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April 4, 2013

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

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Oconee Nuclear Station, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Response to NRC RAI regarding 10 CFR 50.46 Notification of Change in
Peak Cladding Temperature for Large Break Loss of Coolant Accident
Analysis

Reference:

1) Letter, D. C. Culp (Duke Energy) to USNRC, Subject: Oconee Nuclear Station - 30-
   Day Report Pursuant to 10 CFR 50.46, Changes to or Errors in an Evaluation Model,
   March 9, 2012. [ADAMS Accession No. ML12073A354]

2) Email correspondence, John Boska (NRC) to Kent R. Alter (Duke Energy), Subject:
   Oconee Units 1, 2, and 3, NRC Request for Additional Information on Errors
   Reported per 10 CFR 50.46, ME9119, October 26, 2012. [ADAMS Accession No.
   ML12300A201]

3) Letter, Pedro Salas, Director, Regulatory Affairs (AREVA NP Inc.) to USNRC,
   Subject: "Generic RAI Response to a 30-day 10 CFR 50.46 Report of Significant
   PCT Change", (NRC:14:014), December 6, 2012.

4) Email correspondence, John Boska (NRC) to Kent R. Alter (Duke Energy), Subject:
   Oconee Nuclear Station, Units 1, 2, and 3, NRC Request for Additional Information
   on Errors Reported per 10 CFR 50.46 (TACs ME9119, ME9120, ME9121) March 1,
   2013. [ADAMS Accession No. ML13060A417]

5) Letter, Pedro Salas, Director, Regulatory Affairs (AREVA NP Inc.) to USNRC,
   Subject: "Generic RAI Response to a 30-day 10 CFR 50.46 Report of Significant
   PCT Change", (NRC:13:013), March 28, 2013.

On March 9, 2012, (Reference 1) Duke Energy submitted a 30-day report pursuant to
10 CFR 50.46(a)(3)(ii) regarding the impact on Peak Cladding Temperature (PCT) from
two errors in the Emergency Core Cooling System (ECCS) evaluation model used to
assess a postulated Large Break Loss of Coolant Accident (LBLOCA) for Oconee
Nuclear Station. This information is specific to the application of the AREVA ECCS
evaluation model for B&W plants, as applied to Oconee Nuclear Station.

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The NRC transmitted, via email, (Reference 2) Requests for Additional Information
(RAI) regarding the Reference 1 letter, dated March 9, 2012 [ADAMS Accession No.
ML12073A354] concerning the 30-day report pursuant to 10 CFR 50.46(a)(3)(ii).
AREVA provided a generic response to the initial RAI in Reference 3. After considering
AREVA's generic response to the first set of RAI, the NRC transmitted, via email
(Reference 4), new RAI regarding the Reference 3 report. The purpose of this letter is
to provide a response to the March 1, 2013 RAI (see Enclosure).

In response to the March 1, 2013 RAI, AREVA has submitted a generic response to the
NRC in Reference 5.

There are no regulatory commitments contained in this letter.

Please address any comments or questions regarding this matter to Paul Guill at
(704) 382-4753 (paul.guill@duke-energy.com).

Sincerely,

Garry D. Miller
Senior Vice President
Nuclear Engineering

Enclosure: Response to NRC RAI regarding 10 CFR 50.46 Notification of Change in
               Peak Cladding Temperature for Large Break Loss of Coolant Accident
               Analysis

xc (with enclosure):

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DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

ENCLOSURE

RESPONSE TO NRC RAI REGARDING 10 CFR 50.46 NOTIFICATION OF CHANGE IN PEAK CLADDING TEMPERATURE FOR LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS

2 total pages to follow

Response to NRC Requests for Additional Information

NRC RAIs

1. For the analyses completed pertaining to the ECCS bypass error for the lowered loop design, a 2.506-ft peak power location was used, and the analyses for the ECCS bypass error for the raised loop design used a 9.536-ft peak power location. In the December 7, 2012, supplemental letter, the effects of the end-of-bypass timing error are expressed in terms of liquid inventory available to reach the lower plenum and initiate a bottom-up core reflood. The effects of an adiabatic heatup, which is terminated by the core reflood, are also discussed. In consideration of these phenomena, it would appear that a higher elevation in the core would be a more limiting location to evaluate the effects of an error associated with end-of-bypass timing.

Provide information to demonstrate that the bottom-peaked power shape being used for the lowered loop design is conservative and/or appropriate.

2. After evaluating a 177 fuel assembly (FA) lowered loop (LL) plant with column weldments modeled for a 205 FA plant, details of the column weldments for a 177 FA plant were developed. The model for column weldments of a 177 FA plant were then used for the analyses of a raised loop (RL) plant. Two 177 FA raised loop cases showed that the newly developed column weldments increased PCT for an unruptured fuel segment by 3 degrees Fahrenheit.

It was also reported that the column weldments in a lowered loop plant increased PCT by 11.5 degrees Fahrenheit for an unruptured fuel segment and 26.2 degrees Fahrenheit for a ruptured fuel segment. This result was bounded by generically estimating the effect of column weldments to be an increase in PCT of 80 degrees Fahrenheit.

Column weldments in a raised loop plant increased PCT by 8.9 degrees Fahrenheit for an unruptured segment, which is less than the effect seen in the lowered loop design.

a. Provide justification to show that analyzing column weldments modeled for a 177 FA plant has an effect on PCT of the same magnitude in a lowered loop plant as in a raised loop plant.

b. Describe the nodalization for the column weldments used in RELAP5 analyses.

c. Provide simplified drawings to compare the column weldment design for a 205 fuel assembly plant to the column weldments for the 177 fuel assembly plant.

Duke Energy Carolinas Response to NRC RAIs

In response to the RAIs, AREVA has submitted a generic response to the NRC. Please see AREVA Letter from Pedro Salas, Director, Regulatory Affairs to NRC dated March 28, 2013, which transmits AREVA Document ANP-3180, Revision 001, "177 Fuel-Assembly Plant RAI Response to a 30-Day 50.46 Report of Significant PCT Change", March 2013. The AREVA report has been reviewed by Duke Energy and is provided in response to the RAIs for Oconee Nuclear Station.
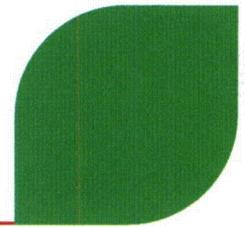
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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
All.	N/A	Initial release addressing RAIs from the fall of 2012.
All.	i through vi, 1-1, 3-1, 4-1 through 5-1, 6-1 through 6-2	Updated the document throughout to address the supplemental RAIs that were sent in the spring of 2013 and replaced the template with the current version.

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Nomenclature

Acronym	Definition
BAP	Boric Acid Precipitation
BAST	Boric Acid Storage Tank
BLN	Bellefonte Nuclear Plant
BOCR	Bottom of Core Recovery
BOL	Beginning-of-Life
BWST	Borated Water Storage Tank
B&W	Babcock and Wilcox
CFR	Code of Federal Regulations
CFT	Core Flood Tank (Accumulator)
CL	Cold Leg
CLPD	Cold Leg Pump Discharge
CLPS	Cold Leg Pump Suction
CR-3	Crystal River Unit 3
CW	Column Weldment (upper support)
DB	Davis-Besse Nuclear Plant
DC	Downcomer
EM	Evaluation Model
ECCS	Emergency Core Cooling System
EPU	Extended Power Uprate
FA	Fuel Assembly
FPC	Florida Power Corporation
HC	Hot Channel
HLI	Hot Leg Injection
HL-BD	Hot Leg Blowdown
HPI	High Pressure Injection
ITS	Improved Technical Specification
LAR	License Amendment Request
LBLOCA	Large Break Loss-of-Coolant Accident
LHR	Linear Heat Rate
LL	Lowered-Loop (plant arrangement)
LOCA	Loss-of-Coolant Accident
LP	Lower Plenum
LPI	Low Pressure Injection

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Acronym	Definition
LTCC	Long-Term Core Cooling
MOL	Middle-of-Life
MWt	Megawatts
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PCT	Peak Cladding Temperature
PSC	Preliminary Safety Concern
RAI	Request for Additional Information
RBS	Reactor Building Spray
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RL	Raised-Loop (plant arrangement)
RV	Reactor Vessel
RVC	Reactor Vessel Nozzle Centerline
RVVV	Reactor Vessel Vent Valve
SBLOCA	Small Break Loss of Coolant Accident
SG	Steam Generator
SNPB	Nuclear Performance and Code Review Branch
UP	Upper Plenum
UPTF	Upper Plenum Test Facility
WebCAP	AREVA Proprietary Web-Based Corrective Action Program

1.0 INTRODUCTION

In late February 2012, AREVA provided each of the B&W-designed 177 fuel-assembly (FA) plants with a 30-day 50.46 report of significant peak cladding temperature (PCT) changes for an error and change in the Evaluation Model (EM) used for plant licensing. The utilities separately reported these to the NRC in March 2012. The NRC has sent two requests for additional information (RAIs) to each utility regarding these letters in October and November 2012 (References [7-11]). The first RAI question requests additional details on the supporting LOCA analyses addressing the two PCT changes. The second RAI question is for the licensees, and it requests justification for not providing a schedule for reanalysis since each change was classified as significant. In March 2013, the NRC sent two additional RAI questions (References [13] and [14]) seeking clarification to the RAI responses from 2012. At the time this report was prepared, Entergy Operations, Inc., the licensee for Arkansas Nuclear One, Unit 1 (ANO-1), did not receive the two additional RAI questions from the NRC. Based on communication between the NRC and AREVA, it is anticipated that the RAIs addressed to Entergy will be equivalent to the RAIs sent to the other 177-FA plants (References [13] and [14]).

The purpose of this material is to provide detailed information related to the error correction and the EM modeling change that the utilities can individually reference for their RAI responses to the RAIs from 2012 and 2013. The information contained in the response (Reference [12]) to the first question in the set of RAIs from 2012 was used by the utilities for responding to the second question from that RAI set. Revision 1 provides supplemental explanation to address the second set of RAIs that were sent in March 2013 (References [13] and [14]).

The generic NRC RAI responses provided in this document do not contain any Proprietary Information to be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390(a)(4).

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2.0   BACKGROUND

A large break loss-of-coolant accident (LBLOCA) analysis sensitivity study with the BWNT LOCA EM (BAW-10192P-A Rev. 0 Reference [1]) for the 205-FA Bellefonte (BLN) plant completion restart efforts uncovered a mathematical error in the emergency core cooling system (ECCS) bypass logic implemented via a control system that used trips and control variables to compute a balance between the steam energy and the ECCS condensation potential energy. The error resulted in an end of ECCS bypass time that was roughly 2 seconds sooner when the control system was corrected. With the correction, the bottom of core recovery (BOCR) time occurred earlier, so the calculated core refill period was shorter and core cooling during reflood began earlier. As a result, lower PCTs were predicted for BLN. The ECCS bypass error was identified in an AREVA Condition Report.

Shortly after the BLN ECCS bypass error was discovered, another 205-FA LOCA sensitivity study was completed with a model that simulated the control rod housing support column weldment over the hot channel. It was discovered that the core cooling during the second half of the blowdown phase was worse when the hot bundle was located under a column weldment. The adverse core cooling effects associated with the column weldment (CW) modeling were identified in a separate AREVA Condition Report.

In evaluating the extent of condition for these two issues, similar errors and modeling choices were found in the 177-FA plant LOCA analyses as well. Given that these two issues were being evaluated at the same time, the ECCS bypass correction and the column weldment EM deficiency were combined in the same 50.46 report that was provided to the B&W plant licensees in February 2012. Separate assessments of each issue were performed via scoping analyses to show the impacts on the calculated cladding temperatures. The scoping analyses with the corrected ECCS bypass model showed that a generic PCT decrease of roughly 80 F was conservative for use in all 177-FA plant analyses. When a generic column weldment was modeled over the hot channel, the PCT increased by roughly the same magnitude that it had decreased with the ECCS bypass error. When considered together generically, the two individual significant PCT changes combined to produce roughly the same PCT from the original analysis, which did not include either of the two corrections.

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3.0   RESPONSE TO FIRST SET OF NRC RAIS FROM 2012

This section repeats the generically paraphrased NRC RAI and provides a generic
response to the question.

1. *There are two changes to peak cladding temperature (PCT) for Large Break*
   Loss-of-Coolant Accident (LBLOCA) analysis discussed in the report submitted
   by the licensee. The first change is an Evaluation Model (EM) application error
   in the determination of the end of Emergency Core Cooling System (ECCS)
   bypass, which resulted in an 80 F decrease in PCT. The second change is an
   EM modeling change to include the effects of the upper plenum column
   weldments, which resulted in an 80 F increase in PCT.

   Provide the analysis that lead to each change having an 80 degree change in PCT.

Generic Response:

Two issues, incorrect end of ECCS bypass time determination and upper plenum
column weldment modeling, were identified from 205 Fuel Assembly (205-FA) B&W
LBLOCA analyses and these also impact the 177-FA B&W LOCA analyses. These
issues were considered and evaluated for the purpose of providing PCT estimates to
support 50.46 reporting for the 177-FA B&W plants.

A mathematical error was identified in the energy balance calculations with control
variables used to determine the complete end of ECCS bypass in LBLOCA applications.
These erroneous control variables were similar and common to all (205-FA and 177-FA)
LBLOCA ECCS bypass models, so the error was propagated to each LBLOCA model.
All ECCS liquid in the reactor vessel or inlet pipes injected during the ECCS bypass
period of the blowdown phase of a LBLOCA is subtracted from the reactor vessel
calculated inventory to be in compliance with 10 CFR 50 Appendix K. The error in the
control variables incorrectly calculated the complete ECCS bypass time and liquid mass
that should have remained in the vessel at the end of blowdown.

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For EM applications associated with B&W-designed reactors, BAW-10192P-A (Reference [1]), "BWNT Loss-of-Coolant Accident-Evaluation Model for Once-Through Steam Generator Plant", defines both partial and complete end-of-bypass periods. The partial end-of-bypass is defined as the time at which some portion of ECCS injected during the Core Flood Tank (CFT) or low pressure injection (LPI) period reaches the lower plenum. The complete end of bypass is achieved when all of the injected ECCS reaches the lower plenum prior to the beginning of the REFLOD3B core reflood analysis.

The complete end-of-bypass time is defined by the earliest of the end-of-blowdown time or the time at which the ECCS flows can condense all the steam flowing into the upper downcomer region. A conservative condensation efficiency based on the Upper Plenum Test Facility (UPTF) results was used when the plant steam flows are within the UPTF test data range. The total ECCS flow following this complete end-of-bypass period is integrated and added to the partial end-of-bypass ECCS liquid to determine the quantity of liquid that should have reached the lower plenum during the end-of-bypass period.

When the energy balance error was corrected, the end-of-bypass time was earlier and the amount of ECCS fluid that is not bypassed increased. The ECCS bypass liquid added to the lower plenum caused the time of BOCR to occur earlier. This timing shift decreases the calculated PCT in all cases. The magnitude of the maximum cladding temperature decrease was dependent on whether the cladding segment was unruptured or if it had ruptured.

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Sensitivity studies were performed to consider the hot channel location relative to a CW modeled in the upper plenum of the RELAP5 models. This study was performed in part due to considerations of how variations in the fluid temperature in the Reactor Vessel Closure Head of B&W plants could be higher than the hot leg temperature. The CWs are important in determining the temperature of the fluid reaching the upper head since they physically allow a portion of the flow exiting from the core channels underneath them to reach the upper head. The CW structure limits mixing with flows exiting other fuel assembles, the flow exiting the guide tubes of the other assemblies, and core baffle flows.

The LBLOCA RELAP5 model development for the Bellefonte plant incorporated the detailed column weldments over top of the hot bundle. Bellefonte is a B&W-designed 205 Fuel Assembly Raised Loop plant and sensitivity studies confirmed that PCTs were higher when a CW is located over the hot bundle. Since these 205-FA plant CW results are not directly applicable to the 177-FA plants, some 177-FA scoping cases were performed. These studies suggested that the results could be worse and that additional evaluation was required. Additional analyses were performed and this document provides details of the analyses used to develop the PCT impact of the column weldments for the 177-FA plants combined with the ECCS bypass error correction to support 50.46 reporting.

Analytical Modeling Details

LBLOCA analyses are performed in accordance with the NRC approved RELAP5-based EM contained in Reference [1], as amended by any NRC-approved code topical revisions, 10 CFR 50.46 changes made associated with preliminary safety concern (PSC) resolutions, and method changes based on the NRC approved M5 cladding topical report (Reference [6]). The LBLOCA methodology makes use of four computer codes as discussed in the EM (Reference [1]), which are interconnected as depicted in Figure 4-1 of the EM (Volume I). The RELAP5/MOD2-B&W code (Reference [2]) calculates the system thermal hydraulics and cladding temperature response, including the hot assembly and the hot pin during blowdown. The thermal hydraulic transient calculations are continued within the REFLOD3B code (Reference [3]) to determine refill time and core reflooding rates. The BEACH code (Reference [4]) determines the cladding temperature response during the refill and reflood phases of the transient. The CONTEMPT code (Reference [5]) calculates the containment pressure response with time and requires mass and energy release input from both RELAP5 and REFLOD3B codes. The containment pressure response is primarily determined by the mass and energy releases during the transient and these were effectively unchanged so new minimum containment pressure responses were not needed for the ECCS bypass and CW modeling changes.

177-FA LL Plant ECCS Bypass Error Analyses

The effects of the ECCS bypass error were evaluated by performing a specific 2.506-ft peak power location beginning-of-life (BOL) RELAP5 blowdown analysis with corrected bypass control variables for a 177-FA plant. The ECCS bypass did not change the fuel pin heat removal in RELAP5 blowdown analysis but it does change the inputs to REFLOD3B and it reduced the end of ECCS bypass time by roughly 2 seconds.

The previous REFLOD3B case was reanalyzed with this earlier end of bypass and it shortened adiabatic heatup period by 1.85 seconds. The BOCR timing and the new flooding rate from this case was provided as a boundary condition to a new BEACH analysis. This analysis predicted the limiting PCT decreased by 90.3 F for the ruptured node. The peak unruptured segment decreased by 45.4 F. This PCT difference is consistent with typical observations in which the inside metal-water reaction produces a 2 to 1 ΔPCT variation between ruptured and unruptured segments, respectively. Given that each plant should have similar earlier end of bypass timing, it was concluded that the calculated results for this 177-FA plant case could be applied to each plant and each axial power elevation peak. Based on this execution of a RELAP5, REFLOD3B, and BEACH case, a slightly conservative PCT reduction of 80 F for ruptured segments and 40 F for unruptured segments was assigned generically to all plant LBLOCA analyses.

Since the ECCS bypass model is not used in EM small break LOCA (SBLOCA) applications, there is no impact to any SBLOCA analysis results.

177-FA LL Plant Column Weldment Model Change Analyses

Since the details of the column weldments for the 177-FA plants were not readily available without several months of model development and changes, a simplified approach was developed to expedite the evaluations. The simplified approach used the column weldment model that was developed for the 205-FA BLN plant and incorporated it over the hot channel (HC) for a generic 177-FA plant evaluation. Although not formally documented, this simplified approach was tested on the original 205-FA model to confirm it would provide estimated results that was representative of a detailed CW model. This test confirmed that a simple CW added over the HC could be used to provide inputs to a 50.46 evaluation. This simplified model was justified as reasonable to determine CW modeling impacts on the transient progression and ultimately the overall estimated PCT.

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The modeling of the CW over the HC in a 177-FA plant in the RELAP5 blowdown analysis resulted in a ~40 F (calculated 35.6 F) increase in the end-of-blowdown (EOB) fuel temperature. The fuel temperature increase observed at EOB is generally maintained throughout the refill period because the fuel heats up nearly adiabatically during this phase. The actual fuel temperature increase at BOCR was calculated to be 30.2 F for the plant specific case analyzed.

The generic estimate of a 40 F increase in fuel temperature at the time of BOCR generally leads to a doubling of the ruptured cladding segment temperature increase driven by the fuel-to-clad heat transfer during the adiabatic heatup period and the additional metal water reaction energy deposited directly into the cladding. Therefore, the generic ruptured-limited cladding segments should experience a PCT increase of 80 F, estimated based on a 40 F increase in EOB fuel temperature. The unruptured segment cladding PCT will increase in proportion to the EOB fuel temperature, so it was estimated to increase by roughly 40 F. The actual BEACH calculation predicted a 26.2 F increase in rupture-segment PCT due to a favorable shift of rupture time into the blowdown phase. The BEACH calculation predicted an 11.5 F increase in the unruptured segment PCT. While these RELAP5 and BEACH cases predicted smaller increases than was estimated, a favorable rupture time is not assured in all cases, so the generic estimate is conservative. This penalty was assigned to all plants generically.

The detailed CW modeling used for LBLOCA influenced the fuel pin temperatures after the flow direction reversed (flow from top to bottom of the core). The SBLOCA break size is smaller and the net core flows remain upward during the transient. Given the flow is upward for SBLOCA applications and there is a slower progression during the SBLOCA transient, column weldments are not expected to affect the SBLOCA fuel temperature response. Therefore, including column weldments in the SBLOCA model was judged to be unnecessary as it will not affect the limiting SBLOCA PCT.

177-FA LL Plant Combined ECCS Bypass and Column Weldment PCT Changes

The changes in the lowered loop calculated PCTs were considered independently and they were also combined and extended to all 177-FA plant types and fuel designs. While EM-type calculations with RELAP5, REFLOD3B, and BEACH were performed for only one plant design and axial power peak, the changes observed were considered and it was deemed that they were directly applicable to or conservative for all plants and axial peaking locations.

Some key comparison plots were generated for the 177-FA LL plants in Figure 3-1, Figure 3-2, Figure 3-3, and Figure 3-4. Each figure shows the results for three cases:

1) The base model with the ECCS bypass error and no CWs,

2) The corrected ECCS bypass result (no separate CWs modeled), and

3) The corrected ECCS bypass with CWs.

Figure 3-1 provides the fuel volume-average temperature for the limiting PCT location, while Figure 3-2 shows the ruptured cladding temperatures at the limiting PCT location for the entire LBLOCA transient. In these analyses, the reduction in fuel temperature and PCT from the ECCS bypass correction begins around 28 seconds (time of BOCR) versus the base case. The ECCS bypass case fuel and cladding temperatures are lowest near the PCT time. The ECCS bypass and CW case (3) is in between the base case (1) and the ECCS bypass case (2).

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The cladding temperatures from the PCT location are shown in Figure 3-3 for the blowdown and early reflood period. In this plot, the base case and the ECCS bypass case have identical cladding temperatures. The CW study shows that the temperatures remained similar during the first 7 seconds, which is during the core up flow period. Some larger temperature differences are observed in around 7 and then again at 13 seconds. These temperature changes are associated with variations in core flow near the time when the core flow reverses or during the reverse flow period shortly after the CFT flow and its condensation begins. Figure 3-4 provides the HC hot spot flow for the blowdown and early reflood period to confirm the cladding temperature changes are associated with the core flow variations.

When both changes were considered together to define an overall generic PCT change, variations in allowed linear heat rate (LHR) limits were included in the assessments to ensure all plants were covered in the evaluations. The generic analysis that was performed used the Mark-B-HTP fuel with a linear heat rate of 17.8 kW/ft. The fuel heat up rate for Mark-B-HTP fuel with a linear heat rate of 17.8 kW/ft is roughly 21.7 F/s during the refill period from the base case. With a 1.85 second shorter refill period, the fuel temperature will decrease by approximately 40 F (21.7 F/s x 1.85 s) due to the ECCS bypass error correction. If the LHR limit was 15.8 kW/ft, the approximate decrease in fuel temperature is estimated to be 35.6 F (21.7 F/s x 15.8 kW/ft x 1.85 s / 17.8 kW/ft). This 35.6 F decrease calculated in fuel temperature from the ECCS bypass change is equivalent to the analyzed fuel temperature increase for the CW cases. Therefore, for UO_2 LHR limits between 15.8 and 17.8 kW/ft that produce limiting PCTs, the reduction in the ECCS bypass covers the fuel temperature increase observed with the CW. All of the peak LHR limits used to calculate the limiting PCTs for all B&W-designed plants fall within this LHR range.

If the combination of ECCS bypass with a CW over the HC does not change the fuel temperature at BOCR, then the estimated PCT will effectively be unchanged. This BOCR fuel temperature evaluation concludes that the LBLOCA PCT change is near zero at the LHR limits used for the PCT limited cases on a B&W-designed plant.

Supplemental Assessments

Following receipt of the NRC RAIs on the ECCS bypass and column weldment changes, AREVA reviewed the previous assessments based on the core inlet power peaking cases for a lowered-loop plant. AREVA confirmed that the conclusions drawn appeared to be valid given the conservative nature of the generic estimates based on the analyses performed; however, there were questions regarding extension of the core inlet peak analysis results to the core exit or possibly for the 177-FA raised-loop (RL) plant. To address the questions, a series of EM-type analyses for the core exit power peak analysis was performed for the raised-loop plant to defend the validity of the core inlet peak for all other power peak locations and plants.

177-FA RL Plant ECCS Bypass Error Analyses

The effects of the ECCS bypass error were evaluated by performing a specific core exit 9.536-ft peak power location at BOL. The RELAP5 blowdown analysis used corrected bypass control variables for a 177-FA RL plant. The ECCS bypass did not change the fuel pin heat removal in RELAP5 blowdown analysis but, as it did for the 177-FA LL analyses, it did change the inputs to REFLOD3B and it reduced the end of bypass time by roughly 2 seconds.

The previous REFLOD3B case was reanalyzed with this earlier end of bypass and it shortened adiabatic heatup period by 1.9 seconds. The BOCR timing and the new flooding rate from this case was provided as a boundary condition to a new BEACH analysis that predicted the limiting PCT, which was for the unruptured node, decreased by 51.7 F. The peak ruptured segment temperature, which was not limiting, decreased by 96.7 F. This LBLOCA PCT reduction is slightly larger than that observed for the 177-FA LL plant core inlet peaking, and it confirms that the previous generic reductions were reasonable to conservative for the core exit power peaks and the raised-loop plant. These conclusions are based on this execution of RELAP5, REFLOD3B, and BEACH cases.

177-FA RL Plant Column Weldment Model Change Analyses

Since the details of the column weldments for the 177-FA plants were not readily available when the CWs were first assessed for the condition report, a simplified approach was applied using the 205-FA BLN plant CW over the HC for the 177-FA plant evaluations. Additional details of 177-FA CWs have been developed and they were used to confirm the simplified 205-FA CW approach was appropriate. Two 177-RL cases showed that the 205-FA CW EOB peak fuel temperature was 3 F lower than the 177-FA CW model EOB peak fuel temperature. This comparison shows that there was little difference, but the difference observed for the 177-FA to 205-FA CW differences was amply covered by the PCT conservatism applied to the generic PCT penalty.

The modeling of the CW over the HC in a 177-FA RL plant resulted in a lower EOB peak fuel temperature prediction by RELAP5, but the location of the peak fuel temperature also changed. However, the unruptured segment PCT increased by 8.9 F for the plant specific RL BEACH calculation. The generic CW estimate of a 40 F increase in the unruptured PCT is confirmed by the analysis to be bounding and this result is consistent with the previous results based on the 177-FA LL core inlet peak evaluations. These RELAP5 and BEACH cases predicted smaller increases than was estimated for the unruptured PCT, so the generic penalty remains bounding. The non-limiting ruptured segment cladding temperature decreased for this analyzed case over the previous cases. A favorable shift in the timing of lock-in to film boiling and its influence on the cladding rupture conditions and timing confirmed that the generic estimate of an 80 F increase was bounding.

177-FA RL Plant Combined ECCS Bypass and Column Weldment PCT Changes

The changes in the lowered loop calculated PCTs were considered independently and they were also combined and extended to all 177-FA plant types and fuel designs. While EM-type calculations with RELAP5, REFLOD3B, and BEACH were performed for only one plant design and axial power peak, additional work has been completed to show the changes assessed were applicable to or conservative for all plants and axial peaking locations.

Some key comparison plots were generated for the 177-FA RL plants in Figure 3-5, Figure 3-6, and Figure 3-7. Each figure shows the results for three cases:

1) The base model with the ECCS bypass error and no CWs,

2) The corrected ECCS bypass result (no separate CWs modeled), and

3) The corrected ECCS bypass with CWs.

Figure 3-5 provides the fuel volume-average temperature for the limiting PCT location, while Figure 3-6 shows the unruptured cladding temperatures at the limiting PCT location for the entire LBLOCA transient. In these analyses, the reduction in fuel temperature and PCT from the ECCS bypass correction begins around 28 seconds (time of BOCR) versus the base case. The ECCS bypass case fuel and cladding temperatures are lowest near the PCT time. As with the 177-FA LL plants, the ECCS bypass and CW case (3) is in between the base case (1) and the ECCS bypass case (3).

The cladding temperatures from the PCT location are shown in Figure 3-7 for the blowdown and early reflood period. In this plot, the base case and the ECCS bypass case have identical cladding temperatures, but the CW shows some small differences around 2 seconds associated with a short term flow reversal at that time.

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Figure 3-1: 177-FA LL 2.506-ft BOL LBLOCA RELAP5 Analysis,

Comparison of HP Fuel Temperatures (PCT Node 6)

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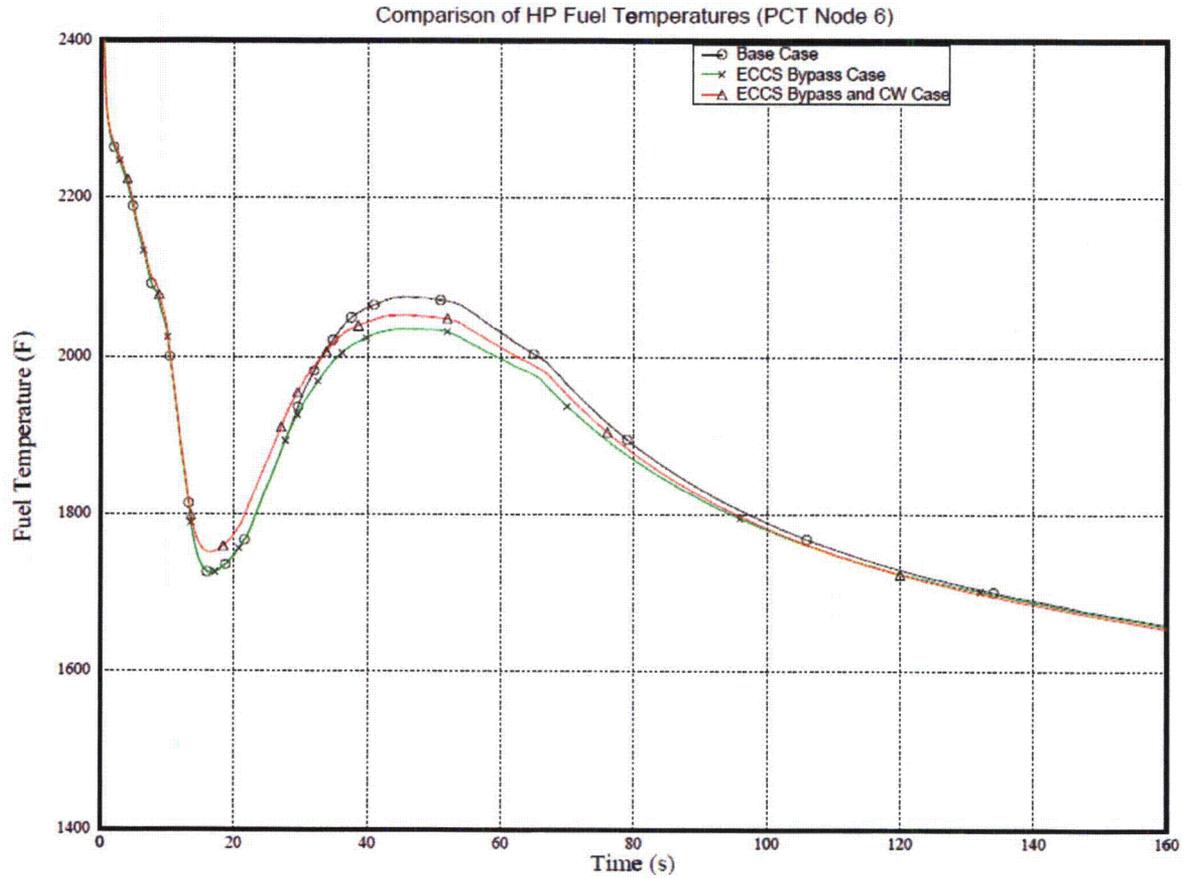
Figure 3-2: 177-FA LL 2.506-ft BOL LBLOCA RELAP5 Analysis,

Comparison of HP Clad Temperatures (PCT Node 6)

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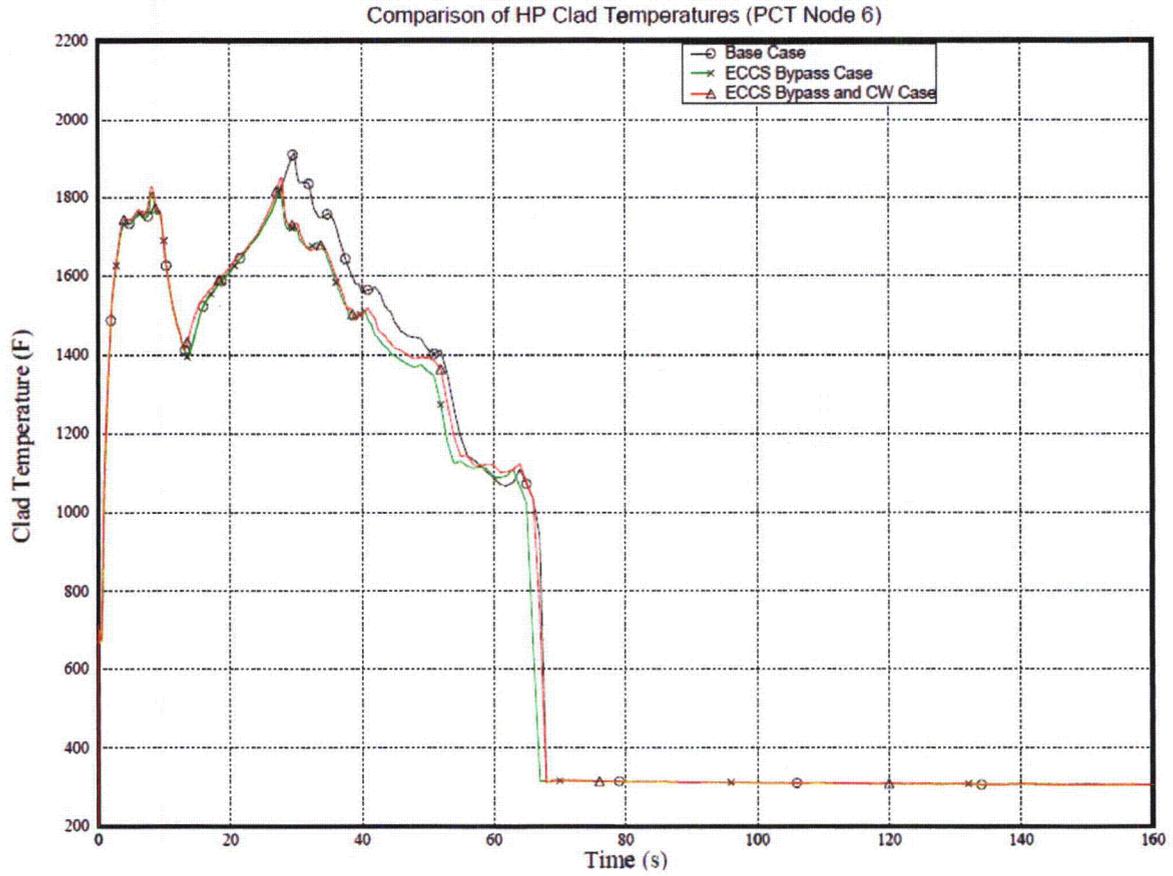
Figure 3-3: 177-FA LL 2.506-ft BOL LBLOCA RELAP5 Analysis,

Comparison of HP Clad Temperatures (PCT Node 6)

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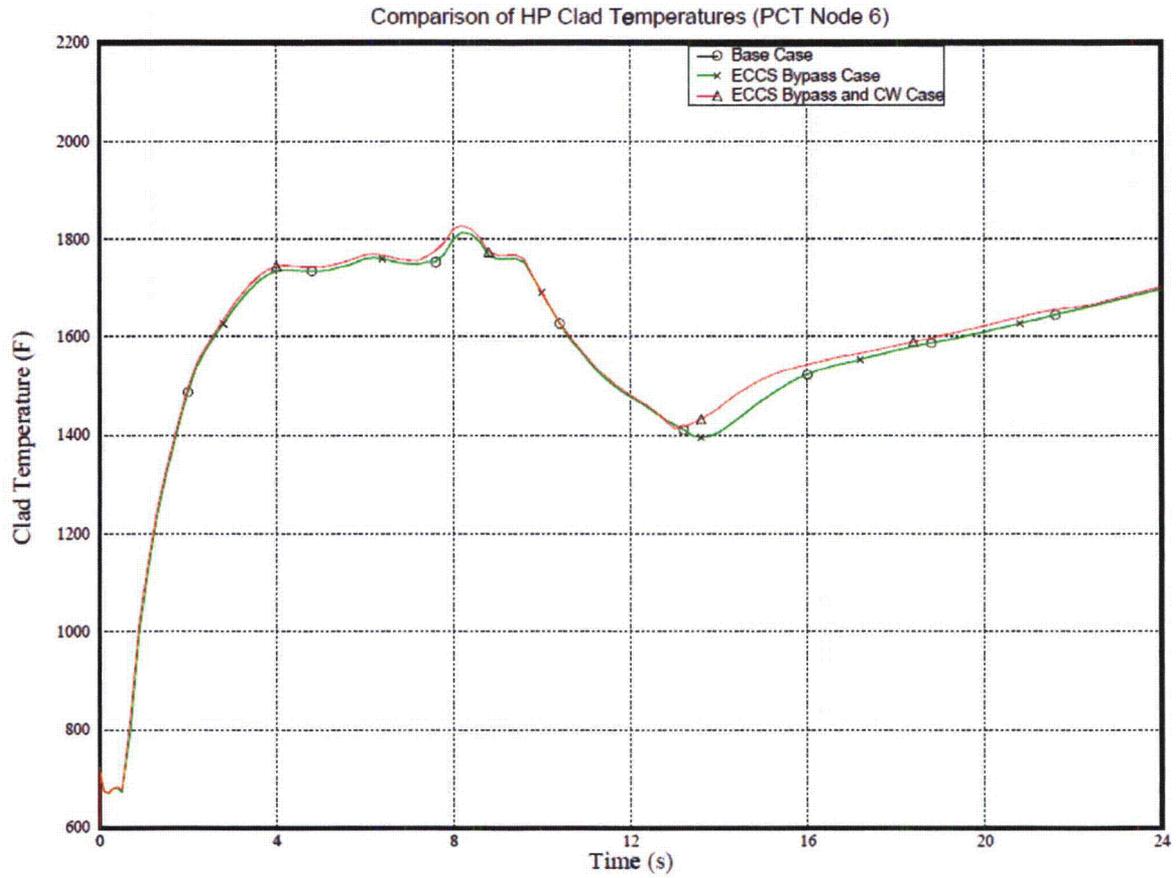
Figure 3-4: 177-FA LL 2.506-ft BOL LBLOCA RELAP5 Analysis,

Comparison of HC Flow Rate (Exit of PCT Node 6)

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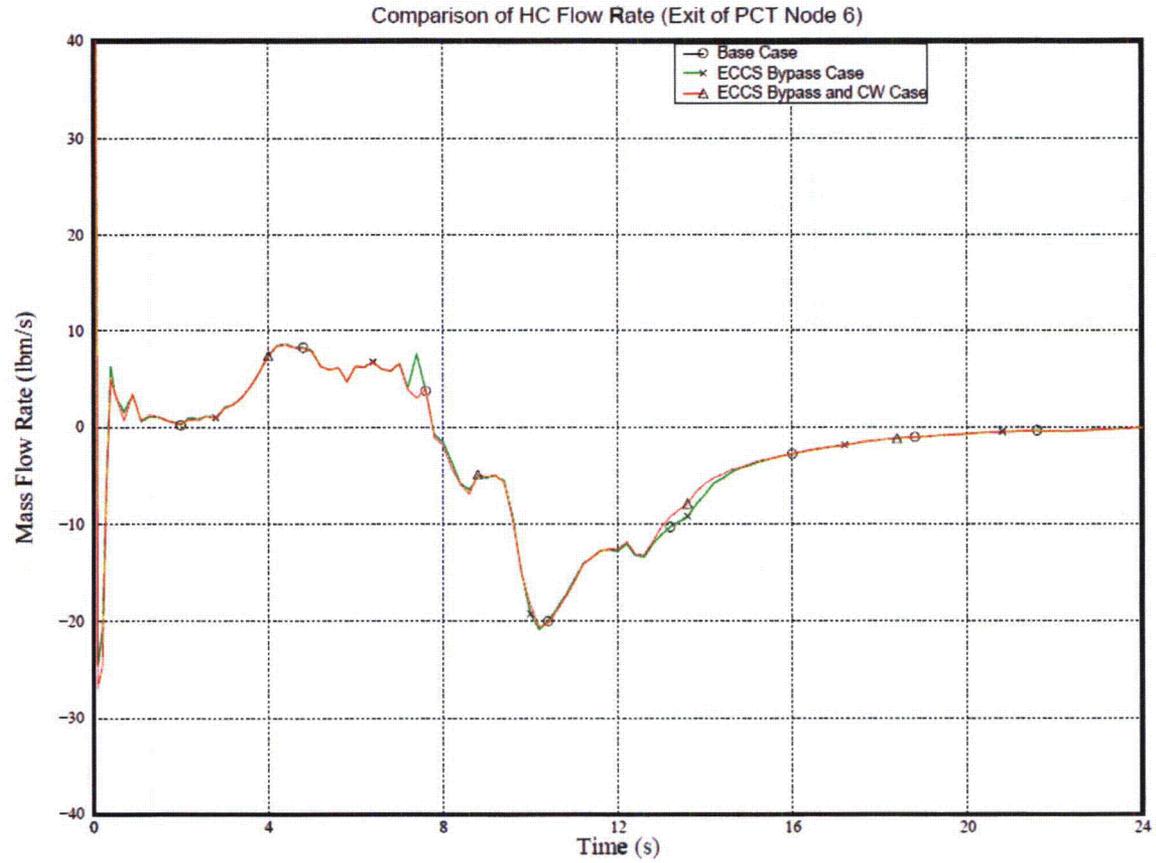
Figure 3-5: 177-FA RL 9.536-ft BOL LBLOCA RELAP5 Analysis,

Comparison of HP Fuel Temperatures (PCT Node 16)

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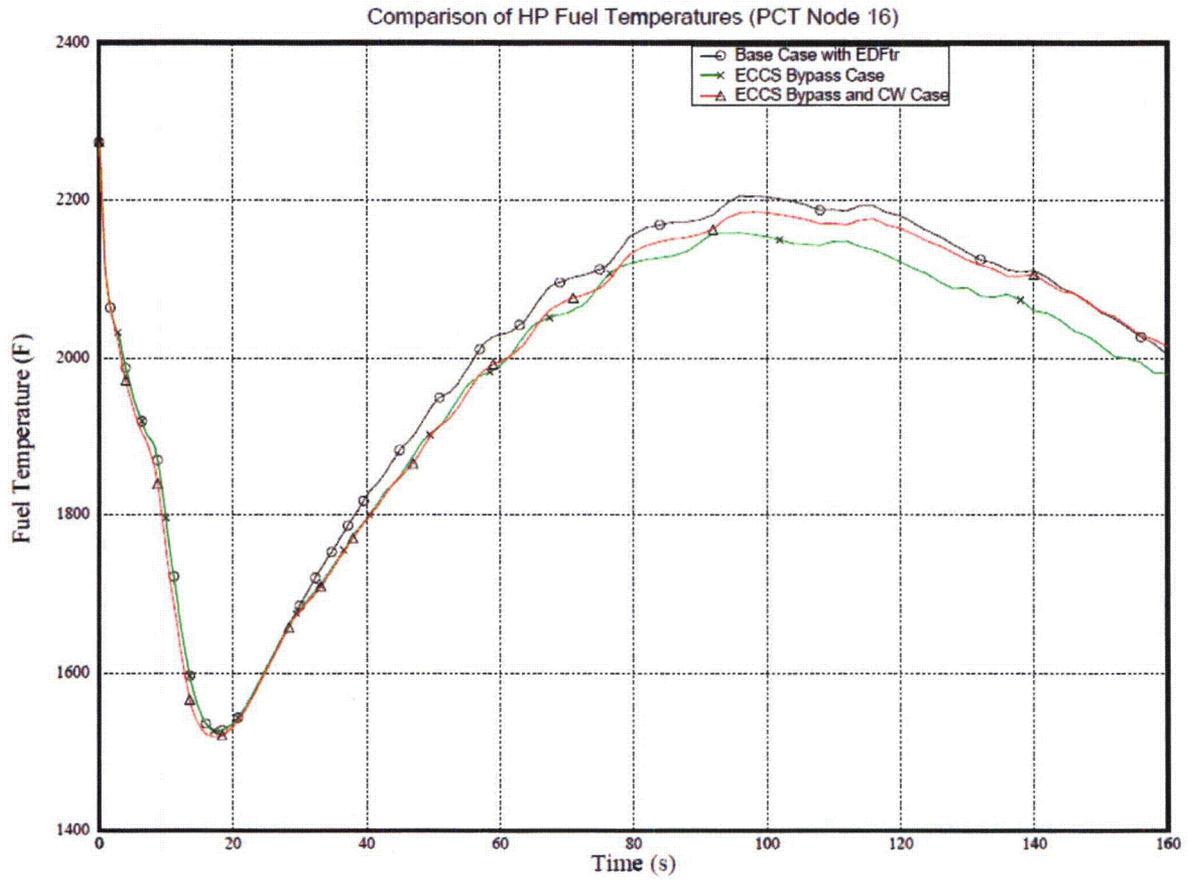
Figure 3-6: 177-FA RL 9.536-ft BOL LBLOCA RELAP5 Analysis,

Comparison of HP Clad Temperatures (PCT Node 16)

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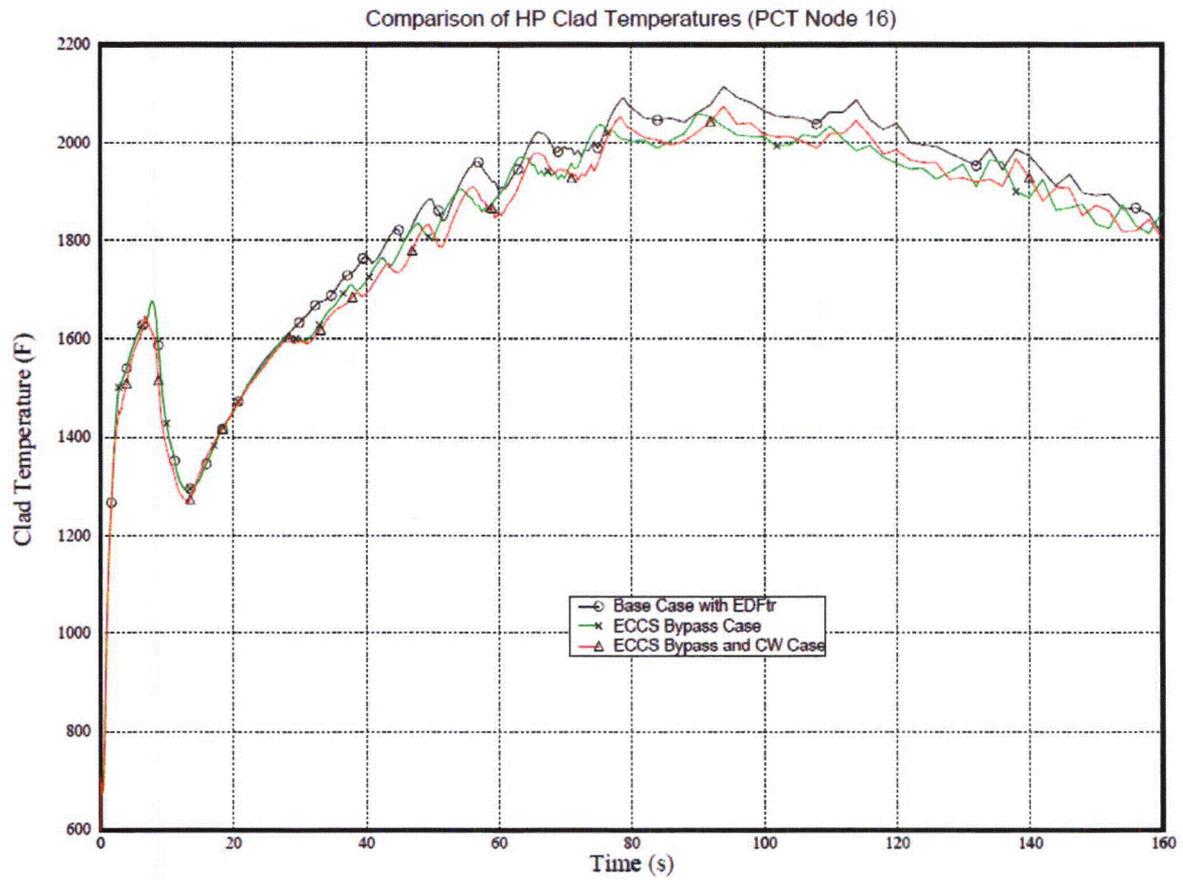
Figure 3-7: 177-FA RL 9.536-ft BOL LBLOCA RELAP5 Analysis,

Comparison of HP Clad Temperatures (PCT Node 16)
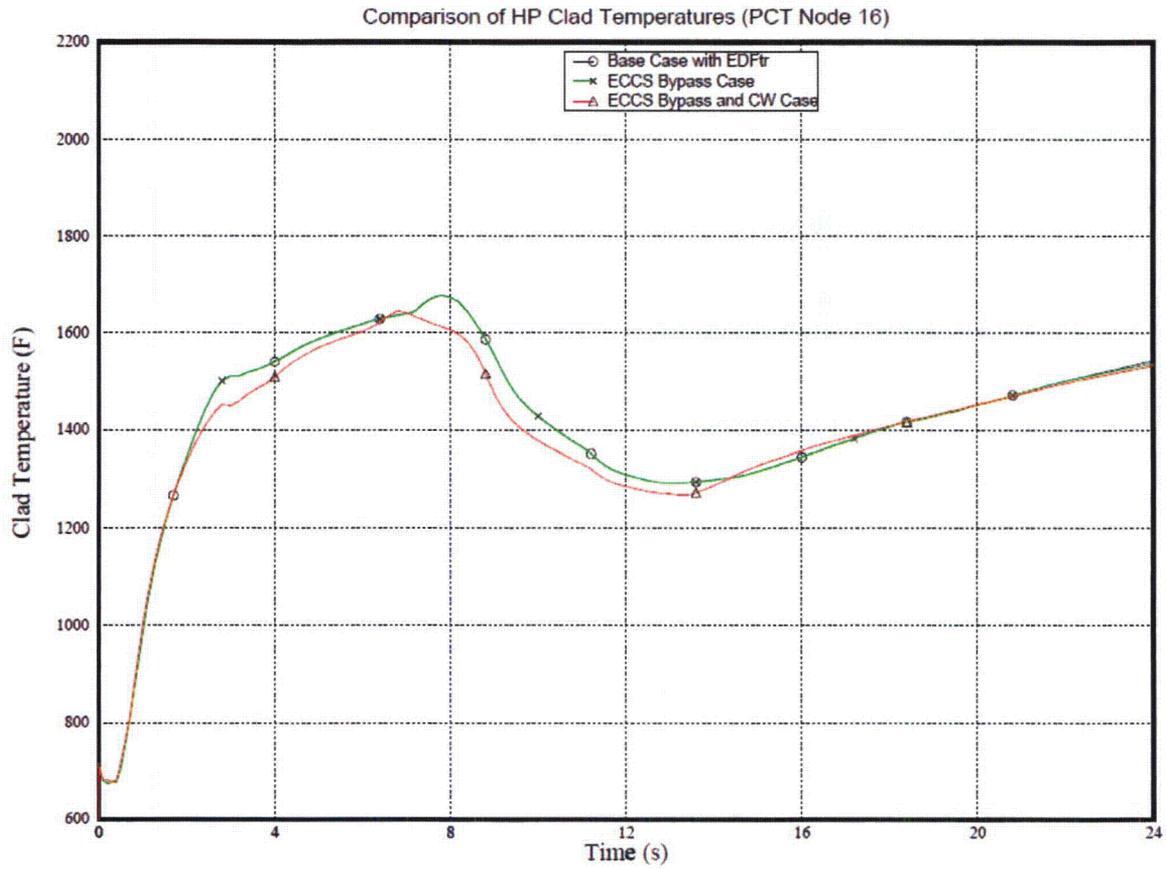
4.0   RESPONSE TO SECOND SET OF NRC RAIS FROM 2013

This section repeats the generic NRC RAIs and provides a generic response to the questions.

1. *For the analyses completed pertaining to the Emergency Core Cooling System (ECCS) bypass error for the lowered loop design, a 2.506-ft peak power location was used, and the analyses for the ECCS bypass error for the raised loop design used a 9.536-ft peak power location. In the December 6, 2012, supplemental letter, the effects of the end-of bypass timing error are expressed in terms of liquid inventory available to reach the lower plenum and initiate a bottom-up core reflood. The effects of an adiabatic heatup, which is terminated by the core reflood, are also discussed. In consideration of these phenomena, it would appear that a higher elevation in the core would be a more limiting location to evaluate the effects of an error associated with end-of-bypass timing.*

   Provide information to demonstrate that the bottom-peaked power shape being used for the lowered loop design is conservative and/or appropriate.

Response to RAI 1:

The limiting LBLOCA PCT for the raised-loop design occurred in the unruptured segment at the 9.536-ft elevation. After correcting the bypass error, the PCT decreased by 96.7 F and 51.7 F in the ruptured and unruptured segments, respectively. The PCT reduction was slightly less pronounced in the lowered-loop (LL) ECCS bypass evaluation using a 2.506-ft peak power location. The PCT for the lowered-loop plant was reduced by 90.3 F in the ruptured segment and 45.4 F in the unruptured segment after correcting the ECCS bypass error. The PCT decrease for the bottom-peaked power shape from the lowered-loop analysis is more conservative because correcting the ECCS bypass error resulted in a smaller PCT reduction compared to the raised-loop analyses for the top skewed peak power elevation. Limiting the PCT reduction to 80 F for ruptured segments and 40 F for unruptured segments generically for all plant LBLOCA analyses is conservative and appropriate. These temperature changes are shown in Table 4-1.

2. *After evaluating a 177 fuel assembly (FA) lowered loop plant with column weldments modeled for a 205 FA plant, details of the column weldments for a 177 FA plant were developed. The model for column weldments of a 177 FA plant were then used for the analyses of a raised loop plant. Two 177 FA raised loop cases showed that the newly developed column weldments increased PCT, for an unruptured fuel segment, by 3 degrees Fahrenheit.*

   It was also reported that the column weldments in a lowered loop plant increased PCT by 11.5 degrees Fahrenheit for an unruptured fuel segment and 26.2 degrees Fahrenheit for a ruptured fuel segment. This result was bounded by generically estimating the effect of column weldments to be an increase in PCT of 80 degrees Fahrenheit.

   Column weldments in a raised loop plant increased PCT by 8.9 degrees Fahrenheit for an unruptured segment, which is less than the effect seen in the lowered loop design.

  a. Provide justification to show that analyzing column weldments modeled for a
  177 FA plant has an effect on PCT of the same magnitude in a lowered loop
  plant as in a raised loop plant.

  b. Describe the nodalization for the column weldments used in the RELAP5
  analyses.

  c. Provide simplified drawings to compare the column weldment design for a
  205 FA plant to the column weldments for the 177 FA plant.

Response to RAI 2a:

When column weldments are explicitly modeled in the upper plenum model of the LBLOCA analyses, the structures impede coolant flow entering the top of the core after the core flow reverses direction. The crossflow into the lower CW holes and slots from the upper plenum is restricted in the reverse-flow direction due to the form losses. The hole and slot sizes in the lower CWs are identical between the two 205-FA and 177-FA plant designs; therefore, the controlling resistances and flow areas for crossflow into or out of the lower column weldments are the same.

The flow areas inside the column weldment for the 177-FA plants are 7 percent smaller than those in the 205-FA plant. This is related to the 15x15 versus 17x17 fuel bundle arrangements and a different number of control rods. Compared to a case with only the corrected ECCS bypass calculation, it was observed that modeling the 177-FA CW in a RL plant increased PCT by 8.9 F for an unruptured segment. For the LL plant, the PCT increased 11.5 F for an unruptured segment, after modeling a simplified version of the 205-FA CW, compared to a case with only the corrected ECCS bypass calculation. The variation in flow resistance, due to the different flow area, resulted in a 3 F PCT deficit, in the unruptured segment, when comparing the analysis performed using 177-FA column weldment model in the RL plant to the simplified 205-FA column weldment model in the LL plant.

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It is recognized that the estimated +80 F PCT penalty assigned to ruptured segments for all plants includes more than a factor of three margin over the +26.2 F PCT penalty that was calculated using the simplified 205-FA column weldment model in a 177-FA LL plant with the corrected ECCS bypass calculation. The +40 F PCT penalty for unruptured segments is also more than four times greater than the calculated PCT of 8.9 F using the 177-FA column weldment model in the 177-FA RL plant with the corrected ECCS bypass calculation. This demonstrates that the +80 F assigned PCT is conservative and has sufficient margin; therefore, the 3 F PCT deficit between the 177-FA column weldment model and the 205-FA column weldment model does not need to be included as a separate addition. Therefore, the assigned PCT change is independent of the CW design used in the analyses since the PCT difference of 3 F is within the margin between the calculated and assigned PCTs, as shown in Table 4-1.

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Table 4-1: Calculated and Assigned ECCS Model PCT Changes

Plant Type	Core Peaking Elevation	ECCS Bypass Model PCT Changes (F)		Simplified Column Weldment PCT Changes (F)		Limiting PCT Change (F)
		Ruptured Segment	Unruptured Segment	Ruptured Segment	Unruptured Segment	PCT Change
177-FA LL	2.506-ft	<u>-90.3</u>	-45.4	<u>+26.2</u>	+11.5	-64.1 (ruptured segment)
177-FA RL	9.536-ft	-96.7	<u>-51.7</u>	***	<u>+8.9</u>	-42.8 (unruptured Segment)
Assigned	Independent	-80	-40	+80	+40	0

The underlined PCT changes indicate which PCT was limiting. For the LL plant analysis, the ruptured segment was limiting, while the unruptured segment was limiting for the RL plant analysis.

***This non-limiting, ruptured segment PCT is not provided because the PCT decreased as a result of a favorable rupture timing shift.

Response to RAI 2b:

The 177-FA column weldment model was implemented using two axial control volumes to approximate the full height of the control rods contained within a single column weldment above the hot channel. The lowest control volume begins at the top of the unheated fuel region (located at the bottom of the mounting block on the lower side of the upper core support plate). It is the same height as the lowermost upper plenum control volume in the LOCA EM model from BAW-10192. The second column weldment control volume in these analyses extends to the bottom of the reactor vessel upper head.

Response to RAI 2c:

A simplified drawing of the column weldments is shown in Figure 4-1. The column weldments for the 177-FA plants and 205-FA plants are similar. Table 4-2 provides some relative comparative ratios of select parameters rather than proprietary dimensions.

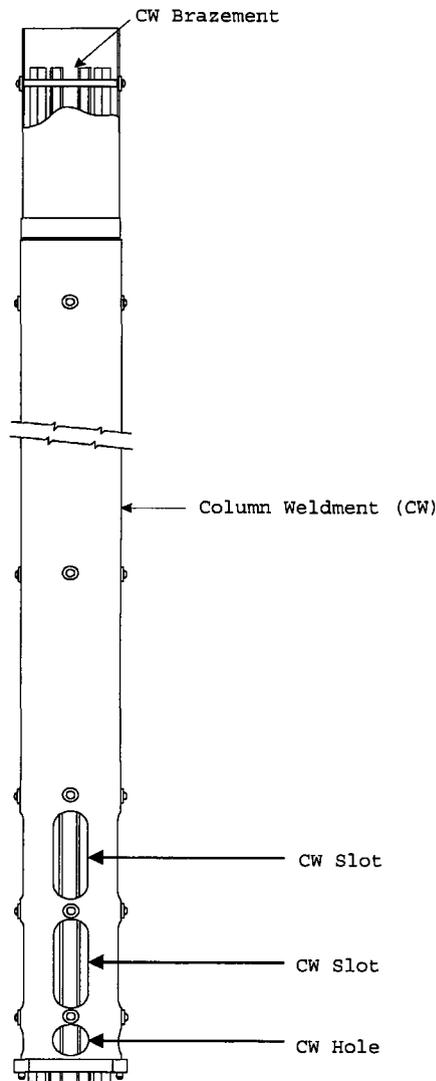**Figure 4-1: Simplified Column Weldment Drawing**

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Table 4-2. 177-FA and 205-FA Column Weldment Comparison

Parameter	177-FA Column Weldment	205-FA Column Weldment
Total CW Hole Flow Area	$A_{CW\,Hole}$	$A_{CW\,Hole}$
Total CW Slot Flow Area	$A_{CW\,Slot}$	$A_{CW\,Slot}$
Outer Diameter of the CW	$D_{o\,CW}$	104% $D_{o\,CW}$
Vertical Flow Area inside the CW not at a Spacer	A_{CW}	107% A_{CW}

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5.0   SUMMARY

Multiple sets of EM-based analyses were performed to compute the LBLOCA PCT changes for the ECCS bypass error and the column weldment modeling change for two different plant types, along with consideration for axial peaking at the top and bottom of the core. The analyses were performed with simplified column weldment models to generate bounding estimated PCT changes. These analyses and engineering evaluations assigned, and subsequently confirmed, generic PCT changes for all cases and plant types based on these two sets of studies. This information is provided in response to the Requests for Additional Information by the NRC transmitted in October and November 2012 References ([7], [8], [9], [10], and [11]) and March 2013 (References [13] and [14]). Analyses were performed and a cumulative generic PCT change of 0 F was assigned based on analyses and generic PCT assessments. While the net estimated change of 0 F is conservative versus the net reduction in PCT that was predicted by the documented studies in this RAI response, a conservative generic estimate is used to cover the LBLOCA results for all plants and axial peaking variations that were not analyzed.

Since there is no estimated change in PCT or time at elevated cladding temperatures, there is also no net change in the local oxidation or whole core hydrogen generation rates. The engineering judgment conservatively assigned a -80 F (ECCS bypass) / +80 F (CW) ruptured node temperature change and a -40 F (ECCS bypass) / +40 F (CW) unruptured node temperature change based on analyses. Given that analyses were performed, and these cases show that the actual PCTs are applicable to but lower than the previously analyzed values, no new LBLOCA cases are deemed necessary to show compliance to 10 CFR 50.46 at this time. However, the correction of the ECCS bypass error and modeling of the column weldments will be incorporated into the LBLOCA model for application in future analyses.

The SBLOCA model results were also reviewed considering these two changes. Since the ECCS bypass is not used for SBLOCA, it does not impact the SBLOCA results. An evaluation of CW model impact was performed for SBLOCA and it was concluded that these analyses are unaffected by the CW modeling as the net flow remains upward for these slower evolving transients. Nonetheless, for consistency with the LBLOCA model, the CW model will be included in future SBLOCA analyses.

6.0   REFERENCES

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2. BAW-10164P-A, Revision 6, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis", AREVA NP, Lynchburg, Virginia, June 2007.

3. BAW-10171P-A, Revision 3, "REFLOD3B – Model for Multinode Core Reflooding Analysis", B&W Nuclear Technologies, Lynchburg, Virginia, December 1995.

4. BAW-10166P-A, Revision 5, "BEACH – A Computer Program for Reflood Heat Transfer during LOCA", Framatome ANP, Lynchburg, Virginia, November 2003.

5. BAW-10095P-A, Revision 1, "CONTEMPT – Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident", Babcock & Wilcox, Lynchburg, Virginia, January 1975.

6. BAW-10227P-A, Revision 0, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel", February 2000.

7. Letter from Kaly Kalyanam (NRC) to David Bice (ANO-1), Request for Additional Information Regarding Arkansas Nuclear One, Unit 1 30-day Report for Emergency Core Cooling System Model Changes Pursuant to the Requirements of 10 CFR 50.46 Docket No. 50-313 (TAC NO. ME9719) (ML12300A365), October 26, 2012.

8. Letter from Farideh Saba (NRC) to Daniel Westcott (CR-3), Request for Additional Information Regarding Crystal River Unit 3 Nuclear Generating Plant 30-day Report for Emergency Core Cooling System Model Changes Pursuant to the Requirements of 10 CFR 50.46 Docket No. 50-302 (TAC NO. ME8408) (ML12304A068), October 26, 2012.

9. Letter from Michael Mahoney (NRC) to Ray Lieb (DB-1), Davis-Besse Nuclear
    Power Station, Unit No. 1 - Request for Additional Information Regarding Davis-
    Besse Notification of Significant Change to the Large-Break Loss-of-Coolant
    Accident Emergency Core Cooling Model in Accordance with Title 10 of the Code of
    Federal Regulations, Sections 50.46(a)(3) (TAC NO. ME8411) (ML12310A012),
    November 16, 2012.

10. Letter from John Boska (NRC) to Kent R. Alter (ONS), Oconee Units 1, 2, and 3,
    NRC Request for Additional Information on Errors Reported per 10 CFR 50.46 (TAC
    NO. ME9119) (ML12300A201), October 26, 2012.

11. Letter from Peter Bamford (NRC) to Michael J. Pacilio (TMI-1), Three Mile Island
    Nuclear Station, Unit 1 – Request for Additional Information Regarding 30-day
    Report for Emergency Core Cooling System Model Changes Pursuant to the
    Requirements of 10 CFR 50.46 (TAC NO. ME8237) (ML12310A175), November 13,
    2012.

12. ANP-3180, Revision 000, "177 Fuel-Assembly Plant RAI Responses to a 30-Day
    50.46 Report of significant PCT Change", AREVA NP, Lynchburg, Virginia,
    December 5, 2012.

13. Letter from Peter Bamford (NRC) to Michael J. Pacilio (TMI-1), Three Mile Island
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