental data. This ratio of predicted to measured rods in boiling transition is [ ] used in safety limit analyses.

4.0 Assessment of ACE/ATRIUM-10 With Defining Data Set Critical Power

The performance of the ACE/ATRIUM-10 critical power correlation compared to the defining data set is provided in this section. The following topics are presented:

- ECPR trend plots comparing the critical power to the defining data set as a whole (Section 4.1)
- [ ] performance (Section 4.2)
- Statistical analysis by single variable subset, test subset, and in some cases two variable subsets of the defining data set (Section 4.3)
- ECPR as a function of mass flow rate by test (Section 4.4)
- Additive constant statistical distribution (Section 4.5)

4.1 *Overall Critical Power and ECPR Behavior (Defining)*

The ACE/ATRIUM-10 correlation is compared to measured data with respect to critical power, mass flow rate, pressure, inlet subcooling, axial power shape, and K-Factor to examine overall trends.

The correlation predicted critical power is plotted against the measured critical power in Figure 4.1. The data fall in a narrow, well-defined band about the expected value, consistent with the overall standard deviation. No trends are evident.

Figure 4.2 shows the ECPR as a function of mass flow rate. Mass flow rate is the most significant parameter in the critical power correlation. Examination of the data shows that no trend is evident. A trend line representing a linear least squares fit of the ECPR as a function of mass flow rate is shown on the plot, confirming that no significant trends with mass flow rate exist.

The ECPR is plotted as a function of pressure in Figure 4.3. The data show no significant trends in pressure. Figure 4.4 shows the ECPR as a function of the inlet subcooling. There is no apparent trend with inlet subcooling. The ECPR is plotted as a function of axial power shape in Figure 4.5. There is no apparent trend with axial power shape. Figure 4.6 shows the ECPR as a function of K-Factor. There is no apparent trend with K-Factor.

Figure 4.1 Calculated vs. Measured Critical Power (Defining)

Figure 4.2 ECPR as Function of Mass Flow Rate (Defining)

Figure 4.3 ECPR as Function of Pressure (Defining)

Figure 4.4 ECPR as Function of Inlet Subcooling (Defining)

Figure 4.5 ECPR as Function of Axial Power Shape (Defining)

Figure 4.6 ECPR as Function of K-Factor (Defining)

4.3 *ACE/ATRIUM-10 Statistical Analysis of Defining Data Set*

Overall statistics of the fit of the ACE/ATRIUM-10 correlation fit to the defining data set are provided in Table 4.1. [ ] are in excellent agreement with the experimental data.

Higher moments for the ACE/ATRIUM-10 correlation analysis of ECPR are computed. Reference 10 provides the relationships for computing the higher order moments about the mean. The second moment about the mean is calculated

$$m_2 = \frac{1}{n}\sum_{i=1}^{n}(x_i - \bar{x})^2 \tag{4.1}$$

The third moment about the mean is calculated

$$m_3 = \frac{1}{n}\sum_{i=1}^{n}(x_i - \bar{x})^3 \tag{4.2}$$

The fourth moment about the mean is calculated

$$m_4 = \frac{1}{n}\sum_{i=1}^{n}(x_i - \bar{x})^4 \tag{4.3}$$

A measure of skewness is given by

$$\sqrt{\beta_1} = \frac{m_3}{m_2^{1.5}} \tag{4.4}$$

A measure of kurtosis is given by

$$\beta_2 = \frac{m_4}{m_2^2} \tag{4.5}$$

These statistics, computed for the ECPR from the ACE/ATRIUM-10 correlation, are summarized in Table 4.2. [ ]. The distributional character of the ACE/ATRIUM-10 critical power ratios are shown in Figure 4.9 and Figure 4.10. Figure 4.9 is a histogram of the frequency of occurrence of CPR while Figure 4.10 shows that the distribution [

].

The number of degrees of freedom [

] The conclusion is that the correlation has not been over-fit.

Table 4.1. Overall Statistics (Defining)

Table 4.2. Higher Moments of ECPR Mean (Defining)

Figure 4.9 Frequency Distribution of ECPR (Defining)

Figure 4.10 Expected Value for [ ] Distribution of ECPR (Defining)

4.3.1 Statistics by Single Variable Subsets

The descriptive statistics for the overall data can be examined by several subgroups of data. Mean, standard deviation, and number of data are presented. The correlation predictions are compared to the defining data set.

Table 4.3 provides a summary of descriptive statistics by mass flow rate and overall across the defining data set.

[

]. The conclusion that there are no significant trends with mass flow rate are supported by the data in this table.

Table 4.4 provides statistics by subgroups of pressure. There are no significant trends. [

]

[

]. The conclusion that there are no significant trends with pressure is supported by the data in this table.

Table 4.5 provides statistics by subgroups of inlet subcooling. There are no significant trends [ ] or outliers.

Table 4.6 provides statistics by subgroups of axial power shape. There are no significant trends [ ] or outliers. [

]

Table 4.7 provides statistics by subgroups of K-Factor. There are no significant trends [ ] or outliers.

Table 4.3. Statistics by Binned Mass Flow Rate (Defining)

Table 4.4. Statistics by Binned Pressure (Defining)

Table 4.5. Statistics by Binned Inlet Subcooling (Defining)

Table 4.6. Statistics by Axial Power Shape (Defining)

Table 4.7. Statistics by Binned K-Factor (Defining)

4.3.2 Statistics by Test

The descriptive statistics for the overall data can be examined by several subgroups of data (Table 4.8). Mean, standard deviation, and number of data are presented. The correlation predictions are compared to the defining data set.

[

]

The conclusion is that the defining data set, as a whole and by individual test, confirm the fit of the critical power correlation to the defining data set.

Table 4.8. Statistics by Test (Defining)

4.3.3 Statistics by Subgroups of Two Variables

The next group of statistics examines the correlation behavior by paired statistics. The defining data set is large enough to permit the examination of data by paired statistics in mean and standard deviation. However, care should be taken in interpreting these statistics because the size of the data samples in paired statistic bins is quite small.

Table 4.9 provides statistics by subgroups of test and mass flow rate. [

]

Therefore, there are no significant trends in the ACE/ATRIUM-10 correlation with respect to the paired variables mass flow rate and test peaking pattern.

Table 4.10 provides statistics by subgroups of mass flow rate and pressure. [

] Therefore, there are no significant trends in the ACE/ATRIUM-10 correlation with respect to the paired variables mass flow rate and pressure.

Table 4.11 provides statistics by subgroups of mass flow rate and inlet subcooling. [

]

Table 4.12 provides statistics by subgroups of inlet subcooling and pressure. [

]

The conclusion is that the ACE/ATRIUM-10 critical power correlation is in excellent agreement with the defining data set, not only as a whole, but also when binned by one or two variables.

Table 4.9. Statistics by Test and Mass Flow Rate (Defining)

Table 4.9. Statistics by Test and Mass Flow Rate (Defining) – cont.

Table 4.10. Statistics by Mass Flow Rate and Pressure (Defining)

Table 4.11. Statistics by Mass Flow Rate and Inlet Subcooling (Defining)

Table 4.12. Statistics by Inlet Subcooling and Pressure (Defining)

4.4 *ECPR - Mass Flow Plots*

The mass flow rate is the most significant variable in determining the critical power of the fuel assembly. For this reason, the ECPR of each test is plotted as a function of mass flow rate. The overall view of ECPR versus mass flow is shown in Figure 4.2 and no trend is observed.

[

]

4.5 *Additive Constants*

A residual additive constant for each state point can be obtained by subtracting the final additive constant from the test state point additive constant (described in Section 3.10.1) for the limiting rod. A frequency plot of these residuals in additive constant for the defining data set is provided in Figure 4.30. **[** **]**

[redacted]

Figure 4.30 Additive Constant Residual

5.0 ACE/ATRIUM-10 Correlation Validation

The development of the ACE/ATRIUM-10 correlation required that the database be divided into two sets, one for correlation development and the other for correlation validation. When the correlation development was complete, the defining data set was used to verify that the correlation had a proper fit to the data. In this section, the validation data set is applied to examine the behavior of the ACE/ATRIUM-10 critical power correlation.

Additional validation is also performed. The ACE/ATRIUM-10 critical power correlation was further validated by comparing its prediction with the measurements made for transient critical power tests.

Finally, validation is also performed using two other assembly designs. [

]

This section covers the following topics:

- Overall statistical performance of the validating data set, with comparison to defining data set and combined data set (Section 5.1).
- ECPR trend plots comparing the critical power to the validating data set as a whole (Section 5.2)
- [ ]
- Statistical analysis by single variable subsets (Section 5.4)
- Validation to other assembly designs (Section 5.5)
- Validation of [ ] (Section 5.6)
- ACE/ATRIUM-10 critical power correlation benchmark to transient test data (Section 5.7)

5.1 *Overall Statistical Performance*

The overall statistics of the correlation are presented in the following table:

The standard deviation of the validating data set is [ ] than the standard deviation of the defining data set. [ ] The differences are insignificant. On the basis of these statistics, the ACE/ATRIUM-10 critical power correlation is in excellent agreement with the validating data set and with the combined data set.

5.2 *Overall ECPR Trends*

One of the requirements imposed on the correlation is that there are no significant trends in the correlation with measured variables. Therefore, the key experimental variables are plotted against the ECPR to examine trend behavior.

The calculated critical power is plotted against the measured critical power of the validating data set in Figure 5.1. The data fall in a narrow, well-defined band about the expected value, consistent with the overall standard deviation. No trends are evident.

The ECPR as a function of inlet mass flow rate is shown in Figure 5.2. Mass flow rate is the most significant variable in the critical power correlation. A trend line representing a linear least square fit of the ECPR with mass flow rate is shown on the plot confirming that no significant trends exist.

The ECPR as a function of pressure is shown in Figure 5.3. The data show no significant trends with pressure.

The ECPR as a function of inlet subcooling is shown in Figure 5.4. [ ]

[

]

The ECPR as a function of axial power shape is shown in Figure 5.5. There is no apparent trend with axial power shape.

The ECPR as a function of K-Factor is shown in Figure 5.6. There is no apparent trend with K-Factor.

On the basis of the observed trends in the validating data set, the ACE/ATRIUM-10 critical power correlation is in excellent agreement with the experimental data and exhibits no significant trends.

Figure 5.1 Calculated vs. Measured Critical Power (Validating)

Figure 5.2 ECPR as Function of Mass Flow Rate (Validating)

Figure 5.3 ECPR as Function of Pressure (Validating)

Figure 5.4 ECPR as Function of Inlet Subcooling (Validating)

Figure 5.5 ECPR as Function of Axial Power Shape (Validating)

Figure 5.6 ECPR as Function of K-Factor (Validating)

5.4 *Statistical Analysis*

The frequency distribution of ECPR for the validating data set is shown in Figure 5.9. [

]

The validating data set is binned by mass flow rate in Table 5.1. It shows that each mass flow rate, along with the validating data set as a whole, is in excellent agreement with the correlation, and without significant bias.

The validating data set is binned by pressure in Table 5.2. [

]

The validating data set is binned by inlet subcooling in Table 5.3. No unusual trends are observed. [

]

Similar statistical examinations are performed for the axial power shape, in Table 5.4. [

] The difference is insignificant.

The ECPR is binned as a function of K-Factor in Table 5.5. No significant trends are observed in the ECPR.

The results by test can be examined to determine of the correlation predicts the ECPR for each test treated as a subset of the population. The calculated results are provided in Table 5.6. The structure of the table and the entries are the same as those described in Section 4.3.

[

]

The conclusion is that the validating data set, as a whole, by important variables and by individual test, confirm the applicability of the critical power correlation to the population.

Table 5.1. ECPR Binned by Mass Flow Rate (Validating)

Table 5.2. ECPR Binned by Pressure (Validating)

Table 5.3. ECPR Binned by Inlet Subcooling (Validating)

Table 5.4. ECPR Binned by Axial Power Shape (Validating)

Table 5.5. ECPR Binned by K-Factor (Validating)

Table 5.6. Statistics By Test (Validating)

Figure 5.9 Frequency Distribution of ECPR (Validating)

Figure 5.10 Expected Value for [ ] Distribution of ECPR (Validating)

5.5 *Validation to Other Fuel Assembly Designs*

5.5.1 ATRIUM-10P

The ACE/ATRIUM-10 critical power correlation is further validated by evaluating the critical power performance for [ ] The characteristic swirl vanes of the ATRIUM-10 are incorporated on the spacer design used in the validation test and the rod diameter is identical to the ATRIUM-10 design. The ATRIUM (central water canister) remains in the same position. [ ] The data set contains [ ] data points.

Because additive constants are affected by the number and locations of part length fuel rods in the assembly, the additive constants were recomputed for the ATRIUM-10P design. The correlation overall performance against this design is summarized in the following table

[ ]

The resulting correlation is in excellent agreement with the experimental data.

5.5.2 Design 10x10-8

The 10x10-8 design is quite different from the ATRIUM-10 designs. It contains a single large round water rod occupying the four center rod positions in the fuel assembly. 4 small water rods are arrayed around this position. [

]

Additive constants are calculated for this design prior to application of the ACE/ATRIUM-10 critical power correlation to this design. The resulting correlation performance is summarized in the following table:

5.6 *[ ] Bounds*

[

]

[

]

5.7 *Evaluation of Transient Critical Power Data*

An industry-accepted standard in BWR transient methodology is that steady-state dryout correlations are appropriate to use in transient methodology. Transient dryout tests [ ] were performed to reconfirm this for ATRIUM-10 when using the ACE/ATRIUM-10 critical power correlation.

The ATRIUM-10 transient critical power tests were performed [ ] For comparison, the steady-state performance of the ATRIUM-10 as measured and as predicted by ACE/ATRIUM-10 correlation is given in Figure 5.11. [ ]

The transient tests of interest are simulated load rejection without bypass (LRNB) events that consist of power and pressure ramps and flow decay; and simulated loss of flow events that consist of flow decay and power decay. The flow, pressure, and power are controlled by a function generator. The forcing functions were programmed to produce the transient rod surface heat flux typical of the various events. Figure 5.12 shows the forcing function characteristics for a typical LRNB test while Figure 5.13 shows the comparable forcing function characteristics for a typical loss of flow event.

Figure 5.14 illustrates the response of thermocouples attached to the interior of the heater rod tubing. Initially, the clad temperature rises in response to the pressure and power ramps. The transition point, where the heat transfer mode changes from nucleate to film boiling, is characterized by a sudden, rapid increase in clad temperature. This point defines the onset of boiling transition and shows that boiling transition has occurred. [

]

Parameters monitored during the tests include power, inlet flow, system pressure, inlet temperature, and clad temperatures [

]

The AREVA transient thermal hydraulic code XCOBRA-T (References 13 and 14), is used to predict the transient test results using the ACE/ATRIUM-10 steady-state critical power correlation. The test power forcing function provides the boundary condition of power, which appears immediately as heat flux (i.e., no time delay) from the surface of the rods. The critical power is calculated at each time step, then compared to the corresponding assembly power. The ratio of the critical power to the assembly power is CPR. A MCPR of unity during the transient signifies boiling transition. Although applying the steady-state critical power correlation to the transient is considered conservative, the ACE/ATRIUM-10 correlation is a best fit correlation, and for a given steady-state condition shown to be in boiling transition by test, the correlation may under- or over predict a boiling transition state within the range of defined uncertainties. Thus, during transient test conditions, dryout may not be measured or predicted for all cases because of the defined uncertainties.

The [ ] transient tests modeling the ATRIUM-10 geometry in simulated LRNB or loss of flow events were evaluated. [

]

[ ] Table 5.7 summarizes initial state conditions for all the transient tests.

The results are summarized in Table 5.8. Figure 5.15 compares the measured and calculated time of boiling transition after the start of the transient. The comparisons demonstrate that the [ ] transient tests are either exactly predicted or conservatively predicted by the ACE/ATRIUM-10 correlation.

The XCOBRA-T analyses calculated dryout in [ ] evaluated at [ ]. In every case, the time of dryout is calculated to be less than or equal to the time of dryout measured in the transient tests. This validation confirms that the use of the ACE/ATRIUM-10 steady-state dryout correlation is appropriate for use in evaluation of transient events.

Table 5.7. Transient Initial Conditions

Table 5.8. XCOBRA-T Transient Dryout Benchmark

Figure 5.11 Critical Power vs. Flow for Tests 17.8 and 29.5

Figure 5.12 Typical LRNB Transient Forcing Function

Figure 5.13 Typical Loss of Flow Transient Forcing Function

Figure 5.14 Thermocouple Response for Typical Transient Test

Figure 5.15 Comparison of Measured and Calculated Time of Dryout

6.0 ACE/ATRIUM-10 Database

The ACE/ATRIUM-10 database contains [

] to validate the correlation. All data were taken at the AREVA KATHY thermal hydraulic test facility located in Karlstein, Germany.

6.1 *Facility Description*

Figure 6.1 shows that the thermal hydraulic test facility is a high-pressure-water heat-transfer loop containing a test vessel, as shown in Figure 6.2, with the test assembly and upper and lower bus bars, high-pressure coolers, a direct-contact condenser, a pressurizer, and the main circulation pumps. The test loop is rated at 2683 psia and 680°F. The DC power supply consists of four thyristor controlled rectifiers, each rated at 20,750 amps, with a design power of 15 MW.

The data acquisition system uses a DATA GENERAL MV 7800 computer to sample the analog signals of the loop instrumentation, digitize them, and store the signals on hard disc. The system has 176 channels available and a sample rate of 20 samples per second per channel. After the test, the data is archived on magnetic tapes. Table 6.1 shows the test loop uncertainties.

During the dryout test, the dryout power is determined when the temperature of a heater rod thermocouple rises more than [ ]. Additionally, after the test, the data obtained from each thermocouple is evaluated to determine the maximum value. The point of data evaluation for critical power is considered to be between 24.6 seconds and 34.2 seconds of the total file record. Dryout is defined to have occurred if the maximum value of the thermocouple reading is more than [ ] than the arithmetic mean value of the first five temperature values from the beginning of the defined time window. If a thermocouple has an increase in temperature of greater than [ ] the thermocouple is defined as defective and excluded from data evaluation.

Using the time of dryout defined from the thermocouple evaluation, the arithmetic mean values of 11 consecutive power measurements are determined. The maximum mean is defined as the critical power.

6.2 *Test Assembly Descriptions*

The dryout test assemblies are full array assemblies designed to represent the production fuel assembly as close as possible. The rod assembly is housed in a ceramic liner fabricated from alumina ceramic with a purity of 99 percent or better. The inner dimension of the liner is 5.276 in. with the corners rounded to a radius of 0.39 in. The liner serves to simulate the flow channel and electrically insulate the spacers from each other. The ceramics are housed in a stainless steel outer channel assembly.

The heater rods used in the testing are direct heaters; that is, the current flowing through the rod wall provides the heating. Therefore, the thickness of the heater wall determines the relative power of the rod and the variation in wall thickness determines the axial power profile. The high-powered rods, where critical heat flux is expected to occur, are equipped with thermocouples for dryout detection (see Figure 6.3). The thermocouples are located radially to point to the subchannel of interest and axially about 0.5 in. below the top three spacers of the active length.

6.2.1 ATRIUM-10

The ATRIUM-10 test assembly consists of a square array of rods supported at fixed axial locations by ULTRAFLOW spacers and with one 1.378 in square cross section water channel. The array contains 83 full length rods and 8 part length rods.

The test assemblies had the following characteristics (Table 6.2 summarizes the physical characteristics of the ATRIUM-10 test assembly):

During testing, the test assembly is shimmed to its most conservative lateral position by placing shims on the top three spacers.

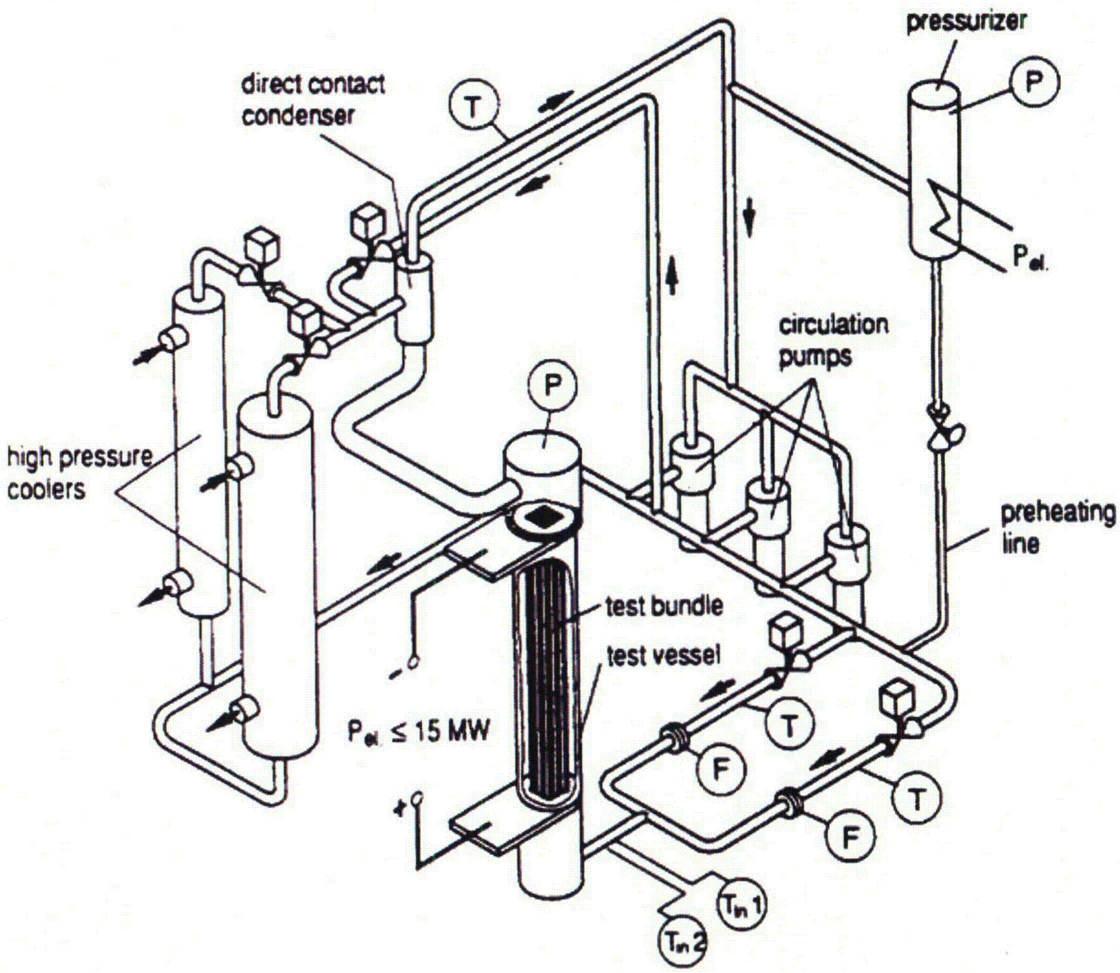

Figure 6.1 KATHY Thermal Hydraulic Test Loop

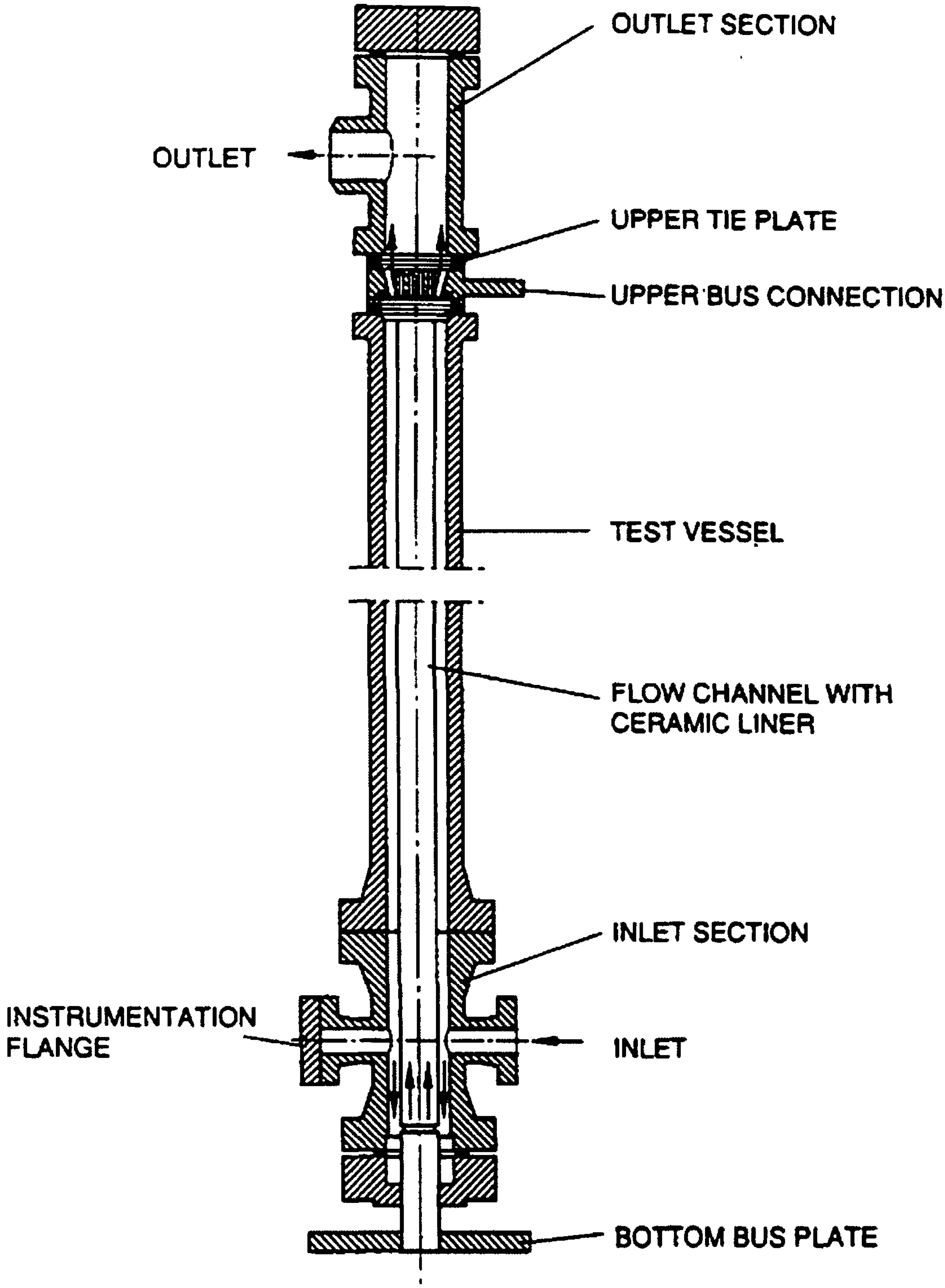

Figure 6.2 Test Vessel

Figure 6.3 Thermocouple Locations

Figure 6.4 Assembly Spacer Locations

Table 6.1. Test Loop Uncertainties

Table 6.2. Physical Characteristics of the ATRIUM-10 Test Assembly

6.3 *Test Strategy*

The development of a dryout correlation requires the acquisition of an appropriate database, where an appropriate database is defined as one that fills the applicable domain with acceptable density, displays acceptable uncertainty everywhere, and provides repeatability. This presents a particular challenge for dryout correlation development. Radial peaking, axial power profile, pressure, flow, and inlet subcooling have been considered in developing the testing strategy to ensure that the number of assemblies used in the correlation is sufficient.

6.3.1 Radial Peaking Profiles

The usual practice is for the local peaking of the test rods to vary between 0.9 and 1.2, with peaking as low as 0.6 used occasionally. Because the purpose of the variation in local peaking is to determine the dryout characteristics of a particular rod position, no effort is made to simulate any particular neutronic design.

The testing program takes advantage of the symmetry of the test assembly. The ATRIUM-10 has half-assembly symmetry along the diagonal of the assembly. In all, [ ]. All major positions of the fuel assembly were tested.

Specific tests were performed during the testing of the ATRIUM-10 assembly to demonstrate the effect of radial peaking on additive constants. The test series STS-17.5 and STS-17.6 peaked rod [ ] respectively. Then STS-32.1 was performed to peak rod [ ]. With the completion of these tests, the representative locations in the assembly were driven into dryout at different local peaking factors.

6.3.2 Axial Power Profile

Three axial power profiles were tested during the ATRIUM-10 dryout test series. The STS-17 and STS-32 series were performed on a [ ] peak to average chopped cosine axial, the STS-28 series was [ ] peak to average downskew axial, and the STS-29 and STS-48 series was performed on a [ ] peak to average upskew axial power profile. Figure 6.5 represents the rod axial power profiles. For the part length rods, the axial power shape is the same as a full length rod, except that is it truncated at the end of the part length rod.

Dryout occurs only after the peak of an axial power profile. For ATRIUM-10, upskew axial power shapes dryout occurs only under the topmost spacer of the heated length. For a cosine axial power shape, dryout may occur under the top or second from the top spacer of the heated length. The same happens for a downskew axial power shape for a fully rodded assembly; for the ATRIUM-10 the dryout may occur as low as the third spacer from the top. For any fuel assembly, the upskew axial power profile will have less critical power than the cosine axial power shape for the same local peaking, and the downskew axial power shape will have higher critical power than the cosine. In general, for the same peak-to-average power shape, for a fully rodded assembly the increase in critical power of a downskew axial will be about the same as the loss of critical power for the upskew relative to the cosine axial power shape.

6.3.3 Thermal Hydraulic Test Conditions

The database for the ATRIUM-10 was obtained prior to ACE/ATRIUM-10 correlation development (Reference 6). It contains data over a range of [ ] a subcooling of [ ] and pressures ranging from [ ]. Table 6.3 summarizes the tests and test conditions used in the development, verification, and validation of the ACE/ATRIUM-10 correlation.

Figure 6.5 ATRIUM-10 Test Rod Axial Power Shapes

Table 6.3. Dryout Test Data

6.3.4 Test Design

The methodology developed for performing dryout testing is fairly standard. The testing is performed by setting pressure and flow. The inlet subcooling is then set and the power is slowly increased until dryout is achieved. The inlet subcooling is then decreased or increased and the process is repeated. After one flow condition is tested, the flow is reset to the desired rate and the entire process is repeated. After all inlet subcoolings and flows are tested, the pressure is changed and testing continued. To ensure that this did not introduce a systematic error, the test process was changed for a few points. In this change, the flow and power were held constant and the inlet subcooling varied until dryout was reached. This process reproduced the standard test procedure.

Because the dryout test results are somewhat ordered, most errors in the test are immediately evident. When the flow is set, the critical power will vary directly with the inlet subcooling. The slope of the line increases as the flow increases. This is seen in any of the plots at the end of this section. During the test series for each day, some test points are repeated to ensure reproducibility.

The development of the test plan is dependent on the use of the data. Statistical design of experiments was used for some of the tests of ATRIUM-10.

6.4 *ACE/ATRIUM-10 Data*

The database for ACE/ATRIUM-10 contains [ ] peaking patterns performed on test sections with cosine, upskew, and downskew axial power profiles for the ATRIUM-10 design. The correlation database contains [ ] data points taken over the range of applicability of the ACE/ATRIUM-10 correlation. Of the [ ] data points, [ ] form the information used during the correlation process and [ ] data points validate the correlation. Table 6.4 contains the measured and calculated critical power ratio of the verification and validation database. [ ] present the dryout test peaking pattern and its associated inlet subcooling versus critical power plot for both the test data and the ACE/ATRIUM-10 prediction of the test data.

Table 6.4. ACE/ATRIUM-10 Data and Analysis Results

Table 6.4. ACE/ATRIUM-10 Data and Analysis Results (cont.)

Table 6.4. ACE/ATRIUM-10 Data and Analysis Results (cont.)

Table 6.4. ACE/ATRIUM-10 Data and Analysis Results (cont.)

Table 6.4. ACE/ATRIUM-10 Data and Analysis Results (cont.)

Table 6.4. ACE/ATRIUM-10 Data and Analysis Results (cont.)

Table 6.4. ACE/ATRIUM-10 Data and Analysis Results (cont.)

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Table 6.4. ACE/ATRIUM-10 Data and Analysis Results (cont.)

Table 6.4. ACE/ATRIUM-10 Data and Analysis Results (cont.)

Table 6.4. ACE/ATRIUM-10 Data and Analysis Results (cont.)

7.0 References

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Appendix A Theoretical Development of ACE/ATRIUM-10 Correlation Form

Appendix B K-Factor Examples

Appendix C List of Symbols