FRAMATOME ANP

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2001 – Annual Reporting of Changes and Errors in ECCS Evaluation Models

Attached is a summary report of the changes and error corrections implemented in the Framatome ANP ECCS evaluation models for the period of January 1, 2001 to December 31, 2001.

FRA-ANP considers the BWR and 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 changes to LOCA evaluation models and input changes are reported on a plant specific basis by FRA-ANP 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.

Very truly yours,

J F Holm for

James F. Mallay, Director
Regulatory Affairs

/lmk

Attachments

cc: J. S. Cushing (w/Attachment)
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D045

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Attachment A

Listing of Framatome ANP LOCA Evaluation Models

EXEM BWR Large and Small Break LOCA Evaluation Model

This model is applicable to all boiling water reactors for both large and small break LOCA analyses. The NRC approved topical report for this evaluation model is ANF-91-048PA Supplements 1 and 2.

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 Rev 0.

CRAFT2 PWR Large Break LOCA Evaluation

This model is applicable to all B&W designed pressurized water reactors for large break LOCA analyses. The NRC approved topical report for this evaluation model is BAW-10104PA Rev 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. The NRC approved topical report for this evaluation model is BAW-10154PA Rev 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. The NRC approved topical report for this evaluation model is BAW-10192PA Rev 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 Rev 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 Rev 0.

ANF-RELAP PWR 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 small break LOCA analyses. The NRC approved topical report for this evaluation model is XN-NF-82-49PA Rev 1 Supplement 1.

S-RELAP5 PWR 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 small break LOCA analyses. The NRC approved topical report for this evaluation model is EMF-2328PA Rev 0.

Attachment B
Annual Reporting of Framatome ANP LOCA Evaluation Model Changes and Error Corrections (2001)

EXEM BWR Large and Small Break LOCA Evaluation Model

This model is applicable to all boiling water reactors for both large and small break LOCA analyses. The NRC approved topical report for this evaluation model is ANF-91-048PA Supplements 1 and 2.

The Evaluation Model consists of four computer codes: (1) RELAX to compute the system and hot channel response during blowdown, (2) FLEX to calculate the time for refill of the lower plenum and reflood of the core, (3) HUXY to calculate the heatup of the peak power plane, and (4) RODEX2 to determine the rod conditions at the start of the transient.

The code or methodology changes to this Evaluation Model implemented during this reporting period are described below.

Errors Discovered During RODEX2 V&V

Ref.: 1. Letter, J. H. Nordahl (SPC) to Director, Office of Enforcement (NRC), "Siemens Power Corporation-Nuclear Division Responses to the Demand for Information, Notice of Nonconformance, and Unresolved Items (Inspection Report 9990081/97-01)," February 24, 1998.

Ref.: 2. XN-NF-81-58(P)(A) Revision 2 Supplements 1 & 2(P)(A) Revision 2, *RODEX2 Fuel Rod Thermal-Mechanical Responses Evaluation Model*, Siemens Power Corporation, March 1984.

Ref.: 3. ANF-81-58(P)(A) Revision 2 Supplements 3 & 4, *RODEX2 Fuel Rod Thermal-Mechanical Responses Evaluation Model*, Siemens Power Corporation, April 1990.

Ref.: 4. XN-NF-85-74(P)(A), *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model*, Siemens Power Corporation, October 1994.

Ref.: 5. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Models*, Siemens Power Corporation, February 1998.

Ref.: 6. Letter, Suzanne C. Black, Chief Quality Assurance, Vendor Inspection and Maintenance Branch, Division of Reactor Controls and Human Factors, Office of Nuclear Reactor Regulation to Chris M. Powers, Vice President Quality and Regulatory Affairs, Siemens Power Corporation - Nuclear Division, "NRC Inspection Report No. 99900081/98-01," September 4, 1998.

In Reference 1 Framatome ANP (formerly Siemens Power Corporation) committed to perform additional V&V and to update or create users manuals, theory manuals, and programmers manuals for Framatome ANP's primary codes. Specifically, the pertinent commitments in Reference 1 are numbers 24 and 25. Three of the primary codes identified were RODEX2, RODEX2-2A and RDX2LSE. The work on these three codes has been completed. RODEX2-2A has two modes: RODEX2 and RODEX2A. The RODEX2 mode is used for PWR mechanical analyses and the RODEX2A mode is used for BWR mechanical analyses. The code RDX2LSE is a separate code which implements the RODEX2 mode for use in BWR and PWR safety analyses. The code RODEX2 was approved by the NRC for use in BWR and PWR analysis (mechanical and safety) in References 2 and 3. The code RODEX2A was approved by the NRC for use in BWR mechanical analysis in References 4 and 5.

The NRC had expressed concern in the Reference 6 inspection about maintaining multiple versions of an NRC-approved code. However, the NRC concluded that the user guidance for both versions of the code clearly distinguished which options are to be used for specific analyses and were consistent with the SER for RODEX2. Subsequently, Framatome ANP decided to merge the RODEX2 code versions into a single code.

While performing the additional V&V, two errors were identified that are described below.

Documentation Error

Reference 2 states that 75 percent of the pellet dish volume is assumed to be available to accommodate gaseous swelling, but the version of RODEX2 used for mechanical analyses actually uses a smaller fraction. This error dates back to the mid-1980s and involves a mistake in a RODEX2 equation that resulted in 57 percent of the actual dish volume being available to accommodate swelling.

This error in RODEX2 was corrected and the assumption of the pellet dish volume available to accommodate gaseous swelling was reduced from 75 to 57 percent in order to maintain agreement with the value implicitly used for the benchmark data. The Reference 2 document should therefore state that 57 percent of the pellet dish volume is assumed to be available to accommodate gaseous swelling.

Code Error

The mistake discussed above was not corrected in the RDX2LSE version (which is used in safety analysis) at the time it was corrected in the version of RODEX2 used for mechanical analysis. In RDX2LSE, the error caused the model to become overly sensitive to the number of radial nodes (NRD) used to model the central dished part of the pellet.

The underestimate of the volume allocated for swelling becomes magnified as NRD is reduced. The reduced volume directly affects the calculated temperatures and gap conductances.

BWR safety analyses are performed with the maximum value of NRD and are not affected by this sensitivity, whereas PWR analyses use a smaller value of NRD, and the results could be in error.

Correcting the error and merging the two codes has been shown to have negligible effect on the safety and mechanical analyses

The following actions were taken on the RODEX2 codes:

- The portion of the pellet dish that is assumed to be available to accommodate swelling (as described in Reference 2) was made consistent; it should be 57 percent instead of 75 percent.
- The error in RDX2LSE has been corrected.
- The various versions of RODEX2 were merged into a single code.
- The results from the new code were verified against the original benchmark results submitted to the NRC in References 2, 3, 4, and 5.
- The new code was assessed for any impact on safety and mechanical analyses, and the results show negligible differences.

The impact of this change on the PCTs for those plants for which Framatome ANP performs BWR LOCA analyses using this evaluation model was estimated to be +1 degrees F.

Gadolinia Conductivity Model in HUXY

A fuel conductivity error was discovered in which the gadolinia-bearing fuel conductivity equation described in XN-NF-85-92 (P)(A), and approved by the NRC, was not incorporated in the code HUXY. The previous gadolinia-bearing fuel conductivity equation, described in XN-NF-79-56(P)(A), was not replaced upon the approval of the gadolinia-bearing fuel conductivity equation described in XN-NF-85-92(P)(A). The impact of this error on the RODEX2 code was reported in a previous year.

The impact of this change on the PCTs for those plants for which Framatome ANP performs BWR LOCA analyses using this evaluation model was estimated to range from 0 to +3 degrees F.

BULGEX Model Implementation

The BULGEX model in HUXY should be used for strain calculations when the cladding temperature is within 200°F of the rupture temperature. HUXY prints a message that allows the user to determine if the BULGEX model was turned on soon enough. It was determined that the message might not be correct for all situations. The HUXY code was modified to ensure that the message identified the correct time for when the limiting rod is within 200°F of the rupture temperature.

The impact of this change on the PCTs for those plants for which Framatome ANP performs BWR LOCA analyses using this evaluation model was estimated to range from 0 to -37 degrees F.

Rupture Temperature Calculation for Temperatures above 950 Degrees C

A code review determined that a coefficient used in the HUXY rupture temperature calculation for temperatures greater than 950°C was rounded in an incorrect and non-conservative manner. The HUXY code was revised to correct the error.

The impact of this change on the PCTs for those plants for which Framatome ANP performs BWR LOCA analyses using this evaluation model was estimated to be +1 degrees F.

Incorrect Zircaloy Heat of Reaction as a Function of Temperature

A code review determined that HUXY incorrectly calculated the Zircaloy heat of reaction as a function of temperature. The calculation did not account for the variation of heat capacity in the alpha-beta transformation temperature range. The HUXY code was revised to correct the error.

The impact of this change on the PCTs for those plants for which Framatome ANP performs BWR LOCA analyses using this evaluation model was estimated to be +3 degrees F.

Automation of LOCA Calculation Process

An older plant analysis was replaced using the current LOCA heatup analysis automated process. The automated process evaluates several input parameters slightly different from the previous hand calculations.

The impact of this change on the PCT for the plant for which Framatome ANP has used the automated BWR LOCA analyses process was estimated to be -24 degrees F.

Incorrect Pump Junction Area

For the EXEM BWR evaluation model, the junction area that should be used for the recirculation pump junctions is the minimum of the connecting pipe area or the pump eye area. It was determined that the current analysis supporting one plant used one-half the connecting pipe area for the pump junctions. The full pipe area should have been used in this analysis. Sensitivity studies indicate that the analysis would be impacted only if choked flow occurred at the pump junction and that does not occur until the area is reduced to much less than one-half the pipe area.

The impact of this change on the PCT for the plant for which Framatome ANP has used the incorrect pump junction area was estimated to be 0 degrees F.

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 Rev 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 evaluation model changes or error corrections made during 2001 for the EXEM BWR evaluation model also apply to the EXEM BWR-2000 evaluation model.

The EXEM BWR-2000 Evaluation Model has not yet become the licensing basis for any 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. The NRC approved topical report for this evaluation model is BAW-10104PA Rev 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 2001.

CRAFT2 PWR Small Break LOCA Evaluation Model

This model is applicable to all B&W designed pressurized water reactors for small break LOCA analyses. The NRC approved topical report for this evaluation model is BAW-10154PA Rev 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 2001.

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. The NRC approved topical report for this evaluation model is BAW-10192PA Rev 0.

The large break LOCA Evaluation Model consists of four 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, (3) CONTEMPT to compute the containment pressure response, and (4) 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) RELAP5/MOD2-B&W to compute the system, core, and hot rod response during the transient and (2) CONTEMPT to compute the containment pressure response, if needed. An NRC-approved fuel code (currently TACO3) is used to supply the fuel rod steady-state conditions at the beginning of the small or large break LOCA.

References:

1. FTI Topical Report BAW-10192P-A, Rev. 0, "BWNT LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," June 1998.
2. Letter to BWOG Analysis Committee, "PSC 2-00 Status and Summary", FTI-00-2268, September 11, 2000.
3. Letter to USNRC, "Report of Preliminary Safety Concern Related to Core Flood Line Break with 2-Minute Operator Action Time", FTI-00-2433, September 26, 2000.
4. Letter to USNRC, " Transmittal of Final Report on the Evaluation of PSC 2-00 Relating to Core Flood Line Break with 2-Minute Operator Action Time", FANP-01-988, April 2, 2001.
5. FTI Topical Report BAW-10166PA-04, "BEACH – Best Estimate Analysis Core Heat Transfer; A Computer Program for Reflood Heat Transfer During LOCA", February 1996.
6. FTI Topical Report BAW-10166P-05, "BEACH – Best Estimate Analysis Core Heat Transfer; A Computer Program for Reflood Heat Transfer During LOCA", December 2001.

Correction of RCP Two-Phase Degradation Model for SBLOCA

The NRC-approved SBLOCA EM (Volume II of Ref. 1) prescribes that the two-phase RELAP5 head difference and degradation multipliers, derived from the Semiscale pump tests, be used with the reactor coolant pump (RCP) performance curves. Examination of the Semiscale pump degradation curves, which are based upon tests run at relatively low pressures, indicates that the RELAP5 model can overpredict the amount of head degradation during the first several minutes of a SBLOCA transient with continued RCP operation (as analyzed in resolution of Preliminary Safety Concern 2-00). Comparison of the RELAP5 curves to representative data, specifically the CE 1/5-scale steam-water tests (which were run at higher pressures) confirms that the RELAP5 model overpredicts pump head degradation during two-phase flow early in the event. Since less pump degradation results in additional RCS liquid loss that leads to more extensive core uncovering and higher PCTs, the approved RELAP5 model cannot be judged to be conservative for application to continued RCP operation during a SBLOCA. When a bounding pump performance curve (the lower bound "M3-modified" curve used in the approved large break LOCA model) is modeled, the predicted consequences are much more severe. Therefore, the SBLOCA EM must be corrected to specify that the selection of a RCP two-phase degradation model in future SBLOCA analyses must be justified by sensitivity studies. This approach, which is similar to that used for LBLOCA applications, performs or makes reference to applicable studies that determine which RCP degradation model is conservative for application to plant-specific SBLOCA analyses.

RCP degradation model sensitivity studies were performed to resolve PSC 2-00 with RCP trip at two minutes after loss of subcooling margin. It was concluded from these analyses that the consequences for the CFT line break are most influenced by this change in degradation. CFT line break sensitivity studies confirmed that the M3-modified curve produces the most conservative results.

The PCT change associated with the correction to the SBLOCA RCP degradation model for the B&W-designed lowered loop plants is highly plant specific as determined by the ECCS configurations and flow capacities. For some plants there was no change in overall SBLOCA PCT, however, for others the PCT change was sufficient for the PCT to exceed 2200 F if the pumps were not tripped until two minutes. RCP trip at one-minute results in no core uncovering for the cases that exceed 2200 F with a two-minute trip. The utilities affected by the RCP trip time for analyses performed at 100 percent full power were notified on 9/11/00 (Ref. 2). The NRC was initially notified on 9/26/00 (Ref. 3). During the resolution of the PSC, the LOCA analyses that support off-nominal plant operation at 75 percent power with one HPI pump out of service were also found to have unacceptable PCT consequences with respect to PSC 2-00. Acceptable results were achieved at a reduced power of 50 percent with credit for a two-minute RCP trip. The final report summarizing the analyses and sensitivity studies for all operating B&W plants was sent to the NRC on 4/2/01 (Ref. 4).

<u>BEACH Initial Cladding Temperature Range</u>

Revisions to the BEACH topical report (Ref. 5) were submitted to the NRC in December of 2001. The revision (Ref. 6) seeks to extend ranges on the SER restrictions currently in place on the use of BEACH.

Appendix H was added to the BEACH topical to extend the limitation of the cladding temperature at the onset of reflood. The BEACH SER restriction on Revision 2 limited the temperature to between 950 and 1640 F. All of the demonstration cases presented in the BWNT LOCA EM (Ref. 1) and subsequently approved by the NRC had cladding temperatures that exceeded this range. Nonetheless, Framatome ANP completed an additional benchmark to assure that the code application could be extended to higher cladding temperatures. This benchmark (FLECHT-SEASET Test 34420) provides additional confirmation that the mechanistic modeling of BEACH is adequate and acceptable for reflood heat transfer prediction for plant LOCA application temperatures to 2045 F. This material was provided to the utilities for 10 CFR 50.46 reporting for calendar year 1999. After the NRC was informed that the BWNT LOCA EM demonstration cases were outside of this initial temperature range, they stated that this was an error and they requested that a formal revision to the BEACH code topical report be prepared and submitted for review and approval. The extension of the SER restriction on the initial cladding temperatures effectively corrects all application analyses that may have had temperatures outside of the initially approved range.

The PCT change for increasing the acceptable range of initial cladding temperatures in BEACH analyses is 0 F for all plants because no alteration of the code results or formulation was required.

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 Rev 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. An NRC-approved fuel code (currently TACO3) 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 2001.

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 Rev 0.

The SEM/PWR-98 LBLOCA 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 error corrections and model changes to the LBLOCA Evaluation Model implemented during this reporting period are described below.

Inappropriate Heat Transfer in TOODEE2

TOODEE2 calculates a radiation heat transfer coefficient which is used during the refill period of the TOODEE2 calculation. In addition, after reflood begins, the radiation heat transfer coefficient at each node is used until the FCTF reflood heat transfer becomes larger than the radiation heat transfer. Additionally, if at any point in the calculation, the radiation heat transfer coefficient becomes larger than the FCTF calculated value, the radiation model heat transfer coefficient is used. However, once a convective heat transfer coefficient is used in the calculation for an individual node, assumptions used in the radiation heat transfer model are no longer valid. This results in erroneous radiation sink temperatures to be calculated and, thereby, erroneous calculation of radiation heat transfer coefficients. It has been found that when extended calculations are run to evaluate maximum oxidation, the radiation model produces erroneous negative sink temperatures and large heat transfer coefficients which exceed the FCTF predicted values. When this occurs, the erroneous heat transfer coefficient values are used.

The TOODEE2 code was modified to turn off the radiation model at each node once the convective heat transfer at the node exceeds the radiation value.

The impact of this change on the PCTs for those plants for which Framatome ANP performs LBLOCA analyses using this evaluation model was estimated to be 0 degrees F.

End-of-Bypass Prediction by TEOBY

The time at end-of-bypass is based on the time of sustained flow reversal at the junction between the upper and lower downcomer or the junction between the broken cold leg and upper downcomer. The SEM/PWR-98 methodology defined sustained flow reversal as a reversal which occurs for 0.5 seconds or greater. It has been found that under certain circumstances, when a flow reversal occurs at the junction between the upper downcomer and the broken cold leg, a sustained flow reversal may begin between the upper and lower downcomer before 0.5 seconds has passed which may cause a momentary positive flow from the upper downcomer to the broken cold leg. As a result, the end-of-bypass time is based on the flow reversal time for upper to lower downcomer junction instead of earlier reversal at the junction between the upper downcomer and the broken cold leg.

The methodology was modified to choose the earlier reversal at the break to define end-of-bypass time. The short period of positive flow is an artifact of the 1 dimensional homogenous equilibrium model used in the methodology and does not represent the resumption of bypass flow.

The impact of this change on the PCTs for those plants for which Framatome ANP performs LBLOCA analyses using this evaluation model was estimated to be 0 degrees F.

Errors Discovered During RODEX2 V&V

This change is discussed under the EXEM BWR evaluation model since it applies to both BWR and PWR evaluation models.

The impact of this change on the PCTs for those plants for which Framatome ANP performs LBLOCA analyses using this evaluation model was estimated to be 0 degrees F.

Accumulator Parameters

The SEM/PWR-98 evaluation model was changed to require the use of average values for the parameters used to describe the Accumulators, loss coefficients, volumes, and areas.

The impact of this change on the PCTs for those plants for which Framatome ANP performs LBLOCA analyses using this evaluation model was estimated to be +4 degrees F.

Isolation Valve Modeling

The guideline for the SEM/PWR-98 evaluation model was changed to require the use of instantaneous stroke times versus the use of realistic stroke times. This additional detail was added to the guideline to ensure consistency amongst plants as well as to avoid potential numerical instabilities. Since this change is a model change, and not an error correction, the change will be reported in conjunction with the first use of the revised guideline for an individual plant. For some plants this change in the guideline was made prior to the first use of the SEM/PWR-98 evaluation model for the plant and the change was considered part of the change to the SEM./PWR-98 evaluation model.

The impact of this change on the LBLOCA PCT for one plant was estimated to be about +40 degrees F.

Safety Injection Nozzle Location Modeling

The guideline for the SEM/PWR-98 evaluation model was changed to require that the intact cold leg be divided at the safety injection nozzle location (nodalization). This additional detail was added to the guideline to ensure consistency amongst plants. Since this change is a model change, and not an error correction, the change will be reported in conjunction with the first use of the revised guideline for an individual plant. For some plants this change in the guideline was made prior to the first use of the SEM/PWR-98 evaluation model for the plant and the change was considered part of the change to the SEM./PWR-98 evaluation model.

The impact of this change on the LBLOCA PCT for one plant was estimated to be about +20 degrees F.

ANF-RELAP PWR 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 small break LOCA analyses. The NRC approved topical report for this evaluation model is XN-NF-82-49PA Rev 1 Supplement 1.

The ANF-RELAP SBLOCA 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 code or methodology changes to this Evaluation Model implemented during this reporting period are described below.

Errors Discovered During RODEX2 V&V

This change is described under the EXEM BWR Evaluation Model.

The impact of this change on the PCTs for those plants for which Framatome ANP performs SBLOCA analyses using this evaluation model was estimated to be 0 degrees F.

Automation of ANF-RELAP Input

The SBLOCA analysis performed for one plant used an automated tool to generate the ANF-RELAP input. The previous analysis was performed with manually-generated ANF-RELAP input. There are a number of methods used by the automated tool which result in nodalization and input model changes when compared to the previous input.

The impact of this change on the PCTs for those plants for which Framatome ANP performs SBLOCA analyses using this evaluation model was estimated to be 0 degrees F.

S-RELAP5 PWR 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 small break LOCA analyses. The NRC approved topical report for this evaluation model is EMF-2328PA Rev 0.

The evaluation model changes or error corrections made during 2001 for the ANF-REALP SBLOCA evaluation model are also applicable to the S-REALP5 SBLOCA evaluation model.

The S-RELAP5 SBLOCA Evaluation Model has not yet become the licensing basis for any plants.