

EMF-2209(NP)(A)
Revision 2

SPCB Critical Power Correlation

September 2003

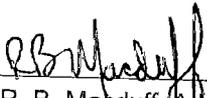
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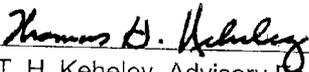
Framatome ANP, Inc.

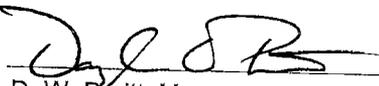
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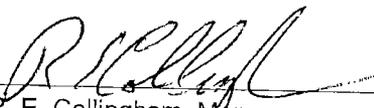
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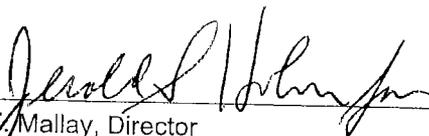
SPCB Critical Power Correlation

Prepared: 
R. B. Macduff, Advisory Engineer
Systems Analysis Methods
Date 5 NOV 03

Reviewed: 
T. H. Keheley, Advisory Engineer
Systems Analysis Methods
Date 6 Nov '03

Concurred: 
D. W. Pruitt, Manager
Systems Analysis Methods
Date November 11 '03

Approved: 
R. E. Collingham, Manager
Methods Development
Date Nov 11, '03

Approved: 
J. F. Mallay, Director
Regulatory Affairs
Date 11/17/03

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**U.S. Nuclear Regulatory Commission
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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

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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,

A handwritten signature in black ink, appearing to read 'Herbert N. Berkow', written in a cursive style.

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

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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,

A handwritten signature in black ink, appearing to read "Stuart A. Richards", with a large, sweeping flourish extending to the right.

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_{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 (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^2), 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^2) 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.

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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.

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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.

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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.

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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.

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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 ²)	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, Ink, 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.

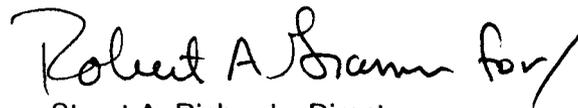
Pursuant to 10 CFR 2.790, we have determined that the enclosed SE does not contain proprietary information. However, we will delay placing the SE in the public document room for a period of ten (10) working days from the date of this letter to provide you with the opportunity to comment on the proprietary 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.

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,

A handwritten signature in black ink that reads "Robert A. Richards for". The signature is written in a cursive, flowing style.

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-ft²), 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.

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

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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 ²)	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, Ink, 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



FRAMATOME ANP

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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 Ω for the SPCB database provide support.

Question 2

For SPCB Critical Power Correlation, Revision 0, page 2-8, equation 2.14, Ω 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 Ω decreasing. As Ω 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 Ω is dependent on FEFF, and Tong Factor is dependent on Ω , and Tong Factor is inversely proportional to the critical heat flux, i.e., critical power is inversely proportional to FEFF.

Response 3

Ω 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

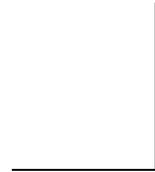
It is not clear to the staff what the 3rd. paragraph on page A-2 of the June 20 submittal is trying to communicate, and there are too many concerns to list in one question. Please be prepared to provide a much needed clearer depiction of what it is being conveyed by this paragraph.

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 Ω is shown at the bottom of the page, please provide further qualitative and quantitative background for the new equation for Ω_{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 UO₂ 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 UO₂ 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.



FRAMATOME ANP

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FRAMATOME ANP, Inc.

June 20, 2003

NRC:03:039

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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-ft². In reality, the critical heat flux should not decrease at all for this situation; it should remain at approximately 0.13 MBtu/hr-ft².

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



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September 4, 2002
NRC:02:042

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ATTN: Chief, Planning, Program and Management Support Branch
U.S. Nuclear Regulatory Commission
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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



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, " Ω " (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. [

]

SIEMENS

April 24, 2000
NRC:00:024

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

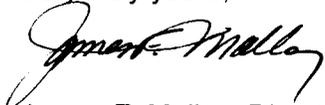
SER Condition for EMF-2209(P) Revision 1, "SPCB Critical Power Correlation"

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,



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

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:

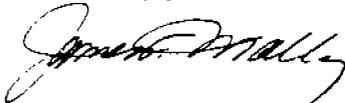
1. The SPCB correlation is applicable to Siemens Power Corporation ATRIUM¹-9B and ATRIUM-10 fuel designs with a local peaking factor no greater than 1.5.
2. If, in the process of calculating the MCPR safety limit, the local peaking factor exceeds 1.5, an additional uncertainty of 0.026 for ATRIUM-9B 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 ²)	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

¹ ATRIUM is a trademark of Siemens.

Siemens Power Corporation

2101 Horn Rapids Road
Richland, WA 99352

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

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.

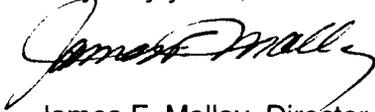
In Reference 1, the NRC requested additional information to facilitate the completion of its review of the Siemens Power Corporation topical report on the SPCB correlation (see Reference 2). Responses to this request are provided in two attachments: one proprietary and one nonproprietary.

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><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>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><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>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><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><i>a. What is the basis for these lower and upper limits?</i></p> <p><i>b. Provide technical justification for the interpolation between these bounds.</i></p> <p>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>The fitting of the coefficients for the A and B functions was performed in the following manner:</p> <p>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-ft² using the low flow set of coefficients and A would be determined at the high mass velocity lower bound of 0.42 Mlb/hr-ft² 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 \bar{G}.</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	<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> Response: The values of the limits are depicted in Table 1.2 on page 1-3.
10	<i>On page 2-36, second paragraph, it is stated that "corresponding quality distributions are artificially increased." What are "quality distributions?"</i> Response: For each nodal enthalpy, a thermodynamic quality can be determined. As quality is a parameter in the Hensch-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.
11	<i>Is "inlet mass velocity" inter-changeable with "inlet mass flow rate?"</i> 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.
12	<i>Figure 3.8 on page 3-15, the x-axis is labeled "Active flow." What is "Active Flow?"</i> 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.
13	<i>Does Figure 3.39 on page 3-41 include all the data for the ATRIUM-9B and that for the ATRIUM-10?</i> 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.
14	<i>Page 4-9, Figure 4.7, please provide justification for the difference in magnitude at high inlet flow.</i> 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

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.	<p>SPCB-Correlation Development RAIs.</p> <p>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).</p>																								
Q	<p>SPCB Statistical RAIs</p>																								
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> <p>Response: See Tables 3.1, 4.1, 4.2, and 4.3 in Revision 1 of EMF-2209(P).</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" data-bbox="305 989 1239 1117"> <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> <p>Comparison of means: $0.0204/\sqrt{(1876)} = 0.0005$ $0.0241/\sqrt{(781)} = 0.0009$ $0.0233/\sqrt{(316)} = 0.0013$</p> <table border="0" data-bbox="305 1244 842 1383"> <thead> <tr> <th>Group</th> <th></th> </tr> </thead> <tbody> <tr> <td>Verification</td> <td>0.992 ± 0.0005</td> </tr> <tr> <td>Validation</td> <td>0.983 ± 0.0009</td> </tr> <tr> <td>Validation2</td> <td>1.002 ± 0.0013</td> </tr> </tbody> </table> <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 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>	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		Verification	0.992 ± 0.0005	Validation	0.983 ± 0.0009	Validation2	1.002 ± 0.0013
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Q.	SPCB-Correlation Development RAIs.
3	<p data-bbox="302 374 1321 463"><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="302 495 1321 708">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="302 719 1321 787"><i>Pages 2-4 thru 2-10 (Tables 2.1 thru 2.7). Is "G" in these tables the same as G bar (\bar{G}) given in, say, equation (2.7)?</i></p> <p data-bbox="302 819 1321 863">Response: Yes. Revision 1 to EMF-2209(P) has (\bar{G}) in these tables.</p>
5	<p data-bbox="302 874 1321 902"><i>Page 2-6, Equation 2.12. The coefficient "f1" in equation 2.12 is not defined.</i></p> <p data-bbox="302 934 1321 1021">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="302 1032 1321 1100"><i>Page 2-13, Equation 2.23. Identify/explain how the coefficients 0.624 and 0.314 were obtained.</i></p> <p data-bbox="302 1153 1321 1304">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="302 1315 1321 1342"><i>Page 2-26, Figure 2.3. Symbols (like X, square, and diamond) need a legend.</i></p> <p data-bbox="302 1374 1321 1402">Response: See response to question 7 of SPCB-Correlation Development RAIs.</p>
8	<p data-bbox="302 1412 1321 1481"><i>Page 3-1, Table 3.1. Show how Sigma (given as 0.021) for ATRIUM-9B was derived.</i></p> <p data-bbox="302 1495 1321 1617">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>Page 3-1, Table 3.1:</p> <ul style="list-style-type: none"> 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. <p>Provide similar entries, separately for each fuel, and for all tests of the same profile.</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>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?</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"> σ_{TRUE}^2 = True between mean variance $\sigma_{\text{BETWEEN}}^2$ = Total between test variance $\sigma_{\text{WithinTest}}^2$ = Within test variance ESS = Effective sample size <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="302 670 1313 946"> <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																														
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11	<p><i>Pages 3-2 and 3-3, equations for m_2, m_3, m_4, β_1, and β_2. 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>$m_2 = 0.000491$ $\sqrt{\beta_1} = 0.004488$ $\beta_2 = 2.83$</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><i>Pages 3-4 thru 3-29, Figures 3.1 through 3.36. Indicate the sample size associated with each figure.</i></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="305 368 1313 427"><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="305 459 1313 676">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="305 680 1313 710"><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="305 742 1313 804">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="305 808 1313 902"><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="305 934 1313 1023">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="305 1027 1313 1057"><i>Were there any outliers in the evaluation, and if so, what was their disposition?</i></p> <p data-bbox="305 1089 1313 1183">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="305 1215 1313 1338">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>









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 , β_1 , and β_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

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)

EMF-2209(NP)
Revision 2

SPCB Critical Power Correlation

November 2003

Nature of Changes

Item	Page	Description and Justification
1.	All	Siemens has been changed to Framatome ANP
2.	2-6	Added three terms to Equation 2.12
3.	2-7	Added values for the constants in the above three terms to Table 2.5
4.	2-8	Footnotes have been updated
5.	2-8	Revision 2 describes a reduction in the conservatism included in the SPCB correlation for designs with an upper natural blanket. Because Revision 1 describes a more conservative implementation of SPCB, it can continue to be used and referenced until it is convenient to transition to Revision 2.

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1.0 Introduction and Summary

This document describes SPCB, Framatome ANP, Inc.'s (FANP) 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 Framatome ANP.

peak-to-average upskew and downskew. The database was compiled from tests performed exclusively at the Framatome ANP 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 ²)	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_{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_{bt} is the enthalpy at the plane of boiling transition. For the SPCB application another term, C, is added to h_{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

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

In this expression, Ω is a function of mass velocity and heat flux gradient, $q''(Z)$ is the axial heat flux, ℓ_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







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, " Ω " (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. [

]

The base non-uniform axial factor is given by

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

where

- $q''(Z)$ = Axial heat flux
- Z = Axial position on fuel rod
- Ω^\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_{Base} , by the following pressure, flow, and enthalpy gradient term.

[_____]

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

† Ω is taken as the []

‡ []

§ []



* []
† []

]

]

2.2.2 Gradient Function

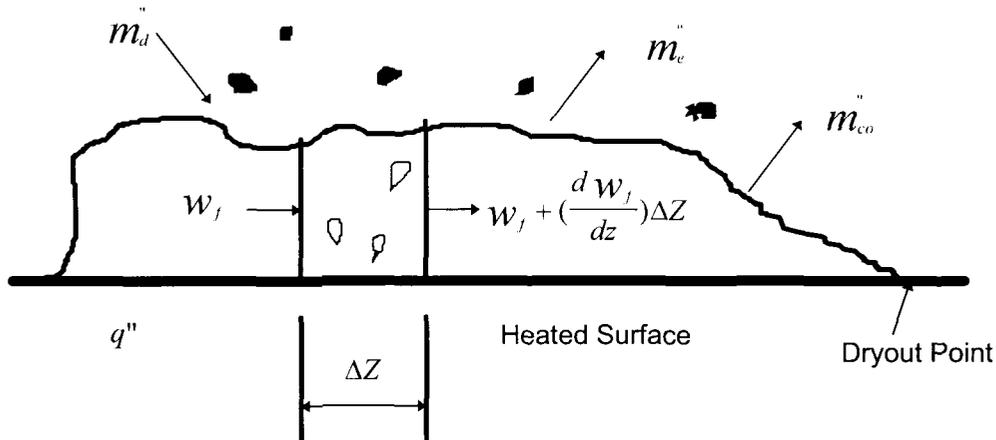
This section describes the GRAD1^{*} term and how it is calculated and used in evaluating critical power. A gradient type of term was suggested as being important for dryout in References 2.4 and 2.5.

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.

[

]

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

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

Therefore, the spatial derivation of the flow gradient is

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

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.

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

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, ℓ , 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^{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

SPCB Critical Power Correlation



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

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



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, ℓ . 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.



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 Δ CPR Analysis

The Δ 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 Δ 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 CHF. This CHF 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

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

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.







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



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.

[

]

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).

3.6.4 SPCB Conservatisms

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 FEFB values for the assembly with the FEFB 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.



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.

SPCB Critical Power Correlation

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.



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.



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 CHFR. A MCHFR 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 Framatome ANP test facility at Karlstein, Germany.

5.1 Facility Description

All dryout testing for the ATRIUM-9B and ATRIUM-10 assembly was performed at the Framatome ANP 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.

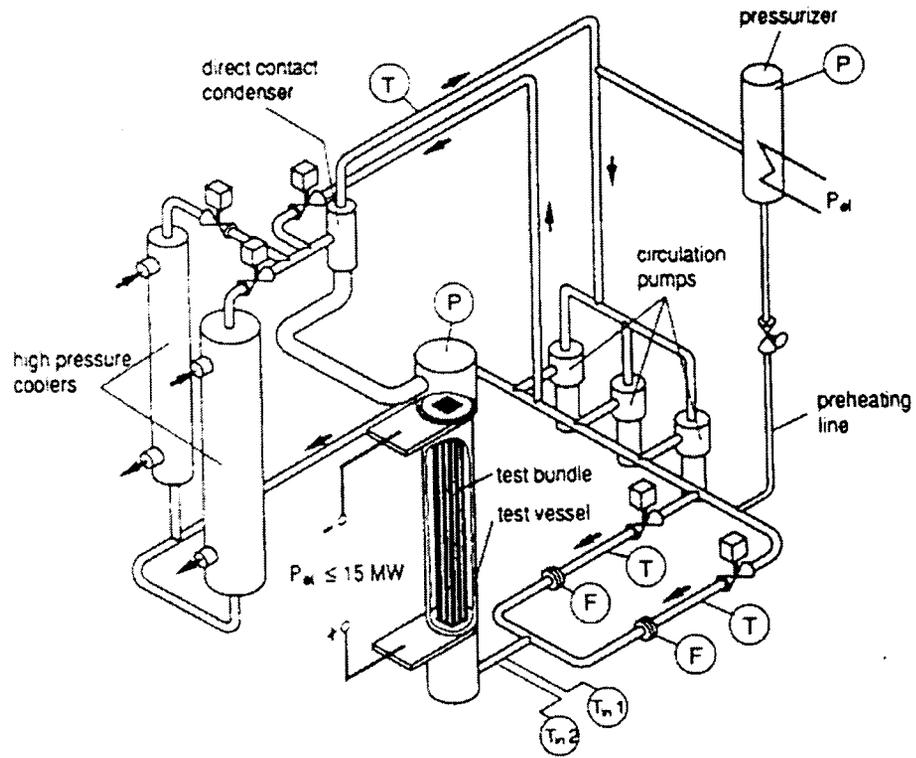


Figure 5.1 Karlstein Thermal Hydraulic Test Loop

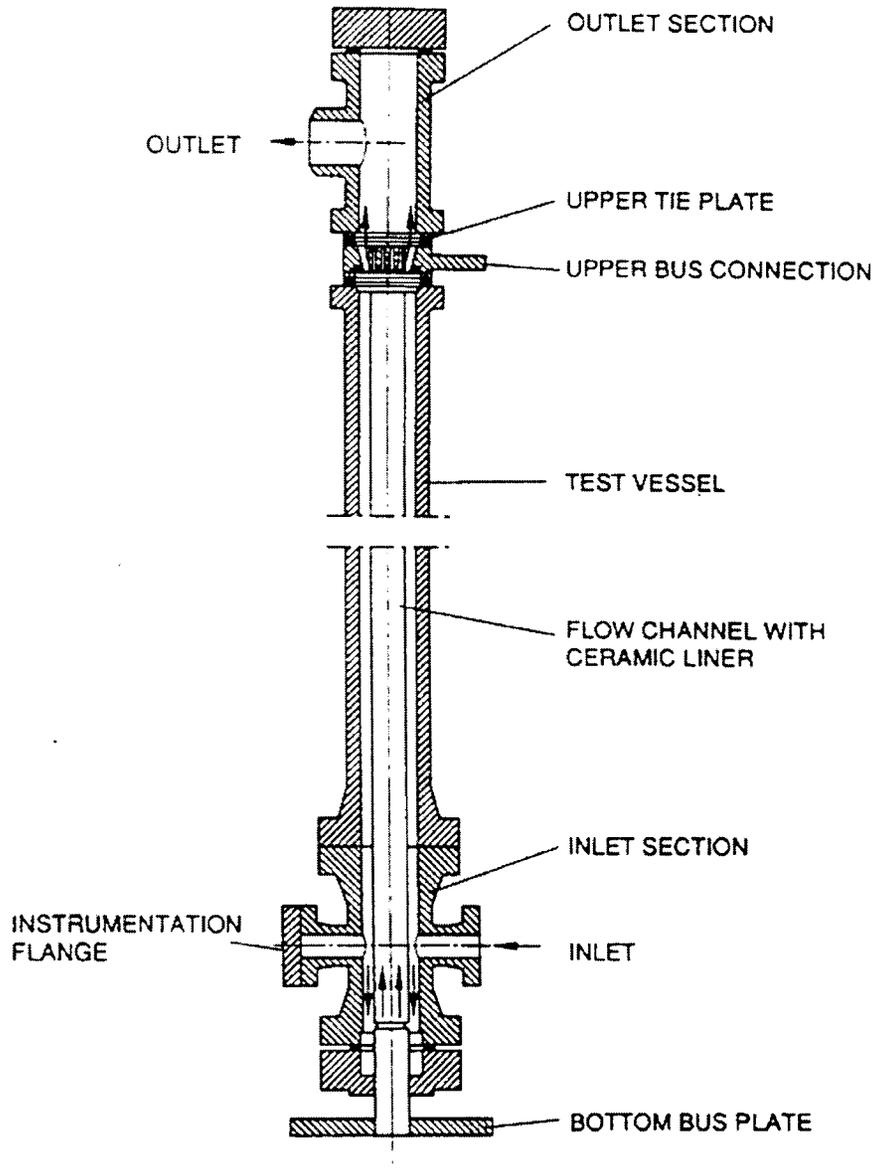


Figure 5.2 Test Vessel





1
1
1





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.

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