AREVA

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U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Annual Reporting of Changes and Errors in Emergency Core Cooling Systems (ECCS) Evaluation Models

Ref. 1: Letter, Pedro Salas (AREVA NP Inc.) to Document Control Desk (NRC), "2010 – Annual Reporting of Changes and Errors in Emergency Core Cooling Systems (ECCS) Evaluation Models," NRC:11:123, December 20, 2011.

Attached is a summary report of changes and error corrections implemented in the AREVA NP Inc. (AREVA NP) Emergency Core Cooling Systems (ECCSs) evaluation models for the period of November 1, 2011 to December 31, 2012. Note that the previous reporting letter in Reference 1 incorrectly identified the reporting period for that letter. The Reference 1 letter indicated that the reporting period was from January 1, 2010 to December 31, 2010. It actually contained reporting for the period from January 1, 2010 to October 31, 2011. Therefore, this letter brings reporting up to date through December 31, 2012.

AREVA NP considers the Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) ECCS evaluation models to include both the codes and the methodology for using the codes. Changes to inputs that result from fuel or plant changes, and that are treated according to the methodology, are not considered model changes and, therefore, are not reported in the attachment. Changes in peak cladding temperatures (PCTs) due to loss of coolant accident (LOCA) evaluation model changes and errors are reported on a plant specific basis by AREVA to the affected licensees. The licensees have the obligation under 10 CFR 50.46 to report the nature of changes and errors affecting PCT. The report in this letter is provided for information only.

Sincerely,

Pedro, Salas
Director, Regulatory Affairs
AREVA NP Inc.

Attachments:
1. Attachment A- Listing of AREVA NP LOCA Evaluation Models
2. Attachment B- Annual Reporting of AREVA NP LOCA Evaluation Model Changes and Error Corrections (November 1, 2011- December 31, 2012)

cc: J. G. Rowley
Project 728

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NRR

Attachment A

Listing of AREVA NP LOCA Evaluation Models

EXEM BWR-2000 Large and Small Break LOCA Evaluation Model

This model is applicable to jet-pump boiling water reactors for both large and small break LOCA analyses. The NRC approved topical report for this evaluation model is EMF-2361PA, Revision 0.

CRAFT2 PWR Large Break LOCA Evaluation Model

This model is applicable to all B&W designed pressurized water reactors for large break LOCA analyses of zircaloy clad fuel. The NRC approved topical report for this evaluation model is BAW-10104PA, Revision 5.

CRAFT2 PWR Small Break LOCA Evaluation Model

This model is applicable to all B&W designed pressurized water reactors for small break LOCA analyses of zircaloy clad fuel. The NRC approved topical report for this evaluation model is BAW-10154PA, Revision 0.

RELAP5/MOD2-B&W Once Through Steam Generator Large and Small Break LOCA Evaluation Model

This model is applicable to all B&W designed pressurized water reactors for large and small break LOCA analyses of zircaloy or M5 clad fuel. The NRC approved topical report for this evaluation model is BAW-10192PA, Revision 0. The NRC has approved this evaluation model for M5 clad fuel in BAW-10227PA, Revision 0.

RELAP5/MOD2-B&W Re-Circulating Steam Generator Large and Small Break LOCA Evaluation Model

This model is applicable to Westinghouse designed 3 and 4 loop pressurized water reactors and Combustion Engineering designed pressurized water reactors for large and small break LOCA analyses. The NRC approved topical report for this evaluation model is BAW-10168PA, Revision 3.

SEM/PWR-98 PWR Large Break LOCA Evaluation Model

This model is applicable to Westinghouse designed 3 and 4 loop pressurized water reactors and Combustion Engineering designed pressurized water reactors for large break LOCA analyses. The NRC approved topical report for this evaluation model is EMF-2087PA, Revision 0.

ANF-RELAP PWR Small Break LOCA Evaluation Model

This model is applicable to Westinghouse designed 2, 3, and 4 loop pressurized water reactors and Combustion Engineering designed pressurized water reactors for small break LOCA analyses. The NRC approved topical report for this evaluation model is XN-NF-82-49PA, Revision 1, Supplement 1.

S-RELAP5 PWR Small Break LOCA Evaluation Model

This model is applicable to Westinghouse designed 2, 3, and 4 loop pressurized water reactors and Combustion Engineering designed pressurized water reactors for small break LOCA analyses. The NRC approved topical report for this evaluation model is EMF-2328PA, Revision 0.

Realistic PWR Large Break LOCA Model

This model is applicable to Westinghouse designed 3 and 4 loop pressurized water reactors and Combustion Engineering 2x4 designed pressurized water reactors for large break LOCA analyses. The NRC approved topical report for this evaluation model is EMF-2103PA, Revision 0.

Attachment B

Annual Reporting of AREVA NP LOCA Evaluation Model Changes and Error Corrections

(November 1, 2011- December 31, 2012)

EXEM BWR-2000 Large and Small Break LOCA Evaluation Model

This model is applicable to jet-pump boiling water reactors for both large and small break LOCA analyses. The NRC approved topical report for this evaluation model is EMF-2361PA, Revision 0.

The Evaluation Model consists of three computer codes:(1) RELAX to compute the system and hot channel response during blowdown and to calculate the time for refill of the lower plenum and reflood of the core, (2) HUXY to calculate the heatup of the peak power plane and (3) RODEX2 to determine the rod conditions at the start of the transient.

The following evaluation model changes or error corrections were made during the reporting period.

1. Fuel Thermal Conductivity Degradation as a Function of Exposure

   Exposure dependent degradation of fuel thermal conductivity over the approved burnup range was not supported by experimental data when older generation codes, like RODEX2, were approved. Hence, it was not explicitly modeled. Evaluations were performed to determine the impact of this phenomenon. It was determined that the use of the RODEX2 code (which provides inputs to RELAX and HUXY) results in conservatively high fuel temperatures at low burnup (<15 GWd/MTU) and an under prediction of fuel temperatures at higher exposures. EXEM BWR-2000 Licensing calculations predict the limiting PCT to occur at beginning of life (BOL). The effect of degraded thermal conductivity at higher burnup was found to be insufficient to result in more limiting PCTs at higher exposure.

   The impact of Fuel thermal conductivity degradation as a function of exposure has been assessed for all plants supported by LOCA analyses using the BWR EXEM-2000 methodology. The impact on PCT was 0°F for all plants.

2. Modified Analysis Approach

   The NRC staff has recently questioned how certain cooling phenomena are modeled in the EXEM BWR-2000 ECCS evaluation model during some LOCA events. In order to address these concerns AREVA NP developed the modified analysis approach with impact and implementation strategy presented in NRC:11:096, Letter P. Salas (AREVA NP Inc.) to Document Control Desk (NRC), "Proprietary Viewgraphs and Meeting Summary for Closed Meeting and Application of the EXEM BWR-2000 ECCS Evaluation Methodology," September 22, 2011. The NRC communicated that the proposed modified analysis approach was more conservative and generally acceptable in the Letter T. McGinty (NRC) to P. Salas (AREVA NP Inc.), "Response to AREVA NP Inc. (AREVA NP) Proposed Analysis Approach for its EXEM Boiling Water Reactor (BWR)-2000 Emergency Core Cooling System (ECCS) Evaluation Model," July 5, 2012.

CRAFT2 PWR Large Break LOCA Evaluation Model

This model is applicable to all B&W designed pressurized water reactors for large break LOCA analyses of zircaloy clad fuel. The NRC approved topical report for this evaluation model is BAW-10104PA, Revision 5.

The Evaluation Model consists of five computer codes:
(1) CRAFT2 - to compute the system and core response during blowdown.
(2) REFLOD3 - to calculate the time for refill of the lower plenum and core reflood rate.
(3) CONTEMPT - to compute the containment pressure response.
(4) FLECSET - to calculate the hot pin heat transfer coefficients.
(5) THETA1-B - to determine the hot pin thermal response for the entire transient. An NRC-approved fuel code (currently TAC03) is used to supply the fuel rod steady-state conditions at the beginning of the transient.

There were no evaluation model changes or error corrections made during the reporting period.

CRAFT2 PWR Small Break LOCA Evaluation Model

This model is applicable to all B&W designed pressurized water reactors for small break LOCA analyses of zircaloy clad fuel. The NRC approved topical report for this evaluation model is BAW-10154PA, Revision 0.

The Evaluation Model consists of three computer codes:
(1) CRAFT2 to compute the system and core response during blowdown,
(2) FOAM2 to calculate the core mixture level and average channel steaming rate, and
(3) THETA 1-B to determine the hot pin thermal response for the entire transient. An NRC-approved fuel code (currently TAC03) is used to supply the fuel rod steady-state conditions at the beginning of the transient.

There were no evaluation model changes or error corrections made during the reporting period.

RELAP5/MOD2-B&W Once Through Steam Generator Large and Small Break LOCA Evaluation Model

This model is applicable to all B&W designed pressurized water reactors for large and small break LOCA analyses of zircaloy or M5 clad fuel. The NRC approved topical report for this evaluation model is BAW-10192PA, Revision 0. The NRC has approved this evaluation model for M5 clad fuel in BAW-10227PA, Revision 0.

The large break LOCA Evaluation Model consists of four computer codes:

(1) BAW-10164PA, RELAP5/MOD2-B&W to compute the system, core, and hot rod response during blowdown.
(2) BAW-10171PA, REFLOD3B to calculate the time for refill of the lower plenum and core reflood rate.
(3) BAW-10095A, CONTEMPT to compute the containment pressure response, and
(4) BAW-10166PA, BEACH (RELAP5/MOD2-B&W reflood heat transfer package) to determine the hot pin thermal response during refill and reflood phases.

The small break LOCA Evaluation Model consists of two codes:

(1) BAW-10164PA, RELAP5/MOD2-B&W to compute the system, core, and hot rod response during the transient
(2) BAW-10095A, CONTEMPT to compute the containment pressure response, if needed. An NRC-approved fuel code (currently BAW-10162PA, TAC03 or BAW-10184PA, GDTACO) is used to supply the fuel rod

steady-state conditions at the beginning of the small or large break LOCA. These codes are approved for use with M5 cladding via the safety evaluation report on BAW-10227PA.

The following evaluation model changes or error corrections were made during the reporting period.

1. End of Bypass Time Calculation Error

   While performing a LBLOCA sensitivity study for the 205 FA Bellefonte plant a mathematical error was discovered in the RELAP5/MOD2-B&W blowdown model control variables that calculate the time for total end of bypass. BAW-10192PA describes how the end of bypass calculations determine when an 80 percent condensation efficiency on the core flood tank (CFT) injected liquid could condense all the steam reaching the upper downcomer region. The control variables incorrectly calculated the steam energy flowing into the upper downcomer region. When the control variables were corrected, the end of bypass time was predicted approximately 2 seconds earlier, resulting in a shorter lower plenum refill period with a quicker onset of lower core quench and lower PCT for the 205 FA plant limiting LBLOCA analyses. In evaluating the extent of condition for this error, a similar error was found to be in all 177 FA plant LBLOCA analyses as well. AREVA NP corrected the control variable error and performed a new limiting LBLOCA analysis for the Oconee 177 Fuel Assembly (177 FA) lowered-loop plant.

   The corrections also shortened the lower plenum refill period by roughly 2 seconds for the LL plant and this change decreased the ruptured segment cladding temperature by 80 F, a value rounded in the conservative (lower) direction, relative to the previously calculated value with the ECCS bypass error. In comparison, the limiting unruptured segment cladding temperature change was a decrease of 40 F, a value also rounded in the conservative (lower) direction. Since the same error was in all the 177 FA models and the CFT flows and plant geometry are similar in all models, the refill period will shorten by approximately the same interval with the expectation that cladding temperatures decrease similarly for all 177 FA plants. Therefore, a generic LBLOCA PCT change of -80 F (reduction) is assigned to the ruptured cladding segments and a -40 F is assigned to limiting unruptured cladding segments. ECCS bypass is not used for SBLOCA so these analyses are not affected by this error.

2. Control Rod Guide Tube Column Weldment (CW) Modeling Enhancement

   During the assessment of the ECCS bypass error, another LBLOCA sensitivity study was being performed for the 205 FA Bellefonte plant with a revised upper plenum and upper head modeling that considered the changes in core cooling when upper plenum column weldments are explicitly modeled. This revised modeling reflects a more detailed noding arrangement in the reactor vessel upper plenum than was used and approved for application in the EM. A simplified column weldment model was developed for the 177 FA model based on approximations from the 205 FA model. When this simplified model was used, the scoping case with the column weldment modeled over the top of the hot channel resulted in reduced cooling during portions of the blowdown phase. As a result, the end of blowdown fuel temperatures increased resulted in a 40 F increase in the unruptured segment PCTs, while the ruptured segment is increased by 80 F. Both estimated values were rounded in the conservative (higher) direction.
   This modeling change was considered for SBLOCA. It was concluded that it will not affect the limiting results because the SBLOCA is a slower evolving transient with up flows in the core hot bundles such that there is no net change from the presence of a column weldment in the upper plenum.

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.

RELAP5/MOD2-B&W Re-Circulating Steam Generator Large and Small Break LOCA Evaluation Model

This model is applicable to Westinghouse-designed 3 and 4 loop pressurized water reactors and Combustion Engineering-designed pressurized water reactors for large and small break LOCA analyses. The NRC approved topical report for this evaluation model is BAW-10168PA, Revision 3.

The large break LOCA Evaluation Model consists of three computer codes:

(1) RELAP5/MOD2-B&W - to compute the system, core and hot rod response during blowdown,
(2) REFLOD38 to calculate the time for refill of the lower plenum and core reflood rate, and
(3) BEACH (RELAP5/MOD2-B&W reflood heat transfer package) to determine the hot pin thermal response during refill and reflood phases. The small break LOCA Evaluation Model consists of one code: RELAP5/MOD2 B&W - to compute the system, core and hot rod response during the transient. A NRC-approved fuel code (currently TAC03 or GDTACO) is used to supply the fuel rod steady state conditions at the beginning of the small or large LOCA transient.

There were no evaluation model changes or error corrections made during the reporting period.

SEM/PWR-98 PWR Large Break LOCA Evaluation Model

This model is applicable to Westinghouse designed 3 and 4 loop pressurized water reactors and Combustion Engineering designed pressurized water reactors for large break LOCA analyses. The NRC approved topical report for this evaluation model is EMF-2087PA, Revision 0.

The SEM/PWR-98 large break LOCA Evaluation Model consists of four primary computer codes:

(1) RELAP4 - to compute the system and hot channel response,
(2) RFPAC - to compute the containment pressures, reflood rates, and axial shape factors,
(3) TOODEE2 - to calculate the hot rod heatup, and (4) RODEX2 to determine the rod conditions at the start of the transient.

There were no evaluation model changes or error corrections made during the reporting period.

ANF-RELAP PWR Small Break LOCA Evaluation Model

This model is applicable to Westinghouse designed 2, 3, and 4 loop pressurized water reactors and Combustion Engineering designed pressurized water reactors for small break LOCA analyses. The NRC approved topical report for this evaluation model is XN-NF-82-49PA, Revision 1, Supplement 1.

The ANF-RELAP small break LOCA Evaluation Model consists of three computer codes: (1) ANF-RELAP to compute the system response, (2) TOODEE2 to calculate the hot rod heatup, and (3) RODEX2 to determine the rod conditions at the start of the transient.

There were no evaluation model changes or error corrections made during the reporting period.

S-RELAP5 PWR Small Break LOCA Evaluation Model

This model is applicable to Westinghouse designed 2, 3, and 4 loop pressurized water reactors and Combustion Engineering designed pressurized water reactors for small break LOCA analyses. The NRC approved topical report for this evaluation model is EMF-2328PA, Revision 0.

The following evaluation model changes or error corrections were made during the reporting period.

1. Sleicher-Rouse Correlation

The Sleicher-Rouse correlation is one of the correlations used to define the heat transfer between the fuel and coolant. This correlation is applicable to both Large and Small Break analyses performed with the S-RELAP5 computer code.

The error relates to the form of the equation calculating the exponent of the temperature ratio correction term. The Sleicher-Rouse correlation is as follows:

The S-RELAP5 form is:

$$Nu_b = 5 + 0.012*Re_b^{0.83} * (Pr_b + 0.29)*(T_w/T_b)^n \qquad \text{(Form A)}$$

with

$$n = - \log10(Tw/Tb)1/4 + 0.3$$

The alternative form used in other industry codes is:

$$Nub = 5 + 0.012*Reb0.83 * (Prb + 0.29)*(Tw/Tb)n \qquad \text{(Form B)}$$

with

$$n = -[\log10(Tw/Tb)]1/4 + 0.3$$

The alternative form is more consistent with other heat transfer correlations and expected physical trends.

The alternative form of the Sleicher-Rouse correlation has been incorporated into the S-RELAP5 code. Scoping calculations were used to support the engineering judgment for the determination of PCT impact. The following tables show the PCT impact for all plants which are using or may use their current SBLOCA analysis.

Plant Type	10 CFR 50.46 Reportable Impact, °F
W 4 Loop	-89
W 3 Loop	-19
CE 2x4 Loop	83

This model is applicable to Westinghouse designed 3 and 4 loop pressurized water reactors and Combustion Engineering 2x4 designed pressurized water reactors for large break LOCA analyses. The NRC approved topical report for this evaluation model is EMF-2103PA, Revision 0.

The following evaluation model changes or error corrections were made during the reporting period.

1. Upper Plenum Modeling

   A recent methodology change was implemented to address the potential for non-physical flow behavior in the upper plenum (UP) during the core reflood phase. This non-physical flow behavior may manifest itself by producing both liquid and vapor flow spikes from upper plenum to the hot channel (HC) as well as to the surrounding ring that represents the six fuel assemblies. Even though CCFL modeling was applied at the hot channel exit junction, the code does not prevent liquid fall back when the flow is co-current downward. To correct the non-physical flow behavior, high reverse form loss coefficients are applied to the hot channel-to-UP and central core-to-UP junctions at the beginning of the core reflood phase.

   In addition to correct this non-physical behavior, changes to the upper plenum noding are implemented. The UP nodalization was not consistent with plant geometry which contributed to the non-physical flow behavior.

   Scoping calculations were used to support the engineering judgment for the determination of PCT impact. The following table shows the PCT impact for all plants which are using or may use their current RLBLOCA analysis.

Plant Type	10 CFR 50.46 Reportable Impact, °F
W 4 Loop	0
W 3 Loop	8
CE 2x4 Loop	0

2. Sleicher-Rouse Correlation

   The Sleicher-Rouse correlation is one of the correlations used to define the heat transfer between the fuel and coolant. This correlation is applicable to both Large and Small Break analyses performed with the S-RELAP5 computer code.

   The error relates to the form of the equation calculating the exponent of the temperature ratio correction term. The Sleicher-Rouse correlation is as follows:

   The S-RELAP5 form is:

$$Nu_b = 5 + 0.012*Re_b^{0.83} * (Pr_b + 0.29)*(T_w/T_b)^n \qquad \text{(Form A)}$$

   with

$$n = -\log_{10}(T_w/T_b)^{1/4} + 0.3$$

The alternative form used in other industry codes is:

$$Nu_b = 5 + 0.012*Re_b^{0.83} * (Pr_b + 0.29)*(T_w/T_b)^n \qquad \text{(Form B)}$$

with

$$n = -[\log_{10}(T_w/T_b)]^{1/4} + 0.3$$

The alternative form is more consistent with other heat transfer correlations and expected physical trends.

The alternative form of the Sleicher-Rouse correlation has been incorporated into the S-RELAP5 code. The estimated impacts of this change are reported on a plant type basis as follows:

Plant Type	10 CFR 50.46 Reportable Impact, °F
W 4 Loop	-35
W 3 Loop	14
CE 2x4 Loop	8

3. Cathcart-Pawel

In realistic large break loss of coolant accident (RLBLOCA) analyses, energy released through the oxidation of cladding is calculated from the Cathcart-Pawel correlation for oxide growth. The correlation has the form:

$$\delta^2/2 = A * exp(-Q/R*1/T)$$

Where A and Q are experimentally determined constants and R and T are the gas constant and temperature, respectively. The uncertainty parameter for the A value is given in terms of the natural logarithm: ln(A). The value of ln(A) follows a normal distribution and the value of A follows a log-normal distribution. RLBLOCA applications implement the Cathcart-Pawel uncertainty using a log-normal function for the uncertainty multiplier, B, applied to a constant, A. The equation to determine the uncertainty multiplier, B, was determined to be incorrect. However, the incorrect equation still has a log-normal distribution like the corrected equation for the uncertainty multiplier, B. In addition, the range of sampled values for B falls within the range expected for the corrected equation for the uncertainty multiplier, B. The estimated impact of this change on the RLBLOCA PCT is +0°F.