Entergy

Entergy Operations, Inc.
River Bend Station
5485 U.S. Highway 61N
St. Francisville, LA 70775
Tel 225-381-4149

Jerry C. Roberts
Director, Nuclear Safety Assurance

April 14, 2009

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: License Amendment Request
Main Turbine Bypass System
River Bend Station, Unit 1
Docket No. 50-458
License No. NPF-47

REFERENCES:
1. Letter RBG-46690 from Entergy to USNRC dated January 25, 2008, regarding Main Turbine Bypass System
2. USNRC letter to Entergy dated December 19, 2008, regarding Request for Additional Information for Main Turbine Bypass System License Amendment Request (TAC NO. MD7966)

RBG-46901

Dear Sir or Madam:

On January 25, 2008, Entergy Operations, Inc. (Entergy) submitted an amendment request to revise the Operating License for River Bend Station, Reference 1. This request would revise Technical Specification (TS) 3.7.5 "Main Turbine Bypass System," and provide an alternative to the existing Limiting Condition for Operation (LCO).

After discussions with the NRC staff, a Request for Additional Information (RAI) was issued by the NRC, Reference 2. Further discussions on January 30, 2009, identified additional information needed for the NRC staff review.

Attachment 1 to this letter contains the proprietary version of the responses to the NRC questions. A non-proprietary version of the document is also enclosed as Attachment 2.

The proprietary information is requested to be withheld from public disclosure in accordance with 10 CFR 9.17(a)(4) and 10 CFR 2.390 (a)(4). An affidavit attesting to the proprietary nature of the information is provided in Attachment 3.

The original no significant hazards consideration is not affected by any information contained in the supplemental letter.

There are no new commitments contained in this letter.

A001
NRR

If you have any questions or require additional information, please contact David Lorfing at 225-381-4157.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 14, 2009.

Sincerely,

Director, Nuclear Safety Assurance
River Bend Station - Unit 1

JCR/DNL/bmb

Attachments:
1. Responses to NRC Request for Additional Information (Proprietary Version)
2. Responses to NRC Request for Additional Information (Non-Proprietary Version)
3. Affidavit attesting to the proprietary nature of the information

cc: Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
612 E. Lamar Blvd., Suite 400
Arlington, TX 76011-4125

NRC Senior Resident Inspector
P. O. Box 1050
St. Francisville, LA 70775

U. S. Nuclear Regulatory Commission
Attn: Mr. Carl F. Lyon
OWFN 8 B1
Washington, DC 20555-0001

Mr. Jeffrey P. Meyers
Louisiana Department of Environmental Quality
Office of Environmental Compliance
Attn. OEC - ERSD
P. O. Box 4312
Baton Rouge, LA 70821-4312

Attachment 2

RBG-46901

(Non-Proprietary Version)

Response to River Bend Station, Unit 1 - Request for Additional Information for Main Turbine Bypass System License Amendment Request (TAC NO. MD7966) – Non-Proprietary

The following constitutes the Entergy responses to the NRC request for additional information transmitted by letter dated December 19, 2008. This response contains information proprietary to AREVA NP Inc. with such information contained in brackets within the individual responses.

1. **Specify the "certain accidents" that will be affected by the proposed change. Describe the rationale for choosing the accidents such that it is clear that the most limiting scenarios were chosen to determine the penalties for the operating limits. Specify the NRC-approved code(s) that are utilized for the chosen accidents.**

   Response:

   Of the events evaluated in RBS USAR Chapter 15 only a subset of the events are re-analyzed each cycle to demonstrate conformance to acceptance limits. These selected events represent limiting or near limiting events. Of the events selected for reanalysis each cycle only the feedwater controller (FWCF) event and the slow core flow runout event are impacted by operation with the main turbine bypass system out-of-service.

   The FWCF event with main turbine bypass system out-of-service is analyzed using the same methods and computer programs as the base feedwater controller failure event with the only exception that the bypass valves are analytically inhibited. The primary analytical tools used are COTRANSA2 and XCOBRA-T.

   The slow core flow runout is evaluated for determination of flow dependent thermal limits. The slow core flow runout event could result in vessel steam flow exceeding the capacity of the turbine control valves, and thus the turbine bypass valves would open to control reactor pressure. With the main turbine bypass system out-of-service, the reactor pressure would increase relative to the nominal slow flow runout analysis. The slow core flow runout analysis with main turbine bypass system out-of-service is analyzed using the same methods and computer programs as the base slow core flow runout event with the main difference being the determination of the power/flow state point at the end of the event. This state point includes the affect of pressurization of the reactor pressure vessel due to reduced steam flow capability resulting from main turbine bypass system out-of-service. The runout analyses are primarily performed using the XCOBRA computer program for flow dependent minimum critical power ratio ($MCPR_F$) limits and CASMO-4/MICROBURN-B2 core simulator codes for the flow dependent linear heat generation rate factor ($LHGRFAC_F$) limits.

2. **Describe the out-of-service analysis, results, and effect on the revised reactor operating limits.**

   Response:

   The events analyzed and the methods used to analyze the events are discussed in the response to the previous question. The following describes the impact of assuming the main turbine bypass system out-of-service on the calculated minimum critical power ratio (MCPR) and linear heat generation rate (LHGR).

   The operating limits for main turbine bypass system out of service were determined using the NRC approved AREVA methodology as listed in RBS Technical Specification 5.6.5 (Core Operating Limits Report). This NRC approved methodology is designed to ensure that operating limits are established that protect the fuel from particular events. In the case of the main turbine bypass system out of service, power and flow dependent MCPR and linear heat generation rate factor (LHGRFAC) limits are established to protect the fuel from damage during an anticipated operating occurrence (i.e., moderate frequency event).For Cycle 15, the impact of the main turbine bypass system out-of-service on calculated ΔMCPR and heat flux ratio (HFR) for the feedwater controller failure event are small and in the range of 0.00 to 0.02 depending upon the core exposure and initial core thermal power. At higher exposures, the impact on the power dependent operating limit MCPR ($MCPR_P$) are similarly small in the range of 0.01 to

0.02 for initial core power levels above 70%. Below 70% core power other events, which are not impacted by the main turbine bypass system out-of-service, set $MCPR_P$.

Operation with the Main Turbine Bypass system out-of-service results in a greater transient heat flux ratio (HFR) for the FWCF event. However when they are combined with other equipment out of service options, the required $LHGRFAC_P$ is either limited by the steady state HFR limit or the required $LHGRFAC_P$ from another equipment out of service condition.

The slow core flow runout event was analyzed for Cycle 15 with and without the main turbine bypass system out-of-service. As discussed above, the main turbine bypass system out-of-service cases include pressurization effects if the steam flow terminal point exceeds the system capacity without bypass valves. The analysis results indicate a maximum increase of 0.10 in the flow dependent MCPR. The Cycle 15 $LHGRFAC_F$ multipliers were established to support base case operation and operation in the EOOS scenarios for all cycle exposures.

3. **Describe the expected differences between the cycle-specific analysis with the proposed changes and the most recent cycle-specific analysis. Describe what additional information will be included in the core operating limits report (COLR).**

Response:

The cycle specific analysis will address the events discussed above as an equipment out-of-service option. Appendix A of the current RBS Core Operating Limits Report (COLR) contains the equipment out-of-service thermal limits curves and will be modified to include the main turbine bypass system out-of-service $MCPR_P$, $MCPR_F$, $LHGRFRAC_P$, and $LHGRFRAC_F$ as appropriate in accordance with Section 5.6.5 of the RBS Technical Specifications.

4. **In the licensee's draft response to Question 1, it appears that the NRC-approved topical report to be used to re-calculate the average planer linear heat generation rate (APLHGR), maximum critical power ratio (MCPR), and linear heat generation rate (LHGR) limits for an inoperable MTBS is stated to be XN-NF-80-19(P), Volume 3, Revision 2. There are 4 volumes of XN-NF-80-19(P). This volume appears to address abnormal operational occurrences (AOOs), but does not address accidents. Volume 4, for example, appears to address AOOs and accidents. Also, since the APLHGR limit is derived from the emergency core cooling system (ECCS) analysis for loss-of-coolant accidents (LOCAs), it appears that Volume 3 would be inappropriate for calculating the APLHGR limit.**

For the APLHGR, MCPR, and LHGR limits for an inoperable MTBS, provide the volume of XN-NF-80-19(P) that is being used to calculate each limit.

Response:

APLHGR. APLHGR is determined by averaging the LHGR over each fuel rod in a plane. The limit for APLHGR is expressed as the maximum APLHGR (MAPLHGR) for any plane in the fuel assembly. The MAPLHGR limit is chosen such that 10 CFR 50.46 criteria are met (peak cladding temperature for design basis LOCA will not exceed 2200°F, etc.). The criteria are evaluated with the ECCS analysis methodology.

Volume 4 summarizes the MAPLHGR analyses. Volumes 2, 2A, 2B and 2C address ECCS analyses; however, these volumes have been superseded by Reference 4-1. An inoperable MTBS does not result in an increase in severity of results associated with the ECCS analyses; therefore, the MAPLHGR limit is unchanged for an inoperable MTBS.

MCPR. Volume 3 provides the overall methodology for determining a MCPR operating limit and Anticipated Operation Occurence (AOO) analyses.

LHGR. Volume 4 provides the methodology for LHGR limits. LHGR limits are defined based on References 4-2 and 4-3.

Refer to the response of Question 5 for additional information on superseded reports.

References:

4-1. EMF-2361(P)(A) Revision 0, EXEM BWR-2000 ECCS Evaluation Model, Framatome ANP, May 2001.

4-2. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model, Siemens Power Corporation, February 1998.

4-3. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, March 1984.

5. ***In each volume of XN-NF-80-19(P), there is the following: (1) the conditions and limitations on the use of the topical report in the NRC safety evaluation (SE) approving the use of the topical report, which is enclosed in the topical report, and (2) a discussion of the appropriate AOOs and/or accidents or both to be addressed in calculating the core limits. Discuss how (1) the conditions and limitations in the appropriate NRC SE are met, (2) the affected AOO and accident analyses are identified, and (3) appropriate AOOs and accidents are being considered consistent with the discussions in the applicable volume of XN-NF-80-19(P). Provide a list of the affected safety analyses. Provide justification for any AOOs and accidents specified in the applicable volume of XN-NF-80-19(P) that are not re-analyzed for the MTBS being inoperable.***

Response:

Reference 5-1 provides information on the use and applicability of the AREVA licensing topical reports including how the conditions and limitations in the appropriate NRC SE are met. Furthermore, Reference 5-1 discusses how appropriate AOOs and accidents are applicable to each licensing topical report.

The transient analyses are identified in the COTRANSA2 and XCOBRA-T topical reports (References 5-2 and 5-3). NRC concurrence with clarifications related to SER issues concerning the topical report was requested in References 5-4 and 5-5. The NRC concurrence with these clarifications was provided in Reference 5-6. These references clarify that COTRANSA2/XCOBRA-T is approved for the analysis of the events in Table 5.1, which correlates the transient analyses to USAR sections.

For an inoperable MTBS, the affected AOO and accident analyses are identified by a review of the USAR events. Table 5.2 provides a list that identifies the affected safety analyses. Also in the table is justification for any AOOs and accidents that are not reanalyzed for the MTBS being inoperable.

In summary, the events and analyses impacted by MTBS being inoperable are:

- Feedwater controller failure event
- Slow core flow runout event

References:

5-1. ANP-2637 Revision 2, Boiling Water Reactor Licensing Methodology Compendium, AREVA NP, December 2007.

5-2. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, Advanced Nuclear Fuels Corporation, August 1990.

5-3. 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.

5-4. Letter, James F. Mallay (SPC) to Document Control Desk (NRC), "Request for Concurrence on SER Clarifications," NRC:99:030, July 28, 1999.

5-5. Letter, James F. Mallay (SPC) to Document Control Desk (NRC), "Revision to Attachment 1 of Letter NRC:99:030, Request for Concurrence on SER Clarifications," NRC:99:045, October 12, 1999.

5-6. Letter, Stuart Richards (NRC) to James F. Mallay (SPC), "Siemens Power Corporation Re: Request for Concurrence on Safety Evaluation Report Clarifications (TAC No. MA6160)," May 31, 2000.

Table 5.1 Applicable Transient Analysis Events With COTRANSA2/XCOBRA-T

SRP Section	Chapter 15 Analysis
15.1.1 – 15.1.3	Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Demand
15.2.1 – 15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)
15.2.7	Loss of Normal Feedwater Flow
15.3.1 – 15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions
15.3.3 – 15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
15.4.4 – 15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory
15.6.1	Inadvertent Opening of a PWR Pressure Relief Valve and BWR Pressure Relief Valve
15.8	Anticipated Transients Without Scram (the Initial Pressurization Only)

[redacted]

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted]

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted]

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	
[redacted]	[redacted]	[redacted] [redacted]
[redacted]	[redacted]	[redacted] [redacted]
[redacted]	[redacted]	[redacted] [redacted] [redacted]
[redacted]	[redacted]	[redacted] [redacted] [redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	
[redacted]	[redacted]	[redacted] [redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted]		
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

	[redacted]	
[redacted]	[redacted]	[redacted]
	[redacted]	
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

6. **To support the review of a license amendment for a power uprate, the NRC staff conducted a review of the applicability of the NRC-approved AREVA suite of nuclear design and transient analysis methods. As part of this review, an audit of the AREVA codes was conducted. The result of the audit is that the staff needs additional information for clarification on the use of the AREVA suite of methods. Until this is resolved with AREVA, these questions are being requested on plant-specific applications that use this suite of methods. These questions for clarification of the River Bend application are the following:**

6.1 **Verify that the upstream transient COTRANSA2 analysis: (1) includes the 110% integral thermal power multiplier, (2) biases all relevant input parameters to the limiting values allowable by TSs as appropriate, (3) biases non-TS controlled input parameters to the most conservative value based on their associated uncertainty, and (4) is representative of the limiting plant configuration allowable for equipment out-of-service in the TSs.**

Response:

Based on additional clarification of the question from an NRC telecom with AREVA and Entergy on December 4, 2008, the following response is provided.

The 110% integral thermal power multiplier is applied to the output of COTRANSA2 that is used as the input to XCOBRA-T; therefore, the 110% integral power multiplier is included in the transient analyses. Important input parameters are biased in a conservative direction in licensing calculations. For TS controlled input parameters, the biasing is either the limiting value allowable by TS, or an analytical limit that is beyond the limiting value allowable by TS. If a particular equipment out-of-service is applicable to a particular transient event, the transient analysis is performed with the limiting plant configuration for the allowable equipment out-of-service.

6.2 **If COTRANSA2 or another one-dimensional code is used to determine the transient reactor power, please describe how appropriate axial planar average fuel rod parameters are determined for the analysis. This discussion should address: gap conductance, thermal conductivity, pellet size, and heat capacity. Justify that these parameters are acceptably accurate or conservative. Demonstrate that the conservatism in the COTRANSA2 analysis is sufficient to bound any bias in the transient peak heat flux calculation as a result of known [[redacted]] for ATRIUM-10 fuel.**

Response:

Based on additional clarification of the question from an NRC telecom with AREVA and Entergy on December 4, 2008, the following response is provided.

Thermal-Mechanical Model

The RODEX2 computer code provides initial input information relative to core average fuel-to-cladding gap heat transfer coefficients for the COTRANSA2 computer code. As such, RODEX2 uses steady-state heat conduction models. The heat conduction model employed by COTRANSA2 includes transient terms.

The fuel thermal conductivity correlations used by COTRANSA2 are equivalent to the RODEX2 models.

COTRANSA2 computes a fuel temperature for each axial plane in the core. Based on the assumption of a core composition primarily consisting of uranium dioxide, COTRANSA2 does not account for gadolinium in the fuel thermal conductivity calculation.

Heat capacities of fuel components (uranium dioxide, gadolinium, and cladding) are not required for the RODEX2 steady-state calculations but are used in the COTRANSA2 transient calculations.

A gap conductance sensitivity study was performed for the load rejection with no bypass (LRNB) transient event for a BWR. The purpose of the sensitivity study was to show the ΔCPR trend for changes in gap conductance for COTRANSA2 versus XCOBRA-T. The gap conductance change considered was [[redacted]]. The results are provided in Table 6.2-1. As seen from the results, an increase in COTRANSA2 core average gap conductance results in a decrease in ΔCPR; whereas an increase in XCOBRA-T hot channel gap conductance results in an increase in ΔCPR. A decrease in gap conductance shows the opposite trend. The XCOBRA-T ATRIUM-10 hot channel model is slightly more sensitive to the change in gap conductance than the COTRANSA2 ATRIUM™-10 average core model. When both COTRANSA2 and XCOBRA-T gap conductance are changed by an equivalent amount, the net impact is no significant change in ΔCPR.

Void Quality Correlation

In a BWR, the core power and power distribution are tightly coupled with the void fraction and a large error in predicted core void fraction would have a significant effect on the predicted power distribution measurements obtained from operating reactors. If the error in void fraction was significant, the effect would be observed in comparisons of predicted to measured power distributions obtained from operating reactors.

Integral power is a parameter obtainable from test measurements that is directly related to ΔCPR and provides a means to assess code uncertainty by increasing heat flux during the duration of the event. The COTRANSA transient analysis methodology was a predecessor to the COTRANSA2 methodology. The integral power figure of merit was introduced with the COTRANSA methodology as a way to assess (not account for) code uncertainty impact on ΔCPR. From COTRANSA analyses of the Peach Bottom turbine trip tests, the mean of the predicted to measured integral power was 99.7% with a standard deviation of 8.1%. AREVA (Exxon Nuclear at the time) initially proposed to treat integral power as a statistical parameter. However, following discussions with the NRC, it was agreed to apply a deterministic 110% integral power multiplier (penalty) on COTRANSA calculations for licensing analyses. That increase was sufficient to make the COTRANSA predicted to measured integral power conservative for all of the Peach Bottom turbine trip tests.

COTRANSA2 (Reference 6.2-1) was developed and approved as a replacement for COTRANSA in the AREVA thermal limits methodology (Reference 6.2-2). Initially it was not planned to use the 110% integral power multiplier with the COTRANSA2 methodology. COTRANSA2 predictions of integral power were conservative for all Peach Bottom turbine trip tests. The minimum conservatism was [[redacted]] and the mean of the predicted to measured integral power was [[redacted]]. The comparisons to the Peach Bottom turbine trip tests demonstrated that the 110% integral power multiplier was not needed for COTRANSA2. However, because the thermal limits methodology that was approved independently of COTRANSA2 included discussion of the 110% integral power multiplier, the use of the multiplier was retained for COTRANSA2 licensing calculations. With the 110% multiplier, the COTRANSA2 predicted to measured mean integral power is [[redacted]] for the Peach Bottom turbine trip tests. Applying a [[redacted]] integral power multiplier provides an operating limit MCPR (OLMCPR) conservatism of [[redacted]]. The 110% integral power multiplier is just one part of the conservatism in the COTRANSA2 methodology and application process that covers methodology uncertainties.

Based on a sensitivity study for a BWR, the potential impact of a bias with uncertainty in the void correlation was determined to be in the range of [[redacted]] for ΔCPR.

COTRANSA2 is not a statistical methodology and uncertainties are not directly input to the analyses. The 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. Justification that the integrated effect of all the conservatisms in COTRANSA2 licensing analyses is adequate for methodology uncertainties is provided below.

- The COTRANSA2 methodology results in predicted power increases that are bounding ([[redacted]] on average) relative to Peach Bottom benchmark tests. In addition, for licensing calculations a 110% multiplier is applied to the calculated integral power to provide additional conservatism. This approach adds significant conservatism to the calculated heat flux and OLMCPR as discussed previously.
- Biasing of important input parameters in licensing calculations provides additional conservatism in establishing the OLMCPR. The Peach Bottom turbine trips were performed assuming the measured performance of important input parameters such as control rod scram speed and turbine valve closing times. For licensing calculations, these (and other) parameters are biased in a conservative bounding direction. These conservative assumptions are not combined statistically; assuming all parameters are bounding at the same time produces very conservative results.
- As discussed previously, the core axial power distribution is tightly coupled with the void fraction. A large error in predicted void fraction would have a significant effect on the predicted axial power distribution measurements obtained from operating reactors. The very good comparisons between predicted and measured axial power distributions obtained from operating reactors indicates that the void distribution within the core is being predicted well.

Based on the above discussions, the impact of void correlation uncertainty is inherently incorporated in the analytical methods used to determine the thermal limits. No additional adjustments to the thermal limits are required to address void correlation uncertainty.

AREVA sensitivity analyses have determined that the known [[redacted]] for the ATRIUM-10 could result in a small underprediction of peak vessel pressure. A sensitivity study was performed for a BWR to assess the bias. The impact of the bias of the void-quality correlation on peak pressure is expected to be more than offset by the model conservatisms. Independent of other model conservatisms, the sensitivity analyses determined a 10-psi increase to the peak vessel pressure for the ATWS overpressure analysis and a 7-psi increase to the peak vessel pressure for the ASME overpressure analysis to account for the bias. These increases in peak vessel pressure do not challenge the vessel pressure limits for RBS.

References:

6.2-1. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, Advanced Nuclear Fuels Corporation, August 1990.

6.2-2. XN-NF-80-19(P)(A) Volume 3 Revision 2, Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, Exxon Nuclear Company, January 1987.

Table 6.2-1 Gap Conductance Study

Gap Conductance Condition	Δ(ΔCPR)
***Increase** in Gap Conductance*	
Core average [[redacted]]	−0.011
Hot channel [[redacted]]	+0.012
Core average and hot channel [[redacted]]	0.000
***Decrease** in Gap Conductance*	
Core average [[redacted]]	+0.015
Hot channel [[redacted]]	−0.016
Core average and hot channel [[redacted]]	−0.001

6.3 For the transient analysis, is the thermal power assumed to be 102% of the licensed thermal power at the initiation of the transient?

Response:

For the transient analysis, two different approaches are used to account for the power uncertainty. Most of the analyses were performed at 100% power level and the impact of the power uncertainty is accounted for either statistically or through the inherent conservatism of the methodology. For some of the analyses, (i.e., ASME overpressurization and LOCA-ECCS analyses) the effects of the power uncertainty are not directly included in the methodology used for the analyses; therefore, these analyses were performed at increased power to account for the power uncertainty. It should be noted that River Bend Station has undergone an Appendix K power uprate. The LOCA analyses were performed at 102% pre-Appendix K rated power. The ASME overpressurization analyses were conservatively performed at 102% of the Appendix K power uprate.

6.4 Specify the code that is used to determine the transient LHGR limit relative to the 1% plastic strain criterion and fuel centerline melt criterion if this code is not RODEX2.

Response:

RODEX2A (Reference 6.4-1) is used to evaluate the fuel centerline temperature. RODEX2 in conjunction with the RAMPEX code (Reference 6.4-2) are used to evaluate cladding transient strain. The methodology for evaluating the cladding transient strain criterion and the fuel centerline melt criterion is described in Reference 6.4-1. Further clarification is contained in the TER for Reference 6.4-1 with regard to the cladding strain and fuel melt methods.

References:

6.4-1. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model, Siemens Power Corporation, February 1998.

6.4-2. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, March 1984.

6.5 During cycle operations, please describe what surveillances or checks are performed by the licensee to ensure that actual plant operations are within the bounds of the COLR analysis in terms of meeting the 1% plastic strain and fuel centerline melt criteria.

Response:

The core monitoring system is available during plant operations to monitor core performance parameters like maximum LHGR (MLHGR). The system models MLHGR explicitly, and the 24 hour surveillance requirement ensures that the result is routinely checked. Since the MLHGR is explicitly monitored during core operations, there are no additional surveillances or checks required to ensure that operations are within the bounds of the COLR analysis.

Regarding the fuel centerline melt criteria, the licensing analyses are performed to ensure the criteria will not be exceeded. For example, the flow run-up excursion analyses are performed employing all conservative assumptions with regard to the starting point on the power/flow map, i.e. the analyses are performed starting from a rod line that results in the most conservative result. Given the conservative bounding nature of the analyses, no further surveillances or checks are required.

6.6 Verify that conformance with the operating limit MLHGR is performed accounting for channel bow. If not, justify why not.

Response:

The Item 13 response in Supplement 2 of Reference 6.6-1 (correspondence, R.A. Copeland to Robert C. Jones, "Additional Information Regarding Loss of Thermal Margin Caused by Channel Box Bow," May 3, 1990) discusses the account for channel bow in determining the LHGR limits. The response is applicable to the ATRIUM-10 design since the local peaking and LHGR limits continue to provide substantial margin to 1% strain and fuel centerline melt.

Additional information requested by the NRC on the 1% plastic strain limit, the fuel centerline melt limit, the LHGR at beginning of life (BOL), the degree of anticipated bow, and the LPF (local peaking factor) impact of that bow for the ATRIUM-10 design at River Bend are described below for comparison to the values previously provided in Reference 6.6-1 for the 8x8 and 9x9 designs.

The respective LHGRs at fuel centerline melt and 1% cladding strain for the ATRIUM-10 design are [[redacted]] kW/ft and [[redacted]] kW/ft. The [[redacted]] value is for a [[redacted]] that is assumed to be at the allowed concentration of [[redacted]].To provide an estimate of the LHGR at fuel centerline melt and at 1% cladding strain, it is necessary to extrapolate the results from existing analyses for River Bend since the current analyses do not go high enough in power to reach either design limit. Note that the LHGRs at fuel melt and 1% strain for the 1990 response were likely obtained in the same way since the approved codes and methods used at RBNS have not changed. For centerline melt, the fuel temperature [[redacted]]. A similar estimate is done for 1% cladding strain.

The ATRIUM-10 BOL LHGR limit is [[redacted]] kW/ft. Figure 6.6-1 shows the LHGR limit for the ATRIUM-10 fuel at River Bend.

Figure 6.6-1 ATRIUM-10 LHGR Limit for RBNS

Note that the limit is now specified as a function of pellet exposure while previous limits are based on an assembly planar exposure.

A conservative value to use for the degree of anticipated bow for the ATRIUM-10 design is [[redacted]] mils. This value is conservative for two reasons. One, the [[redacted]] mils, when combined with bulge, is sufficient to result in contact with the control rod. Thus, higher values would be detectable in operation. Two, the [[redacted]] mils is conservative because of the manner (described below) in which the bow value is applied in calculating the impact on the LPF.

A maximum change in lattice limiting LPF of [[redacted]]

Based on the information summarized above, the conclusion provided in the Item 13 response in Supplement 2 of Reference 6.6-1 for 8x8 and 9x9 fuel designs continues to apply for 10x10 fuel.

Reference:

6.6-1. ANF-524(P)(A) Revision 2 and Supplements 1 and 2, ANF Critical Power Methodology for Boiling Water Reactors, Advanced Nuclear Fuels Corporation, November 1990.

6.7 Verify that conformance with the operating limit MLHGR is performed accounting for LPRM rod power biases. If not, justify why not.

Response:

The explicit LPRM model is used in the River Bend core monitoring, hence LPRM rod power biases are accounted for.

6.8 Verify that conformance with the operating limit MLHGR is performed accounting for PLFR fission gas plena. If not, justify why not.

Response:

Monitoring for conformance with the operating limit MLHGR is not performed accounting for the fission gas plena, because sensitivity studies show the plena has negligible effect on LHGR.

Lattices occupying the node directly above the top of the PLFR active fuel length were evaluated in a 3-D full core equilibrium cycle model to determine the impact of modeling the PLFR upper plena regions as coolant versus modeling the plena explicitly. The difference in core limiting margin to the LHGR thermal limit due to the modeling of the PLFR varies throughout the cycle. The largest decrease in LHGR margin (conservative) due to modeling the PLFR upper plena as coolant is 0.019. The largest increase in LHGR margin (nonconservative) due to modeling the PLFR upper plena as coolant is 0.003. The changes in LHGR margin due to using the coolant model for the PLFR upper plena are small.

6.9 Clarify if the relevant rod power histories used in the thermal mechanical analysis come from calculated off-line or [[redacted]] power histories. Justify the approach used.

Response:

The LHGR limit is established to support plant operation while satisfying the fuel mechanical design criteria. The LHGR limit is translated into power history inputs as described in the Reference 6.9-1 topical report (see the response to RAI, Question 3, in Reference 6.9-1). In other words, the power history inputs are developed from the LHGR limit. Then, the power histories are used as input to the RODEX2A, RAMPEX (with RODEX2) and COLAPX codes in evaluating the fuel rod thermal-mechanical criteria (References 6.9-2 and 6.9-3).

The profiles selected for the analyses are conservatively peaked to result in higher rod average power levels while attaining the LHGR limit. Separate studies (Reference 6.9-4) have shown the current methodology, which makes use of a bounding power history, to be conservative.

References:

6.9-1. XN-NF-85-67(P)(A) Revision 1, Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, Exxon Nuclear Company, September 1986.

6.9-2. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model, Siemens Power Corporation, February 1998.

6.9-3. ANF-89-98(P)(A) Revision 1 and Supplement 1, Generic Mechanical Design Criteria for BWR Fuel Designs, Advanced Nuclear Fuels Corporation, May 1995.

6.9-4. BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, April 2008.

6.10 Verify that the power shapes used in the thermal-mechanical calculations of the operating limit and transient limit are conservative for the plant-specific application. If these shapes are different from the shapes reported in BAW-10247(P)(A), justify why they are different.

Response:

Please see the response to the preceding question (6.9) that describes the power history methodology that was used for RBS.

6.11 The NRC staff is aware that the transient analysis is performed using off-line simulations. For the COLR analysis, verify that the steady state off-line cycle analysis used to determine the EOC axial power shape is conservative relative to the operational flexibility allowed by the flow control window along the licensed thermal power line (LTPL) of the approved operating domain.

Response:

The transients are analyzed based on an assumed aggressive burn of the core at lower flow than is projected to be the case during actual plant operations. This provides a conservative margin to cover differences between the cycle step-out projection and actual plant operations. AREVA provides guidance for the reactor engineers in the form of a licensing basis axial power shape that the reactor engineers use to monitor actual operations. The reactor engineers project actual operations to the end of cycle all-rods out condition and compare the resulting axial power shape with that which is the basis of the licensing analyses, confirming compliance with the requirement.

6.12 The staff is aware that the transient analysis is performed using off-line simulations. Verify that the steady state off-line cycle tracking analysis is sufficiently detailed to meet the uncertainty requirements imposed on CASMO-4/MICROBURN-B2, in the SER for EMF-2158(P)(A), which References Tables 2.1 and 2.2 in the LTR. The response should provide operational data to verify the accuracy of the offline analysis against plant-specific axial, radial, and nodal TIP data.

Response:

For confirmation that the measured power distributions from the core are consistent with the power distribution uncertainty requirements imposed on CASMO-4/MICROBURN-B2, it is appropriate to perform a statistical test to confirm that actual TIP response uncertainties are consistent with expected TIP uncertainties. The χ^2 (chi-squared) test is an appropriate consistency test for TIP measurements. The χ^2 test has been performed for River Bend TIP data.

Five TIP measurements have been taken at River Bend during Cycle 15. Two of the measurements were taken with symmetric control rod patterns. The χ^2 test results are presented in Tables 6.12-1 through 6.12-4. All of the tests meet the statistical criteria which demonstrates that the actual TIP data taken at River Bend complies with the topical report EMF 2158(P)(A).

Table 6.12-1 River Bend Cycle 15
2D Measured Vs. Calculated χ^2
Statistical Test

TIP Measurement	χ^2	CritVal	Pass/Fail
08APR07_14.08.45	22.44	54.84	Pass
08MAY21_17.56.33	21.28	54.84	Pass
08JUL23_10.47.36	18.62	54.84	Pass
08OCT28_15.46.44	20.14	54.84	Pass
09JAN12_15.52.48	17.63	54.84	Pass

Table 6.12-2 River Bend Cycle 15
3D Planar Measured Vs. Calculated χ^2
Statistical Test

TIP Measurement	χ^2	CritVal	Pass/Fail
08APR07_14.08.45	513.90	747.60	Pass
08MAY21_17.56.33	527.53	747.60	Pass
08JUL23_10.47.36	495.44	747.60	Pass
08OCT28_15.46.44	467.58	747.60	Pass
09JAN12_15.52.48	490.13	747.60	Pass

Table 6.12-3 River Bend Cycle 15
2D Symmetric χ^2
Statistical Test

TIP Measurement	χ^2	CritVal	Pass/Fail
08APR07_14.08.45	8.23	29.14	Pass
08OCT28_15.46.44	6.02	29.14	Pass

Table 6.12-4 River Bend Cycle 15
3D Planar Symmetric χ^2
Statistical Test

TIP Measurement	χ^2	CritVal	Pass/Fail
08APR07_14.08.45	187.95	338.08	Pass
08OCT28_15.46.44	160.84	338.08	Pass

6.13 Specify the code that is used to determine the transient LHGR during simulated FWCF events if this code is not XCOBRA-T.

Response:

XCOBRA-T is used.

6.14 Describe any differences between the XN-NF-84-105(P)(A) licensing topical report description of XCOBRA-T and the current standard production code version that supports the use of this methodology for modern fuel designs such as ATRIUM-10, the response should address axial geometry changes and modern fuel spacers.

Response:

As identified in the question, the differences between the XN-NF-84-105(P)(A) licensing topical report description of XCOBRA-T and the current standard production code version is axial geometry changes and modern fuel spacers.

Axial Geometry Changes

XCOBRA-T calculates the fuel rod surface heat flux using a fuel rod heat conduction model, the power generated in the fuel rod, and the fluid conditions at the surface of the rod. The power generated in the fuel rod is described in Reference 6.14-1 Section 2.5.5. The power generated in each axial section of a fuel rod is calculated using Equation 2.130 from Reference 6.14-1. Although Reference 6.14-1 states that Equation 2.130 is calculated for each axial node, the equation itself does not denote which variables are axially dependent. Because the equation is for each axial node, the variables for heat generation rate, axial peaking factor, and number of rods are axial dependent. At the time Reference 6.14-1 was prepared, the number of rods at each axial plane was a constant for the fuel designs being supplied. For the ATRIUM-10 fuel design with part-length fuel rods (PLFRs), the number of rods became axial dependent and the code was modified to make application of Equation 2.130 correct and consistent with the NRC-approved Reference 6.14-1. For application to current fuel designs, a better definition of the variable N_r in Equation 2.130 would be "number of *heated* rods per assembly *at the axial plane*" (italic indicates added text).

For bundles with part-length fuel rods (PLFRs), the rod heat flux calculation begins by computing the time-dependent heat flux generation rate at each axial section in the fuel rod. The updated equation, corresponding to equation 2.130 of Reference 6.14-1 is:

$$q''(t) = \frac{P(t)}{\pi D_{rod,i}} \frac{1}{L N_a N_{ri}} (f_f + f_c) F_{ri} F_{li} F_a$$

where

$P(t)$	=	transient reactor power
f_f	=	fraction of power produced in the fuel
f_c	=	fraction of power produced in the cladding
N_a	=	total number of assemblies in the core
N_{ri}	=	total number of heated rods for type i assembly at the axial plane

F_{ri} = radial peaking factor of type i assembly

F_{li} = local peaking factor of type i assembly

F_a = axial peaking factor at the axial plane

$D_{rod,i}$ = fuel rod diameter of type i assembly

L = axial heated length

This equation differs from that in Reference 6.14-1 by replacing the initial reactor power in the denominator with π. In addition, the variable definitions have been modified to identify that the total number of heated rods is dependent on both the assembly type and axial elevation and the definition of L has been corrected to the axial heated length of the assembly. This equation is substituted into equations 2.129a and 2.129b in Section 2.5.5 of Reference 6.14-1 to define the volumetric heat deposition rate for the fuel pellet and cladding, respectively. This volumetric heat deposition rate is used in the right-hand side of equation 2.85 of Reference 6.14-1 to iteratively solve the transient heat conduction equation and the hydraulic conservation equations for the new time step temperatures and surface heat flux. The heat flux is introduced into the channel energy equation (2.2 of Reference 6.14-1) through the term q'. This linear heat deposition rate is a summation of the energy added by direct energy deposition and surface heat flux:

$$q'(t) = \left\{ \frac{P(t)}{N_a L} f_{cool} F_{ri} F_a + H_{surf} \cdot (T_{NodesT} - T_{fluid}) \cdot \pi \cdot D_{rod,i} \cdot N_{ri} \right\} N_i$$

where

f_{cool} = fraction of power produced in the coolant

H_{surf} = film heat transfer coefficient at the axial plane

T_{NodesT} = cladding surface temperature at the axial plane

T_{fluid} = fluid temperature at the axial plane

N_i = number of fuel assemblies in channel i

In addition to axially varying number of heated rods, proper modeling of PLFRs also requires axial variations in the active flow area, the heated perimeter, and the wetted perimeter and these parameters are now defined as axially dependent quantities in AREVA methods. Consequently, all references to these parameters or parameters derived from the basic geometry data in the approved topical reports should be interpreted as being axially dependent variables. The pressure drop due to the area expansion at the end of the PLFRs (or anywhere in the active flow path) is modeled using the specific volume for momentum as expressed in equations 2.78 and 2.79 of Reference 6.14-1. For current designs, area contractions occur in the single phase region, but the coding was generalized to address area contractions in the two-phase region based on a solution of the two-phase Bernoulli equation.

Modern Fuel Spacers

During the development of the ATRIUM-10 fuel design, the single phase loss coefficients resulted in an under prediction of the pressure drop data as shown in Figure 6.14-1. The spacer loss coefficients (K) used to generate the results presented in Figure 6.14-1 are of the form

$$K = A + B\,Re^{C}$$

where A, B, and C are constants and Re is the Reynolds number based on local fluid conditions and geometry.

A means of adjusting loss coefficients to better predict the pressure drop data was developed. [[redacted]] are shown in Figure 6.14-2. The spacer loss coefficients (K) used to generate the results presented in Figure 6.14-2 are of the form

[[redacted]]

where [[redacted]] for the ATRIUM-10 design.

Assessments of the predicted pressure drop relative to measured two-phase pressure drop data confirmed the applicability of the [[redacted]] for use with spacer pressure loss coefficients. Results of analyses for each region of the bundle (lower, transition, upper) when using the PHTF spacer loss coefficients [[redacted]] are shown in Figures 6.14-3, 6.14-4, and 6.14-5.

On May 4, 1995, a meeting was held with the NRC to describe the ATRIUM-10 design and the application of the approved AREVA methodology for the design. Two view graphs extracted from those presented at the meeting are provided in Figures 6.14-6 and 6.14-7. A summary of the May 4, 1995 meeting was provided to the NRC in Reference 6.14-2.

References:

6.14-1. 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.

6.14-2. Correspondence, R.A. Copeland (Siemens) to R.C. Jones (NRC), "ATRIUM-10 Presentations," RAC:95:080, May 4, 1995 (38-9091703-000).

Figure 6.14-1 ATRIUM-10 Bundle Pressure Drop
[[redacted]]

Figure 6.14-2 ATRIUM-10 Bundle Pressure Drop
[[redacted]]

Figure 6.14-3 ATRIUM-10 Lower Region Spacer Pressure Drop Using PHTF Loss Coefficients
[[redacted]]

Figure 6.14-4 ATRIUM-10 Transition Region Spacer Pressure Drop Using PHTF Loss Coefficients
[[redacted]]

Figure 6.14-5 ATRIUM-10 Upper Region Spacer Pressure Drop Using PHTF Loss Coefficient
[[redacted]]

Figure 6.14-6 Viewgraph From May 4, 1995
Presentation to NRC
Regarding ATRIUM-10 Fuel

Figure 6.14-7 Viewgraph From May 4, 1995
Presentation to NRC
Regarding ATRIUM-10 Fuel

6.15 Describe how gamma smearing, decay heat, and direct energy deposition are treated in XCOBRA-T. Provide justification for any assumptions in the analysis. If historical (non-ATRIUM-10 or non-cycle loading specific) parameters are used to model the event, justify the use of these values.

Response:

Based on additional clarification of the question from an NRC telecom with AREVA and Entergy on December 4, 2008, the following response is provided.

The decay heat is calculated by COTRANSA2 and is included in the total core power versus time provided as a boundary condition to XCOBRA-T. The decay heat model used in COTRANSA2 is a curve fit (11 groups) to the 1973 ANS standard decay heat model. The COTRANSA2 core power boundary condition includes the decay heat contribution based on the core average power density. The decay heat power remains essentially constant during the transient. Therefore, the decay heat during the transient is primarily a function of initial power density. Application of power peaking factors (axial, radial, local) to the COTRANSA2 average power properly accounts for local decay heat in the XCOBRA-T hot channel analysis.

Gamma smearing does not affect the XCOBRA-T hot channel calculation. The hot channel calculation models an average fuel rod (the average is not affected by flattening of the distribution). The calculation process for determining the peak transient LHGR is equivalent [[redacted]] and is not dependent of the actual rod local peaking factor.

The total core power calculated by COTRANSA2 is distributed between the fuel rod, the active channel coolant, and the core bypass coolant. The fraction deposited in each component is based on fuel type specific calculations performed with the CASMO computer code. For XCOBRA-T, generic power fractions were used for River Bend that approximate the values calculated by CASMO.

An XCOBRA-T deposited power fraction sensitivity study was performed for a BWR for the LRNB transient event. The purpose of the sensitivity study was to show the impact on ΔCPR from using generic ATRIUM-10 power fractions versus case-specific power fractions. The case-specific power fractions are used in COTRANSA2 and are obtained from CASMO-4/MICROBURN-B2. AREVA is in the process of automating the transfer of the case-specific power fractions into XCOBRA-T such that the generic values will no longer be used. [[redacted]] The power that would have been deposited [[redacted]]. A review of an ATRIUM-10 power deposition study showed that the [[redacted]]. A study was performed by taking [[redacted]]. The results are provided in Table 6.15-1. The study shows no significant change in ΔCPR. [[redacted]] This study demonstrates that the ATRIUM 10 generic power fractions in XCOBRA-T are adequate.

Table 6.15-1 Deposited Heat Study

Condition	Fuel Heat	Cladding Heat	Moderator Heat	Bypass Heat	Δ(ΔCPR)
Generic power fractions	[[redacted]]	[[redacted]]	[[redacted]]	[[redacted]]	NA
Case-specific power fractions	[[redacted]]	[[redacted]]	[[redacted]]	[[redacted]]	−0.0004
Case-specific power fractions [[redacted]]	[[redacted]]	[[redacted]]	[[redacted]]	[[redacted]]	+0.0008

6.16 If XCOBRA-T or another one-dimensional code is used to perform the FWCF event analysis, justify the appropriateness of the assumption to hold the radial power shape constant. This justification should consider the sensitivity of the local sub-bundle radial pin power distribution to the instantaneous void fraction.

Response:

Based on additional clarification of the question from an NRC telecom with AREVA and Entergy on December 4, 2008, the following response is provided.

The local sub-bundle radial pin power is accounted for in the SPCB critical power correlation in the form of Feff. Evaluations were performed to assess the impact on ΔCPR for a change in Feff resulting from the variation in the lattice void fraction during a pressurization event for a BWR.

MICROBURN-B2 analyses were performed using the nominal void correlation and an adjusted void correlation to assess the change in Feff as void changes. The MICROBURN-B2 cases were run to reflect an instantaneous change in core average void fraction of +0.05. For the limiting MCPR bundle in the core, the changes in void, local peaking factor (LPF), and Feff were:

Δvoid = +0.0441 (node 24)

Δvoid = +0.0456 (node 23)

ΔLPF = −0.0026 (node 24)

ΔLPF = −0.0030 (node 23)

ΔFeff = 0.0000 (assembly)

For other potentially limiting bundles (10% highest powered bundles) in the core, the change in Feff was between -0.0002 and +0.0011 for a +0.05 core average Δvoid. In general, an increase in void fraction resulted in an increase in Feff for high power, low exposure (end of first cycle) assemblies and a decrease in Feff for low power, high exposure assemblies.

A decrease in Feff during the transient will improve the CPR during the transient and result in a reduced ΔCPR. The converse is true for an increase in Feff during the transient. The sensitivity of MCPR to Feff is about 2 to 1; therefore, the sensitivity of ΔCPR is about twice the ΔFeff during the transient. The change in ΔCPR would be between 0.000 and +0.002 for a +0.05 core average Δvoid.

During a pressurization event, the core void will initially decrease followed by an increase in core void. Therefore, the effect of the change in void on fuel rod peaking factors (and Feff) will tend to be offset during the transient.

The assessment above for the impact of a void change on ΔFeff and Δ(ΔCPR) is based on assuming the nuclear power is instantly converted to surface heat flux. Because the time of MCPR (~1.25 sec) is less than the fuel rod thermal time constant (~ 5 sec), the actual impact on Feff and ΔCPR from the void change will be much less. Likewise, the time of the peak heat flux (~0.8 second) is less than the fuel rod thermal time constant. At the boiling transition plane, there is an insignificant change in void until after the time of peak power. Because the increase in void and the corresponding increase in Feff occur close to the time of MCPR, the slight change in rod power will not significantly change the rod heat flux at the time of MCPR. Therefore, the effect on ΔCPR will be much less than estimated based on the MICROBURN-B2 analyses.

In summary, the above results show that the effect of the variation in void fraction during a transient on the Feff has an insignificant effect on ΔCPR.

6.17 If XCOBRA-T or another one-dimensional code is used to determine the transient hot rod heat flux, please describe how appropriate fuel rod parameters are determined for the analysis. This discussion should address the following: gap conductance, thermal conductivity, pellet size, and heat capacity. Justify that these parameters are acceptably accurate or conservative.

Response:

Based on additional clarification of the question from an NRC telecom with AREVA and Entergy on December 4, 2008, the following response is provided.

The fuel rod models in XCOBRA-T are consistent with RODEX2 (thermal conductivity and heat capacity) and the fuel rod gap conductance values input to XCOBRA-T are obtained from RODEX2 analyses. The gap conductance includes the effect of pellet geometry changes (densification, swelling, etc.) The XCOBRA-T analyses are performed to ensure that fuel rod thermal mechanical limits established with RODEX2 are not exceeded during AOOs. Refer to the response of 6.2 for a similar discussion for COTRANSA2.

For the pressurization transients, gap properties from RODEX2 are used to model the core average gap conductance in COTRANSA2 and the hot channel gap conductance in XCOBRA-T. The core average gap conductance in the system model is not the same as the gap conductance of the hot channel. The gap conductance of the system model is based on the average of all fuel in the core; whereas the hot channel gap conductance is based on a limiting assembly. The gap conductance is a function of exposure.

The COTRANSA2 system model includes the neutronic feedback from a change in the thermal time constant. [[redacted]] The XCOBRA-T hot channel model uses the boundary conditions from COTRANSA2. A higher hot channel gap conductance in XCOBRA-T is conservative for pressurization events since higher values increase the heat flux and coolant quality, and thereby decrease the margin to boiling transition. These trends have been confirmed with AREVA sensitivity calculations.

For fuel rods early in life, the gap is not closed. AREVA applies the following conservatism for gap conductance for fuel rods with significant open gaps: [[redacted]

[redacted]]

Refer to Table 6.2-1 for the gap conductance study.

6.18 If XCOBRA-T or another three-equation thermal hydraulic code is used to perform the FWCF event analysis, please justify the appropriateness of utilizing the [[redacted]]. In particular, pressurization in the FWCF may result in significant changes to the fluid saturation temperature. The code treats these temperature changes as [[redacted]] and may result in changes in the cladding heat flux that are nonphysical relative to the expected behavior based on a more detailed two-fluid representation of the liquid film and vapor fields. Provide detailed transient analyses to demonstrate that the predicted transient peak heat flux is accurately calculated or conservative relative to the limitations in the thermal hydraulic model.

Response:

Based on additional clarification of the question from an NRC telecom with AREVA and Entergy on December 4, 2008, the following response is provided.

The XCOBRA-T hot channel analysis is used to determine the peak LHGR during the transient. The XCOBRA-T hot channel analysis determines the assembly radial peaking factor that results in dryout during the transient. The peak axial LHGR in the fuel assembly during the transient is also calculated in this analysis. However, because the maximum allowed steady state LHGR may be higher then the initial LHGR used in the XCOBRA-T hot channel analysis, the peak transient LHGR may be higher than calculated in the XCOBRA-T analysis. The peak LHGR for the transient is determined [[redacted]] If necessary, the steady state LHGR limit is adjusted to ensure that the peak transient LHGR remains below the transient LHGR limit established from RODEX2 analyses.

Except for post-dryout conditions, which are not experienced for XCOBRA-T analyses, the thermal resistance of the fluid surrounding the cladding is much less than the resistance of the cladding, gap and pellet. A detailed two-fluid representation of the liquid film and vapor fields may determine a different fluid thermal resistance; however, the magnitude of the difference in the fluid resistance will not be significant relative to the total resistance and would not result in a significant impact on the cladding heat flux. Therefore, the limitations of the fluid model in XCOBRA-T would not impact the evaluations of the transient peak LHGR.

6.19 If XCOBRA-T is the code used to perform the transient LHGR analysis, confirm that the thermal hydraulic conditions simulated during the FWCF event do not exceed the application range of the critical heat flux correlation. If these bounds are exceeded the staff is aware that XCOBRA-T [redacted]. If the bounds are exceeded provide justification of the application of the analysis to demonstrate acceptable thermal-mechanical performance.

Response:

Based on additional clarification of the question from an NRC telecom with AREVA and Entergy on December 4, 2008, the following response is provided.

Bounds checking is provided in the XCOBRA-T coding to ensure the conditions provided to the SPCB correlation are within the correlation limits as specified in Table 1.1 of EMF-2209(P)(A), *SPCB Critical Power Correlation.* Should any of the condition limits be violated, the behavior will be as specified in Section 2.6 of this document. In the specific case where the pressure limit is exceeded, XCOBRA-T will write an appropriate error message and terminate the calculation as specified in Section 2.6.3.

With respect to the remaining parameters, the behavior for transient calculations is summarized in the following table. Also, XCOBRA-T evaluates Reynolds number for each node for each step of the calculation. If the flow becomes negative at any node, the code stops the calculation.

Refer to Response 6.18 for a discussion on the insensitivity of fluid conditions on heat flux.

The out-of-bounds corrections affect the [[redacted]] used in the evaluation of the transient LHGR. Therefore, the corrections do not impact the evaluation of the thermal-mechanical performance.

[redacted]		
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

Attachment 3

RBG-46901

Affidavit attesting to the proprietary nature of the information

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

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

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

3. I am familiar with the AREVA NP information contained in 51-9101813-000 entitled, "Reponses to NRC RAI – RBS Main Turbine Bypass System (MTBS) Inoperable," dated January 2009 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 26th

day of January 2009.

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

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2010