Progress Energy

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January 24, 2008

SERIAL: BSEP 08-0009
TSC-2006-06

10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Docket Nos. 50-325 and 50-324/License Nos. DPR-71 and DPR-62
Additional Information in Support of Request for License Amendments Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA Fuel (NRC TAC Nos. MD4063 and MD4064)

References:

1. Letter from James Scarola to the U.S. Nuclear Regulatory Commission (Serial: BSEP 06-0129), "Request for License Amendment Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel," dated January 22, 2007 (ADAMS Accession Number ML070300570)
2. Letter from James Scarola to the U.S. Nuclear Regulatory Commission (Serial: BSEP 07-0053), "Additional Information in Support of Request for License Amendment Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel," dated June 21, 2007 (ADAMS Accession Number ML071840214)
3. Letter from Ben Waldrep to the U.S. Nuclear Regulatory Commission (Serial: BSEP 07-0067), "Additional Information in Support of Request for License Amendment Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel," dated July 18, 2007 (ADAMS Accession Number ML072070305)
4. Letter from James Scarola to the U.S. Nuclear Regulatory Commission (Serial: BSEP 07-0108), "Additional Information in Support of Request for License Amendments Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel," dated October 15, 2007 (ADAMS Accession Number ML072950366)

Ladies and Gentlemen:

By letter dated January 22, 2007, as supplemented by letters dated June 21, 2007, July 18, 2007, and October 15, 2007, Carolina Power & Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc., requested a license amendment to revise the

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NRR

Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed changes are being made to support the BSEP transition to AREVA manufactured fuel assemblies.

On November 28, 2007, and January 7, 2008, via electronic mail, the NRC requested additional information regarding the changes being proposed. Enclosure 1 provides responses to the information requested. The responses in Enclosure 1 include information that AREVA considers to be proprietary, as defined in 10 CFR 2.390. AREVA, as the owner of the proprietary information, has executed the affidavit provided in Enclosure 2, which identifies that the enclosed proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. AREVA requests that the enclosed proprietary information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390. A non-proprietary version of the responses is provided in Enclosure 3.

There are no regulatory commitments associated with this submittal. Please refer any questions regarding this submittal to Mr. Randy C. Ivey, Manager - Support Services, at (910) 457-2447.

I declare, under penalty of perjury, that the foregoing is true and correct. Executed on January 24, 2008.

Sincerely,

Benjamin C. Waldrep

WRM/wrm

Enclosures:

1. Response to November 28, 2007, and January 7, 2008, NRC Requests for Information
   (Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)
2. AREVA Affidavit Regarding Withholding Proprietary Information from Public Disclosure
3. Response to November 28, 2007, and January 7, 2008, NRC Requests for Information (Non-Proprietary Version)

cc (with Enclosures 1, 2, and 3):

U. S. Nuclear Regulatory Commission, Region II
ATTN: Mr. Victor M. McCree, Regional Administrator (Acting)
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
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U. S. Nuclear Regulatory Commission
ATTN: Mr. Joseph D. Austin, NRC Senior Resident Inspector
8470 River Road
Southport, NC 28461-8869

U. S. Nuclear Regulatory Commission **(Electronic Copy Only)**
ATTN: Mr. Stewart N. Bailey (Mail Stop OWFN 8B1)
11555 Rockville Pike
Rockville, MD 20852-2738

cc (with Enclosures 2 and 3):

Chair - North Carolina Utilities Commission
P.O. Box 29510
Raleigh, NC 27626-0510

Ms. Beverly O. Hall, Section Chief
Radiation Protection Section, Division of Environmental Health
North Carolina Department of Environment and Natural Resources
3825 Barrett Drive
Raleigh, NC 27609-7221

AREVA Affidavit Regarding
Withholding Proprietary Information
from Public Disclosure

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

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

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

3. I am familiar with the AREVA NP information contained in the attachment to Progress Energy letter BSEP 08-0009 with subject entitled "Additional Information in Support of Request for License Amendments Regarding Linear Heat Generation Rate and Core Operating Limits Report," dated January 24, 2008 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

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

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

accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

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

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

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

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

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

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

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

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

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

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

SUBSCRIBED before me this 24th

day of January, 2008.

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

SHERRY L. MCFADEN
COMMONWEALTH
VIRGINIA
NOTARY PUBLIC

Response to November 28, 2007, and January 7, 2008,
NRC Requests for Information
(Non-Proprietary Version)

Responses to November 28, 2007, and January 7, 2008, NRC Requests for Information

By letter dated January 22, 2007, as supplemented by letters dated June 21, 2007, July 18, 2007, and October 15, 2007, Carolina Power & Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc., requested a license amendment to revise the Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed changes are being made to support the BSEP transition to AREVA manufactured fuel assemblies.

On November 28, 2007, and January 7, 2008, via electronic mail, the NRC requested additional information regarding the changes being proposed. The responses below include information that AREVA considers to be proprietary as defined in 10 CFR 2.390. The AREVA proprietary information is identified by a double underline inside double square brackets. [[This sentence is an example.]] AREVA, as the owner of the proprietary information, has executed an enclosed affidavit which identifies that the identified proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. AREVA requests that the identified proprietary information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390. A non-proprietary version of these responses is also provided.

NRC Question 1 - General

Based on the information provided for the license amendment (References 1 through 8), it appears that certain generic and specific analyses were performed for one BSEP unit and applied to both. With regard to those analyses intended to apply to both units, please provide a description of key differences in operation and system configuration between Units 1 and 2, and show that the analyses bound both units.

For Unit 2 transition, the NRC requests that cycle specific analysis reports (i.e. thermal-hydraulic design, fuel cycle design, reload safety analysis) be provided for staff confirmation 3 months prior to startup from the first refueling outage that uses AREVA fuel.

Response 1

The key differences in system configuration between BSEP Unit 1 and Unit 2 are in the core inlet region and the turbine bypass system. The orifice diameter in Unit 2 is smaller than Unit 1, 2.09 inches compared to 2.43 inches, and the turbine bypass system for Unit 2 has 10 valves whereas Unit 1 has 4 valves. Small variations in cycle energy requirements may introduce differences in neutronic design and operation; however, these differences would be minimal since both Units operate on 24 month fuel cycles.

The following analyses performed to support BSEP Unit 1 are applicable to BSEP Unit 2:

Long Term Anticipated Transient Without Scram (ATWS) Evaluation – The cycle to cycle variation in void coefficient and boron worth due to neutronic design changes will not significantly affect the long term ATWS evaluation.

Criticality Analysis – There is nothing in the operation or system configuration differences between the two units that impact the criticality evaluations presented in References 9 and 10. The reload fuel design for each cycle will be checked to ensure that the enrichment and gadolinia concentration are within the limits presented in Reference 9, and the enrichment and gadolinia concentration or k-effective are within the limits presented in Reference 10.

Radiological Accident Analyses – Fuel assembly and reactor core isotopic inventories used as input to design basis radiological accident analyses are applicable to BSEP Units 1 and 2, and bound both the ATRIUM™-10 and GE14 fuel designs.

Loss-of-Coolant Accident (LOCA) Analysis – The results of the LOCA analyses presented in References 3 and 4 are applicable for ATRIUM™-10 fuel at both Unit 1 and Unit 2. Unit specific analyses were performed to ensure that the system configuration differences were included in the analysis. As indicated in Reference 2, the LOCA analysis for Unit 2 is limiting. This is consistent with the GE14 LOCA analysis discussion presented in the Updated Final Safety Analysis Report, Section 6.3. The heatup portion of the analysis will be performed on a cycle specific basis to ensure that the acceptance criteria continue to be met.

The disposition of events will be reviewed for continued applicability for Unit 2. Except for the analyses identified above, other potentially limiting events will be analyzed for Unit 2. Based on the minimal differences between Units 1 and 2 and the justification for applicability of generic and specific analyses to both Units provided above, CP&L will include, for information only, the thermal-hydraulic design, fuel cycle design and reload safety analysis reports with our transmittal of the Core Operating Limits Report prior to startup from the first Unit 2 refueling outage that uses AREVA fuel.

NRC Question 2 - Power Distribution Uncertainty

During a recent review of a BWR extended power uprate and associated AREVA fuel methodologies, the adequacy of benchmark data associated with neutronic power prediction methods was questioned. The issue was resolved by increasing the power distribution uncertainties and propagating them into the SLMCPR calculation. Please discuss the applicability of this issue to BSEP and, if applicable, discuss the approach that will be taken to resolve it.

Response 2

Information concerning the applicability of AREVA methods, including radial power uncertainties, to the operating conditions of BSEP has been provided in the Reference 5 report. This issue is not applicable to BSEP because specific benchmarking analysis of both BSEP Units 1 and 2 operating at the extended power uprate (EPU) conditions demonstrated that the radial power uncertainty values presented in the Reference 11 topical report are very conservative for BSEP. Even if one uses a reduced correlation coefficient, as was done for other EPU licensing analyses, the BSEP specific uncertainty is significantly less than the value reported in the topical report. BSEP Safety Limit Minimum Critical Power Ratio (SLMCPR) analyses use the value based upon data from the topical report for D-lattice plants rather than a BSEP specific value. The use of a reduced correlation coefficient increases the BSEP specific value by 0.27%. This leaves a margin of 1.62% to the value reported in the topical report.

The primary reason for this conservatism is that BSEP has implemented gamma traversing incore probes (TIPs) for Local Power Range Monitor (LPRM) calibration. Gamma TIP systems have been demonstrated to result in smaller uncertainties as in the BSEP benchmarking results. The additional margin provided by the BSEP gamma TIP system to the D-lattice topical report uncertainty applicable to BSEP is greater than the penalty assigned to other EPU licensing analyses. Increasing the power distribution uncertainty is not necessary for the SLMCPR analysis of BSEP.

NRC Question 3 - Void-Quality Correlation

Please justify the application of the [[ ]] void-quality correlation to co-resident GE14 fuel.

Response 3

The [[ ]] void-quality correlation has been qualified by AREVA against both the FRIGG void measurements and ATRIUM™-10 measurements. The cross-sectional views of these two test channels are illustrated in Figure 3.1 along with the geometry of the GE14 fuel design. Despite the significantly different geometrical configurations between FRIGG and ATRIUM™-10, the behavior of the [[ ]] calculations when compared to the measured data is remarkably similar as illustrated in Figure 3.2. This similarity of results indicates that the [[ ]] void-quality correlation is applicable to a range of geometries larger than the differences between ATRIUM™-10 and GE14 and thus is equally applicable to the GE14 fuel design.

NRC Question 4 - ATWS Overpressure

Please explain the 28 psi decrease in the peak ATWS pressure provided in Reference 8. Based on the General Electric Report, NEDC-33039P, "Safety Analysis Report For

Brunswick Steam Electric Plant Units 1 and 2 Extended Power Uprate," August 2001, the peak ATWS overpressure result for EPU conditions is 1492 psig.

The staff recognizes that the potential biases between different code sets may potentially be significant; however, past ATWS overpressure results (i.e. Browns Ferry EPU) between GE and AREVA were in better agreement. Please attempt to quantify the effects of any bias, input, and/or assumption differences between the analyses in Reference 8 and NEDC-33039P.

Response 4

A review of the AREVA ATWS overpressurization analyses confirm that the same methodology was used in both the Browns Ferry and BSEP analyses. It is noted that for the Browns Ferry analysis, the same inputs, including operating conditions and set points, were used in both the General Electric (GE) and AREVA analyses (i.e., they were performed using the same EPU design values). While an effort was made to remain consistent with the GE EPU analyses, the AREVA analyses were performed using parameters consistent with the current BSEP configuration and plant operation.

The most significant difference between the analysis inputs is the modeling of the low main steam line pressure setpoint that initiates the main steam isolation valve (MSIV) closure during the pressure regulator failure open event. The GE BSEP EPU analyses conservatively compared the low pressure setpoint against the steam dome pressure to initiate the MSIV closure. The AREVA analyses compared the low pressure setpoint against the turbine header pressure, consistent with the actual location of the transmitters that are connected to the steam line headers as described in the BSEP Technical Specifications Bases, Section B.3.3.6.1. Preliminary calculations indicate that delaying the MSIV actuation until the steam dome pressure reaches the low pressure setpoint accounts for the majority of the difference between the GE and AREVA ATWS peak pressure results.

Other minor differences in some of the plant characteristics, as well as cycle design characteristics, result in additional small changes to the peak pressure result.

NRC Question 5 - ATWS Containment

Please provide the suppression pool temperature and containment pressure limits (i.e. acceptance criteria) and the corresponding licensing values of record.

Response 5

The suppression pool temperature and containment pressure limits and the corresponding licensing values of record are presented in the following table.

ATWS Criteria	Limit	Licensing Value
Suppression pool temperature (°F)	220	195.5
Containment pressure (psig)	62	12.9

NRC Question 6 - GE14 LOCA PCT

Please justify the continued applicability of the GE14 licensing basis PCT and associated MAPLHGR limits for the projected mixed core operation with ATRIUM 10. Please confirm that no system modifications were made at BSEP that would invalidate the reactor system response assumed in the GE14 LOCA analysis of record.

Response 6

As presented in Reference 2, the thermal hydraulic characteristics of the GE14 and ATRIUM™-10 fuel designs are similar. Therefore, the core response during a LOCA will not be significantly different for a full core of GE14 fuel or a mixed core of GE14 and ATRIUM™-10 fuel. In addition, since about 95% of the reactor system volume is outside the core region, slight changes in core volume and fluid energy due to fuel design differences will produce an insignificant change in total system volume and energy. Therefore, the current GE14 LOCA analysis and resulting licensing peak clad temperature (PCT) and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits remain applicable. No system modifications have been made at BSEP that would invalidate the reactor system response assumed in the GE14 LOCA analysis of record.

NRC Question 7 - Stability

RAMONA5-FA code is used to calculate the critical power ratio response during core oscillations. While the RAMONA5-FA method has not been generically approved by staff, its application has been previously accepted on an application-specific basis. Please confirm that the RAMONA5-FA methodology has been applied to Brunswick in a manner consistent with previously accepted applications.

Response 7

The RAMONA5-FA methodology has been applied to BSEP in a manner consistent with previously accepted applications.

NRC Question 8 - Bypass Boiling

Please discuss the impact of bypass boiling on the OPRM setpoints, and whether bypass voiding was considered in the analyses for BSEP.

Response 8

Bypass voiding is not considered in BSEP Oscillation Power Range Monitor (OPRM) setpoint analyses. Consideration of bypass voiding is not necessary due to the expected impact discussed below, as well as the overall conservatism of the approved Reference 12 methodology. Examples of conservatism in the Reference 12 methodology approved for BSEP EPU application include:

- The reactor is assumed to be operating at the operating limit minimum critical power ratio (OLMCPR) prior to the stability event,
- A conservative ΔCPR response (limiting time in cycle) is assumed following a pump trip,
- Equilibrium feedwater temperature for off-rated conditions is assumed to be instantaneously achieved following a pump trip and
- The statistical analysis selects the hot channel oscillation magnitude value at the 95% probability, 95% confidence level.

In addition, as noted in Response 2, the BSEP SLMCPR protected by the OPRM setpoints also reflects a conservative bundle power uncertainty.

The impact of localized bypass boiling is a reduction of the LPRM signal due to the decreased local moderation of the fast flux. [[

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Therefore, no degradation in the OPRM signal is expected due to bypass boiling.

NRC Question 9 - MCPR Part 21

Please discuss any impact of AREVA's October 8, 2007, Part 21 report regarding MCPR calculation to projected Brunswick Units 1 and 2 operation with ATRIUM-10 fuel and AREVA fuel methodologies.

Response 9

The subject Part 21 report discussed an error in the local power peaking distribution in the ATRIUM™-10 test assemblies used to determine critical power performance in the BWR KATHY test loop. The local power peaking distribution is used to develop the additive constants for the SPCB critical power correlation. BSEP will start up with POWERPLEX®-III CMSS input decks that incorporate the impact of the changes in additive constants. The operating limits presented in Reference 8 include the expected impact of this issue.

NRC Question 10 - Shutdown Margin

Please describe qualitatively and quantitatively, the analysis procedure used to ensure that the shutdown margin is within the TS limit through out the transition cycles. In particular please address how the eigenvalue biases and uncertainties are determined and accounted for the first and second transition cycles.

Response 10

In order to accurately determine shutdown margins during transition cycles, AREVA performs detailed benchmarking analyses of the three to five cycles previous to insertion of AREVA fuel in that reactor. This benchmarking is performed with the CASMO-4/MICROBURN-B2 3-D core simulator code system. Hot depletions are performed using actually operated state conditions including as-loaded core configurations, as-operated control rod patterns, and operating power, pressure, flow, and inlet subcooling. To confirm the validity of the hot depletions, comparison of eigenvalue trends and predicted versus measured TIP distributions for the benchmark cycles are performed. These results are used to establish the hot-operating target k-effective for design of the first transition cycle. Cold critical measurements taken during the benchmarking cycles are also modeled in MICROBURN-B2 by restarting from the hot cycle depletions discussed above. The results from the cold critical benchmarks are used to define a cold critical k-effective target. Once the cold target is determined based on the benchmarking, a statistically based design shutdown margin limit is chosen to bound the uncertainty observed by comparing the critical k-effectives computed with MICROBURN-B2 to the target selected from the benchmarking. A typical design target is 1% $\Delta k/k$. This ensures that the transition loading fuel design will support the 0.38% $\Delta k/k$ technical specification cold shutdown margin requirement with additional margin to cover the uncertainty in the design target chosen based on the benchmarking results. Past AREVA experience indicates that the variation in the target cold critical k-effective when transitioning from GE14 to ATRIUM™-10 is small (i.e., ≤ 0.001).

During the design of each transition cycle, shutdown margin is computed by performing restart solutions based on a shuffled core from a short window previous cycle condition. This means that the previous cycle is assumed to shutdown earlier than the nominal planned shutdown for the cycle. The short window shutdown of the previous cycle results in additional carryover reactivity for the shutdown margin analysis of the cycle being designed. Setting the gadolinia design of the fresh fuel and the loading plan to meet the design shutdown margin based on the assumed short window shutdown of the previous cycle assures that adequate shutdown margin exists for the entire cycle at the design stage. Prior to actual startup of the cycle, shutdown margin is recomputed based on the actual previous cycle shutdown exposure. At startup, when each designed cycle reaches cold critical conditions, comparison of the predicted point of criticality to the actual point of criticality is made. High accuracy of the predicted versus actual critical eigenvalue demonstrates the validity of the shutdown margin design for that cycle.

The initial critical and any subsequent cold critical data points achieved in each transition and follow-on cycle are fed back into the cold critical eigenvalue database for the reactor unit, and the target is revised as needed for the design of the subsequent cycle. This method assures continued accuracy in predicting the cold shutdown margin as new fuel is transitioned into the reactor core during the first and second transition cycles and all subsequent cycles.

Question 11 - Void-Quality Correlation

During a recent review of a BWR extended power uprate and associated AREVA fuel methodologies, the staff questioned the void-quality correlation bias and uncertainties. The issue was addressed by performing a plant specific calculation to assess the impact of the uncertainties on the OLMCPR. Please discuss how the void-quality correlation bias and uncertainties are addressed for the projected Brunswick Units 1 and 2 operations.

Response 11

The [[ ]] void-quality correlation has been qualified by AREVA against both the FRIGG void measurements and ATRIUM™-10 void measurements (i.e., see Figure 3-2). The basis for application of the correlation for GE14 fuel is discussed in Response 3.

The OLMCPR is determined based on the SLMCPR methodology and the transient analysis (i.e., ΔCPR) methodology. Void-quality correlation uncertainty is not a direct input to either of these methodologies; however, the impact of void-correlation uncertainty is inherently incorporated in both methodologies as discussed below.

The SLMCPR methodology explicitly considers important uncertainties in the Monte Carlo calculations performed to determine the number of rods in boiling transition. One of the uncertainties considered in the SLMCPR methodology is the bundle power uncertainty. This uncertainty is determined through comparison of calculated to measured core power distributions. Any miscalculation of void conditions will increase the error between the calculated and measured power distributions and be reflected in the bundle power uncertainty. Therefore, void-quality correlation uncertainty is an inherent component of the bundle power uncertainty used in the SLMCPR methodology.

The transient analysis methodology is not a statistical methodology and uncertainties are not directly input to the analyses. The transient analysis methodology is a deterministic, bounding approach that contains sufficient conservatism to offset uncertainties in individual phenomena. Conservatism is incorporated in the methodology in two ways: (1) computer code models are developed to produce conservative results on an integral basis relative to benchmark tests, and (2) important input parameters are biased in a conservative direction in licensing calculations.

The transient analysis methodology results in predicted power increases that are bounding relative to benchmark tests. In addition, for licensing calculations a 110% multiplier is applied to the calculated integral power to provide additional conservatism to offset uncertainties in the transient analysis methodology. Therefore, uncertainty in the void-quality correlation is inherently incorporated in the transient analysis methodology.

Based on the above discussions, the impact of void-quality correlation uncertainty is inherently incorporated in the analytical methods used to determine the OLMCPR. Biasing of important input parameters in licensing calculations provides additional conservatism in establishing the OLMCPR. No additional adjustments to the OLMCPR are required to address void-quality correlation uncertainty.

A sensitivity calculation was previously performed for another plant to assess the impact of a bias in the void-quality correlation on the OLMCPR. The sensitivity calculation used an alternate void-quality correlation that results in the prediction of lower void fractions than the [[ ]] correlation. These sensitivity calculations demonstrated that the void-quality correlation bias had small and offsetting impacts on SLMCPR and ΔCPR; there was no impact on the OLMCPR.

<u>References</u>

1. ANP-2637, Revision 1, "Boiling Water Reactor Licensing Methodology Compendium," AREVA NP, June 2007.
2. ANP-2646(P), Revision 0, "Brunswick Unit 1 Thermal-Hydraulic Design Report for ATRIUM™-10 Fuel Assemblies," AREVA NP, June 2007.
3. ANP-2625(P), Revision 0, "Brunswick Units 1 and 2 LOCA Break Spectrum Analysis for ATRIUM™-10 Fuel," AREVA NP, June 2007.
4. ANP-2624(P), Revision 0, "Brunswick Units 1 and 2 LOCA-ECCS Analysis MAPLHGR Limit for ATRIUM™-10 Fuel," AREVA NP, June 2007.
5. ANP-2638(P) Revision 0, "Applicability of AREVA NP BWR Methods to Extended Power Uprate Conditions," AREVA NP, July 2007.

6. ANP-2658(P) Revision 0, "Brunswick Unit 1 Cycle 17 Fuel Cycle Design," AREVA NP, July 2007.

7. AREVA Presentation Slides for Brunswick Fuel Transition License Amendment Request, August 2007.

8. ANP-2674(P), Revision 0, "Brunswick Unit 1 Cycle 17 Reload Safety Analysis," AREVA NP, September 2007.

9. ANP-2661(P) Revision 0, "Brunswick Nuclear Plant New Fuel Storage Vault Criticality Safety Analysis for ATRIUM™-10 Fuel," AREVA NP, September 2007.

10. ANP-2642(P) Revision 0, Brunswick Nuclear Plant Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM™-10 Fuel, AREVA NP, September 2007

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

12. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.

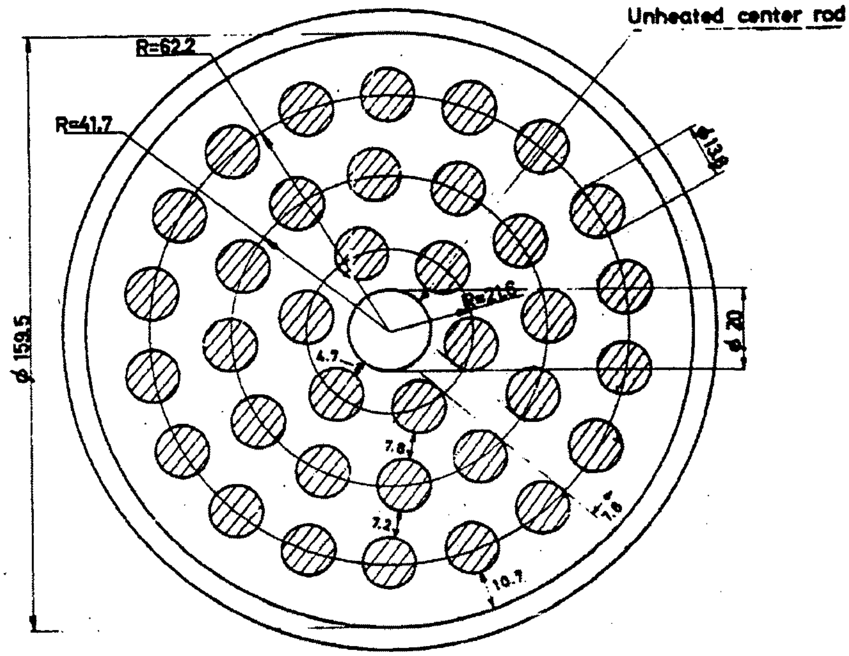

Figure 3.1a: FRIGG Test Cross Section

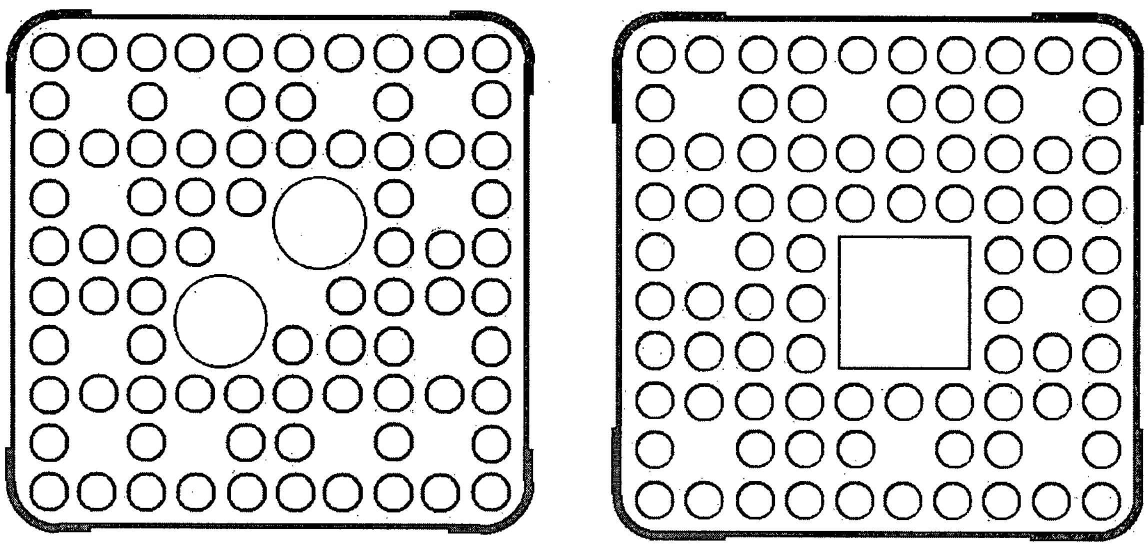

Figure 3.1b: GE14 (left) and ATRIUM™-10 (right) Top Cross Section

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Figure 3.2: Void Fraction Correlation Comparison
to FRIGG and ATRIUM™-10 Test Data