

An AREVA and Siemens company

EMF-2209(NP)(A)  
Revision 3  
SPCB Critical Power Correlation

September 2009

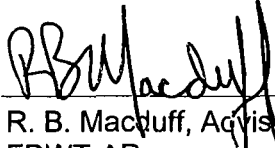


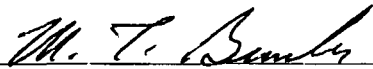
AREVA NP Inc.

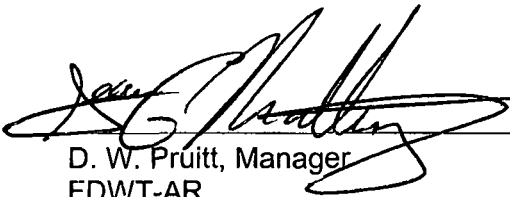
ISSUED IN ON-LINE-  
DOCUMENT SYSTEM  
DATE: 22 DEC 09


EMF-2209(NP)(A)  
Revision 3

**SPCB Critical Power Correlation**

Prepared:  12-21-2009  
R. B. Macduff, Advisory Engineer  
FDWT-AR Date

Reviewed:  12-22-2009  
M. T. Bunker, Supervisor  
FDWT-AR Date

Approved:  for DW Pruitt 12/22/09  
D. W. Pruitt, Manager  
FDWT-AR Date

Approved:  12/22/2009  
A. B. Meginnis  
Regulatory Affairs Date

skm

This document contains 224 pages.

**U.S. Nuclear Regulatory Commission  
Report Disclaimer**

**Important Notice Regarding the Contents and Use of This Document**

***Please Read Carefully***

This technical report was derived through research and development programs sponsored by AREVA NP Inc. It is being submitted by AREVA NP Inc. to the U.S. Nuclear Regulatory Commission as part of a technical contribution to facilitate safety analyses by licensees of the U.S. Nuclear Regulatory Commission which utilize AREVA NP Inc. fabricated reload fuel or technical services provided by AREVA NP Inc. for light water power reactors and it is true and correct to the best of AREVA NP Inc.'s knowledge, information, and belief. The information contained herein may be used by the U.S. Nuclear Regulatory Commission in its review of this report and, under the terms of the respective agreements, by licensees or applicants before the U.S. Nuclear Regulatory Commission which are customers of AREVA NP Inc. in their demonstration of compliance with the U.S. Nuclear Regulatory Commission's regulations.

AREVA NP Inc.'s warranties and representations concerning the subject matter of this document are those set forth in the agreement between AREVA NP Inc. and the Customer pursuant to which this document is issued. Accordingly, except as otherwise expressly provided in such agreement, neither AREVA NP Inc. nor any person acting on its behalf:

- a. makes any warranty, or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method, or process disclosed in this document will not infringe privately owned rights;

or

- b. assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this document.

AREVA NP Inc.

EMF-2209 (NP)(A)  
Revision 3



## **SPCB Critical Power Correlation**



September 2009





AREVA NP Inc.

EMF-2209(NP)(A)  
Revision 3

Copyright © 2009

AREVA NP Inc.  
All Rights Reserved





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.

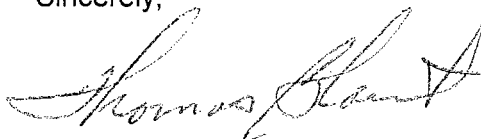
In accordance with the guidance provided on the NRC website, we request that AREVA publish accepted proprietary and non-proprietary versions of the TRs within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, 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.

R. Gardner

- 2 -

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 rodged 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-conservatism 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



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

September 17, 2003

Mr. James F. Mallay  
Director, Regulatory Affairs  
Framatome ANP  
3815 Old Forest Road  
Lynchburg, VA 24501

SUBJECT SAFETY EVALUATION – REVISION TO FRAMATOME ANP TOPICAL  
REPORT EMF-2209(P)(A), REVISION 1 (TAC NO. MB9719)

Dear Mr. Mallay:

By letter dated June 20, 2003, Framatome ANP (FANP) submitted a revision to Topical Report (TR) EMF-2209(P), Revision 1, "SPCB Critical Power Correlation," to the NRC for review and approval. "SPCB Critical Power Correlation" designates a critical power correlation for boiling water reactors originally developed by Siemens Power Corporation. The submittal describes modifications to the NRC-approved Siemens Power Corporation critical power correlation in the region of the uranium blanket at the top six inches of the fuel. The revision will enhance the behavior of the SPCB correlation in the reflector region of the fuel while reducing some of the conservatism inherently built into the correlation in the region.

The NRC staff has completed its review of the revision to the TR and FANP's response to the staff's July 14, 2003, request for additional information (RAI). The TR is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and in the associated NRC staff's safety evaluation, which is enclosed. The safety evaluation defines the basis for acceptance of the TR.

If the NRC staff's criteria or regulations change so that its conclusion in this letter, that the TR is acceptable, is invalidated, FANP and/or the applicant referencing the TR will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the TR without revision of the respective documentation.

In accordance with the guidance provided on the NRC website, we request that FANP publish an accepted version within three months of receipt of this letter. The modifications in the current submittal would be implemented as modifications in the documentation of the original TR (EMF-2209(P)(A), Revision 1). FANP would modify the footnote on page 2-8 of that TR to specify the maximum value of the Omega Function and would add a footnote on page 2-8 to specify the maximum value of the Tong Factor. FANP shall incorporate (1) this letter and the enclosed SE with the initial SE between the title page and the abstract, (2) all RAIs from the staff and all associated responses into a revised report and publish the revised report as EMF-2209(P)(A), Revision 2. The proposed modifications will have no effect on the statistical aspects of the SPCB data base evaluations, and hence, will not impact the minimum critical power ratio safety limit in any way. The actual statement of the amended footnote is provided in your letter dated June 20, 2003.

J. Mallay

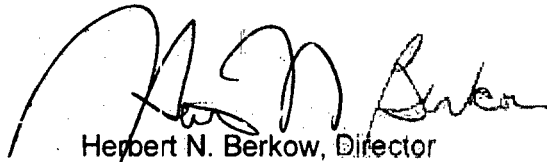
-2-

Pursuant to 10 CFR 2.790, we have determined that the enclosed safety evaluation does not contain proprietary information. However, we will delay placing the safety evaluation in the public document room for a period of ten working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

We do not intend to repeat our review of the matters described in the subject topical report and found acceptable, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. Our acceptance applies only to matters approved in the report.

In the event that any comments or questions arise, please contact Drew Holland at (301) 415-1436.

Sincerely,



Herbert N. Berkow, Director  
Project Directorate IV  
Division of Licensing Project Management  
Office of Nuclear Reactor Regulation

Project No. 728

Enclosure: Safety Evaluation



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REVISION TO TOPICAL REPORT EMF-2209(P)(A), REVISION 1

"SPCB CRITICAL POWER CORRELATION"

FRAMATOME ANP

PROJECT NO. 728

1.0 BACKGROUND

By letter dated June 20, 2003, (Reference 1), Framatome-ANP (FANP) submitted a revision to Topical Report (TR) EMF-2209(P)(A), Revision 1, "SPCB Critical Power Correlation," to the NRC for review and approval. "SPCB Critical Power Correlation" designates a critical power correlation for boiling water reactors originally developed by Siemens Power Corporation. The submittal describes a modification to the NRC-approved Siemens Power Corporation B (SPCB), critical power correlation (Reference 2) in the region of the uranium blanket at the top six inches of the fuel. The revision will enhance the behavior of the SPCB correlation in the reflector region of the fuel, while reducing some of the conservatism inherently built into the correlation in that region. Upon review of the SPCB critical power correlation, the staff asked specific questions regarding the determination of the maximum value for the Tong Factor as well as the maximum value for the Omega Function in a request for additional information (RAI) (Reference 3). The staff also met with FANP in Richland, Washington on July 22, 2003, and July 23, 2003, to discuss FANP's responses to the RAIs (Reference 4).

Reference 2 is the TR originally submitted by the Siemens Power Corporation and approved by the NRC in July 2000. The TR described the methodology behind the development and application of the SPCB critical power correlation to FANP ATRIUM-9B and ATRIUM-10 fuel designs.

Application of the SPCB correlation to D-lattice plants indicated that the correlation is overly conservative for nuclear designs with top natural uranium blankets. This conservatism arises from deriving the SPCB correlation without accounting for the effect of the natural uranium at the top six inches of the fuel rods. Although reflector blankets have always been a part of FANP boiling water reactor (BWR) designs, the details of the power distributions within the reflector region were not fully considered in development of the SPCB Revision 1. Reference 1 proposes revisions to two parameters within the correlation to fully account for the natural uranium in the reflector blanket, and without affecting the correlation behavior in the remaining areas of the fuel.

2.0 REGULATORY EVALUATION

Title 10 of the Code of Federal Regulations (10 CFR) Section 50.34, "Contents of Applications; Technical Information," requires that safety analysis reports be submitted that analyze the

design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. As part of the core reload design process, licensees (or vendors) perform reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle.

To confirm that the analyses remain bounding, licensees confirm that key inputs to the safety analyses are conservative with respect to the current design cycle. If key safety analysis parameters are not bounded, reanalysis or reevaluation of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

Reference 1 describes FANP's methodology for implementing two improvements to the existing SPCB critical power correlation. Since the NRC staff has previously reviewed and approved the SPCB correlation, the staff's review of Reference 1 focused on the two improvements in the application of the critical power correlation to FANP ATRIUM-9B and ATRIUM-10 fuel. There are no specific regulatory requirements or guidance available for the review of TR revisions. As such, the staff reviewed the revisions based on the technical merit and its compliance with any applicable regulations.

The staff validated that the Tong Factor and Omega Function are appropriately defined and that the data supports the proposed modifications.

### 3.0 TECHNICAL EVALUATION

FANP developed the SPCB correlation to address the critical power behavior of the Siemens Power Corporation ATRIUM-9B and ATRIUM-10 fuel designs. The SPCB correlation is applicable in steady-state, transient, and loss-of-coolant accident critical heat flux (CHF) calculations for the ATRIUM-9B and ATRIUM-10 fuel designs.

Typically, natural  $UO_2$  blankets are added to BWR fuel to prevent neutron leakage and improve fuel cycle economics. As FANP pointed out in Reference 4, replacing the top and bottom 6 inches of the active fuel with a natural uranium blanket reduces the overall enrichment of the bundle.

In theory, moving more U-235 toward the center of the core improves U-235 utilization by reducing neutron leakage out of both the top and bottom of the enriched fuel. The top blanket is worth more than the bottom blanket in enrichment savings, since most neutronic activity occurs in the top of the core. However, relatively little power is produced in natural  $UO_2$  blankets themselves.

Natural blankets have always been a part of FANP BWR designs. However, FANP did not fully consider the details of the blanket power distributions when developing the SPCB CHF correlation. Unlike the enriched sections of the fuel assembly, where the radial enrichment distribution is tailored to minimize radial peaking factors, the natural blankets utilize a uniform radial enrichment. The result is that the local peaking depends primarily on the moderator distribution in the vicinity of the blanket.

After investigation of Reference 2, FANP determined that the correlation is overly conservative in the top six inches of the fuel rod when a natural uranium blanket is present. This



conservatism arises from the fact that the test data used for deriving the Omega Function did not appropriately account for the non-linearity of the data in the top six inches of the fuel. The two parameters (the Tong Factor and the Omega Function) within the correlation are revised, but not the correlation itself. In developing the technical basis for the Tong Factor and Omega Function in the SPCB Revision 1 correlation, FANP did not make full use of the available raw test data. Framatome ANP assigned a bounding fixed value to the Omega Function, while in actuality the test data suggested that the factor should be non-linear.

For a D-lattice plant with asymmetric water gaps between assemblies, the relative local peaking and consequently the maximum lattice F-effective can be quite high despite the low planar powers. The high F-effectives for the D-lattice natural blanket, coupled with the unbounded Tong Factor result in the assembly being limited by the natural blanket region.

For a C-lattice plant the results show similar trends but are not as exaggerated. C-lattice plants exhibit the same unbounded Tong behavior. The local pin-peaking of the C-lattice for the natural lattice is not as great as that of the D-lattice; therefore, the F-effective is not as limiting for the C-lattice as for the D-lattice.

FANP determined that the Tong Factor, as defined by equations 2.14 and 2.15 in Reference 2, was the source of the observed behavior. Specifically, the Tong Factor was observed to take on values significantly larger than is typically observed for other axial shapes used in the correlation data base for the exit plane. The sudden increase in the Tong Factor in the six inch blanket results in a calculated critical power for the top node which is overly conservative.

The reduced axial power peaking in the top blanket also introduces a step change in the Omega Function as defined in equation 2.15 of Reference 2. While the Omega Function was forced to assume a fixed value in the original correlation, the locally computed values as a function of mass flow, may drop significantly below the minimum value imposed on the function as defined in RAI response #8 of Reference 4. The value of the Omega Function has an inverse relation to memory length/effect. Memory length refers to the influence of the upstream thermal hydraulic behavior on the local heat flux. A larger value of memory length implies a greater importance of upstream fluid conditions. The evaluation process typically performs evaluations on a six-inch increment. If the memory effect is characterized by some length at one location, then the memory effect at the next location should not be characterized by a memory length that is more than six inches longer.

The SPCB correlation originally accounted only for the observation that the Omega Function should be no less than some minimum value on an absolute basis for the entire database. The insertion of natural uranium in the last 6 to 12 inches of the assembly leads to step changes in the Omega Function that mathematically suggest an increasingly longer memory effect for the natural uranium nodes, well beyond the added 6 inches per node when compared to upstream values. This suggested an improved definition for the Omega Function incorporating values of Omega that bound the correlation data base but allow variation of the minimum value as a function of mass velocity as illustrated in Figure 2 of Reference 1.

The staff examined the literature that describes the development and application of both the Tong Factor and the Omega Function (Reference 2) and the data that FANP used to develop the initial correlation during an onsite audit. After reviewing the responses to the staff's RAI and examining the proposed changes to the Tong Factor and Omega Function at the FANP Richland site, the staff has concluded that the proposed values provide a conservative limit when compared to the data, and the proposed modifications to equations 2.14 and 2.15 are acceptable. Further, the staff agrees with the proposed means of implementing these modifications in the revised TR.

Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," outlined a process that licensees could use to move cycle-specific parameters from the plant-specific TSs to a licensee-controlled document entitled the Core Operating Limits Report (COLR). A necessary element of that process was that licensees include specific types of methodologies (i.e., revision changes) into the TSs. This TR is one such methodology that is required to be listed in the TSs.

#### 4.0 LIMITATION AND CONDITION

Application of this correlation and the proposed revisions to fuel designs other than the ATRIUM-9B and ATRIUM-10 designs requires prior staff approval.

#### 5.0 CONCLUSION

The staff has reviewed FANP's submittal and supporting documentation. Based on the considerations above, the staff has concluded that the proposed revision to TR EMF-2209(P) (A), Revision 1, "SPCB Critical Power Correlation," is acceptable for use in licensing applications for FANP ATRIUM-9B and ATRIUM-10 fuel designs only.

#### 6.0 REFERENCES

1. Letter from J. F. Malley to the U.S. Nuclear Regulatory Commission, "Request for Review of a Revision to EMF-2209(P)A, Revision 1," June 20, 2003.
2. EMF-2209(P)A, Revision 1, "SPCB Critical Power Correlation," July 2000.
3. Fax from U.S. Nuclear Regulatory Commission, "Request for Additional Information (RAI) to Topical Report EMF-2209(P), 'SPCB Critical Power Correlation,' Revision 1 July 14, 2003.
4. Letter from J. F. Malley to the U.S. Nuclear Regulatory Commission, "Responses to Additional Information (RAI) to Revision to Topical Report EMF-2209(P)A, 'SPCB Critical Power Correlation,' Revision 1," July 25, 2003.

Contributors: Patricia Henry  
Anthony Attard

Date: September 17, 2003



UNITED STATES  
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

July 3, 2000

Mr. James F. Mallay  
Director, Nuclear Regulatory Affairs  
Siemens Power Corporation  
2101 Horn Rapids Road  
Richland, WA 99352

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT  
EMF-2209(P) REVISION 1, "SPCB CRITICAL POWER CORRELATION"  
(TAC NO. MA6639)

Dear Mr. Mallay:

Revision 0 of the subject topical report was submitted by the Siemens Power Corporation (SPC) by letter dated September 24, 1999, and Revision 1 was submitted by letter dated April 20, 2000. This topical report describes the analyses conducted by SPC pertaining to the application of the SPCB critical power correlation to the ATRIUM-9B and to the ATRIUM-10 fuel designs. The ATRIUM-9B fuel design is a 9x9 square array, while the ATRIUM-10 fuel design is a 10x10 square array. Both fuel designs are fixed at axial locations by ULTRAFLOW spacers and use an internal square water canister, replacing a 3x3 array of rods. The ATRIUM-9B fuel assembly contains 72 full-length rods (no part-length rods) and the ATRIUM-10 fuel assembly is made up of 83 full-length rods and 8 part-length rods.

The SPCB correlation uses planar average values of coolant mass velocity, enthalpy, and pressure to predict planar average critical heat flux. Although SPCB is a generic correlation (applicable to both ATRIUM-9B and ATRIUM-10), it is very similar to the original ANFB-10 correlation that is currently used to predict critical heat flux for the ATRIUM-10 fuel assemblies.

The staff, after their review, determined the topical report to be acceptable for referencing and conveyed the acceptance along with the safety evaluation (SE) to you by letter dated May 17, 2000. However, in the May 17 letter and accompanying SE, the revision number of Topical Report EMF-2209(P) was referenced as 0. By this letter and the enclosed SE, the revision number is corrected to 1 and additional minor corrections have been made to the SE.

Pursuant to 10 CFR 2.790, we have determined that the enclosed SE does not contain proprietary information. However, we will delay placing the SE in the public document room for a period of ten (10) working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

The staff will not repeat its review and acceptance of the matters described in the report, when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

Mr. James F. Mallay

- 2 -

July 3, 2000

In accordance with the procedures established in NUREG-0390, the NRC requests that SPC publish accepted versions of the report, including the safety evaluation, in the proprietary and non-proprietary forms within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include a "-A" (designating accepted) following the report identification symbol. The accepted versions shall also incorporate all communications between SPC and the staff during this review.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are no longer valid, SPC and the licensees referencing the topical report will be expected to revise and resubmit their respective documentation, or to submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,



Stuart A. Richards, Director  
Project Directorate IV and Decommissioning  
Division of Licensing Project Management  
Office of Nuclear Reactor Regulation

Project No. 702

Enclosure: Safety Evaluation



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT EMF-2209(P), REVISION 1,

"SPCB CRITICAL POWER CORRELATION"

SIEMENS POWER CORPORATION

1.0 BACKGROUND

EMF-2209(P) describes the methodology behind the application of the SPCB correlation to the SPC's ATRIUM-9B and ATRIUM-10 fuel designs (References 1 and 2). EMF-2209(P) provides test data taken specifically at the Siemens test facility at Karlstein, Germany, in support of the application of the SPCB correlation to the ATRIUM-9B and ATRIUM-10 fuel designs and to the determination of the associated correlation, "Additive Constants." The ATRIUM-9B fuel has no part-length rods, but the ATRIUM-10 fuel does.

The additive constants are determined in accordance with the NRC-approved procedure described in References 3 and 4. The uncertainties associated with these additive constants are then used in the approved SPC safety limit methodology for boiling water reactor (BWR) fuel designs. The approved methodology is used to ensure that less than 0.1 percent of the fuel rods are in boiling transition during steady-state operation and during anticipated operational occurrences, in accordance with General Design Criterion 10 and Section 4.4 of the Standard Review Plan.

The SPCB correlation is new but similar to the ANFB-10 correlation, described in References 3 and 5. However, the definitions of the associated parameters (inlet sub-coolant, pressure, and mass flow) as described in Reference 3 are not changed for the application of the new SPCB to the SPC ATRIUM-9B and ATRIUM-10 fuel designs. The technical analysis of the SPCB correlation and its exclusive application to the ATRIUM-9B and ATRIUM-10 fuel designs is presented below.

2.0 TECHNICAL EVALUATION

The SPCB correlation is a new correlation designed and developed to address the critical power behavior of the SPC ATRIUM-9B and ATRIUM-10 fuel designs. The SPCB correlation is designed for application in steady-state, transient, and loss-of-coolant accident critical heat flux (CHF) calculations for the ATRIUM-9B and ATRIUM-10 fuel designs.

The SPCB correlation was developed to predict assembly critical power for the ATRIUM-9B and ATRIUM-10 fuel designs. The correlation was developed to predict the limiting rod in a bundle and account for local spacer effects and bundle geometry on critical power by a set of constants, typically referred to as "Additive Constants," one constant for each rod in the bundle. Each individual fuel design requires a unique set of additive constants.

The SPCB correlation is an empirically derived expression that is a complex function of the input parameters: local coolant enthalpy, mass flow, and pressure. These input parameters cover the ranges of pressure, mass velocity, and inlet cooling, consistent with expected operating and accident conditions. The correlation is based on local coolant conditions predicted from uniform and non-uniform axial power distribution test data. The correlation includes correction factors to account for geometry and non-uniform axial power distributions that deviate from the test data conditions.

Low-flow and high-flow behavior of the correlation are captured by refining the parameters in the correlation equations (Reference 1). These parameters address the impacts of the variations in the local enthalpy from the planar average enthalpy. One of these parameters is the F-effective, which characterizes the fuel rod local behavior, such as enthalpy rise, and which also factors additive constants into the calculations. The additive constants account for the fuel bundle geometry and spacer effects on the critical power behavior of the bundle (References 3 and 4).

## 2.1 SPCB Database and Test Strategy

The SPCB database consists of data taken at the SPC test facility at Karlstein, Germany. The test setup comprises electrically heated bundles that are physically the same as the ATRIUM-9B and the ATRIUM-10 fuel assemblies. The tests are designed to reproduce the local conditions typically present in a BWR fuel assembly and support the full range of applicability for the SPCB correlation.

Different test programs were developed to accumulate a database representative of the appropriate statistical requirements for the ATRIUM-9B and the ATRIUM-10 fuel designs. The tests selected and the number of points required were dictated by the requirements of the statistical design of experiment SDE (References 6 and 7). This approach ensures that an adequate number of tests are performed and that sufficient data are gathered to perform appropriate simulation of the behavior of the ATRIUM-9B and the ATRIUM-10 fuel designs.

Both steady-state and transient tests were performed as part of the validation of the SPCB correlation. In each case, the tests were designed to include test runs with peaked rods located adjacent to the internal water channel.

The database comprises more than 2500 data points taken in a large number of tests performed at the SPC test facility. The database consists of upskew, downskew, and cosine axial power shapes accounting for adjacent rod positions, rods on the interior of an assembly, and rods adjacent to the water canister (channel), a feature unique to the ATRIUM fuel design.

The local power peaking patterns were selected to determine the effects of the upskew axial power profiles as compared to the cosine power profiles in several regions of the test bundle. Local power peaking data were also collected at the corners, the peripheral rows, as well as around the internal water canister to ensure complete understanding of the fuel CHF behavior, particularly in these regions.

The internal water canister is a major and unique characteristic of the SPC's ATRIUM fuel design. It replaces a 3x3 matrix of fuel rods. The rectangular canister is designed so that the subchannels around it are regular in size, typical of those addressed by the original base ANFB correlation. The test matrixes of the ATRIUM-9B and the ATRIUM-10 fuel designs used at the SPC test facility included tests to confirm the behavior of the fuel surrounding the internal water canister. Neither the ATRIUM-9B nor the ATRIUM-10 fuel design showed any abnormal behavior around the internal water canister.

## 2.2 Description of the Additive Constants

Correlation parameters such as F-effective ( $F_{EFF}$ ) account for the local peaking factor effect on the bundle critical power.  $F_{EFF}$  is constructed in two parts. One part depends solely on the peaking factors of the rod of interest and its immediate neighbors ( $F_{EFFO}$ ); the other part, termed the "additive constant," accounts for other local effects, such as bundle geometry and spacer effects. These spacer and bundle geometry effects influence the critical power behavior of the bundle. Therefore, an offset term is applied to each rod in the bundle, subject to the rod's position in the bundle. This offset term is called the "additive constant." The additive constant can be considered as a flow/enthalpy redistribution characteristic of a particular lattice/spacer design, so the additive constants are unique to a particular fuel design. They are explicitly determined for each lattice/spacer design configuration and are utilized in design calculations for the corresponding fuel bundle (Reference 3).

The additive constants are derived from the critical power tests for the ATRIUM-10 fuel and the ATRIUM-9B fuel separately. Specifically, the additive constants are derived from about 80 percent of the data for each fuel. The data includes sufficient radial peaking distributions, sufficient axial shapes, and a representative density of flows and pressures.

## 3.0 STATISTICAL ASPECTS OF THE SPCB CORRELATION

The statistical aspects of the SPCB correlation consist of applying appropriate statistical techniques (References 6 and 7) to the SPCB database. These techniques involve the evaluation of distribution characteristics, figures of critical power ratios (CPRs) with respect to each characteristic within the correlation, descriptive statistics for subgroups of data, descriptive statistics for additive constants and additive constants uncertainty, and conservatism of the SPCB critical power correlation. A good correlation would place the CPR near 1.00 (unity), with a very small associated uncertainty.

The correlation study examined the CPR in a series of tests. A total of 12 tests were performed: 7 tests pertained to the ATRIUM-9B and 5 tests pertained to the ATRIUM-10 fuel designs. For the ATRIUM-9B fuel, three of the seven tests were conducted with a chopped cosine shaped axial power profile, one test with a downskew power profile, and two tests with an upskew power profile. For the ATRIUM-10 fuel, three of the five tests were conducted with a chopped cosine shaped axial power profile, one of the five tests with a downskew power profile, and one test with an upskew power profile. Each test was repeated many times ("runs"). The input variables into each run entered the experimental design at different levels to reflect a diversified operating environment, resulting in a database containing in excess of 2500 data points. Twenty percent of this data was used to validate the correlation, while the remaining 80 percent was used to develop the SPCB correlation.

The multiplicity of runs within each test was required in order to involve various levels of input factors (inlet flow, inlet sub-cooling, and pressure). For most of the runs, these factors were selected at random, following standard statistical procedures (References 6 and 7). For dryout testing, additional runs were made following a two-level, three-factor factorial design to ensure that the entire range of interest (including "corner to corner") was represented.

Review of SPC calculations shows that the average CPR appears to be very near 1.0. That ratio is retained without any apparent trend across inlet mass velocity (Mlb/hr-ft<sup>2</sup>), enthalpy (Btu/lbm), pressure (psia), the best estimate of the  $F_{EFF}$ , or the axial offset. The overall CPR mean for the ATRIUM-9B 1629 data points was calculated to be 0.996, and the CPR mean for the ATRIUM-10 1028 data points was calculated to be 0.996.

To evaluate the quality of the correlation, the staff independently calculated a CPR 95/95 upper tolerance limit (References 8 and 9) for each test, for each profile, and for the entire set of runs. The staff 95/95 calculation was compared to SPC's 95/95 calculation. Apart from rounding errors and conservative table interpolations, the staff's calculation was in total agreement with SPC's calculation. This limit is interpreted to mean that one is 95 percent sure that at least 95 percent of the population of runs yields a CPR value no higher than 1.022 for ATRIUM-9B and a value of 1.034 for ATRIUM-10. SPC's calculations also show that for any test or grouping of tests, the percentage of runs that fall below their associated tolerance limits is at least 95.7 percent for ATRIUM-9B and 96.8 percent for ATRIUM-10.

The submittal contains charts and tables reflecting CPR behavior across different mass velocity (Mlb/hr-ft<sup>2</sup>) for individual tests. Although some tests show higher CPR values associated with high mass velocity, the reverse is true for other tests, and no dependency between CPR and mass velocity is apparent.

Another objective of SPC's study that involves statistical consideration is the determination of the additive constant for both fuel types. The additive constant is a statistical adjustment to the measure of the  $F_{EFF}$  to account for the effect of the rod's geometric position within the assembly. This adjustment has two components: a calculated additive constant and a measure of uncertainty associated with the calculation. In the development of the additive constants, SPC uses only the cosine profile data. However, the measure of the associated uncertainty is calculated from the entire database, containing cosine, upskew, and downskew test data.

The main contributors to this uncertainty are two sources of variability: "within test variability" and "between test variability." The within test variability is given as a weighted average in which the weighting factors are the number of runs per test. The between test variability is given as a weighted average of the difference between the  $F_{EFF}$  for a rod in a test bundle and the average  $F_{EFF}$  for the test bundle. The weighting factors are the number of boiling transitions for a rod in the test bundle. The square root of the sum of the squares (the two sources of variability) give the measure of variability associated with the calculation of the additive constant. In-depth review of the statistical section of the submittal leads the staff to concur with the statistical methods used and the results obtained by the vendor.



#### 4.0 SPCB CORRELATION BEHAVIOR

The SPCB correlation was tested to ensure smooth functions and no significant discontinuities in its behavior over the entire range of operability of the fuel. Flow, enthalpy, and pressure-dependent functions within the correlation, such as the "Tong Factor" correction for both fuels, was investigated for its behavior over the entire applicable range of the fuels. A number of tests were conducted to determine the sensitivity of the major functions within the SPCB correlation to flow, inlet subcooling, pressure variation,  $F_{EFF}$ , and axial power shape.

Review of the data, figures, and tables indicates that the SPCB correlation behaves well over the applicable range of the fuel.

#### 5.0 SPCB CORRELATION VALIDATION

SPC performed several tests to validate the behavior of the SPCB correlation in steady-state and transient events. The validation database consisted of 20 percent of the total steady-state data points that were not included in the correlation database. The remaining 80 percent of the database (the so-called verification set) was used to develop the correlation. In addition, data were collected from tests conducted on an ATRIUM-10P assembly that contained more part-length fuel rods than are usually found in a typical ATRIUM-10 assembly. These tests were conducted to demonstrate the ability of the SPCB correlation to capture the effects of the part-length rods, as well as the correlation agreement with the data. The predicted SPCB correlation between critical power versus the measured critical power for these tests showed very good agreement.

Two sets of transient tests were performed as part of the validation process. Both tests were designed to peak rods around the internal water canister. The difference between the two tests is that the first test had rods with a chopped cosine-shaped axial power profile and the second had rods with an upskew axial power shape. Another purpose of the tests was to validate the concept that the additive constants can be derived from steady-state tests and applied to other axial shapes under transient conditions.

The transient tests performed were the simulated load rejection with no bypass (LRNB) events that consisted of power, pressure ramps, and flow decay. Power forcing functions were programmed to produce transient heat flux on the surface of the rod typical of an LRNB event. Parameters monitored during the tests were power, inlet flow, system pressure, inlet temperatures, and cladding temperatures.

The transient thermal-hydraulic code, XCOBRA-T (References 10 and 11), was used to predict the test results using the SPCB steady-state critical power correlation. XCOBRA-T calculates the fluid conditions at a specified time step. The CHF is calculated at each axial position and time step, then compared to the corresponding measured rod heat flux at the surface of the rod. The ratio of the calculated heat flux to the measured rod heat flux is defined to be the critical heat flux ratio (CHFR). When this ratio is unity, it is referred to as the minimum critical heat flux ratio (MCHFR), and it signifies "boiling transition" in a transient event. Comparison of measured and calculated time-to-boiling transitions for cosine and upskew transient tests shows that the XCOBRA-T calculated time-to-boiling transition values are conservative when

compared to actual boiling transition time. This validation confirms the use of the steady-state SPCB correlation and the associated additive constants in evaluating transient events.

## 6.0 LOCAL PEAKING FACTORS

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 1.5 is exceeded, an additional additive constant uncertainty is applied on a rod-by-rod basis in accordance with Table 3.15 of Reference 1. These conditions have been agreed upon by both the NRC staff and SPC (Reference 12).

## 7.0 NON-CONFORMANCE ISSUES

The submittal, as documented in Reference 1, is SPC's corrective action in response to Part 2 of notice of Nonconformance 99900081/97-01, as stated in Attachment II of SPC's letter to the NRC, dated February 24, 1998 (Reference 13). The Nonconformance stated that: SPC failed to develop an adequate number of tests points and failed to test an adequate range of conditions to justify the uncertainty values for the "additive constants" used in determining the safety limit minimum critical power ratio (SLMCPR) for the ATRIUM-9B fuel design. This statement implies that SPC should have used larger uncertainty values in the SLMCPR determinations in order to reflect the full operability range of the ATRIUM-9B fuel design. In addition, because the results of the ANFB correlation are used as inputs to the safety limit methodology, this has immediate implications regarding the SLMCPR and the operating limit minimum critical power ratio (OLMCPR) of the Commonwealth Edison Company plants (Quad Cities Unit 2, Cycle 15, Dresden Unit 2, Cycle 15, and LaSalle County Unit 2, Cycle 8) and the Washington Public Power Supply System (Washington Nuclear Unit 2, Cycle 13) loaded with ATRIUM-9B fuel.

In response to this notice of Nonconformance, SPC developed and implemented interim methodologies (ANF-1125, Appendixes D and E) (References 14 and 15), while performing additional dryout testing of the ATRIUM-9B design to obtain additional data to cover the extended range of thermal-hydraulic parameters for the ATRIUM-9B fuel design.

The NRC staff contends that with the submittal of EMF-2209(P), the vendor (SPC) has provided the additional data necessary for the SPCB critical power correlation to provide a rigorous treatment over the entire operating range of the ATRIUM-9B fuel. Thus, with the submittal of EMF-2209(P), all problems identified in the inspection report (Nonconformance 99900081/97-01, Part 2) related to the dryout methodology for ATRIUM-9B fuel have been addressed.

## 8.0 TECHNOLOGY TRANSFER

SPC described the technology transfer program (Reference 16) which the licensees must successfully complete in order to perform their own thermal-hydraulic calculations using the SPCB correlation and the XCOBRA-T code in support of reload analyses. The overall process consists of training, benchmarking, and change control. In addition, SPC described the process for a licensee to implement the new correlation (SPCB). This process includes performance of

an independent benchmarking calculation by SPC for comparison to the licensee-generated results to verify that the new CHF correlation is properly applied. The staff has reviewed the process and find it acceptable because training, bench-marking, and change control have been adequately addressed.

## 9.0 CONCLUSION

The staff has reviewed the analyses in Topical Report EMF-2209(P), Revision 1, "SPCB Critical Power Correlation," and concludes that on the basis of its findings presented above, Topical Report EMF-2209(P) is acceptable for licensing applications, in accordance with SPC's agreement, subject to the following conditions:

1. The SPCB correlation (as described in this submittal, Reference 1) is applicable to SPC ATRIUM-9B and ATRIUM-10 fuel designs, with a local peaking factor no greater than 1.5.
2. If, however, in the process of calculating the MCPR safety limit, the local peaking factor of 1.5 is exceeded, an additional uncertainty of 0.026 for ATRIUM-9B and 0.021 for ATRIUM-10 will be imposed on a rod-by-rod basis.
3. The SPCB correlation range of applicability is as follows:

Pressure (psia)	571.4 to 1432.2
Inlet Mass Velocity (Mlb/hr-ft <sup>2</sup> )	0.087 to 1.5
Inlet Subcooling (Btu/lbm)	5.55 to 148.67
Design Local Peaking	1.5
Tested Local Peaking	1.45

4. Technology transfer will be accomplished only through the process described in Reference 16, which includes the performance of an independent bench-marking calculation by SPC for comparison to the licensee-generated results to verify that the new CHF correlation (SPCB) is properly applied for the first application by the licensee.

## 10.0 REFERENCES

1. Letter from H. D. Curet, SPC, to the U.S. Nuclear Regulatory Commission, submitting Topical Report EMF-2209(P), Revision 0, September 24, 1999.
2. Letter from J. F. Mallay to the U.S. Nuclear Regulatory Commission, "Request for Additional Information (RAI) to Topical Report EMF-2209(P), 'SPCB Critical Power Correlation,'" Revision 0, March 20, 2000.
3. Letter from R. A. Copeland, SPC, Transmittal of (A) Version of ANF-1125(P) to the U.S. Nuclear Regulatory Commission, April 27, 1990.

4. Letter from H. D. Curet, SPC, to the U.S. Nuclear Regulatory Commission, submitting Topical Report EMF-1997(P), Revision 0, October 30, 1997, and Letter from H. D. Curet, submitting Topical Report EMF-1997(P), Supplement 1, Revision 0, January 29, 1998.
5. Letter from J. F. Mallay to the U.S. Nuclear Regulatory Commission, "Request for Additional Information (RAI) to Topical Report EMF-1997(P), 'ANFB-10 Critical Power Correlation,'" Revision 0, May 7, 1998.
6. "Statistical Methods for Nuclear Material Management," NUREG/CR-4604, PNL-5849, December 1988.
7. G. J. Hahn and S. S. Shapiro, "Statistical Models in Engineering," Wiley, 1967.
8. H. W. Lilliefors, "On the Kolmogorov Test for Normality with Mean and Variance Unknown," Journal of American Statistical Association, Vol. 62, June 1967.
9. R. E. Odeh and D. B. Owen, "Tables for Normal Tolerance Limits," Table 1, Marcel Dekker, Inc, 1990.
10. "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," XN-NF-84-105(P)(A), Volume 1, Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, February 1987.
11. "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, Void Fraction Model Comparison to Experimental Data," XN-NF-84-105(P)(A), Volume 1, Supplement 4, Advanced Nuclear Fuels Corporation, Richland, WA 99352, February 1987.
12. Letter from J. F. Mallay to the U.S. Nuclear Regulatory Commission, "SER Conditions for EMF-2209(P) Revision 1, SPCB Critical Power Correlation," April 20, 2000.
13. Letter, Samuel J. Collins, NRC, to David G. McLees, SPC, "Demand for Information and Notice of Nonconformance (Inspection Report 99900081/97-01)," EA97-495, dated October 27, 1997.
14. Letter from Don Curet, SPC, to the U.S. Nuclear Regulatory Commission, Requesting Review of ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANFB-1125(P), Supplement 1, Appendix D, April 18, 1997.
15. Letter from J. F. Mallay to the U.S. Nuclear Regulatory Commission, "Request for Review of ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANFB-1125(P), Supplement 1, Revision 0, August 11, 1998.
16. Letter from J. F. Mallay to the U.S. Nuclear Regulatory Commission, "SER Conditions for EMF-2209(P) Revision 1, SPCB Critical Power Correlation," April 24, 2000.

Principal Contributor: A. Attard

Date: July 3, 2000



UNITED STATES  
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

May 17, 2000

Mr. James F. Mallay  
Director, Nuclear Regulatory Affairs  
Siemens Power Corporation  
2101 Horn Rapids Road  
Richland, WA 99352

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT  
EMF-2209(P) REVISION 0, "SPCB CRITICAL POWER CORRELATION"  
(TAC NO. MA6639)

Dear Mr. Mallay:

The subject topical report was submitted by the Siemens Power Corporation (SPC) by letter dated September 24, 1999. This topical report describes the analyses conducted by SPC pertaining to the application of the SPCB critical power correlation to the ATRIUM-9B and to the ATRIUM-10 fuel designs. The ATRIUM-9B fuel design is a 9x9 square array, while the ATRIUM-10 fuel design is a 10x10 square array. Both fuel designs are fixed at axial locations by ULTRAFLOW spacers and use an internal square water canister, replacing a 3x3 array of rods. The ATRIUM-9B fuel assembly contains 72 full-length rods (no part-length rods), and the ATRIUM-10 fuel assembly is made up of 83 full-length rods and 8 part-length rods.

The SPCB correlation uses planar average values of coolant mass velocity, enthalpy, and pressure to predict planar average critical heat flux. Although SPCB is a generic correlation (applicable to both ATRIUM-9B and ATRIUM-10), it is very similar to the original ANFB-10 correlation that is currently used to predict critical heat flux for the ATRIUM-10 fuel assemblies.

The staff has reviewed the topical report and the additional information and finds that the topical report is acceptable for referencing. Our safety evaluation (SE) is provided in Enclosure 1.

Pursuant to 10 CFR 2.790, we have determined that the enclosed SE does not contain proprietary information. However, we will delay placing the SE in the public document room for a period of ten (10) working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

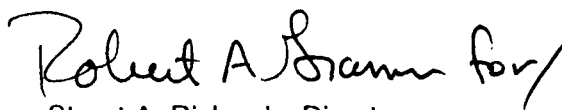
The staff will not repeat its review and acceptance of the matters described in the report, when the report appears as a reference in license applications, except to assure that the material presented is applicable to specific plant involved. Our acceptance applies only to the matters described in the report.

May 17, 2000

In accordance with the procedures established in NUREG-0390, the NRC requests that SPC publish accepted versions of the report, including the safety evaluation, in the proprietary and non-proprietary forms within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include a "-A" (designating accepted) following the report identification symbol. The accepted versions shall also incorporate all communications between SPC and the staff during this review.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are no longer valid, SPC and the licensees referencing the topical report will be expected to revise and resubmit their respective documentation, or to submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,



Stuart A. Richards, Director  
Project Directorate IV and Decommissioning  
Division of Licensing Project management  
Office of Nuclear Reactor Regulation

Project No. 702

Enclosure: Safety Evaluation



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING  
TO TOPICAL REPORT EMF-2209(P), REVISION 0,  
"SPCB CRITICAL POWER CORRELATION  
SIEMENS POWER CORPORATION  
TAC NO. MA6639

1.0 BACKGROUND

EMF-2209(P) describes the methodology behind the application of the SPCB correlation to the SPC's ATRIUM-9B and ATRIUM-10 fuel designs, (Refs. 1 and 2). EMF-2209(P) provides test data taken specifically at the Siemens test facility at Karlstein, Germany, in support of the application of the SPCB correlation to the ATRIUM-9B and ATRIUM-10 fuel designs and to the determination of the associated correlation, "Additive Constants." The ATRIUM-9B fuel has no part-length rods, but the ATRIUM-10 fuel does.

The additive constants are determined in accordance with the NRC-approved procedure described in References 3 and 4. The uncertainties associated with these additive constants are then used in the approved SPC safety limit methodology for boiling water reactor (BWR) fuel designs. The approved methodology is used to ensure that less than 0.1 percent of the fuel rods are in boiling transition during steady-state operation and during anticipated operational occurrences, in accordance with the General Design Criterion 10 and the Standard Review Plan, Section 4.4.

The SPCB correlation is new but similar to the ANFB-10 correlation, described in References 3 and 5. However, the definitions of the associated parameters (inlet sub-coolant, pressure, and mass flow) as described in Reference 3 are not changed for the application of the new SPCB to the SPC ATRIUM-9B and ATRIUM-10 fuel designs. The technical analysis of the SPCB correlation and its exclusive application to the ATRIUM-9B and ATRIUM-10 fuel designs is presented below.

2.0 TECHNICAL EVALUATION

The SPCB correlation is a new correlation designed and developed to address the critical power behavior of the SPC ATRIUM-9B and ATRIUM-10 fuel designs. The SPCB correlation is designed for application in steady-state, transient, and Loss of Coolant Accident critical heat flux (CHF) calculations for the ATRIUM-9B and ATRIUM-10 fuel designs.

The SPCB correlation was developed to predict assembly critical power for the ATRIUM-9B and ATRIUM-10 fuel designs. The correlation was developed to predict the limiting rod in a bundle

Enclosure

and account for local spacer effects and bundle geometry on critical power by a set of constants, typically referred to as "Additive Constants," one constant for each rod in the bundle. Each individual fuel design requires a unique set of additive constants.

The SPCB correlation is an empirically derived expression that is a complex function of the input parameters: local coolant enthalpy, mass flow, and pressure. These input parameters cover the ranges of pressure, mass velocity, and inlet cooling, consistent with expected operating and accident conditions. The correlation is based on local coolant conditions predicted from uniform and non-uniform axial power distribution test data. The correlation includes correction factors to account for geometry and non-uniform axial power distributions that deviate from the test data conditions.

Low-flow and high-flow behavior of the correlation are captured by refining the parameters in the correlation equations (Ref. 1). These parameters address the impacts of the variations in the local enthalpy from the planar average enthalpy. One of these parameters is the F-effective, which characterizes the fuel rod local behavior, such as enthalpy rise, and which also factors additive constants into the calculations. The additive constants account for the fuel bundle geometry and spacer effects on the critical power behavior of the bundle (Refs. 3 and 4).

## 2.1 SPCB Database and Test Strategy

The SPCB database consists of data taken at the SPC test facility at Karlstein, Germany. The test setup comprises electrically heated bundles that are physically the same as the ATRIUM-9B and the ATRIUM-10 fuel assemblies. The tests are designed to reproduce the local conditions typically present in a BWR fuel assembly and support the full range of applicability for the SPCB correlation.

Different test programs were developed to accumulate a database representative of the appropriate statistical requirements for the ATRIUM-9B and the ATRIUM-10 fuel designs. The tests selected and the number of points required were dictated by the requirements of the statistical design of experiment SDE (Refs. 6 and 7). This approach ensures that an adequate number of tests are performed and that sufficient data are gathered to perform appropriate simulation of the behavior of the ATRIUM-9B and the ATRIUM-10 fuel designs.

Both steady-state and transient tests were performed as part of the validation of the SPCB correlation. In each case, the tests were designed to include test runs with peaked rods located adjacent to the internal water channel.

The database comprises more than 2500 data points taken in a large number of tests performed at the SPC test facility. The database consists of upskew, downskew, and cosine axial power shapes accounting for adjacent rod positions, rods on the interior of an assembly, and rods adjacent to the water canister (channel), a feature unique to the ATRIUM fuel design.

The local power peaking patterns were selected to determine the effects of the upskew axial power profiles as compared to the cosine power profiles in several regions of the test bundle. Local power peaking data were also collected at the corners, the peripheral rows, as well as around the internal water canister to ensure complete understanding of the fuel CHF behavior, particularly in these regions.



The internal water canister is a major and unique characteristic of the SPC's ATRIUM fuel design. It replaces a 3X3 matrix of fuel rods. The rectangular canister is designed so that the subchannels around it are regular in size, typical of those addressed by the original base ANFB correlation. The test matrixes of the ATRIUM-9B and the ATRIUM-10 fuel designs used at the SPC test facility included tests to confirm the behavior of the fuel surrounding the internal water canister. Neither the ATRIUM-9B nor the ATRIUM-10 fuel design showed any abnormal behavior around the internal water canister.

## 2.2 Description of the Additive Constants

Correlation parameters such as F-effective ( $F_{EFF}$ ) account for the local peaking factor effect on the bundle critical power.  $F_{EFF}$  is constructed in two parts. One part depends solely on the peaking factors of the rod of interest and its immediate neighbors ( $F_{EFF0}$ ); the other part, termed the "additive constant," accounts for other local effects, such as bundle geometry and spacer effects. These spacer and bundle geometry effects influence the critical power behavior of the bundle. Therefore, an offset term is applied to each rod in the bundle, subject to the rod's position in the bundle. This offset term is called the "additive constant." The additive constant can be considered as a flow/enthalpy redistribution characteristic of a particular lattice/spacer design. So the additive constants are unique to a particular fuel design. They are explicitly determined for each lattice/spacer design configuration and are utilized in design calculations for the corresponding fuel bundle (Ref. 3).

To assert the ability of the correlation to predict steady-state as well as transient upskew and downskew axial power shape, only the cosine test data were used in the determination of the additive constants, thus validating the use of the additive constants in steady-state and transient calculations. The additive constants are experimentally determined from a large data bank representative of the power profile expected during the operational range of the ATRIUM-9B and the ATRIUM-10 fuel designs.

## 3.0 STATISTICAL ASPECTS OF THE SPCB CORRELATION

The statistical aspects of the SPCB correlation consist of applying appropriate statistical techniques (Refs. 6 and 7) to the SPCB database. These techniques involve the evaluation of distribution characteristics, figures of critical power ratios (CPRs) with respect to each characteristic within the correlation, descriptive statistics for subgroups of data, descriptive statistics for additive constants and additive constants uncertainty, and conservatism of the SPCB critical power correlation. A good correlation would place the CPR near 1.00 (unity), with a very small associated uncertainty.

The correlation study examined the CPR in a series of tests. A total of 12 tests were performed: 7 tests pertained to the ATRIUM-9B and 5 tests pertained to the ATRIUM-10 fuel designs. For the ATRIUM-9B fuel, three of the seven tests were conducted with a chopped cosine shaped axial power profile, one test with a downskew power profile, and two tests with an upskew power profile. For the ATRIUM-10 fuel, three of the five tests were conducted with a chopped cosine shaped axial power profile, one of the five tests with a downskew power profile, and one test with an upskew power profile. Each test was repeated many times ("runs"). The input variables into each run entered the experimental design at different levels to reflect a diversified operating environment, resulting in a database containing in excess of 2500 data points. Twenty percent of this data was used to validate the correlation, while the remaining 80 percent was used to develop the SPCB correlation.

The multiplicity of runs within each test was required in order to involve various levels of input factors (inlet flow, inlet sub-cooling, and pressure). For most of the runs, these factors were selected at random, following standard statistical procedures (Refs. 6 and 7). For dryout testing, additional runs were made following a two-level, three-factor factorial design to ensure that the entire range of interest (including "corner to corner") was represented.

Review of SPC calculations shows that the average CPR appears to be very near 1.0. That ratio is retained without any apparent trend across inlet mass velocity (Mlb/hr-ft<sup>2</sup>), enthalpy (Btu/lbm), pressure (psia), the best estimate of the  $F_{EFF}$ , or the axial offset. The overall CPR mean for the ATRIUM-9B 1629 data points was calculated to be 0.996, and the CPR mean for the ATRIUM-10 1028 data points was calculated to be 0.996.

To evaluate the quality of the correlation, the staff independently calculated a CPR 95/95 upper tolerance limit (Refs. 8 and 9) for each test, for each profile, and for the entire set of runs. The staff 95/95 calculation was compared to SPC's 95/95 calculation. Apart from rounding errors and conservative table interpolations, the staff's calculation was in total agreement with SPC's calculation. This limit is interpreted to mean that one is 95 percent sure that at least 95 percent of the population of runs yields a CPR value no higher than 1.022 for ATRIUM-9B and a value of 1.034 for ATRIUM-10. SPC's calculations also show that for any test or grouping of tests, the percentage of runs that fall below their associated tolerance limits is at least 95.7 percent for ATRIUM-9B and 96.8 percent for ATRIUM-10.

The submittal contains charts and tables reflecting CPR behavior across different mass velocity (Mlb/hr-ft<sup>2</sup>) for individual tests. Although some tests show higher CPR values associated with high mass velocity, the reverse is true for other tests, and no dependency between CPR and mass velocity is apparent.

Another objective of SPC's study that involves statistical consideration is the determination of the additive constant for both fuel types. The additive constant is a statistical adjustment to the measure of the  $F_{EFF}$  to account for the effect of the rod's geometric position within the assembly. This adjustment has two components: a calculated additive constant and a measure of uncertainty associated with the calculation. In the development of the additive constants, SPC uses only the cosine profile data. However, the measure of the associated uncertainty is calculated from the entire database, containing cosine, upskew, and downskew test data.

The main contributors to this uncertainty are two sources of variability: "within test variability" and "between test variability." The within test variability is given as a weighted average in which the weighting factors are the number of runs per test. The between test variability is given as a weighted average of the difference between the  $F_{EFF}$  for a rod in a test bundle and the average  $F_{EFF}$  for the test bundle. The weighting factors are the number of boiling transitions for a rod in the test bundle. The square root of the sum of the squares (the two sources of variability) give the measure of variability associated with the calculation of the additive constant. In-depth review of the statistical section of the submittal leads the staff to concur with the statistical methods used and the results obtained by the vendor.

#### 4.0 SPCB CORRELATION BEHAVIOR

The SPCB correlation was tested to ensure smooth functions and no significant discontinuities in its behavior over the entire range of operability of the fuel. Flow, enthalpy, and pressure-dependent functions within the correlation, such as the "Tong Factor" correction for both fuels, was investigated for its behavior over the entire applicable range of the fuels. A number of tests were conducted to determine the sensitivity of the major functions within the SPCB correlation to flow, inlet subcooling, pressure variation,  $F_{EFF}$ , and axial power shape.

Review of the data, figures, and tables indicates that the SPCB correlation behaves well over the applicable range of the fuel.

#### 5.0 SPCB CORRELATION VALIDATION

SPC performed several tests to validate the behavior of the SPCB correlation in steady-state and transient events. The validation database consisted of 20 percent of the total steady-state data points that were not included in the correlation database. The remaining 80 percent of the database (the so-called verification set) was used to develop the correlation. In addition, data were collected from tests conducted on an ATRIUM-10P assembly that contained more part-length fuel rods than are usually found in a typical ATRIUM-10 assembly. These tests were conducted to demonstrate the ability of the SPCB correlation to capture the effects of the part-length rods, as well as the correlation agreement with the data. The predicted SPCB correlation between critical power versus the measured critical power for these tests showed very good agreement.

Two sets of transient tests were performed as part of the validation process. Both tests were designed to peak rods around the internal water canister. The difference between the two tests is that the first test had rods with a chopped cosine-shaped axial power profile and the second had rods with an upskew axial power shape. Another purpose of the tests was to validate the concept that the additive constants can be derived from steady-state cosine tests and applied to other axial shapes under transient conditions.

The transient tests performed were the simulated load rejection with no bypass (LRNB) events that consisted of power, pressure ramps, and flow decay. Power forcing functions were programmed to produce transient heat flux on the surface of the rod typical of an LRNB event. Parameters monitored during the tests were power, inlet flow, system pressure, inlet temperatures, and cladding temperatures.

The transient thermal-hydraulic code, XCOBRA-T (Refs. 10 and 11), was used to predict the test results using the SPCB steady-state critical power correlation. XCOBRA-T calculates the fluid conditions at a specified time step. The CHF is calculated at each axial position and time step, then compared to the corresponding measured rod heat flux at the surface of the rod. The ratio of the calculated heat flux to the measured rod heat flux is defined to be the critical heat flux ratio (CHFR). When this ratio is unity, it is referred to as the minimum critical heat flux ratio (MCHFR), and it signifies "boiling transition" in a transient event. Comparison of measured and calculated time-to-boiling transitions for cosine and upskew transient tests shows that the XCOBRA-T calculated time-to-boiling transition values are conservative when compared to actual boiling transition time. This validation confirms the use of the steady-state SPCB correlation and the associated additive constants in evaluating transient events.

## 6.0 LOCAL PEAKING FACTORS

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 1.5 is exceeded, an additional additive constant uncertainty is applied on a rod-by-rod basis in accordance with Table 3.15 of Reference 1. These conditions have been agreed upon by both the NRC staff and SPC, (Ref. 12).

## 7.0 NON CONFORMANCE ISSUES

The submittal, as documented in Reference 1, is SPC's corrective action in response to Part 2 of notice of Nonconformance 99900081/97-01, as stated in Attachment II of SPC's letter to the NRC, dated February 24, 1998 (Ref. 13). The Nonconformance stated that: SPC failed to develop an adequate number of tests points and failed to test an adequate range of conditions to justify the uncertainty values for the "additive constants" used in determining the safety limit minimum critical power ratio (SLMCPR) for the ATRIUM-9B fuel design. This statement implies that SPC should have used larger uncertainty values in the SLMCPR determinations in order to reflect the full operability range of the ATRIUM-9B fuel design. In addition, because the results of the ANFB correlation are used as inputs to the safety limit methodology, this has immediate implications regarding the SLMCPR and the operating limit minimum critical power ratio (OLMCPR) of the Commonwealth Edison Company plants (Quad Cities Unit 2, Cycle 15, Dresden Unit 2, Cycle 15, and LaSalle County Unit 2, Cycle 8) and the Washington Public Power Supply System (Washington Nuclear Unit 2, Cycle 13) loaded with ATRIUM-9B fuel.

In response to this notice of Nonconformance, SPC developed and implemented interim methodologies (ANF-1125, Appendixes D and E) (Refs. 14 and 15), while performing additional dryout testing of the ATRIUM-9B design to obtain additional data to cover the extended range of thermal-hydraulic parameters for the ATRIUM-9B fuel design.

The NRC staff contends that with the submittal of EMF-2209(P), the vendor (SPC) has provided the additional data necessary for the SPCB critical power correlation to provide a rigorous treatment over the entire operating range of the ATRIUM-9B fuel. Thus, with the submittal of EMF-2209(P), all problems identified in the inspection report (Nonconformance 99900081/97-01, Part 2) related to the dryout methodology for ATRIUM-9B fuel have been addressed.

## 8.0 TECHNOLOGY TRANSFER

SPC described the technology transfer program (Ref. 16) which the licensees must successfully complete in order to perform their own thermal-hydraulic calculations using the SPCB correlation and the XCOBRA-T code in support of reload analyses. The overall process consists of training, benchmarking, and change control. In addition, SPC described the process for a licensee to implement the new correlation (SPCB). This process includes performance of an independent benchmarking calculation by SPC for comparison to the licensee-generated results to verify that the new CHF correlation is properly applied. The staff has reviewed the process and find it acceptable because training, bench-marking, and change control have been adequately addressed.

## 9.0 CONCLUSION

The staff has reviewed the analyses in Topical Report EMF-2209(P), Revision 0, "SPCB Critical Power Correlation," and concludes that on the basis of its findings (presented above), Topical Report EMF-2209(P) is acceptable for licensing applications, in accordance with SPC's agreement, subject to the following conditions:

1. The SPCB correlation (as described in this submittal, Reference 1) is applicable to SPC ATRIUM-9B and ATRIUM-10 fuel designs, with a local peaking factor no greater than 1.5.
2. If, however, in the process of calculating the MCPR safety limit, the local peaking factor of 1.5 is exceeded, an additional uncertainty of 0.026 for ATRIUM-9B and 0.021 for ATRIUM-10 will be imposed on a rod-by-rod basis.
3. The SPCB correlation range of applicability is as follows:

Pressure (psia)	571.4 to 1432.2
Inlet Mass Velocity (Mlb/hr-ft <sup>2</sup> )	0.087 to 1.5
Inlet Subcooling (Btu/lbm)	5.55 to 148.67
Design Local Peaking	1.5
Tested Local Peaking	1.45

4. Technology transfer will be accomplished only through the process described in Reference 16, which includes the performance of an independent bench-marking calculation by SPC for comparison to the licensee-generated results to verify that the new CHF correlation (SPCB) is properly applied for the first application by the licensee.

## 10.0 REFERENCES

1. Letter from H. D. Curet, of SPC to the U.S. Nuclear Regulatory Commission, submitting Topical Report EMF-2209(P), Revision 0, September 24, 1999.
2. Letter from J. F. Malley to the U.S. Nuclear Regulatory Commission, "Request for Additional Information (RAI) to Topical Report EMF-2209(P), "SPCB Critical Power Correlation," Revision 0, March 20, 2000.
3. Letter from R. A. Copeland, of SPC, Transmittal of (A) version of ANF-1125(P) to the U.S. Nuclear Regulatory Commission, April 27, 1990.
4. Letter from H. D. Curet, of SPC to the U.S. Nuclear Regulatory Commission, submitting Topical Report EMF-1997(P), Revision 0, October 30, 1997, and Letter from H. D. Curet, submitting Topical Report EMF-1997(P), Supplement 1, Revision 0, January 29, 1998.
5. Letter from J. F. Malley to the U.S. Nuclear Regulatory Commission, "Request for Additional Information (RAI) to Topical Report EMF-1997(P), ANFB-10 Critical Power Correlation," Revision 0, May 7, 1998.

6. "Statistical Methods for Nuclear Material Management," NUREG/CR-4604, PNL-5849, December 1988.
7. G. J. Hahn, and S. S. Shapiro, "Statistical Models in Engineering," Wiley, 1967.
8. H. W. Lilliefors, "On the Kolmogorov Test for Normality with Mean and Variance Unknown," Journal of American Statistical Association, Vol. 62, June 1967.
9. R. E. Odeh and D. B. Owen, "Tables for Normal Tolerance Limits," Table 1, Marcel Dekker, Inc, 1990.
10. "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," XN-NF-84-105(P)(A), Volume 1, Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, February 1987.
11. "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, Void Fraction Model Comparison to Experimental Data," XN-NF-84-105(P)(A), Volume 1, Supplement 4, Advanced Nuclear Fuels Corporation, Richland, WA 99352, February 1987.
12. Letter from J. F. Malley to the U.S. Nuclear Regulatory Commission, "SER conditions for EMF-2209(P) Revision 1, SPCB Critical Power Correlation," April 20, 2000.
13. Letter, Samuel J. Collins, of NRC to David G. McLees, of SPC "Demand for Information and Notice of Nonconformance (Inspection Report 99900081/97-01)," EA97-495, dated October 27, 1997.
14. Letter from Don Curet, of SPC to the U.S. Nuclear Regulatory Commission, requesting review of ANFB Critical Power Correlation Uncertainty for Limited data Sets, ANFB-1125(P), Supplement 1, Appendix D, April 18, 1997.
15. Letter from J. F. Malley, of SPC to the U.S. Nuclear Regulatory Commission, "Request for review of ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANFB-1125(P), Supplement 1, Revision 0, August 11, 1998.
16. Letter from J. F. Malley to the U.S. Nuclear Regulatory Commission, "SER conditions for EMF-2209(P) Revision 1, SPCB Critical Power Correlation," April 24, 2000.

Principal Contributor: A. Attard

Date: May 17, 2000

# SIEMENS

September 24, 1999  
NRC:99:042

Document Control Desk  
ATTN: Chief, Planning, Program and Management Support Branch  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555-0001

## **Request for Review of EMF-2209(P) Revision 0, "SPCB Critical Power Correlation"**

Ref.: 1. ANF-1125(P)(A) and Supplements 1 and 2, "ANFB Critical Power Correlation," Advanced Nuclear Fuels Corporation, April 1990.

Ref.: 2. EMF-1997(P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998.

Fifteen proprietary and 12 nonproprietary copies of topical report EMF-2209(P) Revision 0, "SPCB Critical Power Correlation" are being submitted to the NRC for review and acceptance for referencing in licensing actions. (NOTE: Three proprietary copies and one nonproprietary copy have been sent directly to Mr. Nageswaran Kalyanam). This topical report presents an improved critical power correlation that was developed to eliminate deficiencies previously noted by the NRC in other SPC critical power correlations (Reference 1 and Reference 2).

Also enclosed are copies of the two references, which are extensively referred to in EMF-2209(P) Revision 0. These copies have been forwarded to Mr. Kalyanam and are provided for the convenience of the reviewer of the subject topical report.

SPC requests approval of this topical report by April 1, 2000, to permit the use of the correlation in support of licensing action on reload fuel. During a meeting with representatives of the Reactor Systems Branch in late August, they suggested that their review would be facilitated if SPC provided a discussion of the topical report. In response to this suggestion, we propose that a meeting be held at the NRC during the week of October 18. The individuals who developed the correlation would explain its unique characteristics, describe its verification and validation data base, and note its improvements over the current SPC critical power correlations.

Some of the information contained in the enclosed topical report is considered to be proprietary to Siemens Power Corporation. As required by 10 CFR 2.790(b), an affidavit is enclosed to support the withholding of this information from public disclosure.


**Siemens Power Corporation**

2101 Horn Rapids Road  
Richland, WA 99352

Tel: (509) 375-8100  
Fax: (509) 375-8402

The affidavits originally submitted for the enclosed copies of References 1 and 2 satisfy the requirements of 10 CFR 2.790(b) to support the withholding of this information from public disclosure.

Very truly yours,

  
James F. Mally, Director  
Regulatory Affairs

cc: Mr. N. Kalyanam (w/Enclosures)  
Mr. J. L. Wermiel  
Project No. 702 (w/Enclosures)





UNITED STATES  
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

February 22, 2000

Mr. James F. Mallay  
Director, Nuclear Regulatory Affairs  
Siemens Power Corporation  
2101 Horn Rapids Road  
Richland, WA 99352

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON SIEMENS TOPICAL  
REPORT, EMF-2209(P), REVISION 0, "SPCB CRITICAL POWER  
CORRELATION" (TAC NO. MA6639)

Dear Mr. Mallay:

By letter dated September 24, 1999, the Siemens Power Corporation (SPC) submitted Revision 0 to Topical Report EMF-2209(P), "SPCB Critical Power Correlation" for staff review. The staff has done a preliminary review and requests the following information identified in the enclosure.

The additional information was discussed with your staff and a mutually agreeable target date of 30 days from the date of this letter for your response was established. If circumstances result in the need to revise the target date, please call me at the earliest opportunity at 301-415-1480.

Sincerely,

N. Kalyanam, Project Manager, Section 1  
Project Directorate IV & Decommissioning  
Division of Licensing Project Management  
Office of Nuclear Reactor Regulation

Project No. 702

Enclosure: Request for Additional Information

REQUEST FOR ADDITIONAL INFORMATION

SIEMENS TOPICAL REPORT, EMF-2209(P)

"SPEC CRITICAL POWER CORRELATION"

SPCB-Correlation Development

1. On page 1-2, second paragraph, it is stated that an additional 316 validation data points were included to the 781 validation data base for the purpose of validating the SPCB correlation. Were the 316 data points obtained for a different fuel assembly? Please clarify.
2. On page 1-2, third paragraph, please explain the need for additional uncertainty required for peaking factors greater than 1.5.
3. On page 2-4, first paragraph, it is stated that two coefficients are used, one for the mass velocity less than or equal to 0.37, and one for mass velocities greater than or equal to 0.42.
  - a. What is the basis for these lower and upper limits?
  - b. Provide the technical justification for the interpolation between these bounds.
4. Tables 2.1 through Table 2.7 provide values for various coefficients, subject lower and upper bounds. Is one to assume that different values for these coefficients are obtained by interpolation as in the case of Table 2.1?
5. Chapter 2, in particular Sections 2.0 to 2.3, contains the mathematical development of the SPCB correlation. As such, it is imperative that one obtains a clear understanding of the various components (variables, parameters, etc.) and their respective use in the formulation of the correlation for the two different fuels. In reviewing Sections 2.0 to 2.3, it became apparent that there are a number of junctions in the road, depending on whether one is addressing the ATRIUM-9B fuel or the ATRIUM-10 fuel. In order to expedite the review of this topical report, please provide a road map (Flow Chart/Event Tree) showing the clear and separate routes taken in developing the two forms of the SPCB correlation for each of the fuels in question.
6. On page 2-23 (item 2, middle of the page), please provide the reasoning for switching from a simple mean to a weighted mean.
7. Please provide additional information for Figure 2.3.
8. On page 2-34, paragraph 2, reference is made to the "ANFB" limits. Should that be the "SPCB" limits?
9. On page 2-35, Section 2.6.2.1, the high and low enthalpy limits are addressed. What are the values of these limits?

10. On page 2-36, second paragraph, it is stated that "corresponding quality distributions are artificially increased." What are "quality distributions"?
11. Is "inlet mass velocity" inter-changeable with "inlet mass flow rate" ?
12. Figure 3.8 on page 3-15. The x-axis is labeled "Active flow". What is "Active Flow"?
13. Does Figure 3.39 on page 3-41 include all the data for the ATRIUM-9B and that for the ATRIUM-10?
14. Page 4-9, Figure 4.7. Please provide justification for the difference in magnitude at high inlet flow.
15. Page 4-9, the last sentence in the second paragraph states that Test 48.1 consists of transient data only. Does that mean that Test 29.5 consist of data other than transient data?
16. In Section 5.3, the third paragraph brings up the subject of peaked rods going into dryout. For both fuel types, what procedure is used in determining which rods are peaked and which rods go into dryout.
17. Page 5-11, last paragraph. Please provide additional discussion regarding as to why only two tests were needed to be performed on the ATRIUM-9B to demonstrate that the ATRIUM-10 additive constant methodology is applicable to the ATRIUM-9B fuel.
18. Page 5-12, the last paragraph refers to "individual case." Is this same as "individual test?"

#### SPCB-Statistical RAIs

1. A general statement: Whenever presenting mean and standard deviation (such as in the bottom sentence of Page 1-1), include the sample size and the associated tolerance limit.
2. Page 1-2, Section 1.1. Provide statistical tests that compare the behavior (mean, variance) of the 1,876 correlation points to the 781 validation points. Also, did the 316 additional validation points differ in behavior from the 781 validation points?
3. Page 1-2, second paragraph. Text states that transient tests are performed on ATRIUM 10 (not mentioning ATRIUM-9B). Page 3-36, second paragraph says that dryout tests were performed on the ATRIUM-9B and ATRIUM-10.
4. Pages 2-4 thru 2-10 (Tables 2.1 thru 2.7). Is "G" in these tables the same as G bar given in, say, equation (2.7)?
5. Page 2-6, Equation 2.12. The coefficient "f1" in equation 2.12 is not defined.

6. Page 2-13, Equation 2.23. Identify/explain how the coefficients 0.624 and 0.314 were obtained.
7. Page 2-26, Figure 2.3. Symbols (like X, square, and diamond) need a legend.
8. Page 3-1, Table 3.1. Show how Sigma (given as 0.021) for ATRIUM-9B was derived.
9. Page 3-1, Table 3.1:
  - expand the table to include for each test the 95/95 upper tolerance limit for ECPR,
  - maximum value obtained for the test,
  - number of data points in the test that exceed the tolerance limit, and
  - the percent number of points below the tolerance limit.

Provide similar entries, separately for each fuel, and for all tests of the same profile.

Suggestion: Follow the style of Table 4.1 for ANFB 10 that you provided in your May 26, 1998 communication to E.Y. Wang.

10. Page 3-2, Table 3.2. Show a complete table of the analysis of variance (ANOVA). Please provide separate analysis for each fuel type. Did any analysis detect significant test differences? Was the data tested for homogeneity of variances prior to constructing the ANOVA?
11. Pages 3-2 and 3-3, equations for  $m_2$ ,  $m_3$ ,  $m_4$ ,  $\beta_1$ , and  $\beta_2$ : What is the numeric value of "n" in each of these calculations?
12. Page 3-5, paragraph 3.2. Justify the use of 1 percent as a level of significance for testing normality. What is the numerical value of Lillifor's statistic?
13. Pages 3-4 through 3-29, Figures 3.1 through 3.36. Indicate the sample size associated with each figure.
14. Page 3-38, last paragraph. Indicate where the upper 95 percent confidence limits on the additive constants are implemented.
15. Page 4-2, Tables 4.1 and 4.2. Expand the tables as requested for Table 3.1.
16. Page 4-8, Figure 4-6 (and others). What do the different lines represent? Provide the necessary labels. Similarly, provide the necessary labels for Figures 5.7 thru 5.89.
17. Were there any outliers in the evaluation, and if so, what was their disposition?

# SIEMENS

March 20, 2000  
NRC:00:019

Document Control Desk  
ATTN: Chief, Planning, Program and Management Support Branch  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555-0001

## **Request for Additional Information to the Topical Report EMF-2209(P) Revision 0, "SPCB Critical Power Correlation"**

- Ref.: 1. Letter, N. Kalyanam (NRC) to James F. Mallay (SPC), "Request for Additional Information – Siemens Topical Report, EMF-2209(P) Revision 0, *SPCB Critical Power Correlation* (TAC No. MA6639)," February 22, 2000.
- Ref.: 2. Letter, James F. Mallay (SPC) to Document Control Desk (NRC), "Request for Review of EMF-2209(P) Revision 0, *SPCB Critical Power Correlation*," NRC:99:042, September 24, 1999.

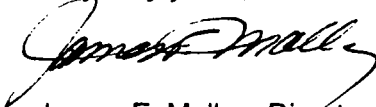
In Reference 1, the NRC requested additional information to facilitate the completion of its review of the Siemens Power Corporation topical report on the SPCB correlation (see Reference 2). Responses to this request are provided in two attachments: one proprietary and one nonproprietary.

These responses, along with several editorial corrections, have made it necessary to change numerous pages in the topical report. These changes have been incorporated in Revision 1 to the topical report and are described on page i, "Nature of Changes," in the report.

In addition to the attached responses, four copies of the proprietary version and two copies of the nonproprietary version of Revision 1 of the topical report are enclosed with this letter. When the revised report is found acceptable for referencing in license applications, SPC will publish the accepted versions of the report in accordance with the procedures established in NUREG-0390.

Siemens Power Corporation considers some of the information contained in the attachments and enclosures to this letter to be proprietary. This information has been noted by enclosing it within brackets. The affidavit provided with the original submittal of the reference topical report satisfies the requirements of 10 CFR 2.790(b) to support the withholding of this information from public disclosure.

Very truly yours,



James F. Mallay, Director  
Regulatory Affairs

cc: N. Kalyanam (3(P) and 1(NP); w/attachment)  
Project No. 702 (1(P) and 1(NP); w/attachment)

## **Siemens Power Corporation**

2101 Horn Rapids Road  
Richland, WA 99352

Tel: (509) 375-8100  
Fax: (509) 375-8402

## Responses to RAI Questions to EMF-2209

Q.	SPCB-Correlation Development RAIs.
1	<p data-bbox="277 411 1357 541"><i>On page 1-2, second paragraph, it is stated that an additional [ ] validation data points were included to the [ ] validation data base for the purpose of validating the SPCB correlation. Were the [ ] data points obtained for a different fuel assembly? Please Clarify.</i></p> <p data-bbox="277 575 1357 741">Response: As shown on page 4-6 and 4-7 SPCB was validated with an alternate (different) fuel design comprised of 12 part-length rods and 79 full-length rods. This alternate fuel design is an ATRIUM-10 design with a spacer that is similar in design to the spacer used in the data base. The distinction of the alternate design is essentially the use of 12 rather than 8 part-length rods.</p>
2	<p data-bbox="277 751 1357 810"><i>On page 1-2, third paragraph, please explain the need for additional uncertainty required for peaking factors greater than 1.5.</i></p> <p data-bbox="277 844 1357 1073">Response: During the dryout testing some tests were performed where the high peaked rods were as high as about 1.45. Because of the trend of the data, one may safely extrapolate to a local peaking of 1.50 for design cases. However, during the safety limit analysis the local peaking may be perturbed so that a design local peaking may increase over the 1.50 limit. If that happens, the safety limit analysis will use the incremental uncertainties for the ATRIUM-9B and ATRIUM-10 designs provided in Table 3.15.</p>
3	<p data-bbox="277 1083 1357 1178"><i>On page 2-4, first paragraph, it is stated that two coefficients are used, one for the mass velocity less than or equal to 0.37 and one for mass velocities greater than or equal to 0.42.</i></p> <p data-bbox="277 1199 1357 1262"> <i>a. What is the basis for these lower and upper limits?</i>  <i>b. Provide technical justification for the interpolation between these bounds.</i> </p> <p data-bbox="277 1293 1357 1566">Response: The behavior of critical power is observed to be strongly influenced by flow rate. The rod indicating boiling transition or dryout may change from a high powered rod to a lower powered rod as the flow rate changes. The intention of the correlation is to provide relationships that help describe the phenomena that are observed while maintaining acceptable uncertainty, no singularity, and well behaved transition between regions. The lower limit is based on providing a good fit of what is occurring at the low flows while the upper limit is based on providing a good fit of what is occurring at the higher flows.</p> <p data-bbox="277 1598 1357 1661">The fitting of the coefficients for the A and B functions was performed in the following manner:</p> <p data-bbox="277 1692 1357 1892">All of the correlation data for the ATRIUM-10 (774 data points) were evaluated to determine coefficients for A and B simultaneously. The result, when applied, provided good estimates for CPR especially for the low flow region. In order to provide improved estimates of CPR for higher flow regions, the data for high flow was correlated (696 data points) and an alternate set of coefficients was determined. In order to provide a smooth transition between the use of the two</p>

Q.	<p>SPCB-Correlation Development RAIs.</p> <p>sets of coefficients, a region was established for which a linear interpolation process would be used. For example a value for the function of A would be determined at the mass velocity corresponding to the usage limit of 0.37 Mlb/hr-ft<sup>2</sup> using the low flow set of coefficients and A would be determined at the high mass velocity lower bound of 0.42 Mlb/hr-ft<sup>2</sup> using the high flow set of coefficients. Then a linear interpolation of A would be performed based on the two flows. Technically the issue is one of avoiding step changes in behavior.</p> <p>Note: Equation 2.7 on page 2.4 shows a double asterisk with the A3 coefficient. This is a typographical error and will be corrected to show a single asterisk in the final approved version of the document.</p>
4	<p><i>Tables 2.1 through Table 2.7 provide values for various coefficients, subject to lower and upper bounds. Is one to assume that different values for these coefficients are obtained by interpolation as in the case of Table 2.1?</i></p> <p>Response: Yes, for the case of flows falling between the various regions of the various tables, a linear interpolation is used to obtain a value of the overall coefficient for use in the correlation.</p>
5	<p><i>Chapter 2, in particular Sections 2.0 to 2.3, contains the mathematical development of the SPCB Correlation. As such it is imperative that one obtains a clear understanding of the various components (variables, parameters, etc.) and their respective use in the formulation of the correlation for the two different fuels. In reviewing Sections 2.0 to 2.3 it quickly became apparent that there are a number of junctions in the road, depending on whether one is addressing the ATRIUM-9B fuel or the ATRIUM-10 fuel. In order to expedite the review of this Topical Report, please provide a road map (Flow Chart/Event Tree) showing the clear and separate routes taken in developing the two forms of the SPCB correlation for each of the fuels in question.</i></p> <p>Response: A flow chart is shown in the attachment that depicts the separate routes taken in developing the two forms of the SPCB correlation for each of the fuel types in question. The notation G in the flow chart is interchangeable with <math>\bar{G}</math>.</p>
6	<p><i>On page 2-23 (Item 2, middle of page), please provide the reasoning for switching from a simple mean to a weighted mean. (Item 2 middle of the page.)</i></p> <p>Response: The use of a simple mean from a power weighted mean was introduced with the ANFB-10 correlation (Reference 2.1) and is retained for the SPCB critical power correlation. The Reference indicated on page 2.23 should be to Reference 2.2, the ANFB correlation.</p>
7	<p><i>Please provide additional information for Figure 2.3.</i></p> <p>Response: Figure 2.3 with a legend is provided in Revision 1 to EMF-2209(P).</p>
8	<p><i>On page 2-34, paragraph 2, reference is made to the "ANFB" limits. Should that be the "SPCB" limits?</i></p>

Q.	SPCB-Correlation Development RAIs.
	Response: Yes, the letters ANFB have been replaced with the letters SPCB in Revision 1 to EMF-2209(P).
9	<p><i>On page 2-35, Section 2.6.2.1, the high and low enthalpy limits addressed. What are the values of these limits?</i></p> <p>Response: The values of the limits are depicted in Table 1.2 on page 1-3.</p>
10	<p><i>On page 2-36, second paragraph, it is stated that "corresponding quality distributions are artificially increased." What are "quality distributions?"</i></p> <p>Response: For each nodal enthalpy, a thermodynamic quality can be determined. As quality is a parameter in the Hench-Levy correlation, then when incrementing the enthalpy, the quality is also incremented. As each node would have a different enthalpy, then within the assembly there would be a distribution of quality.</p>
11	<p><i>Is "inlet mass velocity" inter-changeable with "inlet mass flow rate?"</i></p> <p>Response: The correlation calculation relates only to inlet mass velocity (a mass flow rate per unit area). The correlation or physical behavior can be visually examined using either mass velocity or mass flow rate.</p>
12	<p><i>Figure 3.8 on page 3-15, the x-axis is labeled "Active flow." What is "Active Flow?"</i></p> <p>Response: The test assembly uses an electrically heated model of a fuel assembly. One of the differences between a test assembly and an assembly produced for in-reactor use is the test assembly does not typically allow flow through the water channel while an in-reactor fuel assembly would allow flow through the water channel. Thus the flow rate that is measured and delivered to the inlet of a test assembly is the flow rate that is present in the electrically heated region. Within an in-reactor fuel assembly, the flow that passes through the orifice is progressively decreased due to the need for flow in the bypass and the planned flow within the water rod or water channel structure. The flow adjacent to nuclear fuel rods is referred to as active flow to distinguish it from the total flow that might be delivered at the inlet of the assembly. This terminology was borrowed for use in the test assembly to aid the in-house engineer that is reading the document in understanding that the flow that was delivered in test is in-fact the flow that is in the region where the powered rods are located.</p>
13	<p><i>Does Figure 3.39 on page 3-41 include all the data for the ATRIUM-9B and that for the ATRIUM-10?</i></p> <p>Response: Yes, ATRIUM-9B and ATRIUM-10 validation and verification data are included. Figure 3.39 is an overlay of Figure 1.2 on Figure 1.1.</p>
14	<p><i>Page 4-9, Figure 4.7, please provide justification for the difference in magnitude at high inlet flow.</i></p> <p>Response: The principle reason for differences in magnitude of power at high inlet flow is due to the influence of axial power shape. Axial power shape (as might be visualized by the gradient of heat flux) has a strong influence at the high mass velocities. The fact that data is observed to behave in this manner results in the use</p>



Q.	SPCB-Correlation Development RAIs. of an upstream memory effect such as the Tong-Factor or the non-uniform axial correction factor of SPCB. At the lower mass velocities, the importance of heat flux increases and the axial shape is not as dominant.
15	<p><i>Page 4-9, the last sentence in the second paragraph states that Test 48.1 consists of transient data only. Does that mean that Test 29.5 consist of data other than transient data?</i></p> <p>Response: Yes, as shown on page 3-32 Test assembly 29.5 was used for steady state critical power data and then was used for transient data. Test 48.1 was only used for transient data.</p>
16	<p><i>In Section 5.3, the third paragraph brings up the subject of peaked rods going into dryout. For both fuel types, what procedure is used in determining which rods are peaked and which rods go into dryout.</i></p> <p>Response: The first thing to consider when developing the planned peaking pattern for dryout testing is that the peaking pattern does not necessarily mimic a neutronic design. Instead, by peaking certain rods one may take advantage of the bundle symmetry and minimize the number of tests required. Rods are peaked in groups in an attempt to drive certain locations into dryout. Because of the methodology for determining additive constants, if a group of rods are peaked and only one rod goes into dryout, it is conservatively assumed that the other two rods went into dryout at the same power. If a rod location is not peaked at all during the dryout testing, then that position has its additive constant determined by assuming that it went into dryout during the test that had the highest peaking in this location.</p>
17	<p><i>Page 5-11, last paragraph. Please provide additional discussion regarding as to why only two tests were needed to be performed on the ATRIUM-9B to demonstrate that the ATRIUM-10 additive constant methodology is applicable to the ATRIUM-9B fuel.</i></p> <p>Response: The methodology for determining the effect of local peaking on additive constants was developed in ANF-1125, Supplement 1(P)(A). That methodology has remained the same for both ANFB-10 and SPCB. During the testing of the ATRIUM-10 bundle several tests were run to drive rods into dryout at different local peakings. The positions of these tests were in positions that represented each different bundle location; i.e., 3x3 corner, edge rods, interior, etc. The ATRIUM-9B bundle geometry is therefore represented in the ATRIUM-10 testing. Bear in mind also, that the additive constant methodology considers the effect of the rods surrounding the peaked rod of interest. Therefore, the methodology doesn't know whether the bundle is a 4x4, 5x5, 9x9, or 10x10 assembly.</p>
18	<p><i>Page 5-12, in the last paragraph refers to "individual case." Is this same as "individual test?"</i></p> <p>Response: The word "individual" is used three times in the document. One time it modifies "tests," a second time it modifies "rod," and the third time it modifies "case." The context for which it is used as a modifier to case would be this: each line in Table 5.5 would represent a different individual case.</p>

Q.	SPCB-Correlation Development RAIs.																
	Note: The reference to Table 5.9 on page 5-14 has been corrected to Table 5.5 in Revision 1 of EMF-2209(P).																
Q	SPCB Statistical RAIs																
1	<p><i>A general statement: Whenever presenting mean and standard deviation (such as in the bottom sentence of page 1-1) include the sample size and the associated tolerance limit.</i></p>																
	Response: See Tables 3.1, 4.1, 4.2, and 4.3 in Revision 1 of EMF-2209(P).																
2	<p><i>Page 1-2, Section 1.1. Provide statistical tests that compare the behavior (mean, variance) of the 1,876 correlation points to the [ ] validation points. Also, did the [ ] additional validation points differ in behavior from the [ ] validation points?</i></p>																
	<p>Response: Statistical tests can be performed to compare the means and/or the variances between or among these data groups. Due to the large number of data, the statistical tests would be expected to show statistical difference. However, the nature of the difference would be of no practical significance. As can be observed, the standard deviation of the three groups are similar in magnitude and the mean values of the two groups of importance are less than unity.</p>																
	<table border="0"> <thead> <tr> <th>Group</th> <th>Number</th> <th>Mean ECPR</th> <th>Standard Deviation</th> </tr> </thead> <tbody> <tr> <td>Verification</td> <td>1876</td> <td>0.992</td> <td>0.0204</td> </tr> <tr> <td>Validation</td> <td>781</td> <td>0.983</td> <td>0.0241</td> </tr> <tr> <td>Validation2</td> <td>316</td> <td>1.002</td> <td>0.0233</td> </tr> </tbody> </table>	Group	Number	Mean ECPR	Standard Deviation	Verification	1876	0.992	0.0204	Validation	781	0.983	0.0241	Validation2	316	1.002	0.0233
Group	Number	Mean ECPR	Standard Deviation														
Verification	1876	0.992	0.0204														
Validation	781	0.983	0.0241														
Validation2	316	1.002	0.0233														
	<p>Comparison of means:      <math>0.0204/\sqrt{(1876)} = 0.0005</math>             <math>0.0241/\sqrt{(781)} = 0.0009</math>             <math>0.0233/\sqrt{(316)} = 0.0013</math></p>																
	<table border="0"> <tr> <td>Group</td> <td></td> </tr> <tr> <td>Verification</td> <td><math>0.992 \pm 0.0005</math></td> </tr> <tr> <td>Validation</td> <td><math>0.983 \pm 0.0009</math></td> </tr> <tr> <td>Validation2</td> <td><math>1.002 \pm 0.0013</math></td> </tr> </table>	Group		Verification	$0.992 \pm 0.0005$	Validation	$0.983 \pm 0.0009$	Validation2	$1.002 \pm 0.0013$								
Group																	
Verification	$0.992 \pm 0.0005$																
Validation	$0.983 \pm 0.0009$																
Validation2	$1.002 \pm 0.0013$																
	<p>This implies a slight statistical difference in the means of the groups, i.e., 0.01 difference between the Verification and the Validation group. However, there is not a practical difference.</p>																
	<p>Comparison of variances:</p>																
	<p>For example consider an F test between Verification and Validation</p>																
	<p>Then</p>																
	$F = (0.0241/0.0204)^2 = 1.396$																
	<p>Comparing this with an estimate from a table would suggest a slight statistical difference. However there is not a practical difference in as much as the overall statistics that are used in the analysis process specifically includes the effect of the differences within test and between tests.</p>																

Q.	SPCB-Correlation Development RAIs.
3	<p data-bbox="269 369 1356 464"><i>Page 1-2, Second paragraph. Text states that transient tests are performed on ATRIUM 10 (not mentioning ATRIUM 9-B). Page 3-36, second paragraph says that dryout tests were performed on the ATRIUM-9B and ATRIUM-10.</i></p> <p data-bbox="269 506 1356 730">Response: Distinction is made between transient test and quasi-steady state dryout tests. Quasi-steady state dryout tests maintain parameters such as flow, inlet subcooling, and pressure constant while slowly increasing power to attain a dryout condition. Transient tests are purposely varying power and flow in accord with a desired plan and detecting dryout. The ATRIUM-10 was selected for performing transient tests while both ATRIUM-9B and ATRIUM-10 assemblies are tested for steady state dryout.</p>
4	<p data-bbox="269 741 1356 810"><i>Pages 2-4 thru 2-10 (Tables 2.1 thru 2.7). Is "G" in these tables the same as G bar (<math>\bar{G}</math>) given in, say, equation (2.7)?</i></p> <p data-bbox="269 858 1356 894">Response: Yes. Revision 1 to EMF-2209(P) has (<math>\bar{G}</math>) in these tables.</p>
5	<p data-bbox="269 905 1356 932"><i>Page 2-6, Equation 2.12. The coefficient "f1" in equation 2.12 is not defined.</i></p> <p data-bbox="269 968 1356 1058">Response: The definition of f1 shown on page 2-6 in the first line following Equation 2-11 is intended to be the same f1 as used in Equation 2-12. An explanatory note could be added to the text to indicate this common usage for the approved version.</p>
6	<p data-bbox="269 1068 1356 1138"><i>Page 2-13, Equation 2.23. Identify/explain how the coefficients 0.624 and 0.314 were obtained.</i></p> <p data-bbox="269 1205 1356 1360">Response: Equation 2.23 remains identical to the formulation for FEFF presented in EMF-1997, "ANFB-10 Critical Power Correlation" and ANF-1125, Supplement 1, "ANFB Critical Power Correlation." The empirical formulation developed for the 1990 critical power correlation is used so as to minimize the review of some different or new formulation.</p>
7	<p data-bbox="269 1371 1356 1398"><i>Page 2-26, Figure 2.3. Symbols (like X, square, and diamond) need a legend.</i></p> <p data-bbox="269 1440 1356 1472">Response: See response to question 7 of SPCB-Correlation Development RAIs.</p>
8	<p data-bbox="269 1482 1356 1551"><i>Page 3-1, Table 3.1. Show how Sigma (given as 0.021) for ATRIUM-9B was derived.</i></p> <p data-bbox="269 1577 1356 1698">Response: The value of 0.021 for the ATRIUM-9B is obtained by applying the standard relationship for standard deviation such as found as equation 1.7 on page 9 or equation 1.9 on page 10 of NUREG/CR-4604 for the 1629 data of ATRIUM-9B tabulated in Section 5 of the topical report.</p>

Q.	SPCB-Correlation Development RAIs.
9	<p><i>Page 3-1, Table 3.1:</i></p> <ul style="list-style-type: none"> <li>• <i>Expand the table to include for each test the 95/95 upper tolerance limit for ECPR,</i></li> <li>• <i>maximum value obtained for the test,</i></li> <li>• <i>number of data points in the test that exceed the tolerance limit, and</i></li> <li>• <i>the percent number of points below the tolerance limit.</i></li> </ul> <p><i>Provide similar entries, separately for each fuel, and for all tests of the same profile.</i></p> <p>Suggestion: Follow the style of Table 4.1 for ANFB 10 that you provided in your May 26, 1998 communication to E.Y. Wang.</p> <p>Response: Tables 3.1, 4.1, 4.2, and 4.3 have been revised in Revision 1 of EMF-2209(P).</p>
10	<p><i>Page 3-2, Table 3.2. Show a complete table of the analysis of variance (ANOVA). Please provide separate analysis for each fuel type. Did any analysis detect significant test differences? Was the data tested for homogeneity of variances prior to constructing the ANOVA?</i></p> <p>Response: From a practical viewpoint, data from one test to another are expected to behave statistically differently. Thus, it is appropriate to combine those differences. The actual way these differences are combined for Safety Analysis is through uncertainty in Additive Constants. This Analysis of Variance is provided as supplementary information.</p> <p>The Analysis of Variance process used by SPC produces the information shown below. The input is the Table 3.1 entries by test section for the number of data points, the mean ECPR, and the standard deviation. The output is the information shown in Table 3.2. This table shows the result of the Analysis of Variance. This approach is used to separate the components of variance in order to estimate the population variance and the degrees of freedom. The variance model is:</p> $\sigma_{\text{TRUE}}^2 = \sigma_{\text{Between}}^2 - \frac{\sigma_{\text{WithinTest}}^2}{\text{ESS}}$ <p>where</p> <ul style="list-style-type: none"> <li><math>\sigma_{\text{TRUE}}^2</math> = True between mean variance</li> <li><math>\sigma_{\text{BETWEEN}}^2</math> = Total between test variance</li> <li><math>\sigma_{\text{WithinTest}}^2</math> = Within test variance</li> <li>ESS = Effective sample size</li> </ul> <p>The degrees of freedom are computed using Satterthwaite's formula.</p>

Q.	SPCB-Correlation Development RAIs.																																
10	<p>It is apparent from the data in Table 3.1 that the measurement variances are not from the same statistical populations. Most of the differences can be explained by differences in test equipment and particular test. Inclusion of all valid test data provides a more robust and conservative basis for the analysis that establishes the Safety Limit.</p> <p>As the purpose of the Analysis of Variance is to account for the differences of the data, tests for homogeneity of variances was not performed. The summary of information by design and overall is shown below:</p> <table border="1" data-bbox="277 688 1356 989"> <thead> <tr> <th>Parameter</th> <th>All Data</th> <th>ATRIUM-10</th> <th>ATRIUM-9B</th> </tr> </thead> <tbody> <tr> <td>Within Test Variance</td> <td>0.000349876</td> <td>0.000421355</td> <td>0.000305357</td> </tr> <tr> <td>Between Test Variance</td> <td>0.000133328</td> <td>0.0000742898</td> <td>0.0001763</td> </tr> <tr> <td>Weighted Mean ECPR</td> <td>0.99104</td> <td>0.99495</td> <td>0.98315</td> </tr> <tr> <td>Standard Deviation</td> <td>0.021982</td> <td>0.02226</td> <td>0.021946</td> </tr> <tr> <td>Population Variance</td> <td>0.0004832</td> <td>0.0004956</td> <td>0.0004817</td> </tr> <tr> <td>Equivalent Sample Size</td> <td>51</td> <td>44</td> <td>80</td> </tr> <tr> <td>Degrees of Freedom</td> <td>302</td> <td>439</td> <td>56</td> </tr> </tbody> </table> <p>(Satterthwaite)</p> <p>The correction to Table 3.2 is provided in Revision 1 of EMF-2209(P).</p>	Parameter	All Data	ATRIUM-10	ATRIUM-9B	Within Test Variance	0.000349876	0.000421355	0.000305357	Between Test Variance	0.000133328	0.0000742898	0.0001763	Weighted Mean ECPR	0.99104	0.99495	0.98315	Standard Deviation	0.021982	0.02226	0.021946	Population Variance	0.0004832	0.0004956	0.0004817	Equivalent Sample Size	51	44	80	Degrees of Freedom	302	439	56
Parameter	All Data	ATRIUM-10	ATRIUM-9B																														
Within Test Variance	0.000349876	0.000421355	0.000305357																														
Between Test Variance	0.000133328	0.0000742898	0.0001763																														
Weighted Mean ECPR	0.99104	0.99495	0.98315																														
Standard Deviation	0.021982	0.02226	0.021946																														
Population Variance	0.0004832	0.0004956	0.0004817																														
Equivalent Sample Size	51	44	80																														
Degrees of Freedom	302	439	56																														
11	<p><i>Pages 3-2 and 3-3, equations for <math>m_2</math>, <math>m_3</math>, <math>m_4</math>, <math>\beta_1</math>, and <math>\beta_2</math>. What is the numeric value of "n" in each of these calculations?</i></p> <p>Response: All values of ECPR were used in the analysis. Thus, n=2657 Note: the following corrected values are shown below and will be included in the approved version of the document.</p> <p><math>m_2 = 0.000491</math>  <math>\sqrt{\beta_1} = 0.004488</math>      <math>\beta_2 = 2.83</math></p>																																
12	<p><i>Page 3-5, Paragraph 3.2. Justify the use of 1 percent as a level of significance for testing normality. What is the numerical value of Lillifor's statistic?</i></p> <p>Response: The numerical value of the Lillifor statistic is 0.0173. This compares with the 0.02 critical value for 1 percent and the 0.017 value for the 5 percent.</p> <p>The requirement for the data going into the Safety Analysis is that it be approximately normally distributed. The 1 percent level of significance assures this. A tighter level of significance such as 5 or 10 percent could be harmful in that valid data could be excluded.</p>																																
13	<p>Pages 3-4 thru 3-29, Figures 3.1 through 3.36. Indicate the sample size associated with each figure.</p> <p>Response: The figures were modified in Revision 1 of EMF-2209(P) as requested.</p>																																

Q.	SPCB-Correlation Development RAIs.
14	<p data-bbox="277 344 1323 411"><i>Page 3-38, last paragraph. Indicate where the upper 95 percent confidence limits on the additive constants are implemented.</i></p> <p data-bbox="277 443 1348 674">Response: The implementation of the upper 95 percent confidence limit for an incremental additive constant uncertainty is presented in Table 3.15. The values of additive constant uncertainty and the corresponding incremental value for rods peaked more than 1.5 are implemented in the determination of the safety limit MCPR in the methodology contained in ANF-524 (P)(A), Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," November 1990.</p>
15	<p data-bbox="277 680 1339 711"><i>Page 4-2, Tables 4.1, 4.2 and 4.3. Expand the tables as requested for Table 3.1.</i></p> <p data-bbox="277 743 1314 810">Response: Tables 3.1, 4.1, 4.2, and 4.3 have been revised in Revision 1 of EMF-2209(P).</p>
16	<p data-bbox="277 816 1344 911"><i>Page 4-8, Figure 4-6 (and others): What do the different lines represent? Provide the necessary labels. Similarly, provide the necessary labels for Figures 5.7 thru 5.89.</i></p> <p data-bbox="277 947 1331 1045">Response: The lines represent the SPCB correlation prediction of the data as a function of inlet subcooling. The separation among the lines is due to different flow families for the data.</p>
17	<p data-bbox="277 1052 1281 1083"><i>Were there any outliers in the evaluation, and if so, what was their disposition?</i></p> <p data-bbox="277 1119 1323 1220">Response: In EMF-1997, the comment is made that runs 125.4 and 132.4 for test 29.2 are experimentally determined to be outliers. Several low flow points were excluded in EMF-1997 as being below the flow limit for applicability.</p> <p data-bbox="277 1255 1356 1381">Due to the refitting process of the "A" and "B" coefficients in EMF-2209, these points are included in the evaluation process. That is, the low flow points are in the domain of the correlation and retained and the two experimental outliers were retained. The inclusion of the outliers provides conservatively greater value of uncertainty.</p>









# SIEMENS

April 20, 2000  
NRC:00:023

Document Control Desk  
ATTN: Chief, Planning, Program and Management Support Branch  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555-0001

## SER Conditions for EMF-2209(P) Revision 1, "SPCB Critical Power Correlation"

The NRC has proposed that three conditions be included in the SER for EMF-2209(P), a topical report describing the SPCB critical power correlation. These conditions are:

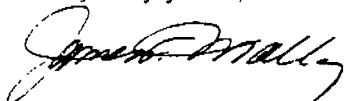
1. The SPCB correlation is applicable to Siemens Power Corporation ATRIUM<sup>1</sup>-9B and ATRIUM-10 fuel designs with a local peaking factor no greater than 1.5.
2. If, in the process of calculating the MCPR safety limit, the local peaking factor exceeds 1.5, an additional uncertainty of 0.026 for ATRIUM-9B and 0.021 for ATRIUM-10 will be imposed on a rod by rod basis.
3. The range of applicability of the SPCB correlation shall be limited to:

Pressure (psia)	571.4 to 1432.2
Inlet mass velocity (Mlb/hr-ft <sup>2</sup> )	0.087 to 1.5
Inlet subcooling (Btu/lbm)	5.55 to 148.67

For clarification, it is noted that the values cited in the second condition are taken from Table 3.15 on page 3-39 of the topical report under "Incremental Uncertainty." The values in condition 3 are found in Table 1.1 on page 1-2. The peaking values in this table are addressed in condition 1.

Siemens Power Corporation has discussed these conditions with the NRC and agrees they are acceptable and appropriate.

Very truly yours,

  
James F. Mallay, Director  
Regulatory Affairs

/arn

cc: A. C. Attard  
N. Kalyanam  
Project No. 702

---

<sup>1</sup> ATRIUM is a trademark of Siemens.

## Siemens Power Corporation

2101 Horn Rapids Road  
Richland, WA 99352

Tel: (509) 375-8100  
Fax: (509) 375-8402



April 24, 2000  
NRC:00:024

Document Control Desk  
ATTN: Chief, Planning, Program and Management Support Branch  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555-0001

**SER Condition for EMF-2209(P) Revision 1, "SPCB Critical Power Correlation"**

Ref.: 1. Generic Letter 83-11, Supplement 1, "Guidelines for Qualifying Licensees to Use Generically Approved Analysis Methods," June 24, 1999.

The NRC has proposed that a condition be included in the SER for EMF-2209(P), a topical report describing the SPCB critical power correlation. This condition addresses the transfer of the technology needed by a licensee to successfully execute methodology developed by Siemens Power Corporation (SPC) and approved by the NRC. To address this condition, SPC has developed Work Practice P104,135, "Guidelines to Qualify a Licensee to Use NRC Approved Analysis Methods." This work practice, as one of the work practices within EMF-1928(P), "Engineering Work Practices," addresses the processes SPC will follow to assist a licensee in meeting the guidelines of Generic Letter 83-11, Supplement 1 (Reference 1). This work practice provides a procedure to help ensure that licensees are adequately trained and are able to comply with the guidelines of Reference 1.

SPC has discussed this condition with the NRC and agrees it is acceptable and appropriate. In addition, SPC believes this work practice fully addresses the intent of the condition.

Very truly yours,

A handwritten signature in black ink, appearing to read "James F. Mallay", is written over a solid black circle.

James F. Mallay, Director  
Regulatory Affairs

/arn

cc: A. C. Attard  
N. Kalyanam  
Project No. 702

**Siemens Power Corporation**

2101 Horn Rapids Road  
Richland, WA 99352

Tel: (509) 375-8100  
Fax: (509) 375-8402

**FRAMATOME ANP, Inc.**

September 4, 2002  
NRC:02:042

Document Control Desk  
ATTN: Chief, Planning, Program and Management Support Branch  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555-0001

**Errata Sheets for EMF-2209(P)(A) Revision 1**

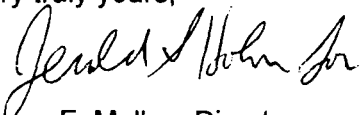
The topical report EMF-2209(P)(A) Revision 1 was reviewed and approved by the NRC in July 2000. The report contains two typographical errors. First, the equation defining the function C on page 2-6, Equation 2.12, is missing three terms. The three terms are  $C16+C17*\bar{G}+C18*(\bar{G})^2$ . Second, "Table 2.5 ATRIUM-10 Coefficients for C" is missing the values for the constants in the three terms. The attached two pages contain the corrected information, with changes marked by vertical bars on the right side.

The typographical errors exist only in the document and do not affect any of the results presented in the topical report and are not present in the computer codes that implement the function.

This letter is provided for information only and no response is required.

Framatome ANP considers some of the information contained in the attachment to be proprietary. An affidavit is enclosed to satisfy the requirements of 10 CFR 2.790(b) to support withholding of this information from public disclosure.

Very truly yours,



James F. Mallay, Director  
Regulatory Affairs

Attachment  
Enclosure

cc: R. Caruso  
D. G. Holland  
J. S. Wermiel  
Project 693

SPCB Critical Power Correlation

---



## 2.2 *Non-Uniform Axial Heat Flux Factor*

The non-uniform axial power corrector (Tong et al., Reference 2.3) used in the ANFB critical power correlation (Reference 2.2) provides the basis of the non-uniform axial correction factor for SPCB. The non-uniform axial correction factor characteristic modified in SPCB is the empirical factor, " $\Omega$ " (Reference 2.4). In addition, a post-multiplier to the non-uniform axial factor is included to provide an adjustment to better address the impact of non-uniform axial shapes. This adjustment factor is appropriate for the steady-state and transient evaluation processes. [

]



**FRAMATOME ANP**

An AREVA and Siemens company

**FRAMATOME ANP, Inc.**

June 20, 2003  
NRC:03:039

Document Control Desk  
ATTN: Chief, Planning, Program and Management Support Branch  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555-0001

**Request for Review of a Revision to EMF-2209(P)(A) Revision 1**

Ref.: 1. EMF-2209(P)(A) Revision 1, *SPCB Critical Power Correlation*, Siemens Power Corporation, July 2000.

Framatome ANP requests the NRC's review and approval for referencing in licensing actions the attached revisions to the SPCB CHF correlation (see Reference 1). This revision describes a reduction in the conservatism included in the SPCB correlation for designs where a uranium blanket is used at the top of the fuel. We request that the NRC approve this revision by August 31, 2003 to support certain fuel reloads.

Framatome ANP will incorporate these changes into EMF-2209(P)(A) Revision 2 following NRC acceptance.

Framatome ANP considers some of the information contained in the enclosed revision to be proprietary. As required by 10 CFR 2.790(b), an affidavit is enclosed to support the withholding of the information from public disclosure. Five copies of the proprietary and non-proprietary versions of the attachment are enclosed.

Very truly yours,

James F. Mallay, Director  
Regulatory Affairs

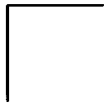
**Attachments**

cc:

D. G. Holland (w/attachments)  
J. S. Wermiel  
Project 728

An investigation of the SPCB critical power correlation has determined that the correlation is overly conservative for nuclear designs with top natural uranium blankets. This conservatism arises from the fact that the test data used for deriving the correlation do not model the effect of the natural uranium at the top of the fuel rods. The top natural blanket significantly lowers the reactivity at the end of the rod and, consequently, the heat flux in the reactor is often nearly an order of magnitude lower at the end of the rod than the heat flux in the test assemblies. The non-uniform axial correction factor derived from the experimental data produces an overly conservative estimate of the local critical heat flux for the very low heat flux in the natural uranium top portion of the fuel rod.

To illustrate the impact of reducing the heat flux in the top node, two axial power distributions are presented in Figure 1. The test axial corresponds to the downskew axial shape used in the CHF tests. A second axial (blanket effect) is constructed from the test axial by reducing the power in the top node (location 0.98) and at the end of the heated length to simulate the natural blanket.



**Figure 1 Test Axial and Modification**



Analyzing these two axial power distributions with SPCB shows that reducing the power in the blanket region results in about a 0.04 reduction in the critical power ratio. The plane indicating boiling transition shifts from 8 feet to the end of the heated length, and the corresponding critical heat flux at the end of the heated length decreases from about 0.13 to 0.0003 MBtu/hr-ft<sup>2</sup>. In reality, the critical heat flux should not decrease at all for this situation; it should remain at approximately 0.13 MBtu/hr-ft<sup>2</sup>.

The investigation of this issue revealed that the Tong Factor, as defined by equation 2.14 in Reference 1, takes on values in the natural blankets that are often more than 100 times as great as the largest values observed for the axial shapes used in the correlation database. The excessively large Tong Factor results in a calculated critical power for the top node which is overly conservative.

The reduced axial power peaking in the top blanket also introduces a step change in the omega function as defined by equation 2.15 in Reference 1. This is because the local computed values may drop significantly below the minimum value imposed on the function. The value of omega has an inverse relation to memory length. The effective memory length may be increased by 3 or more feet for a 6 inch change in the evaluation elevation. The value of the memory length should not increase by more than the 6 inch difference in the evaluation locations. The basis for selecting a minimum value of the omega function was to assure that no memory length should be more than about 10 feet. An improved definition for the omega function may be derived by observing values of omega that bound the correlation database but allow variation of the minimum due to the mass velocity as illustrated in Figure 2.

The proposed modifications to eliminate the excessive conservatism in the SPCB calculation of the critical heat flux in the top natural uranium blanket are:

- c) Limit the maximum value of Tong Factor as given by equation 2.14 in Reference 1 to no more than a prescribed value.
- d) Limit the minimum value of the omega function as given by equation 2.15 to be no less than a mass velocity dependent function.

These modifications would be implemented in the documentation by modifying the footnote on page 2-8 of Reference 1 with respect to the Omega Function and adding a footnote on page 2-8 with respect to the base value of Tong Factor. The amended footnote for the omega function would state:

*Ω is taken as the [*

*]*

The added footnote for the maximum value of the Tong Factor would state:

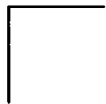
*The maximum value of  $F_{Base}$  is [      ]*

These modified footnotes have no effect on the statistics of the SPCB database evaluation of Reference 1.

A Revision 2 to the document EMF-2209 will be issued following the NRC issuance of a safety evaluation for the proposed change.

References:

2. EMF-2209 (P)(A) Revision 1, *SPCB Critical Power Correlation*, Siemens Power Corporation, July 2000.



**Figure 2 Omega Function and Bound for SPCB Database**



An AREVA and Siemens Company

FRAMATOME ANP, Inc.

July 25, 2003  
NRC:03:048

Document Control Desk  
ATTN: Chief, Planning, Program and Management Support Branch  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555-0001

**Response to Request for Additional Information with Respect to a Revision to EMF-2209(P)(A) Revision 1**

Ref.: 1. Letter, J. F. Mallay (Framatome ANP) to Document Control Desk (NRC), "Request for Review of a Revision to EMF-2209(P)(A) Revision 1," NRC:03:039, June 20, 2003.

The NRC requested by fax additional information to facilitate the completion of its review of a revision to EMF-2209(P)(A) Revision 1 submitted in Reference 1. The response to this request is contained in the attachments to this letter. Proprietary and non-proprietary attachments are provided.

Framatome ANP, Inc. considers some of the information contained in Attachment 1 to this letter to be proprietary. This information has been noted by enclosing it within brackets. The affidavit provided with the original submittal of the reference topical report satisfies the requirements of 10 CFR 2.790(b) to support the withholding of this information from public disclosure.

Very truly yours,

James F. Mallay, Director  
Regulatory Affairs

Enclosures

cc: F. M. Akstulewicz  
A. C. Attard (w/enclosures)  
D. G. Holland (w/enclosures)  
Project 728

**Request for Additional Information on  
SPCB-Mod (MB9719)**

Question 1

*The request to modify the approved SPCB correlation specified in the June 20 submittal is a significant one. In order to expedite the review process, and to obtain a clear understanding of this request, please provide a detailed response to the following:*

- (a) *Provide a step by step development of the current SPCB correlation alongside a step by step development of the modified SPCB correlation.*
- (b) *Indicated in each step of the development of the modified SPCB correlation, the significance of the modification and its impact on the overall correlation, particularly in the six inch reflector region.*

Response 1

The basis of the SPCB CHF correlation has been presented, discussed, and documented in EMF-2209(P)(A) Revision 1, and in a presentation to the NRC staff on July 22, 2003.

The revision to the SPCB Tong factor allowed range was deduced by investigation into the observed behavior of the critical power calculation when a natural uranium node was present. This investigation examined the various terms in the SPCB correlation. The investigation identified equations 2.14 and 2.15 on page 2-8 of the SPCB critical power correlation (EMF-2209(P)(A) Revision 1) as the cause of the observed behavior.

Specifically, the Tong factor was observed to take on values [ ] greater than the values observed in the SPCB database for the exit plane. This examination showed that the maximum value of Tong factor observed in the SPCB database occurred at the exit plane of the test assemblies.  
[ ]

These observations were used to select a maximum value for the Tong factor that would remove the exit plane from being a contributor to the calculation, when natural uranium is present in the exit plane, and provide assurance that no other plane would be influenced by this choice.

The impact on the overall correlation is that a [ ] could be selected and no change would occur for the statistical information that is reported in EMF-2209(P)(A) Revision 1. Calculated values of Tong Factor for assemblies with natural uranium present in the last six inches result in calculated values of Tong factor that are [ ]. Limiting this value to [ ] removes the exit plane from being the portion of the fuel assembly that is setting the limit.

As discussed in the response to Question 6, the minimum value for the upper portion of the assembly should be selected as a function of mass velocity rather than a constant bounding value. Observations of the behavior of  $\Omega$  for the SPCB database provide support.

Question 2

*For SPCB Critical Power Correlation, Revision 0, page 2-8, equation 2.14,  $\Omega$  should increase because the critical length will increase and this will increase the Tong Factor which will decrease critical power. Clarify if this is why critical power is reduced when including the additional six inches.*

Response 2

The increase in the Tong Factor occurs for the natural uranium region because the local heat flux decreases. Coupled with the decrease in local heat flux, the axial peaking factor for the location also decreases. The decrease in the axial peaking factor results in  $\Omega$  decreasing. As  $\Omega$  decreases, the factor  $(1 - \exp(-\Omega l_c))$  decreases. The decrease in the denominator of F-Base is the reason that Tong factor increases for the natural uranium region.

$$F_{\text{Base}} = \frac{\Omega}{q''_{l_c} (1 - e^{-\Omega l_c})} \int_0^{l_c} \left[ e^{-\Omega(l_c - Z)} \right] q''(Z) dZ$$

Question 3

*For SPCB Critical Power Correlation, Revision 0, page 2-12, Clarify  $\Omega$  is dependent on FEFF, and Tong Factor is dependent on  $\Omega$ , and Tong Factor is inversely proportional to the critical heat flux, i.e., critical power is inversely proportional to FEFF.*

Response 3

$\Omega$  is defined by equation 2.15 on page 2-8 of EMF-2209 (P)(A) Revision 1 and shows a dependency on [

]

[

]

Question 4

*In reference to page A-2, first paragraph, second sentence indicates that the plane indicating boiling transition shifts from 8 to 12 feet to the end of the heated length. Please provide a detailed physical description of the "heated length".*

Response 4

The axial extent of interest in the analysis occurs for the length of the fuel rods that transmit energy to the fluid by either direct deposition or conduction heat transfer. This length is the fueled length, including natural blankets, and is referred to as the heated length.

Question 5

*In reference to page A-2, third paragraph it is stated that the value of omega has an inverse relation to memory length. Please provide a definition to the term "memory length".*

Response 5

Lahey and Moody provide a discussion on the memory effect (page 135, *Thermal-Hydraulics of Boiling Water Reactors, Second Edition*) describing how the exponential term dies out strongly for large values of omega – hence, providing more importance to the local heat flux than the upstream memory effect. Furthermore, as quality increases, the values of omega become smaller and the importance of upstream memory becomes more important.

The term "memory length" is employed as a mnemonic device to indicate that the inverse of the omega parameter (which would have length associated with it) represents the relative importance of the upstream fluid conditions effect. A larger value of memory length implies more importance of upstream fluid conditions.

Question 6

*It is not clear to the staff what the 3<sup>rd</sup>. paragraph on page A-2 of the June 20 submittal is trying to communicate, and there are too many concerns to list in one question. Please be prepared to provide a much needed clearer depiction of what it is being conveyed by this paragraph.*

Response 6

This paragraph provides the information to support the request for changing the minimum value for omega from a constant value to a function of mass flow. This change does not affect the comparison of the SPCB correlation to the measured CHF database as reported in EMF-2209(P)(A) Revision 1. Figure 2.0 shows the set of values of omega as calculated for the SPCB database at the predicted location of dryout and superimposes the proposed minimum value function.

An examination of calculated values of the omega function might help to clarify the third paragraph. This parameter attempts to capture the influence of a memory effect – how the behavior upstream influences the local behavior. The evaluation process typically performs evaluations on a six inch increment. If the memory effect is characterized by some length at one location, then the memory effect at the next location should not be characterized by a memory length that is more than six inches longer.

The SPCB correlation originally accounted only for the observation that the omega function should be no less than [ ] on an absolute basis for the entire database. The insertion of natural uranium in the last 6 to 12 inches of the assembly leads to step changes in the omega function that mathematically suggest an increasingly longer memory effect for the natural uranium nodes – well beyond the added six inches per node when compared to the upstream values.



Question 7

*In reference to page A-2, fourth paragraph where the proposed modifications to eliminate excessive conservatism are noted, please provide background theory leading to choices a and b as proposed modifications.*

Response 7

J.G. Collier, *Convective Boiling and Condensation*, provides some guidance pertaining to cold patch tests. Cold patches near the inlet of a tube resulted in power for the critical condition being close to that obtained for the same heated length with uniform heating. A cold patch toward the exit of the tube showed a sharp increase in power that could be greater than that obtained for the same total length with uniform heating. Collier also reports that the range of the Tong factor is typically between 1 and 3.

This guidance suggests that the observed behavior of SPCB, as heat flux decreases significantly at the top end of the heated length (due to presence of natural uranium), is overly conservative as it allows the limiting value to occur at the end of the heated length, a behavior that is counter to observation.

The cause within context of the SPCB correlation has been traced to the Tong Factor. This led to the evaluation that was performed and the proposed limitation of the value of the Tong Factor as a mechanism to reduce the overly conservative result.

Lahey and Moody, *The Thermal Hydraulics of a Boiling Water Reactor, Second Edition*, provides guidance pertaining to  $\omega$  – specifically that  $\omega$  decreases in value as quality increases.

For the application of SPCB, the relation shown for  $\omega$  versus flow is representative of the trend with quality suggested by Lahey.

The investigation of the Tong Factor showed the overly conservative behavior of the  $\omega$  function and led to the suggestion that the minimum value of  $\omega$  be modified.

#### Question 8

*In reference to page A-2 where the modification for  $\Omega$  is shown at the bottom of the page, please provide further qualitative and quantitative background for the new equation for  $\Omega_{min}$ . That is, show how the equation was developed. Also provide quantitative data showing why the maximum value of  $F_{base}$  should be 100.*

#### Response 8

The equation was developed by placing a bounding line below the family of  $\omega$  values that were observed to occur for the SPCB database. Figure 2 shows the family of  $\omega$  values at the SPCB prediction of dryout for the database as a function of mass velocity.

The limiting value of  $F_{base}$  was chosen to be [                      ]. An evaluation of actual power distributions observed in reactor designs indicated that as the power decreased in the top 6-inch segment of the fuel, the critical heat flux decreased significantly. The limitation of [                      ] provided a conservative value for the critical heat flux without producing unreasonably low values. Over the 6-inch segment, the  $F_{base}$  value was changing from numbers of approximately [                      ]

#### Question 9

*In reference to Page A-3, Figure 2, please provide a detailed explanation as to what exactly this plot represents.*

#### Response 9

The values of  $\omega$  are the [                      ] values of  $\omega$  for the SPCB database at the predicted locations of dryout as a function of mass velocity. The bounding line is shown as a function of mass velocity.



Question 10

*How will this modification to the SPCB correlation impact the power production in the rod(s) and the safety margin.*

Response 10

The use of depleted uranium has been implemented for some designs as a means to counteract the conservatism of the SPCB correlation. The changes proposed to the SPCB correlation ranges will allow fuel designs to be constructed using natural uranium. There will be no significant change in safety margin or power production in the rods between these two designs.

A comparison between the use of SPCB with and without the proposed changes shows a difference of about [     ].

Question 11

*On the neutron front, one would expect that accounting for power production in the reflector would impact neutron leakage in the region. One would also expect an increase plutonium production in the same region. Please provide a detailed technical description on a neutronic base, of the impact of operating the fuel in the reflector region.*

Response 11

Natural  $\text{UO}_2$  blankets are added to BWR fuel to improve fuel cycle economics. Indeed, for the same energy output, the replacement of the top and bottom 6-inch segments of the active fuel with natural uranium reduces the overall enrichment of the bundle by about 0.07 w/o U-235 enrichment. The trade-off is a reduction in LHGR margin of the bundle and reduced bundle cold shutdown capability. In theory, moving more U-235 toward the center of the core improves U-235 utilization by reducing neutron leakage out of both the top and bottom of the high enriched fuel. The top blanket is worth more than the bottom blanket in enrichment savings. However, relatively little power is produced in the natural  $\text{UO}_2$  blankets themselves.

Natural blankets have always been a part of Framatome ANP BWR designs. However, the details of the blanket power distributions were not fully considered in developing SPCB CHF correlation. Unlike enriched sections of the fuel assembly, where the radial enrichment distribution is tailored to minimize radial peaking factors, the natural blankets utilize a uniform radial enrichment (0.711). The result is that the local peaking depends primarily on the moderator distribution.

For a D-lattice plant with asymmetric water gaps between assemblies, the relative local peaking and consequently the maximum lattice F-effective can be quite high despite the low planar powers. The high F-effectives for the D-lattice natural blanket, coupled with the unbounded Tong Factor result in the assembly being limited by the natural blanket region.

For C-lattice plants the results show similar trends but are not as exaggerated. C-lattice plants exhibit the same unbounded Tong behavior. The local pin-peaking of the C-lattice for the natural lattice is not as great as that of the D-lattice, therefore the F-effective is not as limiting for the C-lattice as for the D-lattice.



May 1, 2008  
NRC:08:028

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

**Request for Review and Approval of EMF-2209(P), Revision 2, Addendum 1, "SPCB Additive Constants for ATRIUM-10 Fuel,"**

AREVA NP Inc. (AREVA NP) requests the NRC's review and approval of the enclosure, EMF-2209(P), Revision 2, Addendum 1, "SPCB Additive Constants for ATRIUM-10 Fuel." This document presents revised SPCB critical power correlation additive constants for ATRIUM-10 fuel for boiling water reactors.

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.

A proprietary and non-proprietary version of the report is enclosed.

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.

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](mailto:gayle.elliott@areva.com).

Sincerely,

A handwritten signature in cursive script that reads "Ronnie L. Gardner" with the word "for" written below it.

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

Enclosures

cc: H.D. Cruz  
Project 728

**AREVA NP INC.**  
An AREVA and Siemens company

3315 Old Forest Road, P.O. Box 10935, Lynchburg, VA 24506-0935  
Tel.: 434 832 3000 - Fax: 434 832 3840 - [www.areva.com](http://www.areva.com)



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:

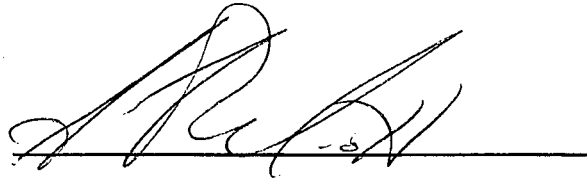
- (a) The information reveals details of AREVA NP's research and development plans and programs or their results.
- (b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or 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.

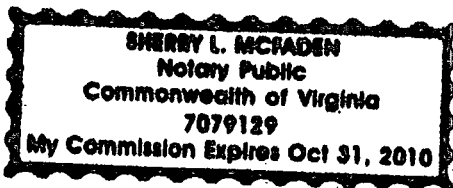
9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

A handwritten signature in black ink, appearing to be "S. L. McFaden", written over a horizontal line.

SUBSCRIBED before me this 1<sup>st</sup>  
day of May, 2008.

A handwritten signature in black ink, appearing to be "Sherry L. McFaden", written over a horizontal line.

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



An AREVA and Siemens company

EMF-2209(NP)  
Revision 2  
Addendum 1  
Revision 0  
SPCB Additive Constants for  
ATRIUM-10 Fuel

April 2008



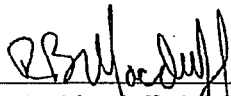
AREVA NP Inc.


ISSUED IN ON-LINE  
DOCUMENT SYSTEM


DATE 4/30/08

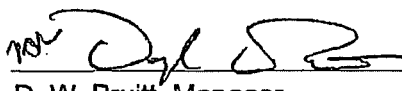
EMF 2209(NP)  
Revision 2  
Addendum 1  
Revision 0

**SPCB Additive Constants for ATRIUM-10 Fuel**

Prepared:  23 Apr 08  
R. B. Macduff, Advisory Engineer  
Thermal Hydraulics Methods  
Date

Reviewed:  4/23/08  
N. A. Carstens, Engineer  
Systems Analysis Methods  
Date

Approved:  for DEG 23 Apr 08  
D. E. Garber, Manager  
Neutronics Richland  
Date

Approved:  29 Apr 08  
D. W. Pruitt, Manager  
Thermal-Hydraulics Richland  
Date

Approved:  30 Apr. 08  
R. L. Gardner, Manager  
Site Operations and Regulatory Affairs  
Date

AREVA NP Inc.

EMF 2209(NP)  
Revision 2  
Addendum 1  
Revision 0

Copyright © 2008

AREVA NP Inc.  
All Rights Reserved



### Nature of Changes

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

## Contents

1.0	Introduction and Summary .....	1-1
2.0	Revised Additive Constants .....	2-1
3.0	Test Data Modifications .....	3-1
3.1	Lattice Peaking Distribution .....	3-1
3.2	Bundle Power .....	3-2
3.3	Axial Power Distribution .....	3-2
4.0	Impact on Additive Constants .....	4-1
5.0	SPCB Conservatism .....	5-1
6.0	Evaluation of Transient Critical Power Data .....	6-1
7.0	References .....	7-1

## Tables

Table 4.1	Additional Additive Constant Uncertainty for High Local Peakings .....	4-1
Table 5.1	FEFF Values for ECPR Evaluation of ATRIUM-10 .....	5-1
Table 6.1	XCOBRA-T Results Using Nominal F-eff .....	6-2

## Figures

Figure 2.1	ATRIUM-10 Additive Constants for SPCB .....	2-1
Figure 2.2	Comparison of Predicted and Measured Critical Power Data with SPCB for ATRIUM-10 Data .....	2-2
Figure 3.1	Peaking Pattern STS 17.1 Dryout detections indicated by heavy outline .....	3-3
Figure 3.2	Revised Peaking Pattern STS 17.1 Dryout detections indicated by heavy outline .....	3-4
Figure 2.1	ATRIUM-10 Additive Constants for SPCB .....	2-1
Figure 2.2	Comparison of Predicted and Measured Critical Power Data with SPCB for ATRIUM-10 Data .....	2-2

*This document contains a total of 20 pages.*

### Nomenclature

<u>Acronym</u>	<u>Definition</u>
BT	Boiling Transition
BWR	Boiling Water Reactor
ECPR	Experimental Critical Power Ratio (Measured / Calculated Critical Power)
FEFFBT	Boiling Transition value of Bundle FEFF
ICC	Inner Copper Conductor
PLR	Part Length Rod
RAI	Request for Additional Information
SER	Safety Evaluation Report

## 1.0 Introduction and Summary

This document presents revised SPCB 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 distribution and the power contained in the part length fuel rods. Evaluations have confirmed that the SPCB critical power correlation constants do not require revision as a result of the error.

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 SPCB 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 derivation of the revised additive constants for SPCB. Included 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 that are shown in Reference 2.

---

1. ATRIUM is a trademark of AREVA NP Inc.

## 2.0 Revised Additive Constants

The revised additive constants for use with the SPCB critical power correlation for the ATRIUM-10 fuel are shown in Figure 2.1. The comparison between the predicted and measured critical power data with SPCB for the ATRIUM-10 database is shown in Figure 2.2.



Figure 2.1 ATRIUM-10 Additive Constants for SPCB



**Figure 2.2 Comparison of Predicted and Measured Critical Power  
Data with SPCB 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 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, both axial and radial, and the total power generated in a bundle required modification to properly account for the power present 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 is included, the relative power delivered by the part length rods in the lower lattice (fully rodded region below the end of the part length rods) of the bundle increases compared to the previously reported powers. 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. 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 SPCB critical power correlation is impacted because the lower lattice power distribution is used as a basis for 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 f-effective values change. The base f-effective 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 342.1) as [                      ]. The adjusted power for this condition becomes:

$$[ \qquad \qquad \qquad ]$$

### 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. SPCB correlation development used a design axial power shape. This revision developed the 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 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 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 f-effective for each test in the data base. The boiling transition values of f-effective (FEFFBT) 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 resulting additive constant uncertainty remains [ ].

The uncertainty in the additive constants associated with high local peaking was also determined. The same process used in Reference 2 was repeated for the corrected data. [

] The resulting ATRIUM-10 additive constant uncertainty for high local peaking also remains unchanged from Reference 2. This is shown in Table 4.1.

**Table 4.1 Additional Additive Constant Uncertainty for High Local Peakings**

## 5.0 SPCB Conservatism

The revised additive constants result in the set of f-effective values shown in Table 5.1 for the tests comprising the SPCB ATRIUM-10 database. This set of values provides an average [                      ]. Comparing these f-effective values with the FEFFBT f-effective 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 FEFF Values for ECPR Evaluation of ATRIUM-10**



## SPCB Predictions Compared with Measurements

The predicted values of critical power compared to the measured values are presented in Figure 2.2 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 f-effective values. The repeat analysis was performed consistent with Reference 2, assuming that all of the power delivered by the electrically heated rods 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 f-effective values did not impact conclusions in Reference 2. The repeat analysis demonstrated the SPCB 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 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 and the calculated time of boiling transition was conservatively predicted.

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 F-eff**



## 7.0 References

1. EMF-2245(P)(A), Revision 0, Applications of SPC Critical Power Correlations to Co-resident Fuel, August 2000.
2. EMF-2209(P)(A), Revision 2, "SPCB Critical Power Correlation," September 2003.



NRC IC: 08:041



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

October 29, 2008

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: AREVA NP, INC (AREVA)  
TOPICAL REPORT (TR) EMF-2209(P), REVISION 2, ADDENDUM 1, "SPCB  
[SIEMENS POWER CORPORATION B] ADDITIVE CONSTANTS FOR  
ATRIUM-10 FUEL" (TAC NO. MD8754)

Dear Mr. Gardner:

By letter dated May 1, 2008 (Agencywide Documents Access and Management System  
Accession No. ML081260452), AREVA submitted for U.S. Nuclear Regulatory Commission  
(NRC) staff review TR EMF-2209(P), Revision 2, Addendum 1, "SPCB Additive Constants for  
ATRIUM-10 Fuel." Upon review of the information provided, the NRC staff has determined that  
additional information is needed to complete the review. On October 9, 2008, Gayle Elliott,  
AREVA Product Licensing Manager, and I agreed that the NRC staff will receive your response  
to the enclosed Request for Additional Information (RAI) questions by November 17, 2008. If  
you have any questions regarding the enclosed RAI questions, please contact me at  
301-415-1053.

Sincerely,

Holly D. Cruz, Project Manager  
Special Projects Branch  
Division of Policy and Rulemaking  
Office of Nuclear Reactor Regulation

Project No. 728

Enclosure: RAI

REQUEST FOR ADDITIONAL INFORMATION  
BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
TOPICAL REPORT (TR) EMF-2209(P), REVISION 2, ADDENDUM 1  
"SPCB ADDITIVE CONSTANTS FOR ATRIUM-10 FUEL"

AREVA NP, INC.

PROJECT NO. 728

1. Section 3.0 of Chapter 3, on page 3-1 of the May 2008 submittal, provides a brief description of the adjustment needed to the voltage power provided to the part-length rods. Since a large number of tests were conducted for the development and validation of the Siemens Power Corporation B (SPCB) correlation, one would expect an equivalent amount of voltage power applied to the part-length rods. Please provide a qualitative and quantitative technical basis as to how this issue was accounted for in your final determination of the adjusted additive constants, listed on page 2-1 of the May submittal.
2. The paragraph in Section 3.2 on page 3-2, attempts to explain the reason behind the higher bundle power reported in Reference 2/3. Did the error in the original calculation lie in assigning full-length to the inner copper conductor instead of a  $\frac{3}{4}$  length, which is the full heated length of the part-length rod? Please provide additional clarification.

ENCLOSURE



November 4, 2008  
NRC:08:088

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

**Response to an RAI on Topical Report EMF-2209(P), Revision 2, Addendum 1, "SPCB Additive Constants for ATRIUM-10 Fuel"**

Ref. 1: Letter, Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), "Request for Review and Approval of EMF-2209(P), Revision 2, Addendum 1, 'SPCB Additive Constants for ATRIUM-10 Fuel'," NRC:08:028, May 1, 2008.

AREVA NP Inc. (AREVA NP) requested the NRC's review and approval of the topical report EMF-2209(P), Revision 2, Addendum 1, in Reference 1. A request for additional information was provided by the NRC in an email on October 1, 2008. The response to the RAI is provided in Attachment A enclosed with this letter. Proprietary and non-proprietary versions of Attachment A are provided.

AREVA NP considers some of the material contained in the enclosed document to be proprietary. The affidavit provided with the original submittal of this topical report satisfies the requirements of 10 CFR 2.390(b) 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](mailto:gayle.elliott@areva.com).

Sincerely,

A handwritten signature in black ink that reads "Ronnie L. Gardner".

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

Enclosures

cc: H.D. Cruz  
Project 728

**AREVA NP INC.**

An AREVA and Siemens company

3315 Old Forest Road, P.O. Box 10935 Lynchburg, VA 24506-0935

An AREVA and Siemens company

ANP-2763Q1NP  
Revision 0  
**RAIs for SPCB Additive Constants for  
ATRIUM-10 Fuel - EMF-2209(P)  
Revision 2, Addendum 1, Revision 0**

October 2008



**RAIs for SPCB Additive Constants for ATRIUM-10 Fuel - EMF-2209(P)  
Revision 2, Addendum 1, Revision 0**

Prepared by:

M. T. Bunker, Supervisor  
Thermal Hydraulic Methods

Reviewed by:

R. B. Macduff  
Thermal Hydraulic Methods

Approved by:

D. W. Pruitt, Manager  
Thermal-Hydraulic Richland

**Copyright © 2008**

**AREVA NP Inc.**

**All Right Reserved**

**Table of Contents**

<u>Question</u>	<u>Page</u>
Question 1. ....	1
Question 2. ....	5

This document contains a total of 9 pages.

**Question 1.** Section 3.0 of Chapter 3, on page 3-1 of the May 2008 submittal, provides a brief description of the adjustment needed to the voltage power provided to the part-length rods. Since a large number of tests were conducted for the development and validation of the SPCB correlation, one would expect an equivalent amount of voltage power applied to the part-length rods. Please provide a qualitative and quantitative technical basis as to how this issue was accounted for in your final determination of the adjusted additive constants, listed on page 2-1 of the May submittal.

**Response 1:**

For each test in the ATRIUM-10 database that was used in the development of the SPCB correlation, the bundle powers as well as the rod powers and relative rod peaking factors at each axial level were re-computed using the method described below. The new local peaking factors were then used in the calculation of new f-effective values which were then used to determine the new additive constants using exactly the same methodology as was used previously.

The part length rods are constructed from full length rods by cutting off the un-needed portion. This assures that the part length rods and the full length rods will have the same axial power profile, within the length of the part length rods. To achieve the expected radial power across all the heated rods, the voltage drop must be equal across the heated length of all the test rods, meaning the part length rods must have the same overall voltage drop as the full length rods. To achieve the same voltage drop, the cut off portion of the part length rods is replaced by a compensation resistance located outside of the test vessel at the KATHY test facility. The current through the part length rods flows back through the rod and out to the compensation resistance through an insulated inner copper conductor that is [ ] .

The rod powers are computed based on the measured electrical resistances for each rod. For part length rods, the total rod resistance is the sum of the outer conductor resistance,  $R_O$ , the inner copper conductor resistance,  $R_{IC}$ , and the compensation rod resistance,  $R_E$ .

$$R_{T,j} = R_{O,j} + R_{IC,j} + R_{E,j} \quad (1)$$

For a full length rod,

$$R_{IC,j} = 0 \quad (2)$$

$$R_{E,j} = 0 \quad (3)$$



Note that the inner copper conductor resistance,  $R_{IC}$ , is composed of 3 parts. The first part consists of the length of conductor within the heated length of the bundle, the second part extends from the bottom of the heated length to the lower plate, and the third part is the length of conductor outside the pressure vessel (from the lower plate to the compensation bundle). The resistances associated with each part of the inner copper conductor are provided by the KATHY test facility and are determined based on the known lengths and a set of defined [ ] respectively, for each part.

An arbitrary voltage  $V_{ref}$  is used to compute the current and then the power (the exact value used is not important as it is normalized out).

From Ohm's law, the current in each rod, in Amperes, is calculated.

$$A_i = \frac{V_{ref}}{R_{T,i}} \times 1000 \quad (4)$$

From current and resistance, the power generated in each rod that is within the heated length of the test assembly is calculated. For a full length rod, the power generated is

$$Q_i = A_i^2 R_{O,i} \quad (5)$$

For a part length rod, the power generated in the outer conductor must be added to the power generated in that part of the inner copper conductor that is within the test assembly.

$$Q_i = A_i^2 (R_{O,i} + R_{IC,i}) \quad (6)$$

The total amount of power generated in the test assembly is

$$Q_T = \sum_{i=1}^{N_R} Q_i \quad (7)$$

If  $N_R$  is the total number of heated rods, then the normalized rod power factor is calculated

$$f_i^T = \frac{N_R Q_i}{Q_T} \quad (8)$$

The axial nodalization is then constructed according to the input with extra junctions added at the location of the end of heated length of the part length rods (in the general case, there can be more than one type of part length rod).

Now, for every rod, and for every axial volume k, the nodal power is calculated as.

$$Q_{i,k} = f_{z,i,k} Q_i \quad (9)$$

Where  $f_{z,i,k}$  is the axial power profile for each rod type, corrected for the presence of the inner copper conductor in part length rods.

Equation (9) provides the three dimensional axial power distribution in the test assembly. For each axial plane k, the local peaking factor of each rod in the volume is calculated as.

$$f_{i,k} = \frac{N_k Q_{i,k}}{\sum_{i=1}^{N_k} Q_{i,k}} \quad (10)$$

Where  $N_k$  is the number of rods at axial level k.

The power deposited in each region of the assembly can be calculated. The average rod power factor for each rod in the lower part of the assembly (the fully rodded part) is calculated as

$$f_{L,i} = \frac{N_L \sum_{k=1}^{N_{LP}} Q_{i,k}}{\sum_{i=1}^{N_L} \sum_{k=1}^{N_{LP}} Q_{i,k}} \quad (11)$$

Where  $N_L$  is the number of rods in the lower part of the bundle.

Similarly, the average rod power factor for each rod in the upper part of the assembly (the part not containing part length rods) is

$$f_{U,i} = \frac{N_U \sum_{k=N_{LP}+1}^N Q_{i,k}}{\sum_{i=1}^{N_U} \sum_{k=N_{LP}+1}^N Q_{i,k}} \quad (12)$$

Where  $N_U$  is the number of rods in the upper part of the bundle.

An example of the impact the correction has on the local peaking factors associated with the bottom lattice (lattice extending from the bottom of the heated length to the end of the heated

length of the part length rods) was provided in figures 3.1 and 3.2 in Reference 1. Note that the correction has no impact on the local peaking factors in the upper lattice.

Using the corrected radial and axial power distributions, as defined above, and the corrected bundle powers, new calculations were performed to determine the boiling transition values of f-effective (FEFF) for each test in the database. The boiling transition values are the values of f-effective which result in a CPR of 1.0 at the measured operating conditions. The new additive constants were then computed using the new data using exactly the methodology as was used previously. This methodology is described in Sections 2.3 and 2.5 of Reference 2.

**Question 2.** The paragraph in Section 3.2 on page 3-2, attempts to explain the reason behind the higher bundle power reported in Reference 2/3. Did the error in the original calculation lie in assigning full length to the inner copper conductor instead of a  $\frac{3}{4}$  length, which is the full heated length of the part length rod? Please provide additional clarification.

**Response 2:**

As discussed in the Q1 response, the part length rod inner copper conductor is composed of 3 parts. The first part consists of the length of conductor within the heated length of the bundle, the second part extends from the bottom of the heated length to the lower plate of the test vessel, and the third part is the length of conductor outside the pressure vessel (from the outside of the lower plate to the compensation bundle).

The error in the previous calculation of the bundle power was that it included all of the power associated with the inner copper conductor. It did not exclude the power associated with the 2 parts of the inner copper conductor outside the heated length of the bundle (i.e. the part extending from the bottom of the heated length to the lower plate of the test vessel and the part extending from the lower plate to the compensation bundle).

**References**

1. SPCB Additive Constants for ATRIUM-10 Fuel, EMF-2209(P), Revision 2, Addendum 1, Revision 0, April 2008.
2. SPCB Critical Power Correlation, EMF-2209(P)(A), Revision 1, July 2000.

An AREVA and Siemens company

EMF-2209(NP)  
Revision 3  
SPCB Critical Power Correlation

December 2009



**U.S. Nuclear Regulatory Commission  
Report Disclaimer**

**Important Notice Regarding the Contents and Use of This Document**

***Please Read Carefully***

This technical report was derived through research and development programs sponsored by AREVA NP Inc. It is being submitted by AREVA NP to the U.S. Nuclear Regulatory Commission as part of a technical contribution to facilitate safety analyses by licensees of the U.S. Nuclear Regulatory Commission which utilize AREVA NP fabricated reload fuel or technical services provided by AREVA NP for light water power reactors and it is true and correct to the best of AREVA NP's knowledge, information, and belief. The information contained herein may be used by the U.S. Nuclear Regulatory Commission in its review of this report and, under the terms of the respective agreements, by licensees or applicants before the U.S. Nuclear Regulatory Commission which are customers of AREVA NP in their demonstration of compliance with the U.S. Nuclear Regulatory Commission's regulations.

AREVA NP's warranties and representations concerning the subject matter of this document are those set forth in the agreement between AREVA NP and the Customer pursuant to which this document is issued. Accordingly, except as otherwise expressly provided in such agreement, neither AREVA NP nor any person acting on its behalf:

- a. makes any warranty, or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method, or process disclosed in this document will not infringe privately owned rights;

or

- b. assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this document.

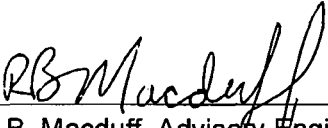
AREVA NP Inc.

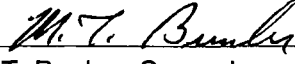
ISSUED IN ON-LINE-  
DOCUMENT SYSTEM


DATE: 12/14/09


EMF-2209(NP)  
Revision 3

### SPCB Critical Power Correlation

Prepared:  10 Dec 09  
R. B. Macduff, Advisory Engineer  
FDWT-AR Date

Reviewed:  12-10-2009  
M.T. Bunker, Supervisor  
FDWT-AR Date

Approved:  12/11/09  
D.W. Pruitt., Manager  
FDWT-AR Date

Approved:  12/10/09  
A.B. Megginis  
Regulatory Affairs Date

skm

AREVA NP Inc.

### Nature of Changes

Item	Page	Description and Justification
1.	All	Framatome ANP has been changed to AREVA NP Inc. (in the Disclaimer, pages 1-1, 1-2, and 5-1, and all the footers)
2.	1-5	Figure 1.2 updated from Addendum 1
3.	2-5	Corrected typographic error in Equation 2.10
4.	2-7	Table 2.5 corrected precision of c13, c14, and c15. (Errata 1)
5.	2-8	Corrected footnote at bottom of page. (Errata 1)
6.	3-30	Table 3-11 Updated from Addendum 1
7.	3-35	Table 3.14 Corrected number of data for STS 17.8 (Errata 2)
8.	3-39	Table 3.16 Updated from Addendum 1
9.	4-16	Table 4.5 Updated from Addendum 1

Revision 3 replaces the items identified above. The information from Addendum 1 to Revision 2 is the basis for the modifications that end with "Addendum 1". Addendum 1 represents the information that is reviewed and approved by the NRC as stated in the SER for Addendum 1. As demonstrated in Addendum 1, the balance of sample analysis results presented in the Revision 2 document remain representative of those that would be obtained with the Addendum 1 modifications. Therefore, the remaining Revision 2 results have not been updated for issuance of Revision 3.

NOTE that the Addendum 1 modifications listed above attain approved status through the SER for Addendum 1. The remainder of the document retains approved status associated with the Revision 2 SER.



## Contents

1.0	Introduction and Summary .....	1-1
1.1	SPCB Database .....	1-1
1.2	SPCB Comparison to the Database .....	1-3
1.3	References .....	1-6
2.0	SPCB Correlation .....	2-1
2.1	SPCB Base Correlation .....	2-2
2.1.1	Functions of A and B .....	2-3
2.1.2	Function C - ATRIUM-9B .....	2-5
2.1.3	Function C - ATRIUM-10 .....	2-6
2.2	Non-Uniform Axial Heat Flux Factor .....	2-7
2.2.1	Non-uniform Factor Corrector .....	2-8
2.2.2	Gradient Function .....	2-10
2.3	Radial Heat Flux Distribution (Rod Centered Local Peaking Function) .....	2-13
2.3.1	FEFFO for Corner Rods .....	2-14
2.3.2	FEFFO for Side Rods .....	2-15
2.3.3	FEFFO for Interior Rods .....	2-17
2.3.4	FEFFO for Interior Rods Adjacent to the ATRIUM Water Canister .....	2-19
2.3.5	Weighting Factors .....	2-22
2.3.6	Bundle Geometry/Spacer Effects (Additive Constants) .....	2-23
2.4	SPCB Correlation Behavior .....	2-24
2.4.1	Functional Behavior of Major Functions within SPCB .....	2-25
2.4.2	Overall Behavior of SPCB .....	2-26
2.5	Additive Constants .....	2-32
2.6	Correlation Range and Applicability .....	2-33
2.6.1	Mass Velocity .....	2-33
2.6.1.1	Steady-State Core Monitoring .....	2-33
2.6.1.2	Transient $\Delta$ CPR Analysis .....	2-34
2.6.1.3	MCPR Safety Limit .....	2-34
2.6.2	Enthalpy .....	2-35
2.6.2.1	Steady-state Core Monitoring .....	2-35
2.6.2.2	Transient $\Delta$ CPR Analysis .....	2-36
2.6.2.3	Safety Limit .....	2-36
2.6.3	Pressure .....	2-37
2.6.4	Inlet Subcooling .....	2-37
2.7	SPCB Application to Other Fuel Designs .....	2-37
2.8	References .....	2-38
3.0	Statistical Analysis of SPCB Critical Power Data .....	3-1
3.1	SPCB Descriptive Statistics .....	3-1
3.2	Statistical Tests .....	3-5
3.3	SPCB Correlation Behavior .....	3-5
3.4	Statistics by Subgroups .....	3-8
3.4.1	Mass Flow Rate .....	3-8

3.4.2	Pressure .....	3-9
3.4.3	Enthalpy at Boiling Transition .....	3-9
3.4.4	FEFF .....	3-10
3.4.5	Axial Offset .....	3-11
3.4.6	Mass Flow Rate and Test Section .....	3-11
3.4.7	Mass Flow Rate and Enthalpy Groups .....	3-13
3.5	ECPR - Mass Flow Plots .....	3-14
3.6	Additive Constant Statistics .....	3-29
3.6.1	Additive Constant Determination .....	3-30
3.6.2	Additive Constant Uncertainty .....	3-33
3.6.3	High Local Peaking .....	3-36
3.6.4	SPCB Conservatisms .....	3-39
3.7	SPCB Predictions Compared with Measurements .....	3-40
3.8	References .....	3-41
4.0	Correlation Validation .....	4-1
4.1	Assessment of ATRIUM-9B/ATRIUM-10 Critical Power Data .....	4-1
4.1.1	Comparison of ATRIUM-9B Verification and Validation Data .....	4-1
4.1.2	Comparison ATRIUM-10 Verification and Validation Data .....	4-4
4.2	Validation with Alternate Design .....	4-7
4.3	Evaluation of Transient Critical Power Data .....	4-9
4.3.1	Conclusions .....	4-17
4.4	References .....	4-17
5.0	SPCB Database .....	5-1
5.1	Facility Description .....	5-1
5.2	Test Bundle Descriptions .....	5-2
5.2.1	ATRIUM-9B .....	5-2
5.2.2	ATRIUM-10 .....	5-3
5.3	Test Strategy .....	5-10
5.3.1	Radial Peaking Profiles .....	5-10
5.3.2	Axial Power Profile .....	5-11
5.3.3	Thermal Hydraulic Test Conditions .....	5-11
5.3.4	Test Design .....	5-12
5.4	SPCB Data .....	5-13
5.5	References .....	5-168

### Tables

1.1	SPCB Range of Applicability .....	1-2
1.2	Enthalpy Bounds for SPCB .....	1-3
2.1	Coefficients for A .....	2-4
2.2	Coefficients of B .....	2-4
2.3	ATRIUM-9B Coefficients for C .....	2-5

## SPCB Critical Power Correlation

2.4	Coefficient Dependence on Mass Velocity.....	2-6
2.5	ATRIUM-10 Coefficients for C.....	2-7
2.6	CORPR Coefficients .....	2-9
2.7	CORPAT10 Coefficients .....	2-10
3.1	Critical Power Ratio Statistics for SPCB Correlation.....	3-1
3.2	Descriptive Statistics .....	3-2
3.3	Inlet Mass Flow Rate Statistics .....	3-9
3.4	Pressure Statistics .....	3-9
3.5	Boiling Transition Enthalpy Statistics .....	3-10
3.6	FEFF Statistics.....	3-10
3.7	Axial Offset Statistics .....	3-11
3.8	Test Section and Flow Statistics .....	3-11
3.9	Flow and Inlet Subcooling Statistics.....	3-13
3.10	SPCB Additive Constants - ATRIUM™-9B .....	3-29
3.11	SPCB Additive Constants - ATRIUM™-10.....	3-30
3.12	Summary of Rods in Boiling Transition .....	3-31
3.13	Test Section FEFFBT Values and Standard Deviations .....	3-32
3.14	Composite Values of FEFFBT.....	3-35
3.15	Additional Additive Constant Uncertainty for High Local Peakings .....	3-39
3.16	FEFF Values for ECPR Evaluation .....	3-39
4.1	ATRIUM-9B Comparison (Verification/Validation).....	4-2
4.2	Additional Validation.....	4-2
4.3	ATRIUM-10 Comparison.....	4-5
4.4	Transient Initial Conditions.....	4-15
4.5	XCOBRA-T Results Using Nominal F-eff.....	4-16
5.1	Test Loop Uncertainties .....	5-9
5.2	Physical Characteristics of the ATRIUM-9B Test Assembly .....	5-9
5.3	Physical Characteristics of the ATRIUM-10 Test Assembly.....	5-9
5.4	Dryout Test Data .....	5-12
5.5	SPCB Data and Analysis Results .....	5-14

## Figures

1.1	Comparison of Predicted and Measured Critical Power with SPCB for ATRIUM™-9B Data.....	1-4
1.2	Comparison of Predicted and Measured Critical Power Data with SPCB for ATRIUM™-10 Data .....	1-5
2.1	Function A.....	2-25
2.2	Function B.....	2-25
2.3	Function (1-B/G).....	2-26
2.4	SPCB Critical Power versus Flow Rate for ATRIUM-10 .....	2-27
2.5	SPCB Critical Power versus Flow Rate for ATRIUM-9B.....	2-27
2.6	SPCB Critical Power Variation with Pressure for ATRIUM-10 .....	2-28
2.7	SPCB Critical Power Variation with Pressure for ATRIUM-9B.....	2-28
2.8	SPCB Critical Power Variation with Inlet Subcooling for ATRIUM-10 .....	2-29
2.9	SPCB Critical Power Variation with Inlet Subcooling for ATRIUM-9B.....	2-29
2.10	SPCB Critical Power Variation with FEFF for ATRIUM-10.....	2-30
2.11	SPCB Critical Power Variation with FEFF for ATRIUM-9B .....	2-30
2.12	Axial Power Shapes for SPCB Sensitivity .....	2-31
2.13	SPCB Critical Power Variation with Axial Power Profiles for ATRIUM-10.....	2-32
2.14	SPCB Critical Power Variation with Axial Power Profiles for ATRIUM-9B .....	2-32
3.1	Distribution of Critical Power Ratio Using SPCB for 2657 Data .....	3-4
3.2	Normality Plot for SPCB for 2657 Data .....	3-5
3.3	SPCB ECPR versus Local Mass Flow Rate for 2657 Data.....	3-6
3.4	SPCB ECPR versus Boiling Transition Enthalpy for 2657 Data.....	3-6
3.5	SPCB ECPR versus Pressure for 2657 Data.....	3-7
3.6	SPCB ECPR versus Assembly FEFF for 2657 Data.....	3-7
3.7	SPCB ECPR versus Axial Offset for 2657 Data.....	3-8
3.8	ECPR versus Mass Flow Rate, ATRIUM-9B with SPCB for 42 Data.....	3-15
3.9	ECPR versus Mass Flow Rate, ATRIUM-9B with SPCB for 41 Data.....	3-15
3.10	ECPR versus Mass Flow Rate, ATRIUM-9B with SPCB for 42 Data.....	3-16
3.11	ECPR versus Mass Flow Rate, ATRIUM-9B with SPCB for 288 Data.....	3-16
3.12	ECPR versus Mass Flow Rate, ATRIUM-9B with SPCB for 297 Data.....	3-17
3.13	ECPR versus Mass Flow Rate, ATRIUM-9B with SPCB for 309 Data.....	3-17
3.14	ECPR versus Mass Flow Rate, ATRIUM-9B with SPCB for 322 Data.....	3-18
3.15	ECPR versus Mass Flow Rate, ATRIUM-9B with SPCB for 44 Data.....	3-18

3.16	ECPR versus Mass Flow Rate, ATRIUM-9B with SPCB for 244 Data.....	3-19
3.17	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 37 Data.....	3-19
3.18	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 31 Data.....	3-20
3.19	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 32 Data.....	3-20
3.20	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 43 Data.....	3-21
3.21	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 34 Data.....	3-21
3.22	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 44 Data.....	3-22
3.23	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 38 Data.....	3-22
3.24	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 117 Data.....	3-23
3.25	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 23 Data.....	3-23
3.26	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 39 Data.....	3-24
3.27	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 34 Data.....	3-24
3.28	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 49 Data.....	3-25
3.29	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 44 Data.....	3-25
3.30	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 47 Data.....	3-26
3.31	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 42 Data.....	3-26
3.32	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 95 Data.....	3-27
3.33	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 102 Data.....	3-27
3.34	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 98 Data.....	3-28
3.35	ECPR versus Mass Flow Rate, ATRIUM-10 with SPCB for 79 Data.....	3-28
3.36	ECPR versus Mass Flow Rate, Upskew Axial Tests with SPCB for 678 Data.....	3-29
3.37	Comparison of STS-40.1 and 33.1.....	3-37
3.38	Comparison of STS-32.1, 17.5 and 17.6.....	3-38
3.39	Comparison of Predicted and Measured Critical Power with SPCB.....	3-41
4.1	ATRIUM-9B Validation Data.....	4-3
4.2	ATRIUM-9B Validation ECPR Characteristics with Flow.....	4-4
4.3	ATRIUM-10 Validation Data.....	4-6
4.4	ATRIUM-10 Validation ECPR Characteristics with Flow.....	4-7
4.5	ATRIUM-10P Validation Data.....	4-8
4.6	Comparison of SPCB with ATRIUM-10P Data.....	4-9
4.7	Steady-State Critical Power versus Inlet Flow for Tests 17.8 and 29.5 at Inlet Subcooling of 20 Btu/lbm.....	4-10
4.8	Typical Pressurization Transient Forcing Function.....	4-11
4.9	Typical Flow Decay Transient Forcing Function.....	4-12

4.10	Thermocouple Response for Typical Transient Test.....	4-13
4.11	Comparison of Measured and Calculated Time of Boiling Transition .....	4-17
5.1	Karlsruhe Thermal Hydraulic Test Loop .....	5-4
5.2	Test Vessel .....	5-5
5.3	Thermocouple Locations .....	5-6
5.4	Bundle Spacer Locations .....	5-7
5.5	Rod Axial Power Profiles Tested.....	5-8
5.6	Peaking Pattern STS 12.1.....	5-84
5.7	STS 12.1 .....	5-85
5.8	Peaking Pattern STS 12.2.....	5-86
5.9	STS 12.2 .....	5-87
5.10	Peaking Pattern STS 12.3.....	5-88
5.11	STS 12.3 .....	5-89
5.12	Peaking Pattern STS 33.1.....	5-90
5.13	STS 33 - 600 psi .....	5-91
5.14	STS 33 - 800 psi .....	5-92
5.15	STS 33 - 1000 psi .....	5-93
5.16	STS 33 - 1200 psi .....	5-94
5.17	STS 33 - 1400 psi .....	5-95
5.18	Peaking Pattern STS 34.1.....	5-96
5.19	STS 34 - 600 psi .....	5-97
5.20	STS 34 - 800 psi .....	5-98
5.21	STS 34 - 1000 psi .....	5-99
5.22	STS 34 - 1200 psi .....	5-100
5.23	STS 34 - 1400 psi .....	5-101
5.24	Peaking Pattern STS 35.1.....	5-102
5.25	STS 35 - 600 psi .....	5-103
5.26	STS 35 - 800 psi .....	5-104
5.27	STS 35 - 1000 psi .....	5-105
5.28	STS 35 - 1200 psi .....	5-106
5.29	STS 35 - 1400 psi .....	5-107
5.30	Peaking Pattern STS 37.1.....	5-108
5.31	STS 37 - 600 psi .....	5-109
5.32	STS 37 - 800 psi .....	5-110

SPCB Critical Power Correlation

---

5.33	STS 37 - 1000 psi .....	5-111
5.34	STS 37 - 1200 psi .....	5-112
5.35	STS 37 - 1400 psi .....	5-113
5.36	Peaking Pattern STS 38.3.....	5-114
5.37	STS 38 .....	5-115
5.38	Peaking Pattern STS 40.1.....	5-116
5.39	STS 40 - 800 psi .....	5-117
5.40	STS 40 - 1000 psi .....	5-118
5.41	STS 40 - 1200 psi .....	5-119
5.42	Peaking Pattern STS 17.1.....	5-120
5.43	STS 17.1 .....	5-121
5.44	Peaking Pattern STS 17.2.....	5-122
5.45	STS 17.2 .....	5-123
5.46	Peaking Pattern STS 17.3.....	5-124
5.47	STS 17.3 .....	5-125
5.48	Peaking Pattern STS 17.4.....	5-126
5.49	STS 17.4 .....	5-127
5.50	Peaking Pattern STS 17.5.....	5-128
5.51	STS 17.5 .....	5-129
5.52	Peaking Pattern STS 17.6.....	5-130
5.53	STS 17.6 .....	5-131
5.54	Peaking Pattern STS 17.7.....	5-132
5.55	STS 17.7 .....	5-133
5.56	Peaking Pattern STS 17.8.....	5-134
5.57	STS 17.8 - Low Pressure .....	5-135
5.58	STS 17.8 - 1000 psi .....	5-136
5.59	STS 17.8 - High Pressure .....	5-137
5.60	Peaking Pattern STS 17.9.....	5-138
5.61	STS 17.9 .....	5-139
5.62	Peaking Pattern STS 17.10.....	5-140
5.63	STS 17.10 .....	5-141
5.64	Peaking Pattern STS 17.11.....	5-142
5.65	STS 17.11 .....	5-143
5.66	Peaking Pattern STS 17.12.....	5-144

## SPCB Critical Power Correlation

5.63	STS 17.10 .....	5-141
5.64	Peaking Pattern STS 17.11.....	5-142
5.65	STS 17.11 .....	5-143
5.66	Peaking Pattern STS 17.12.....	5-144
5.67	STS 17.12 .....	5-145
5.68	Peaking Pattern STS 28.1.....	5-146
5.69	STS 28.1 .....	5-147
5.70	Peaking Pattern STS 28.2.....	5-148
5.71	STS 28.2 .....	5-149
5.72	Peaking Pattern STS 29.1.....	5-150
5.73	STS 29.1 .....	5-151
5.74	Peaking Pattern STS 29.2.....	5-152
5.75	STS 29.2 - 600 psi .....	5-153
5.76	STS 29.2 - 1000 psi .....	5-154
5.77	STS 29.2 - 1400 psi .....	5-155
5.78	Peaking Pattern STS 29.4.....	5-156
5.79	STS 29.4 - 600 psi .....	5-157
5.80	STS 29.4 - 1000 psi .....	5-158
5.81	STS 29.4 - 1400 psi .....	5-159
5.82	Peaking Pattern STS 29.5.....	5-160
5.83	STS 29.5 - 600 psi .....	5-161
5.84	STS 29.5 - 1000 psi .....	5-162
5.85	STS 29.5 - 1400 psi .....	5-163
5.86	Peaking Pattern STS 32.1.....	5-164
5.87	STS 32.1 - 600 psi .....	5-165
5.88	STS 32.1 - 1000 psi .....	5-166
5.89	STS 32.1 - 1400 psi .....	5-167

*This document contains a total of 283 pages.*



## 1.0 Introduction and Summary

This document describes SPCB, AREVA NP Inc.'s (AREVA) critical power correlation for boiling water reactors (BWR). This correlation is designed for application to steady-state, transient, and Loss of Coolant Accident (LOCA) critical heat flux (CHF) predictions for ATRIUM™ -9B and ATRIUM-10 fuel designs.

With the introduction of advanced spacer designs, it was determined that the ANFB (Reference 1.1) critical power correlation (CPR) required modification to properly account for the upstream effects of ULTRAFLOW™ spacers. ULTRAFLOW spacers have swirl vanes at the spacer internal strip intersections which impart a centrifugal force on the two phase mixture. This results in separation of vapor and liquid with the heavier water droplets being deposited on the rod surface, and the steam remaining in the subchannel. The ANFB correlation is comprised of two essential components. One component calculates critical heat flux for axially uniform power conditions. The second component adjusts the calculated heat flux for non-uniform axial conditions. While the correlation could have been modified by changing only the non-uniform axial power corrector, it would have resulted in an increase in the uncertainties associated with the correlation. Therefore, the constants for the correlation were modified to maintain a reasonably low standard deviation.

The SPCB correlation can be used to accurately predict assembly critical power for ATRIUM-9B and ATRIUM-10 fuel designs. The correlation allows accurate prediction of the limiting rod within a bundle and accounts for local spacer effects and bundle geometry on critical power by a set of constants, one constant for each rod in the bundle. These constants are called Additive Constants and are presented in Table 3.10 for the ATRIUM-9B and Table 3.11 for the ATRIUM-10 design. The critical power ratio distribution associated with SPCB is adequately represented with a normal curve using an overall mean of [ ] for ATRIUM-9B and an overall mean of [ ] for ATRIUM-10.

### 1.1 SPCB Database

The SPCB database is comprised of [ ] steady-state data points taken on [ ] different test assemblies. The axial power shapes of the tests were [ ] peak-to-average cosine and [ ]

---

\* ATRIUM and ULTRAFLOW are trademarks of AREVA NP Inc.

SPCB Critical Power Correlation

peak-to-average upskew and downskew. The database was compiled from tests performed exclusively at the AREVA thermal hydraulic test facility at Karlstein, Germany.

During the correlation development, the database was divided into a correlating (verification) set of data and a validation data set. Of the [ ] steady-state data points, [ ] were set aside for validation. In addition, another [ ] validation points taken from steady-state critical power tests not included in the database were analyzed. 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 velocity, enthalpy, axial power profile, and local peaking distribution. Table 1.1 represents the range of parameters tested. [

] Bounding

values for enthalpy are checked at the plane of boiling transition based on ranges shown in Table 1.2.

**Table 1.1 SPCB Range of Applicability**

Pressure (psia)	571.4 to 1432.2
Inlet Mass Velocity (Mlb/hr-ft <sup>2</sup> )	0.087 to 1.5
Inlet Subcooling (Btu/lbm)	5.55 to 148.67
Design Local Peaking	1.5
Tested Local Peaking	1.45

#### 1.2 *SPCB Comparison to the Database*

SPCB has been used to predict the critical power for each data point in the database. The ratio of the predicted critical power to the measured critical power (Experimental Critical Power Ratio, ECPR) has been determined for each test point and the additional validation points and is used along with the standard deviation of the ECPR as the basis to determine the ability of the correlation to predict critical power. Comparisons of the predicted and measured critical power for both the ATRIUM-9B and the ATRIUM-10 are shown in Figures 1.1 and 1.2, respectively.



SPCB Critical Power Correlation

---



1.3 **References**

- 1.1 ANF-1125(P)(A) and Supplements 1 and 2, *ANFB Critical Power Correlation*, Advanced Nuclear Fuels Corporation, April 1990.

## 2.0 SPCB Correlation

A BWR fuel assembly operates into the annular flow regime. A liquid film on the rod and a steam-water mixture in the center region characterizes this regime. As the flow progresses upward, the water film changes because of boiling off and the deposition of water droplets onto the liquid film. A rapid temperature excursion occurs when the liquid film goes to zero thickness. The loss of this liquid film is variously termed dryout, boiling transition, and critical heat flux (CHF).

The SPCB correlation is similar to the ANFB-10 Critical Power Correlation (Reference 2.1) and the ANFB Critical Power Correlation (Reference 2.2). All three correlations use empirical fits to the data that use planar average conditions to predict critical heat flux. The form of the correlations is that developed by Macbeth (Reference 2.3). This correlation form is developed by the transformation from the linear behavior of CHF with inlet subcooling to the linear behavior of CHF with local enthalpy. A plot of inlet subcooling versus critical heat flux (for example, see Figure 2.8) shows that critical heat flux varies linearly with inlet subcooling. For uniform heat flux (Base), this relationship can be expressed

$$q''_{\text{Base}} = A + B(h_{\text{in}}) \quad (2.1)$$

where A and B are functions of pressure and flow and  $h_{\text{in}}$  is the inlet subcooling. However, inlet enthalpy is not an appropriate parameter for a transient application, so the correlation must be converted for local conditions by application of a channel average heat balance. This results in the form

$$q''_{\text{Base}} = \frac{A - B(h_{\text{bt}})}{1 - \frac{B}{G}} \quad (2.2)$$

where G is the mass velocity and  $h_{\text{bt}}$  is the enthalpy at the plane of boiling transition. For the SPCB application another term, C, is added to  $h_{\text{bt}}$ . This parameter is specific to the fuel type being analyzed. Also non-uniform axial power corrector is used which is developed from the

Tong factor (Reference 2.3) used in the analysis of BWR core thermal-hydraulic behavior. This corrector is based on a mass balance on the liquid film and has the form

$$F = \frac{\Omega}{q''_{i,c} (1 - e^{-\Omega \ell_c})} \int_0^{\ell_c} [e^{-\Omega(\ell_c - Z)}] q''(Z) dZ \quad (2.3)$$

In this expression,  $\Omega$  is a function of mass velocity and heat flux gradient,  $q''(Z)$  is the axial heat flux,  $\ell_c$  is the axial plane of interest, and  $Z$  is the position on the fuel rod. [

]

Combining equations 2.2 and 2.3 results in the correlation for a non-uniform (NU) heat flux case,

$$q''_{NU} = \frac{q''_{Base}}{F} \quad (2.4)$$

This formulation is the basis for the SPCB correlation.

### 2.1 **SPCB Base Correlation**

Using the Macbeth form of the critical heat flux equation developed above, SPCB correlation has the following form:





2.1.1 Functions of A and B

The terms A and B are applicable to both the ATRIUM-9B and ATRIUM-10. The functions A and B have the form



SPCB Critical Power Correlation

---



## 2.2 *Non-Uniform Axial Heat Flux Factor*

The non-uniform axial power corrector (Tong, et al., Reference 2.3) used in the ANFB critical power correlation (Reference 2.2) provides the basis of the non-uniform axial correction factor for SPCB. The non-uniform axial correction factor characteristic modified in SPCB is the empirical factor, " $\Omega$ " (Reference 2.4). In addition, a post-multiplier to the non-uniform axial factor is included to provide an adjustment to better address the impact of non-uniform axial shapes. This adjustment factor is appropriate for the steady-state and transient evaluation processes. [

]

SPCB Critical Power Correlation

The base non-uniform axial factor is given by

$$F_{\text{Base}} = \frac{\Omega}{q''_{l_c} (1 - e^{-\Omega l_c})} \int_0^{l_c} \left[ e^{-\Omega(l_c - Z)} \right] q''(Z) dZ \quad (2.14)$$

where

- $q''(Z)$  = Axial heat flux
- $Z$  = Axial position on fuel rod
- $\Omega^\dagger$  = Empirical factor (described below)
- $l_c$  = Axial position of plane of interest

2.2.1 Non-uniform Factor Corrector

An additional correction is obtained by multiplying the base non-uniform axial factor,  $F_{\text{Base}}$ , by the following pressure, flow, and enthalpy gradient term.

[ ]

\* The maximum value of  $F_{\text{Base}}$  is [ ].

†  $\Omega$  is taken as the [ ]

‡ [ ]

§ [ ]

SPCB Critical Power Correlation

---



---

• [ ]  
† [ ]

1

1

### 2.2.2 Gradient Function

This section describes the GRAD1<sup>1</sup> term and how it is calculated and used in evaluating critical power. A gradient type of term was suggested as being important for dryout in References 2.4 and 2.5.

The influence of the axial heat flux shape is affected by the gradient of enthalpy at the location of interest. More specifically, for a steady-state configuration, the gradient of the axial heat flux shape represents the feature being modeled. Enthalpy is used to transform the heat flux shape gradient behavior to the fluid properties used to predict critical heat flux.

---

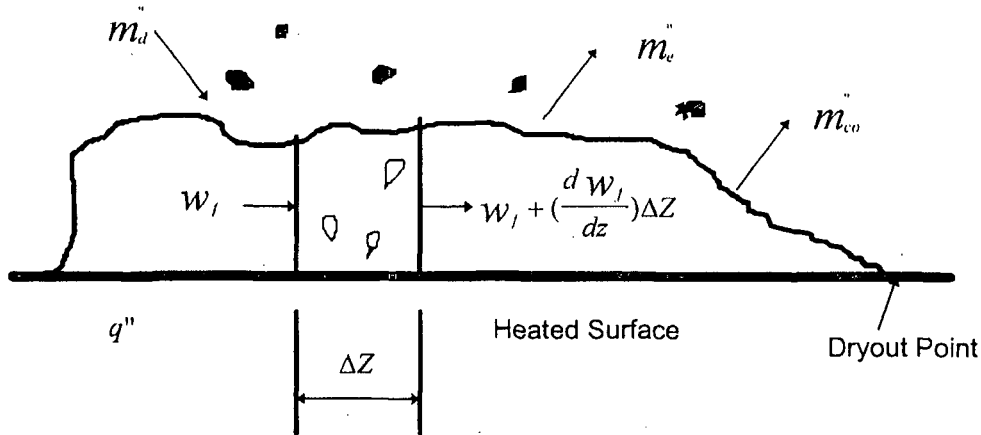
[

]



SPCB Critical Power Correlation

It is an experimentally determined fact that dryout normally occurs in a region of decreasing heat flux. This is shown by (see Reference 2.5)



- Let  $w_f$  = Liquid flow rate  
 $P_H$  = Heated perimeter  
 $m_e''$  = Liquid entrainment mass flux  
 $m_{co}''$  = Liquid carryover mass flux  
 $m_d''$  = Liquid disposition mass flux  
 $A_{x-s}$  = Cross section flow area  
 $W_{le}$  = Entrained liquid flow rate

$$w_f + \frac{dw_f}{dZ} \Delta Z + \frac{P_H \Delta Z q''}{h_{fg}} + m_e'' P_H \Delta Z + m_{co}'' P_H \Delta Z - w_f - m_d'' P_H \Delta Z = 0$$

$$\frac{dw_f}{dZ} = P_H \left( m_d'' - \frac{q''}{h_{fg}} - m_e'' - m_{co}'' \right) \quad (2.19)$$

$$w_f = GA_{x-s} (1 - \langle x \rangle) - W_{le}$$

If one assumes that the occurrence of dryout happens when the film flow gradient is zero, then

$$\begin{aligned} \frac{dw_f}{dZ} &= 0 \\ q_c'' &= h_{fg}(m_d'' - m_e'' - m_{co}'') \\ m_e'' &= m_{co}'' = 0 \\ q_c'' &= h_{fg}m_d'' \end{aligned} \quad (2.20)$$

Therefore, the spatial derivation of the flow gradient is

$$\frac{d^2w_f}{dZ^2} = P_H \left( \frac{dm_d''}{dZ} - \frac{1}{h_{fg}} \frac{dq''}{dZ} \right) > 0 \quad (2.21)$$

Normally,  $\frac{dm_d''}{dZ} < 0$  as dryout is approached since the amount of liquid is decreasing.

Therefore,  $\frac{dq''}{dZ} < 0$  for upstream dryout to occur.

The correlation uses the gradient in the evaluation of the post-multiplier to the non-uniform axial factor, as expressed earlier. The value of gradient is determined for every node. The gradient is based on the Equation 2.22.

$$\text{Grad1} = f \left( \frac{\partial^2 h}{\partial z^2} \right) \quad (2.22)$$

where

h represents the enthalpy (Btu/lbm) at node j

z represents the axial elevation (ft) at node j

[

]

[  
  
]

### 2.3 *Radial Heat Flux Distribution (Rod Centered Local Peaking Function)*

The function C in Equation 2.2 includes the parameter FEFF. The FEFF parameter characterizes the local peaking factor effect on the bundle critical power and is retained from References 2.1 and 2.2 and is defined in the same manner. This section describes how the FEFF calculation is applied to ATRIUM-10 and ATRIUM-9B fuel assemblies. The critical power varies inversely with FEFF. That is, as FEFF increases in value, the critical power decreases in value. FEFF has two parts. One part depends solely on the peaking factors of the rod of interest and its immediate neighbors. The other is termed an Additive Constant,  $\ell$ , which accounts for other local effects from spacing and geometry. The Additive Constant is determined from the experimental data. The definitions of FEFF and examples for several rod locations, including rods located adjacent to part length rods as would be observed for ATRIUM-10 are discussed below.

The portion of FEFF that depends on local peaking distribution is termed FEFFO. FEFFO for the  $i^{\text{th}}$  rod is calculated from the equation

┌

└

2.3.1 FEFFO for Corner Rods

Corner rods in a lower lattice of an assembly are adjacent to three fueled rods. This is illustrated as



### 2.3.2 FEFFO for Side Rods

Side rods could be adjacent to fueled rods and are illustrated as



Similarly, with the unheated portion in a  $k$  position, Equation 2.26 becomes



### 2.3.3 FEFFO for Interior Rods

Interior rods in lower lattices of an ATRIUM-10 fuel assembly are adjacent to other heated rods. In the lattice that considers the plenum of the part length rod, one rod position is treated as unheated. In the lattice above the top of the part length rod, one rod position continues to be treated as unheated. Interior rods adjacent to the ATRIUM water canister are addressed in Section 2.3.4.

The rod configuration examined for an interior rod is illustrated as

$$\begin{array}{ccc} k & j & k \\ j & i & j \\ k & j & k \end{array}$$

The application of Equation 2.23 for rod  $i$  becomes







[

]

#### 2.3.4 FEFFO for Interior Rods Adjacent to the ATRIUM Water Canister

The essential character for rods adjacent to the ATRIUM water canister is similar to the character for rods adjacent to the side (Section 2.3.2). That is, fewer rods are taken into consideration in both the numerator and the denominator of Equation 2.14. For the lower lattice of the ATRIUM-10 design, rods in the middle of the channel see three  $j$  rods and two  $k$  rods, similar to the side rods described in Section 2.3.2. Interior rods adjacent to the middle rod see three  $j$  rods and three  $k$  rods, while rods on the corner of the ATRIUM water canister see four  $j$  rods and three  $k$  rods. The calculation for FEFFO corresponding to these three cases is

$$\frac{j \quad i \quad j}{k \quad j \quad k}$$



SPCB Critical Power Correlation

---



2.3.5 Weighting Factors

The weighting factors used in Equation 2.22 of Section 2.3 determined in the work reported in Reference 2.1 continue to be appropriate and have not been modified.

### 2.3.6 Bundle Geometry/Spacer Effects (Additive Constants)

Spacers and bundle geometry influence the critical power behavior of the bundle. [

] These

[ ] are termed the Additive Constants,  $\ell$ . Additive Constants can be considered as a flow/enthalpy redistribution characteristic for a given bundle or spacer design.



[

]

The described averaging process ensures that the ECPR (ratio of calculated critical power to measured critical power, sometimes abbreviated CPR) is always 1 for the set of experiments used to determine the Additive Constants.

Once values of  $FEFF(i)$  are available, critical power can be determined on a rod-specific basis. The critical bundle power is the critical power calculated based on the limiting rod  $FEFF$ . The averaging process for the Additive Constants can yield values of  $FEFFBT$  that are lower than the limiting  $FEFF$ . All rods with an  $FEFF(i)$  exceeding the  $FEFFBT$  are considered to be in boiling transition. The maximum  $FEFF$  value for an assembly is determined using the Additive Constants and local peaking of the assembly. This maximum  $FEFF$  may exceed the  $FEFFBT$  for some test sections. Using this maximum value of  $FEFF$  provides an appropriate view of the mean ECPR and standard deviation characteristic of the population. The observation that some assemblies will have some rod locations where actual  $FEFF$  exceeds  $FEFFBT$  provides for conservatism in the application of the SPCB correlation; i.e., more rods would be predicted to be in boiling transition.

#### 2.4 ***SPCB Correlation Behavior***

The SPCB critical power correlation was investigated functionally to ensure smooth functions and no discontinuities. Section 2.4.1 describes the functional behavior of the major components of the correlation. The correlation was also investigated for its behavior over a wide range of conditions; this is described in Section 2.4.2.

2.4.1 Functional Behavior of Major Functions within SPCB

Functions A and B are smooth and have a weak dependence on pressure but a strong dependence on flow. Figure 2.1 and Figure 2.2 show the behavior of the A and B functions respectively. The symbols are based on pressures between 600 and 1400 psi.



Equation 2.1 shows that the function [ ] in the correlation. This term is always positive and does not cause any discontinuity. Figure 2.3 shows the behavior of this term. The symbols in Figure 2.3 account for a range of pressures (600 - 1400 psi). [

] Note that functions A and B are similar to the functions A and B described in References 2.1 and 2.2.

#### 2.4.2 Overall Behavior of SPCB

The critical power calculated by the SPCB correlation behaves well throughout its range of validity. This section provides results of sensitivity studies for critical power with respect to flow, pressure, inlet subcooling, FEFF, and axial power shape.



SPCB Critical Power Correlation

---



**Pressure**

Figures 2.6 and 2.7 show the behavior of SPCB with pressure. These figures show that pressure is only a minor contributor to critical power. The reduction in critical power with increasing pressure becomes significant at higher flow rates. [

]



### **Inlet Subcooling**

Figures 2.8 and 2.9 show the gain in critical power with increasing subcooling reduces with a reduction in flow rate. The figures show a nearly linear impact of inlet subcooling on critical power. [

]



**FEFF**

The effect of FEFF on critical power is shown in Figures 2.10 and 2.11. [

]



### **Axial Power Profile**

Figure 2.12 shows a sequence of simulated axial power profiles with peaks varying in both location and absolute magnitude. The corresponding changes in assembly critical power with respect to the variation in axial power shape are shown in Figures 2.13 and 2.14 for five different inlet flow rates. The sensitivity of critical power with respect to the variation of axial power profile is captured by the correlation through the non-uniform axial factor and through the gradient parameter. [

]



## 2.5 *Additive Constants*

Additive Constants were determined using the procedure described in Section 2.3.6. These were determined for the ATRIUM-9B and for the ATRIUM-10 fuel assemblies. The verification data was used to determine the Additive Constants; the combined verification and validation data were used to determine the uncertainty of Additive Constants. The Additive Constants for

the ATRIUM-9B fuel assemblies are given in Table 3.10; those for the ATRIUM-10 are given in Table 3.11.

## 2.6 *Correlation Range and Applicability*

Dryout tests are performed with electrically heated test assemblies. The test assemblies have controls for power, inlet flow, pressure, and inlet subcooling. The specified test parameters are flow, pressure, and subcooling. Determining the power at which dryout occurs is the desired test result. The conditions of the test are used to determine a dryout power. The calculated dryout power divided by the measured power defines the ECPR. In addition, the calculation determines fluid conditions at every location of the bundle.

### 2.6.1 Mass Velocity

The range of applicability for nodal mass velocity at the plane of boiling transition is presented in Table 1.2. [

]

#### 2.6.1.1 Steady-State Core Monitoring

##### High Mass Velocity Limit

If the nodal mass velocity exceeds the high mass velocity applicability limit, an overly optimistic critical heat flux for the node might be predicted because of using the correlation beyond its test range. To avoid this situation and to provide a conservative estimate of the bundle critical power ratio, assemblies with mass velocities greater than the high mass velocity limit are analyzed with the mass velocity conservatively reduced to the high mass velocity limit. This results in conservatively high enthalpy and quality distributions in the bundle and the use of the SPCB correlation within its range of applicability to produce a conservative CHF and bundle critical power ratio (CPR).

### Low Mass Velocity Limit

The critical power calculation is not performed for a bundle with a nodal mass velocity below the low mass velocity applicability limit. Appropriate messages are printed in calculational output.

#### 2.6.1.2 Transient $\Delta$ CPR Analysis

The  $\Delta$ CPR during a transient is the difference between the steady-state CPR before the transient and the minimum CPR during the transient. The transient calculation is performed with an initial assembly power that results in boiling transition (critical heat flux ratio = 1.0) at the worst point throughout the transient simulation. The computer program checks the coolant conditions at the time of boiling transition against the SPCB applicability limits.

### High Mass Velocity Limit

If the nodal mass velocity exceeds the high mass velocity limit at the worst point in the transient, CHF is calculated with both the actual nodal mass velocity, and again with the nodal mass velocity set to the critical power correlation high mass velocity limit. The transient CHF is determined from the more conservative of the two calculations.

### Low Mass Velocity Limit

If the nodal mass velocity is below the low mass velocity limit at the point of boiling transition, the transient CHF is determined from the Hench-Levy correlation (Reference 2.8).

#### 2.6.1.3 MCPR Safety Limit

### High Mass Velocity Limit

The same logic that is applied in the steady-state core monitoring is used for the minimum critical heat flux ratio (MCHFR) node in the MCPR safety limit calculation. The difference is that the resultant lower CHF is used to determine the number of rods in boiling transition rather than the bundle CPR.



### Low Mass Velocity Limit

If the MCHFR nodal mass velocity is below the low mass velocity limit, every rod in the bundle is assumed to be in boiling transition.

### 2.6.2 Enthalpy

The range of applicability for nodal enthalpy at the plane of boiling transition is presented in Table 1.2 as a function of nodal mass velocity. These ranges represent the boiling transition enthalpies envelope observed during SPC testing. Because the boiling transition enthalpy is a primary hydraulic parameter that characterizes the bundle CHF performance for the current hydraulic conditions and the neutronic power distribution within the bundle, specific checks are made within the SPC methodology for exceeding these limits.

#### 2.6.2.1 Steady-state Core Monitoring

### High Enthalpy Limit

If the boiling transition nodal enthalpy exceeds the high enthalpy applicability limit, dryout is assumed to occur when the bundle power is elevated to produce the boiling transition nodal enthalpy equal to the high enthalpy applicability limit. Even though the SPCB correlation used at the high enthalpy applicability limit would predict margin to dryout, no credit is taken and the CPR is limited by the high enthalpy applicability limit.

### Low Enthalpy Limit

If the boiling transition nodal enthalpy is below the low enthalpy applicability limit, the bundle enthalpy and corresponding quality distributions are artificially increased to compute a CHF corresponding to a boiling transition nodal enthalpy within the correlation applicability limits. Because CHF decreases with increasing enthalpy, this results in a conservative CHF based on the use of the SPCB correlation within its range of applicability. This CHF is then used in the critical power calculation for the bundle.

#### 2.6.2.2 Transient $\Delta$ CPR Analysis

##### High Enthalpy Limit

If the boiling transition nodal enthalpy exceeds the high enthalpy applicability limit, the transient simulation is repeated with a lower radial power factor (higher initial critical power ratio) until the worst point in the transient results in a nodal enthalpy below the high enthalpy limit. This treatment results in a conservative transient simulation because the hot bundle will not experience boiling transition (i.e., minimum critical heat flux ratio > 1.0) and the SPCB correlation is applied within its range of applicability.

##### Low Enthalpy Limit

If the boiling transition nodal enthalpy is below the low enthalpy applicability limit, the transient CHF is determined from the Hensch-Levy correlation (Reference 2.8) provided the calculated CHF from Hensch-Levy is greater than the calculated CHF from the SPCB correlation; otherwise, the bundle enthalpy and the corresponding quality distributions are artificially increased to compute a CHF corresponding to a boiling transition nodal enthalpy that is within the correlation applicability limits. Because CHF decreases with increasing enthalpy, this results in a conservative CHF, which is based on the use of the critical power correlation within its range of applicability.

#### 2.6.2.3 Safety Limit

##### High Enthalpy Limit

If the MCHFR nodal enthalpy is above the high enthalpy limit, every rod in the bundle is assumed to be in boiling transition.

##### Low Enthalpy Limit

If the MCHFR nodal enthalpy is below the low enthalpy limit, the enthalpy and quality distributions are artificially increased, as in the steady-state core monitoring calculation to determine a conservative CHFR. This CHFR is then used to compute the number of rods in boiling transition.

### 2.6.3 Pressure

The range of applicability for pressure is presented in Table 1.1. If the pressure falls outside this range, appropriate messages are printed and the calculation is stopped.

### 2.6.4 Inlet Subcooling

The test range for inlet subcooling is presented in Table 1.1. The SPC methodology checks the inlet subcooling against this range; if the subcooling falls below the test range minimum, the calculation is stopped. If the subcooling exceeds the test range maximum, the inlet subcooling is set to the maximum subcooling limit.

## 2.7 ***SPCB Application to Other Fuel Designs***

The SPCB Critical Power Correlation performs well for the ATRIUM-9B and ATRIUM-10 fuel designs. The SPCB correlation may also be applicable to other vendor fuel designs or future SPC fuel designs which have design changes influencing the critical power characteristics. The performance of the SPC correlation for other vendors' fuel designs or future SPC fuel designs with different critical power performance requires appropriate assessment, determination of uncertainties, and determination of boundaries. With sufficient measured data, including a broad range of flows, pressures, subcoolings, axial power shapes, and local peaking configurations, the process used for determining Additive Constants for the ATRIUM-9B or ATRIUM-10 fuel can be directly applied.

With data that are calculated based on an alternative critical power correlation, then the process described in Reference 2.6 for ANFB and submitted as one generic process in Reference 2.7 could be used to obtain appropriate characterization. The use of the generic process in Reference 2.7 requires the use of the appropriate ratio of ECPR standard deviation to Additive Constant standard deviation. [

]

2.8 **References**

- 2.1 EMF-1997(P)(A) and Supplement 1, *ANFB-10 Critical Power Correlation*, Siemens Power Corporation, July 1998.
- 2.2 ANF-1125(P)(A) and Supplements 1 and 2, *ANFB Critical Power Correlation*, Advanced Nuclear Fuels Corporation, April 1990.
- 2.3 L. S. Tong, et al., "Influence of Axially Non-Uniform Heat Flux on DNB," *Chemical Engineering Progress Symposium Series*, No. 64, V 62, 1966.
- 2.4 J. G. Collier, *Convective Boiling & Condensation*, Second Edition, McGraw Hill, 1981.
- 2.5 R. T. Lahey, Jr. and F.J. Moody, *The Thermal Hydraulics of A Boiling Water Nuclear Reactor*, Second Edition, ANS Monograph, 1993.
- 2.6 EMF-1125(P)(A) Supplement 1, Appendix C, *ANFB Critical Power Correlation Application for Co-Resident Fuel*, Siemens Power Corporation, August 1997.
- 2.7 EMF-2245(P), *Critical Power Correlation Application for Co-Resident Fuel*, Siemens Power Corporation, July 1999.
- 2.8 Letter, C. A. Carpenter (NRC) to J. F. Mallay (SPC), "Modifications to Procedures for Use of XCOBRA-T," NRC-IC-99:022, June 10, 1999.

### 3.0 **Statistical Analysis of SPCB Critical Power Data**

The SPCB data statistics discussed in this section include descriptive statistics for the data set, an evaluation of the distribution characteristics, figures of ECPR with respect to correlation parameters, and descriptive statistics for Additive Constants and Additive Constant Uncertainty.

The next section addresses the validation data separately from the combined data.

#### 3.1 ***SPCB Descriptive Statistics***

Table 3.1 shows the mean ECPR, standard deviation of the mean and number of data by test section. Additional information in the table includes the fuel design represented (ATRIUM-9B or ATRIUM-10) and the axial power shape represented (Cosine, Downskew, or Upskew).

Higher moments for the SPCB analysis of ECPR are computed and results are presented for the following equations. Reference 3.1 provides the relationships for computing the higher order moments. These higher order moments include the third moment about the mean and the fourth moment about the mean and are given by

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

Third moment about mean

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

Fourth moment about mean

Similarly, the second moment is given by

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

Second moment about mean

A measure of skewness is given by

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

Skewness measure

A measure of kurtosis is given by

$$\beta_2 = \frac{m_4}{(m_2)^2}$$

Kurtosis measure

The values of the measures with respect to ECPR are

$m_2$	=	0.000491
$m_3$	=	0.000000
$m_4$	=	0.000000680450
$\sqrt{\beta_1}$	=	0.004488
$\beta_2$	=	2.83

These statistics are close to those of a normal distribution. The distributional character of the SPCB critical power ratios are viewed in Figures 3.1 and 3.2. Figure 3.1 is a histogram of the frequency of occurrence of CPR while Figure 3.2 shows that the distribution generally follows the linear characteristics of a normal distribution. As is shown later, the correlation over predicts the number of rods in boiling transition. The behavior of the CPR distribution is adequately represented by normal distribution characteristics for the safety limit analysis.







### 3.2 ***Statistical Tests***

The SPCB data were tested using Lilliefors test for normality (Reference 3.2). The test indicates that at the one percent significance level the assumption of normality cannot be rejected.

### 3.3 ***SPCB Correlation Behavior***

Figures 3.3 through 3.8 show the SPCB ECPR values graphically with respect to mass flow rate, boiling transition enthalpy, pressure, FEFF, and other parameters.



SPCB Critical Power Correlation

---





### 3.4 ***Statistics by Subgroups***

The descriptive statistics for the overall data can be examined by several subgroups of data. Mean, standard deviation, and number of data are presented. This section covers all data. Validation data is separated out in Section 4.

#### 3.4.1 Mass Flow Rate

Table 3.3 shows the results of examining representative mass flow rate groups from [

]



3.4.3 Enthalpy at Boiling Transition

Table 3.5 shows the results of examining representative boiling transition enthalpies from [

]





3.4.5 Axial Offset

Table 3.7 gives the results of examining representative axial offset values.







#### 3.4.7 Mass Flow Rate and Enthalpy Groups

Characteristics for the SPCB ECPR can be examined by considering data within different ranges of flow and inlet subcooling. Table 3.9 shows this characterization based on eleven flow groups and three inlet subcooling ranges.

### 3.5 *ECPR - Mass Flow Plots*

The overall view of ECPR versus mass flow is shown in Figure 3.3 and no trend is observed. Figures 3.8 through 3.35 show the behavior of ECPR with mass flow for each test section. Within individual tests, some tests over predict high flow and some under predict high flow. Because the evaluations are performed using the FEFF calculated for the test section based on the composite Additive Constants, some test sections provide under predictions for many flows. Figure 3.36 shows the combined behavior of ECPR with mass flow for the upskew data.

**Pages 3-15 through 3-28 are proprietary in toto.**

### 3.6 *Additive Constant Statistics*

The Additive Constants for the ATRIUM-9B and ATRIUM-10 fuel designs were developed using the process summarized in Section 2. This section presents the determination of the Additive Constants and uncertainty and describes the conservatism for the SPCB critical power correlation. Tables 3.10 and 3.11 provide the Additive Constants for the ATRIUM-9B and ATRIUM-10 fuel designs.

3.6.1 Additive Constant Determination

The steps summarized in Section 2.3.6 are provided here in more detail.

**Step 1: Identify Rods in Boiling Transition**

[

]

**Step 2: Determine FEFFBT for Each Data of Interest**

[

]

[

]

[  
 ]

### Steps 3-6: Additive Constant Determination

[

]

The results of evaluating the Additive Constants for use with the ATRIUM-9B and ATRIUM-10 fuel designs are presented in Tables 3.10 and 3.11.

#### 3.6.2 Additive Constant Uncertainty

Determining Additive Constants Uncertainty provides for combining the standard deviations of the FEFFBT using a propagation of error method. This process employs the approach used for ANFB. Two major factors contribute to the overall uncertainty of the Additive Constants. These are 1) within test variability and 2) between test variability. The process includes contributions from the cosine axial tests and the upskew/downskew axial tests. The propagation of error method used is implemented by taking the square root of the sum of the squares of the errors. Errors in determining the uncertainty of the Additive Constants include the standard deviation of the FEFFBT for each test section and the difference between a specific value of FEFFBT and FEFF for a rod observed in boiling transition. Weighting factors of the number of rods observed in boiling transition and the number of tests are incorporated into the process to determine the total standard deviation of the Additive Constants. Specifically, where



SPCB Critical Power Correlation

---

N	=	Number of test bundles
M	=	Number of rods in these test bundles
FEFFBT	=	Best estimate FEFF for bundle
DFBT(i)	=	Standard deviation of FEFFBT for test bundle i
NOEX(i)	=	Number of experiments with test bundle i
NBT(J,i)	=	Number of boiling transition detected on rod j of test bundle i
DELTA(J,I)	=	Difference between FEFFBT and FEFF of rod j observed in boiling transition in test bundle i
DELTEX	=	Total standard deviation in Additive Constants
NTOT	=	Total number of experiments for all tests
NBTOT	=	Total number of rods in boiling transition

The Additive Constant uncertainty is calculated using



SPCB Critical Power Correlation

---

Evaluating the ATRIUM-9B data based on the verification data statistics (shown in Table 3.13) results in an Additive Constant uncertainty of [ ] for the ATRIUM-9B. Similarly the evaluation of the ATRIUM-10 data based on the verification data statistics results in an Additive Constant uncertainty of [ ] for the ATRIUM-10. The impact of the validation data needs to be included. This is accomplished by evaluating the validation data set for its FEFF values for each validation data point, then including the values in the mean and standard deviation of FEFF for each test section. To maintain a single process between ATRIUM-9B and ATRIUM-10, this process is chosen rather than using a replicate point process as was used for the ANFB-10 correlation.

Using the values of composite values FEFFBT and Standard Deviation of FEFFBT (Table 3.14) for the respective fuel designs, the overall uncertainty for Additive Constants is determined to be [ ]

### 3.6.3 High Local Peaking

A series of dryout tests were performed on the ATRIUM-9B and the ATRIUM-10 to determine the effect of high local peaking on the Additive Constant methodology. Because of physical manufacturing limits on the test section heater rods, the highest local peaking attained was 1.45. The axial power profile for the test assembly was 1.4 peak to average cosine.

[

]



The Additive Constant uncertainties determined by the Additive Constant methodology are  
[ ] While this conclusion may be drawn from examining the plots, a statistical analysis provides a more rigorous process for demonstrating this. Therefore, a Bartlett test (Reference 3.3, page 802) was applied based on data for each test over the same range of test conditions. The Bartlett test considers the null hypothesis that the variance of each data set is an estimate of the same population variance. The result of this test affirms that the null hypothesis cannot be rejected.

[

]

SPCB Critical Power Correlation

---

The uncertainty used for rods peaked greater than 1.5 is then determined by the square root of the sum of the squares of the Additive Constant uncertainty and its respective incremental uncertainty. This is the same method used in the approved ANFB-10 methodology (Reference 3.4).

The Additive Constants for the ATRIUM-9B and ATRIUM-10 designs are applied to the local peaking patterns for the test assemblies [

]

[

] A further measure of the conservatism of the SPCB correlation is observed by comparing the FEF values for the assembly with the FEFBT that places the ECPR of the assembly at 1.0. This allows the determination of the number of rods calculated to be in boiling transition for each data of each test section. Comparing the calculated number of rods in boiling transition with the number of rods in boiling transition, an over estimate of the number of rods in boiling transition is determined. The ratio of the number of rods calculated to be in boiling transition to the number of rods observed to be in boiling transition is equal to [

] when using the SPCB critical power correlation.

### 3.7 *SPCB Predictions Compared with Measurements*

The predicted values of critical power are compared with the measured values and presented in Figure 3.39.

### 3.8 **References**

- 3.1 G. J. Hahn, S. S. Shapiro, *Statistical Models in Engineering*, Wiley, 1967.
- 3.2 H. W. Lilliefors, *On the Kolmogorov-Smirnov Test for Normality with Mean and Variance Unknown*, Journal of the American Statistical Association, Vol. 62, June 1967.
- 3.3 *Statistical Methods for Nuclear Material Management*, NUREG/CR4604, PNL-5849, December 1988.
- 3.4 EMF-1997(P)(A) and Supplement 1, *ANFB-10 Critical Power Correlation*, Siemens Power Corporation, July 1998.



#### 4.0 Correlation Validation

The development of the SPCB 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 correlating data set was used to verify that the correlation had a proper fit to the data. In this context, the data set used for correlation development is termed the verification set.

The process for validating the SPCB critical power correlation contains several steps. In accordance with the criteria set forth in Reference 4.1, 20 percent of the data was set aside and defined to be a validating set of data. The remaining 80 percent was used to develop the critical power correlation. In addition, data acquired during the correlation development process was used only for validation. Further, data obtained for an assembly design that [

] was used only for validation. The SPCB critical power correlation was further validated by comparing its prediction with the measurements made for transient critical power tests.

#### 4.1 *Assessment of ATRIUM-9B/ATRIUM-10 Critical Power Data*

Information presented in Section 3 provided a combined characteristic for the SPCB correlation based on the evaluation of the verification and validation database for the ATRIUM-9B and ATRIUM-10 fuels. Specific evaluation showed that the Additive Constant uncertainties were unchanged when the entire database was considered versus when only the verification database was considered. This section intentionally examines the validation database.

##### 4.1.1 Comparison of ATRIUM-9B Verification and Validation Data

The statistical comparison of the ATRIUM-9B verification and validation data sets is presented in Table 4.1.

Graphically, the predicted versus measured critical power for ATRIUM-9 Validation data is shown in Figure 4.1, while the behavior of the ECPR with flow is shown in Figure 4.2.

SPCB Critical Power Correlation

---



4.1.2 Comparison ATRIUM-10 Verification and Validation Data

The statistical comparison of the ATRIUM-10 verification and validation data sets is presented in Table 4.3.



The behavior of ECPR versus Flow is shown in Figure 4.4.

#### 4.2 *Validation with Alternate Design*

The SPCB 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. Part length rod positions differ in the ATRIUM-10 with [ ] rods and the ATRIUM-10P with [

] All part length rods occur one row in from the channel. The data set contains 316 data points. The result of the tests shows that the mean ECPR is [

] Figure 4.5 shows the predicted versus measured critical power for these tests. Figure 4.6 is an example of one of the test assembly's critical power versus subcooling plots.

SPCB Critical Power Correlation

---







#### 4.3 *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 with cosine and upskew profiles were performed to reconfirm this for the ATRIUM-10 when using the SPCB critical power correlation.

The ATRIUM-10 [ ] transient critical power tests were performed for test assembly 17.8 (cosine power shape), test 29.5 (upskew axial power shape), and 48.1 (upskew axial power shape). Thirty-two transient tests were evaluated: [

] For comparison, the steady-state performance of the ATRIUM-10 as measured and as predicted by SPCB is given in Figure 4.7. The SPCB correlation correlates well with the respective tests.

[

]

┌

└

The transient tests of interest are both simulated load rejection without bypass (LRNB) events that consist of power and pressure ramps and flow decay; and simulated pump trip 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 4.8 shows the forcing function characteristics for a typical LRNB test while Figure 4.9 shows the comparable forcing function characteristics for a typical pump trip event.

SPCB Critical Power Correlation

---



Figure 4.10 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. Boiling transition occurs slightly upstream of [ ] In the STS-17.8 tests, boiling transition normally occurred [ ] In the STS-29.5 and 48.1 tests, it occurred [ ]



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

]

The SPC transient thermal hydraulic code XCOBRA-T (References 4.2 and 4.3), is used to predict the transient test results using the SPCB steady-state critical power correlation. XCOBRA-T calculated the fluid conditions axially at time steps as small as [ ]. 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 CHF is calculated at each axial position and time step, then compared to the corresponding rod heat flux. The ratio of the critical heat flux to the rod heat flux is CHF<sub>R</sub>. A MCHF<sub>R</sub> of unity during the transient signifies boiling transition. Although applying the steady-state critical power correlation is considered conservative, SPCB 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 overpredict a boiling transition state within the range of defined uncertainties. Thus, during transient test conditions, dryout may not be predicted for all cases because of the defined uncertainties.

Thirty-two transient tests modeling the ATRIUM-10 geometry in simulated LRNB or pump-trip events were evaluated. The two STS-17.8 tests had nearly the same forcing functions, with only the initial power level differing. The nine STS-29.5 tests consisted of six 100 percent flow cases and three 80 percent flow cases. Within these groups, there was a variation in inlet subcooling. The 21 STS-48.1 tests consisted of nine LRNB tests with five LRNB at 100 percent flow and four LRNB tests at 80 percent flow, and 12 pump-trip tests. The 12 pump-trip tests included eight at 100 percent flow and four at 80 percent initial flow. The initial subcooling varied among these tests. Table 4.4 summarizes initial state conditions for all the transient tests.



The XCOBRA-T calculations were performed using nominal Additive Constants. The results using nominal (design) Additive Constants are summarized in Table 4.5. Figure 4.11 compares the measured and calculated time of boiling transition. The comparisons demonstrate that the STS-17.8, STS-29.5 and STS-48.1 transient tests are conservatively predicted. These results validate that Additive Constants can be derived from steady-state tests and applied to transient conditions.





#### 4.3.1 Conclusions

The XCOBRA-T analyses calculated dryout in the [ ] evaluated at nominal design conditions. When considering uncertainties, all transient results were conservatively predicted. This validation confirms that the use of the SPCB steady-state dryout correlation is appropriate for use in evaluation of transient events. Furthermore, this evaluation provides a validation that Additive Constants can be derived from steady-state tests and applied to transient conditions.

#### 4.4 **References**

- 4.1 EMF-2022 Revision 0, *Correlation Development Guideline*, Siemens Power Corporation, February 1998.
- 4.2 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 Core Analysis*, Exxon Nuclear Company, February 1987.
- 4.3 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.

## 5.0 SPCB Database

The SPCB database contains [

] to validate the correlation. All data was taken at the AREVA test facility at Karlstein, Germany.

### 5.1 Facility Description

All dryout testing for the ATRIUM-9B and ATRIUM-10 assembly was performed at the AREVA thermal hydraulic test loop at Karlstein, Germany. Figure 5.1 shows that the thermal hydraulic test facility is a high-pressure-water heat-transfer loop containing a test vessel, as shown in Figure 5.2, with the test bundle 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 psi and 680°F. The DC power supply consists of four thyristor controlled rectifier models, 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 and channel. After the test, the data is archived on magnetic tapes. Table 5.1 shows the test loop uncertainties.

During the dryout test, the dryout power is determined manually 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.

## 5.2 *Test Bundle Descriptions*

The dryout test bundles are full array assemblies designed to represent the production fuel assembly as close as possible. The rod bundle 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 5.3). 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.

### 5.2.1 ATRIUM-9B

The ATRIUM-9B test bundle consists of a square array of rods supported at fixed axial locations by ULTRAFLOW spacers and with one 1.516 in. square cross-section water channel. The array contains 72 full length rods.

The test bundles have the following characteristics (Table 5.2 summarizes the physical characteristics of the ATRIUM-9B test assembly).

[

]

Three axial power profiles were tested during the ATRIUM-9B dryout test series. The STS-12, STS-33, STS-37, and STS-40 series were performed on a [ ] peak-to-average chopped cosine axial, the STS-35 series was [ ] peak-to-average downskew axial, and the STS-35 and STS-38 series was performed on a [ ] peak-to-average upskew axial power profile. Figure 5.5 represents the rod axial power profiles.

#### 5.2.2 ATRIUM-10

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

The test bundles had the following characteristics (Table 5.3 summarizes the physical characteristics of the ATRIUM-10 test assembly):

[

]

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

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 series was performed on a [ ] peak to average upskew axial power profile. Figure 5.5 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.

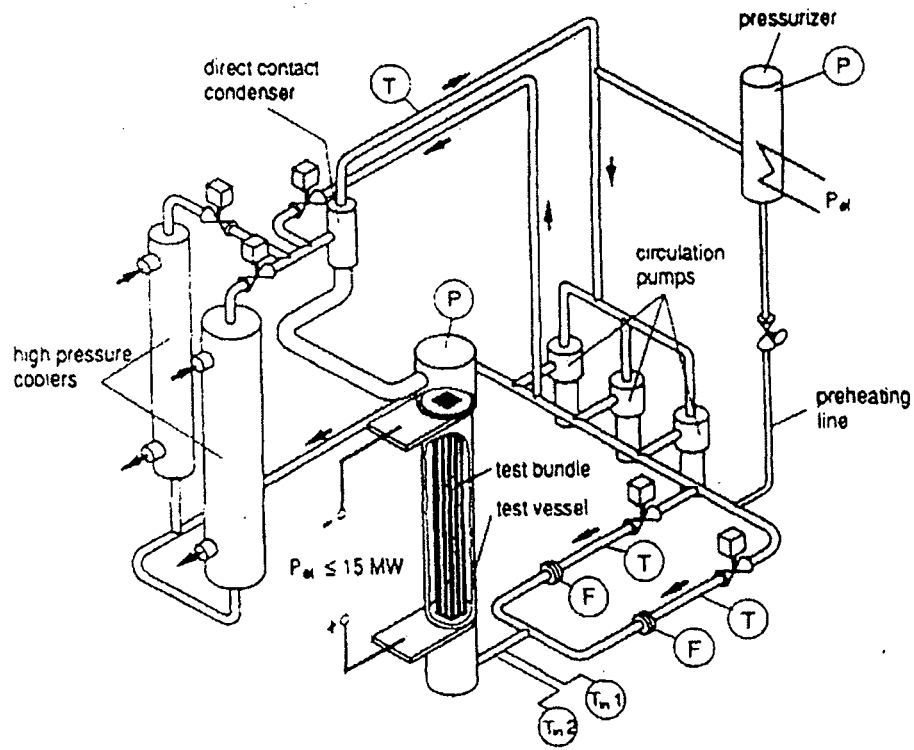


Figure 5.1 Karlstein Thermal Hydraulic Test Loop

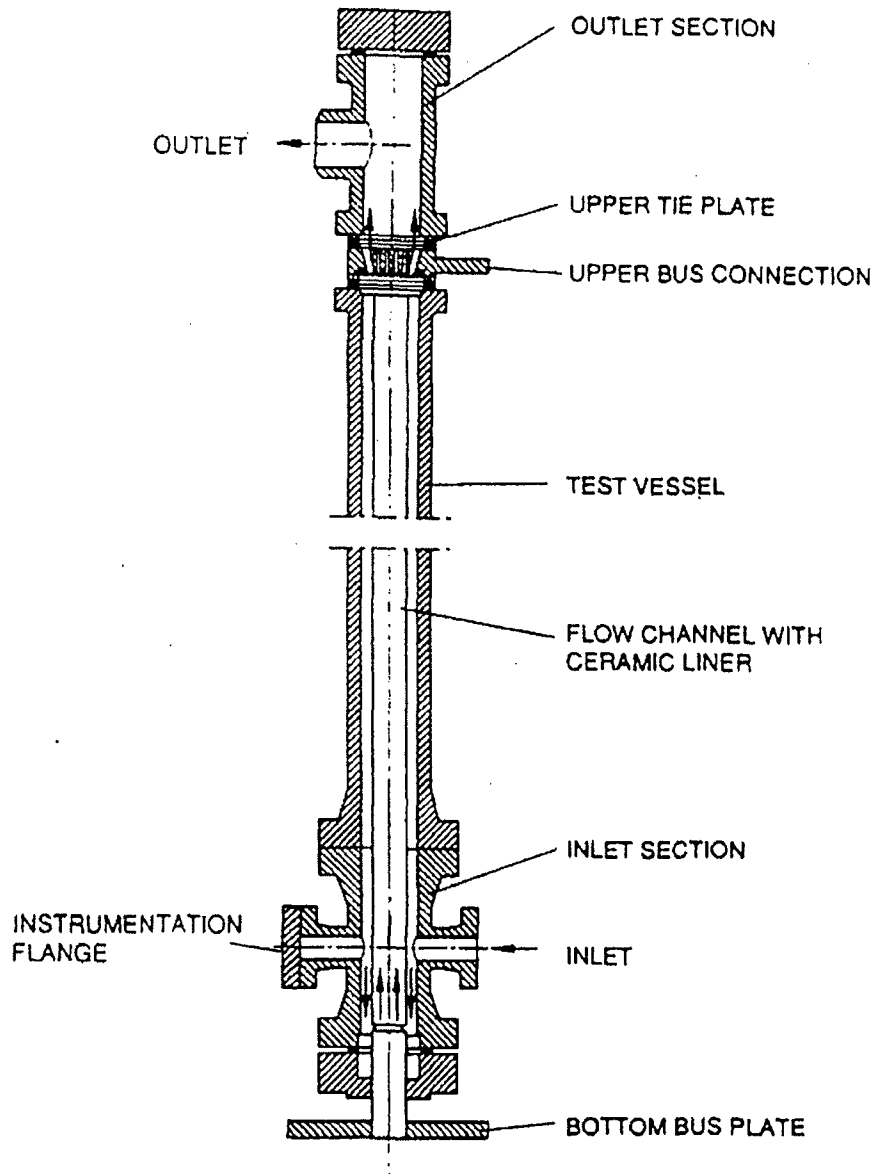


Figure 5.2 Test Vessel

SPCB Critical Power Correlation

---



SPCB Critical Power Correlation

---





SPCB Critical Power Correlation

---





### 5.3 *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.

#### 5.3.1 Radial Peaking Profiles

A conservative assumption is made in the SPC dryout methodology that any rod position in which a symmetric rod is not driven into dryout is assumed to have been in dryout at its highest local peaking. 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 bundle. The ATRIUM-9B bundle has octant symmetry, so that peaking nine individual rods of twelve symmetric positions with five driven into dryout adequately describes the assembly. The rod positions tested were the corner rods, the peripheral rods, the rods in the middle row, and the rods around the internal water canister. The ATRIUM-10 has half-bundle symmetry along the diagonal of the bundle. 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 bundle were driven into dryout at different local peaking factors. To demonstrate that the ATRIUM-9B bundle behaves the same, two tests were run at the same location with a local peaking factor of [ ] Only two tests needed to be performed on the ATRIUM-9B because the purpose was to demonstrate that the algorithm for the Additive Constant methodology behaved the same for the ATRIUM-9B as for the ATRIUM-10. Section 3.1 of this report documents the statistical results of these tests.

### 5.3.2 Axial Power Profile

Three axial power profiles were tested during dryout testing: [ ] peak to average cosine, [ ] peak to average upskew, and [ ] peak to average downskew (see Figure 5.5). Because cosine power shapes are representative of much of the plant operation they are the most prevalent type of testing; other axial power profiles are used to check the axial power corrector used. Dryout occurs only after the peak of an axial power profile. For the ATRIUM-9 and 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 bundle; 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 bundle 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.

The ANFB (Reference 5.1) correlation was developed using uniform axial power data, then the axial power corrector was developed using cosine and upskew data. A uniform axial power profile always results in dryout occurring at the exit of the bundle and, therefore, would easily provide accurate data on the enthalpy at the plane of boiling transition. Because uniform axial power profile rods were not available for the ATRIUM-9B and ATRIUM-10 assemblies and the manufacture of those rods requires a long lead time, the SPCB was developed by calculating the enthalpy at the plane of boiling transition of each individual case.

### 5.3.3 Thermal Hydraulic Test Conditions

The database for the ATRIUM-10 was obtained during the ANFB-10 correlation development (Reference 5.2). It contains data over a range of [ ] a subcooling of [ ] and pressures ranging from [ ] The ATRIUM-9B originally had 125 data points taken over a range of 0.05 Mlb/hr to 0.15 Mlb/hr, a subcooling of 8 Btu/lbm to 90 Btu/lbm at a system pressure of 1000 psia. This database was expanded to include the full range of pressures and flows. Table 5.4 summarizes the tests and test conditions used in the development, verification, and validation of the SPCB correlation.

#### 5.3.4 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 may be 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 may be 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. For example, for the validation test, STS-38.3, a test plan as developed in Reference 5.1 was used. Because of the small amount of data available for the ATRIUM-9B, the testing performed for the SPCB correlation included all pressures and flows. Because the database for the ATRIUM-10 was

larger, not only did some tests have all pressures and flows, statistical design of experiments was also used (Reference 5.2).

#### 5.4. **SPCB Data**

The database for SPCB contains [ ] peaking patterns performed on test sections with cosine, upskew, and downskew axial power profiles for ATRIUM-9B and ATRIUM-10 designs. The correlation database contains [ ] data points taken over the range of applicability of the SPCB correlation. Of the [ ] data points, [ ] form the information used during the correlation process and [ ] data points validate the correlation. Table 5.5 contains the measured and calculated critical power ratio of the verification and validation database. Figures 5.6 through 5.89 present the dryout test peaking pattern and its associated inlet subcooling versus critical power plot for both the test data and the SPCB prediction of the test data.

Pages 5-14 through 5-167 are proprietary in toto.

5.5 **References**

- 5.1 ANF-1125(P)(A), *ANFB Critical Power Correlation*, Advanced Nuclear Fuels Corporation, April 1990.
- 5.2 EMF-1997(P)(A), *ANFB-10 Critical Power Correlation*, Siemens Power Corporation, October 1998.