AREVA

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2010 – Annual Reporting of Changes and Errors in Emergency Core Cooling Systems (ECCS) Evaluation Models

Attached is a summary report of the changes and error corrections implemented in the AREVA NP Inc. (AREVA) Emergency Core Cooling Systems (ECCS) evaluation models for the period of January 1, 2010 to December 31, 2010.

AREVA 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 affected licensees. The licensees have the obligation under 10 CFR Part 50.46 to report the nature of changes and errors affecting PCT. This report is provided for information only.

Sincerely,

Pedro Salas, Manager
Corporate Regulatory Affairs
AREVA NP Inc.

Attachment

cc: H. D. Cruz
Project 728

A002
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

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.

<u>ANF-RELAP PWR Small Break LOCA Evaluation Model</u>

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.

<u>S-RELAP5 PWR Small Break LOCA Evaluation Model</u>

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.

<u>Realistic PWR Large Break LOCA Evaluation Model</u>

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 (2009)

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. Array Bounds Violations in the RELAX Computer Code

   During conversion of the RELAX code to execute on the Linux operating system, array bounds violations were detected in the source code. Four potential issues were identified. Three of the issues were assessed and determined to have no effect on calculations; however, one issue has the potential to affect calculations using the SPCB critical heat flux (CHF) correlation.

   The SPCB correlation is used to predict the occurrence of CHF. SPCB is only used when fluid conditions are within the range of applicability of the correlation. The range of fluid conditions that the SPCB correlation is used for occur early in a LOCA analysis and after that time, other CHF correlations are used by RELAX. The SPCB correlation has no affect on LOCA analyses unless CHF is exceeded before the transfer to other correlations.

   The impact of the code error was 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. Improved View Factor Method for the HUXY Computer Code

   The HUXY code performs fuel rod heatup calculations and provides PCT and local clad oxidation at the axial plane of interest. The code models the radiation heat transfer between the fuel rod of interest and other fuel rods, the internal water channel, and the fuel channel. AREVA has developed a new approach for calculating radiation view factors and has implemented it within the HUXY computer program.

   The original approach for calculating view factors uses the method of cross-strings and results in the derivation of analytical expressions for the radiation view factors that are coded in HUXY for each fuel rod. The view factors are then computed throughout the HUXY heatup analyses based on these analytical expressions and the time dependent fuel rod dimensions. With the evolution of fuel design features such as larger internal water structures and part-length rods, the assembly lattice has become more heterogeneous and the derivation of the analytical expressions for the view factors has become more complex and less precise due to assumptions with the approach. To make the computation of the view factors for current and future design concepts more precise, a numerical computation of the view factors has been introduced. The numerical computation, achieved through a ray tracing algorithm, provides a straight forward approach to compute the view factor from each fuel rod to all other fuel rods, internal water structures, and the external fuel channel.

The impact of the improved radiation view factor calculation approach has been assessed for plants supported by LOCA analyses using the revised HUXY computer code as part of the BWR EXEM-2000 methodology. The impact on PCT is between -1°F and +1°F for these plants.

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, and (5) THETA1-B to determine the hot pin thermal response for the entire transient. An NRC-approved fuel code (currently TACO3) 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) THETA1-B to determine the hot pin thermal response for the entire transient. An NRC-approved fuel code (currently TACO3) 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 and M5 clad fuel. The NRC approved topical report for this evaluation model is BAW-10192PA, Revision 0.

The large break LOCA Evaluation Model consists of four computer codes: (1) BAW-10164P-A, RELAP5/MOD2-B&W to compute the system, core, and hot rod response during blowdown, (2) BAW-10171P-A, REFLOD3B to calculate the time for refill of the lower plenum and core reflood rate, (3) BAW-10095-A, CONTEMPT to compute the containment pressure response, and (4)

BAW-10166P-A, 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-10164P-A, RELAP5/MOD2-B&W to compute the system, core, and hot rod response during the transient and (2) BAW-10095-A, CONTEMPT to compute the containment pressure response, if needed. An NRC-approved fuel code (currently BAW-10162P-A, TACO3 or BAW-10184P-A, 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 (SER) on BAW-10227P-A.

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

1. AREVA identified that the SBLOCA Axial Power Shape defined in BAW-10192P-A Revision 0 may not be bounding for middle- to end-of-cycle (MOC to EOC) conditions. Since the SBLOCA analyses are designed to be independent of time in cycle, a bounding axial power shape should be used to be consistent with the requirements in Section I.A of 10 CFR 50 Appendix K. As a result, a bounding EOC 11-ft axial power shape was developed by AREVA that conservatively meets all requirements, and it results in an increase in the SBLOCA PCT. An assessment was performed, and an estimated SBLOCA PCT increase of 225 F was determined to conservatively account for the 11ft axial shape. Unlike SBLOCA analyses that use a composite set of inputs, LBLOCA analyses are performed at various times in cycle as well as at different peak power elevations. Further, LBLOCA sensitivity studies have been performed with peaks at approximately 1 and 11-ft to support the method for setting the limits at the bottom and top (0 and 12-ft) of the core while keeping the LBLOCA PCT set by an analysis. Therefore, the LBLOCA analyses and resulting PCTs are unaffected by this error.

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 2x4 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) REFLOD3B 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 TACO3 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.

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

1. <u>RODEX2 Thermal Conductivity Degradation</u>

   Burnup dependent thermal conductivity is not accounted for in the RODEX2 code. The code may under-predict the fuel pellet temperatures at burnups beyond approximately 20 GWd/MTU and therefore may under-predict the stored energy initial condition for LBLOCA analyses. The impact on LBLOCA PCT was estimated to be 20°F.

2. <u>Array Bound Violations</u>

   An error was identified dealing with array index issues in the RELAX code used for BWR LOCA calculations using the EXEM BWR-2000 LOCA methodology. It was determined that the index issues also exist for the RELAP4 code which is used as part of the SEM/PWR-98 LBLOCA methodology for PWRs. The impact on LBLOCA PCT was estimated to be 0°F.

<u>ANF-RELAP PWR Small Break LOCA Evaluation Model</u>

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.

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

1. <u>RODEX2 Thermal Conductivity Degradation</u>

   Burnup dependent thermal conductivity is not accounted for in the RODEX2 code. The code may under-predict the fuel pellet temperatures at burnups beyond approximately 20 GWd/MTU and therefore may under-predict the stored energy initial condition for SBLOCA analyses. PWR SBLOCA analyses are insensitive to initial stored energy because sufficient excess cooling capacity exists during the blowdown phase of the transient to effectively remove any excess initial stored energy prior to the extended heatup period when PCT occurs. The impact on SBLOCA PCT was estimated to be 0°F.

2. <u>Point Kinetics Model</u>

   Previously, INL announced an error in the coding of the point kinetics model. More recently, INL announced that the previous error corrections were incorrect and that the recommended convergence criteria supplied with those corrections should be retained. The estimated change in SBLOCA PCT is +8°F.

3. Heat Conduction Solution

   INL announced that the heat conduction solution is incorrectly programmed in the RELAP5 series of codes. The error is associated with using the incorrect heat capacity when evaluating the right boundary mesh point. Instead of using the last (adjacent) mesh interval heat capacity, the code incorrectly uses the next to last mesh interval heat capacity. The affect of this error is maximized in cylindrical and spherical geometries with few mesh points, which can be minimized with an increased number of mesh points. The effect is further minimized by the AREVA SBLOCA methodology which implements close mesh spacing at the left and right boundaries. The estimated change in SBLOCA PCT is 0°F.

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. RODEX2 Thermal Conductivity Degradation

   Burnup dependent thermal conductivity is not accounted for in the RODEX2 code. The code may under-predict the fuel pellet temperatures at burnups beyond approximately 20 GWd/MTU and therefore may under-predict the stored energy initial condition for SBLOCA analyses. PWR SBLOCA analyses are insensitive to initial stored energy because sufficient excess cooling capacity exists during the blowdown phase of the transient to effectively remove any excess initial stored energy prior to the extended heatup period when PCT occurs. The impact on SBLOCA PCT was estimated to be 0°F.

2. Point Kinetics Model

   Previously, INL announced an error in the coding of the point kinetics model. More recently, INL announced that the previous error corrections were incorrect and that the recommended convergence criteria supplied with those corrections should be retained. The estimated change in SBLOCA PCT is +4°F.

3. Heat Conduction Solution

   INL announced that the heat conduction solution is incorrectly programmed in the RELAP5 series of codes. The error is associated with using the incorrect heat capacity when evaluating the right boundary mesh point. Instead of using the last (adjacent) mesh interval heat capacity, the code incorrectly uses the next to last mesh interval heat capacity. The affect of this error is maximized in cylindrical and spherical geometries with few mesh points, which can be minimized with an increased number of mesh points. The effect is further minimized by the AREVA SBLOCA methodology which implements close mesh spacing at the left and right boundaries. The estimated change in SBLOCA PCT is 0°F.

Realistic PWR Large Break LOCA Evaluation 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.

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

1. During development work on EMF-2103 Rev. 2, an issue was discovered that required evaluating the impact of not entraining the appropriate amount of liquid into the steam generator tubes during a LBLOCA event. The Realistic Large Break LOCA (RLBLOCA) methodology uses a bias on interphase friction at the steam generator tube sheet entrance to insure an acceptable amount of liquid is entrained into the steam generator tubes during a large break. The bias determination was performed by comparing calculated results from S-RELAP5 with measured data from the Upper Plenum Test Facility (UPTF) Tests 10 and 29. The UPTF test facility represents a full scale, four loop PWR complete with the necessary hardware that can be used to represent geometry specific phenomena that occurs during a large or small break LOCA. The S-RELAP5 parameter that controls entrainment is interphase friction. The range of interphase friction spans several orders of magnitude between the flow regimes occurring in the hot leg, hot leg riser, steam generator inlet plenum and steam generator tube sheet. Consequently, determining the uncertainty in interphase friction is not feasible so a conservative bias is used instead. The magnitude of the bias is determined by adjusting the S-RELAP5 RLBLOCA Multiplier "FIJ" until S-RELAP5 over-predicts the entrainment observed in UPTF Tests 10 and 29 by an arbitrary amount. Therefore, the FIJ multiplier of 1.75 is invalid and under-predicts the measured entrainment. The re-evaluation of the S-RELAP5 entrainment yielded a value of 5.0 for the FIJ multiplier is appropriate with a modeling change to the steam generator riser angle, greater than 30-degrees, and with the horizontal stratification flag set to off in the hot leg. The estimated changes were reported on a plant type basis as follows:

CE 2x4 Plant Type	0°F
W 3-loop Plant Type	- 4°F
W 4-loop Plant Type	+ 12°F