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0CAN050801

May 7, 2008

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

SUBJECT: Errors or Changes in the Emergency Core Cooling System Evaluation Model: Annual Report for 2007
Arkansas Nuclear One, Units 1 and 2
Docket Nos. 50-313, 50-368, and 72-13 (ISFSI)
License Nos. DPR-51 and NPF-6

Dear Sir or Madam:

10 CFR 50.46(a)(3)(ii) requires licensees to report annually each change to or error discovered in an acceptable evaluation model or in the application of such model for the emergency core cooling system (ECCS) that affects the peak cladding temperature. Included in this submittal is the estimated effect the changes or errors identified during the reporting period of January 1 through December 31, 2007, have on the limiting ECCS analysis for each unit at Arkansas Nuclear One (ANO).

This report fulfills the reporting requirements referenced above. This submittal contains no commitments.

Should you have any questions regarding this report, please contact me.

Sincerely,

Dale E. James

DEJ/rwc

Attachments: 1 ANO-1 Emergency Core Cooling System Report
2 ANO-2 Emergency Core Cooling System Report

cc: Mr. Elmo E. Collins
Regional Administrator
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U. S. Nuclear Regulatory Commission
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Attachment 1 to

0CAN050801

ANO-1 Emergency Core Cooling System Report

ANO-1 Emergency Core Cooling System Report

This section describes the generic Evaluation Model (EM) error corrections or changes that were made in 2007.

1.1 CRAFT2 LOCA EVALUATION MODEL ERROR CORRECTIONS OR CHANGES

No errors or changes were reported in the CRAFT2-based B&W ECCS EM, BAW-10104P-A, Rev. 5 for LBLOCA (Reference 1.5.1) and BAW-10154-A, Rev. 0 for SBLOCA (Reference 1.5.2), during 2007. Currently, all B&W plants rely on the BWNT LOCA EM (Section 1.2) to support the cycle-specific LOCA LHR limits. However, certain plants may rely on evaluations and analyses performed with the CRAFT2-based EM to support evaluation of older fuel designs that may be reinserted into a core design, plant operational guidance, or equipment qualification.

- No changes or errors in 2007.

1.2 BWNT LOCA EVALUATION MODEL ERROR CORRECTION OR CHANGES

This EM is applicable to all B&W-designed pressurized water reactors for large and small break LOCA analyses for zircaloy or M5 cladding. The NRC-approved topical report for this EM is BAW-10192P-A Rev 0 (Reference 1.5.3).

The LBLOCA EM consists of four computer codes:

1. BAW-10164P-A, RELAP5/MOD2-B&W to compute the system, core, and hot rod response during blowdown (Reference 1.5.4);

2. BAW-10171P-A, REFLOD3B to calculate the time for refill of the lower plenum and core reflood rate (Reference 1.5.5);

3. BAW-10095-A, CONTEMPT to compute the containment pressure response (Reference 1.5.7); and

4. BAW-10166P-A, BEACH (the RELAP5/MOD2-B&W reflood heat transfer package) to determine the hot pin thermal response during refill and reflood phases (Reference 1.5.6).

The SBLOCA EM consists of two codes:

1. BAW-10164P-A, RELAP5/MOD2-B&W to compute the system, core, and hot rod response during the transient (Reference 1.5.4), and

2. BAW-10095-A, CONTEMPT to compute the containment pressure response (Reference 1.5.7), if needed.

An NRC-approved fuel code (currently BAW-10162P-A, TACO3 (Reference 1.5.9) or BAW-10184P-A, GDTACO (Reference 1.5.10)) 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 as discussed in BAW-10227P-A (Reference 1.5.8).

It should be noted that the most recently approved versions of the EM and associated topical reports are referenced above. However, the plant- and fuel-type-specific licensing calculations may be based on an earlier approved method and thus do not specifically utilize the enhanced features that were subsequently approved or submitted as a change via 10 CFR 50.46.

1.3 BWNT LOCA EVALUATION MODEL GENERIC ANALYSES

There were no generic B&W evaluations completed in 2007.

1.4 GENERAL INFORMATION

General information related to the EM topical reports and associated modeling guidance, and LOCA-related analyses (sensitivity study or application that supports meeting the remaining 10 CFR 50.46 criteria) that were performed, but do not relate directly to peak clad temperature (PCT) predictions, are discussed in this section. These topics do not constitute EM changes or error corrections, and are provided as information.

1.4.1 Revision 6 to RELAP5 Topical Report BAW-10164

Revision 6 to the RELAP5 topical report (Reference 1.5.4) was prepared and includes typographical corrections, including the BHTP Critical Heat Flux (CHF) option, a user-input BHTP CHF option and the user-input BWU EM CHF option. The BHTP CHF correlation was implemented into the code and reported as a generic EM change in the 2003 and 2005 annual letters (References 1.5.13, 1.5.14 and 1.5.15). The new revision to the RELAP5 topical was submitted to the NRC for review, and gained approval for use (Reference 1.5.4).

- Other – Topical report approval.
- This revision does not deal with PCT effects.

1.4.2 Long-Term Core Cooling GSI-191 Chemical Effects

Concerns have been raised about the potential for debris ingested into the ECCS to affect long-term core cooling when recirculating coolant from the containment sump. This issue is being tracked by the industry as GSI-191. The Pressurized Water Reactor Owners Group (PWROG) undertook a program to address these issues on a generic basis. The objective of the program was to demonstrate that there is reasonable assurance that sufficient long-term core cooling is achieved for PWRs to satisfy the requirements of 10 CFR 50.46 with debris and chemical products that might be transported to the reactor vessel and core by the coolant recirculating from the containment sump. A calculation method was developed by Westinghouse for individual utilities to perform plant-specific analyses of the chemical effects in the core region. The model is described in detail in WCAP-16793-P, Section 5 and Appendix E. Part of the inputs required for this model are core region powers based on plant-specific fuel loading. The appropriate inputs for the B&W-designed plants were provided in Reference 1.5.11.

1.4.3 BAW-2374

AREVA has completed Revision 2 of BAW-2374 (Reference 1.5.16) to request a change to the licensing basis to establish a risk-informed basis for the acceptability of postulated thermal loads on once-through steam generator tubes, tube repair products, and tube-to-tubesheet joints induced by a LOCA in the large-bore piping of the reactor coolant system. This revision was prepared to address compliance to 10 CFR 50.46, 10 CFR 100 and 10 CFR 50.67 for a postulated hot leg U-bend LOCA. Compliance with these criteria has been demonstrated for this LOCA, which can result in consequential steam generator tube rupture of degraded tubes due to the high SG tensile loads during the refill of the reactor coolant system. The revised topical report was submitted to the NRC for review and approval (Reference 1.5.12).

With regard to 10 CFR 50.46, long-term cooling, plant specific net positive suction head (NPSH) calculations provided by each utility justified that the ECCS pumps would continue to operate and provide flow to the RCS under this scenario. The applicability of these NPSH analyses is required to support application of the approved BAW-2374 topical report.

With regard to 10 CFR 100, a generic LOCA analysis was performed specifically for this scenario that could result in consequential steam generator tube rupture. The results of the LOCA analysis justified that the fuel rod cladding would not rupture during this event. Thus it was concluded that this scenario was bounded by the existing dose consequences contained in the licensing basis. The continued assurance that no fuel cladding rupture will occur during this scenario is required to support application of the approved BAW-2374 topical report.

- Other – Revision to BAW-2374.
- There are no PCT changes associated with this topic.

1.5 REFERENCES

1.5.1 AREVA NP Proprietary Topical Report BAW-10104P-A, Rev. 5, "B&W's ECCS Evaluation Model", November 1988.

1.5.2 AREVA NP Topical Report BAW-10154-A, Rev. 0, "B&W's Small-Break LOCA ECCS Evaluation Model", July 1985.

1.5.3 AREVA NP Proprietary Topical Report BAW-10192P-A, Rev. 0, "BWNT LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants", June 1998.

1.5.4 AREVA NP Proprietary Topical Report BAW-10164P-A, Rev. 6, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis", June 2007.

1.5.5 AREVA NP Proprietary Topical Report BAW-10171P-A, Rev. 3, "REFLOD3B – Model for Multinode Core Reflooding Analysis", December 1995.

1.5.6 AREVA NP Proprietary Topical Report BAW-10166P-A, Rev. 5, "BEACH – A Computer Program for Reflood Heat Transfer During LOCA", November 2003.

1.5.7 AREVA NP Topical Report BAW-10095-A, Rev. 1, "CONTEMPT – Computer Program for Predicting Containment Pressure-Temperature Response to a LOCA", April 1978.

1.5.8 AREVA NP Proprietary Topical Report BAW-10227P-A, Rev. 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel", June 2003.

1.5.9 AREVA NP Proprietary Topical Report BAW-10162P-A, Rev. 0, "TACO3 Fuel Pin Thermal Analysis Code", October 1989.

1.5.10 AREVA NP Proprietary Topical Report BAW-10184P-A, Rev. 0, "GDTACO Urania – Gadolinia Fuel Pin Thermal Analysis Code", February 1995.

1.5.11 AREVA NP Document 51-9051889-001, "Core Power Inputs for GSI-191 Chemical Effects", August 2007.

1.5.12 Letter OG:07:3 dated January 4, 2007 from F. P. Schiffley (PWR Owners Group) to Document Control Desk (US NRC), Subject: Submittal of BAW-2374-NP, Revision 2 "Risk-Informed Assessment of Once Through Steam Generator Tube Thermal Loads due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping" (PA-ASC-0255).

1.5.13 AREVA NP Document 47-5039949-00, "2003 DRAFT Annual BWOG 50.46 Letter", March 2004.

1.5.14 AREVA NP Document 47-9012029-001, "2005 DRAFT Annual B&W 50.46 Letter", May 2006.

1.5.15 AREVA NP Document 12-9012652-000, “Supplement to 2005 DRAFT Annual B&W 50.46 Letter”, March 2006.

1.5.16 AREVA NP Topical Report BAW-2374-02, “Risk-Informed Assessment of Once-Through Steam Generator Tube Loads due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping”, December 2006.

2.0 Conclusions

There were no changes to, or errors in, the BWNT LOCA EM to report for calendar year 2007 that have an impact on the Arkansas Nuclear One – Unit 1 (ANO-1) LBLOCA, SBLOCA or LTC Analyses of Record that supported operation for the calendar year 2007 reporting period.

3.0 ANO–1 Specific Information

The following analysis was performed in 2007 and is applicable only to ANO Unit 1.

3.1 EXTENDED CFT SBLOCA

Entergy requested AREVA NP to perform an extension of the CFT line SBLOCA analysis to the time of sump switchover for ANO-1 to provide input (RCS pressure vs. time) to Entergy to be used in their long-term core cooling analysis. The analysis was performed in Reference 3.2.1, and summarized in Reference 3.2.2.

- Other – Extended CFT SBLOCA analysis to provide input to Entergy for use in their long-term core cooling analysis.

- This analysis does not involve PCT effects.

3.2 REFERENCES

3.2.1 AREVA NP Proprietary Document 32-9055562-000, “ANO-1 Mk-B-HTP Extended CFT SBLOCA”, September 2007.

3.2.2 AREVA NP Document 86-9057903-000, “ANO-1 Extended CFT SBLOCA Summary Report”, September 2007.

Attachment 2 to

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ANO-2 Emergency Core Cooling System Report

ANO-2 Emergency Core Cooling System Report

1.0 Evaluation Model Generic Information

Evaluation models (EM) for emergency core cooling system (ECCS) performance analysis of Combustion Engineering (CE) designed PWRs are described in topical reports, are reviewed and approved by the NRC, and are covered by the provisions of 10 CFR 50.46. The evaluation model for the large break loss of coolant accident (LBLOCA) is the 1999 EM. For the small break loss of coolant accident (SBLOCA), the evaluation model is the S2M EM. Post-loss of coolant accident (LOCA) long term cooling (LTC) analyses (which do not affect peak clad temperature) use the LTC evaluation model.

Several digital computer codes are used to do ECCS performance analyses of pressurized water reactors (PWRs) for the evaluation models described above that are covered by the provisions of 10 CFR 50.46. Those for LBLOCA calculations are CEFLASH-4A, COMPERC-II, HCROSS, PARCH, STRIKIN-II, and COMZIRC. CEFLASH-4AS is used in conjunction with COMPERC-II, STRIKIN-II, and PARCH for SBLOCA calculations. The codes for post-LOCA LTC analyses are BORON, CEPAC, NATFLOW, and CELDA.

2.0 Westinghouse Generic Model Changes and Error Corrections

2.1 Appendix K Large Break – 1999 EM Related Items

- Implementation of Optimized ZIRLO™ Cladding Specific Heat
  (Enhancements / Forward-Fit Discretionary Changes)

  <u>Background</u>

  The Appendix K ECCS Performance Analysis for LBLOCA for CE plants is performed with the 1999 EM. The evaluation of fuel designs that use Optimized ZIRLO™ cladding was described in Reference 1 and approved by NRC in Reference 2. In compliance with SER Limitation/Constraint #9 from Reference 2, the Optimized ZIRLO™ cladding specific heat has been implemented into all of the computer codes of the 1999 EM as described by Westinghouse in the response to RAI #21 in Reference 3. The Optimized ZIRLO™ cladding specific heat has been implemented as an option for any future LBLOCA analysis of a Westinghouse fuel rod design that uses this cladding.

  <u>Estimated Effect</u>

  Since this change has already been approved by NRC, the licensed methodology of the 1999 EM is not affected. With the selection of this option for analyses using the 1999 EM, all of the computer codes are brought into compliance with the SER Limitation/Constraint imposed on the modeling of Optimized ZIRLO™ cladding. The impact of the change on PCT becomes integrated with the plant-specific and reload-specific analysis results used in the 10 CFR 50.46 reporting process.

References

1. WCAP-12610-P-A and CENPD-404-P-A Addendum 1, "Addendum 1 to WCAP-12610-P-A and CENPD-404-P-A Optimized ZIRLO^{TM}," LTR-NRC-03-2, February 14, 2003. (ADAMS Accession No. ML030520455)

2. Letter from H. N. Berkow (NRC) to J. A. Gresham (Westinghouse), "Final Safety Evaluation for Addendum 1 to Topical Report WCAP-12610-P-A and CENPD-404-P-A, 'Optimized ZIRLO^{TM}' (TAC No. MB8041)," June 10, 2005.

3. Letter from J. A. Gresham (Westinghouse) to U.S. Nuclear Regulatory Commission," Westinghouse Responses to NRC Request for Additional Information (RAIs) on Optimized ZIRLO^{TM} Topical – Addendum 1 to WCAP-12610-P-A," LTR-NRC-04-44, August 4, 2004. (ADAMS Accession No. ML042240408)

- Path Enthalpy Transport Numerical Stability in CEFLASH-4A
(Enhancements / Forward-Fit Discretionary Changes)

Background

The Appendix K ECCS Performance Analysis for LBLOCA for CE plants is performed with the 1999 EM. Appendix I of CENPD-133 P (Reference 1) documents the flow path enthalpy transport model used in the CEFLASH-4A computer code for calculating the blowdown thermal-hydraulics response during a LBLOCA transient. An upgrade to the flow path enthalpy transport model has been implemented as an optional process improvement to give the user additional numerical stability control other than modifying time step size. The optional improvement has been implemented to automatically control numerical convergence during time periods where the system flow rates are nearly zero, such as near end of blowdown, and the stability of the calculation is more difficult to maintain.

Estimated Effect

This process improvement has no impact on the licensed methodology of the 1999 EM and does not conflict with the SER limitation/constraints imposed on the methodology by NRC. The improvement is made available to the user as an option and is designed to maintain a numerically stable solution. Use of the option will prevent abnormal code operation requiring manual time step adjustments with the current logic to achieve convergence. The impact on PCT for 10 CFR 50.46 reporting purposes is not significant when compared to converged results.

References

1. CENPD-133P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," August 1974

- Upgrade to the Auto-Axial Power Shape Option in STRIKIN-II (Enhancements / Forward-Fit Discretionary Changes)

Background

The Appendix K ECCS Performance Analysis for LBLOCA for CE plants is performed with the 1999 EM. The 1999 EM is required to specify the axial power shape for LBLOCA analyses consistent with the methodology documented in Appendix A of CENPD-132 Supplement 3-P-A. Also, the specification of the axial power shape must be in compliance with an SER Limitation/Constraint on the use of this methodology.

As part of the Advanced Automated Integrated Code System (AAICS) for the 1999 EM, an option was implemented previously to automatically determine the axial power shape that complies with the core design characteristics of the analysis and also meets the methodology requirements. One aspect of the automatic process involves achieving a particular power distribution consistent with the target value for the minimum axial shape index (ASI), the limiting ASI prescribed for top peaked power distributions that are worst for LBLOCA. An upgrade to the computer logic has been implemented to improve the process for matching the prescribed target ASI for the 1999 EM. The improvement more accurately achieves the target ASI over a broader range of conditions than before. This improved computer logic replaces the previous logic. However, it still remains the responsibility of the user to examine the axial power shape generated by the automatic process on a case by case basis.

Estimated Effect

This process improvement has no impact on the licensed methodology of the 1999 EM and does not conflict with the SER limitation/constraints imposed on the methodology by NRC. There is no impact on PCT for 10 CFR 50.46 reporting purposes.

- Implementation of Cathcart-Pawel Oxidation Model and Pre-Transient Oxidation Model (Enhancements / Forward-Fit for LBLOCA Non-Licensing Application)

Background

A new UCI parameter (‘oxidation_model’) is added to the UCI File Parameter List to facilitate the selection of the Cathcart-Pawel model for cladding oxidation as an option for non-licensing applications. The Cathcart-Pawel oxidation model is a best estimate model utilized by the industry as an alternative to the Appendix K required Baker-Just model. This new UCI option is not permitted for licensing applications of the 1999 EM, which must use the Appendix K required Baker-Just model.

A new UCI parameter ('pre_tran_oxidation') is added to the UCI File Parameter List to facilitate the specification of the initial oxide layer thickness through the UCI input file instead of through the base decks for non-licensing applications. This option may also be used to link the input specification to an interface output file from the HIDUTYDRV computer code, which provides maximum oxide thickness as a function of burnup (coordinated with FATES3B cycle numbers) and cladding type. In addition, this UCI parameter contains a new cladding conductivity option, which includes the impact of the oxide layer on the cladding conductivity, thereby directly linking the initialization of the fuel stored energy with the amount of pre-transient oxidation. The initial oxide layer thickness is a user-specified required input to the evaluation model as a constraint on the acceptability of the model for licensing applications. Therefore, this new UCI option is not permitted for licensing applications of the 1999 EM, which must use the required input.

The alternate oxidation model and pre-transient oxidation model are used to study various aspects of new embrittlement criteria being suggested by NRC and to provide an alternate means to address NRC questions regarding the calculation of the peak cladding oxidation percentage in applications of the 1999 EM to CE plants. The new UCI options provide for automatic activation of input vector changes and ensure consistency among the other computer codes of the 1999 EM.

Estimated Effect

This process improvement is for non-licensing applications. Therefore, this change has no impact on the licensed methodology of the 1999 EM and does not conflict with the SER limitation/constraints imposed on the methodology by NRC for licensing applications. For licensing applications, there is no impact on PCT for 10 CFR 50.46 reporting purposes since these changes are intended only for use in non-licensing calculations.

- Implementation of Final SER Limitations and Conditions for the Optional Spacer Grid Steam Cooling Heat Transfer Model Improvement in STRIKIN-II (Non-Discretionary Changes With No PCT Impact for LBLOCA Licensing Applications)

  Background

  The Appendix K ECCS Performance Analysis for LBLOCA for CE plants is performed with the 1999 EM. An improved optional steam cooling heat transfer model for core reflood rates less than 1 in/sec was submitted to NRC in May 2006, to calculate the effects of spacer grids on steam cooling heat transfer mechanisms, CENPD-132, Supplement 4-P-A, Addendum 1-P. NRC acceptance of this improvement to the 1999 EM was documented in the final SER received by Westinghouse in June 2007.

The final SER from NRC requires that several changes be made to the methodology. All future licensing applications that utilize the optional spacer grid steam cooling heat transfer model will be required to implement these changes by using STRIKIN-II, Version STR.2.11 or higher. The following changes bring this version of STRIKIN-II into full compliance with the SER limitations and conditions on the use of the optional steam cooling model as imposed by NRC:

- Grid Rewet Temperature Criterion – The final model approved by NRC requires that the spacer grid rewet temperature criterion be reset to a slightly different value than originally proposed.

- Grid Model Output File – The final model approved by NRC requires that additional information be plotted and submitted for NRC review prior to the initial application of the model to a particular CE plant. To facilitate this documentation requirement, an additional output file is added to streamline generation of the needed plots.

- Update Reynolds Number Formulation – The final model approved by NRC requires that the Reynolds number formulation for the wetted spacer grid heat transfer calculation be revised.

- Diagnostic Edit Statements – The final model approved by NRC requires that the blockage fraction and Reynolds number ranges of applicability be confirmed for each application of the model. Diagnostic edit statements to display the range of validity checks are upgraded to alert the user if the use of the model is outside the range of applicability.

These changes have an insignificant impact on the overall calculated results with no impact on PCT. The revised methodology is activated with the selection of the user-specified option that is referred to as the "NRC-approved model." These changes have no effect on the calculated results if the optional steam cooling model is not used in the analysis.

Estimated Effect

These NRC-required changes to the optional steam cooling model have no impact on PCT for 10 CFR 50.46 reporting purposes for any application that utilizes the optional methodology.

- Additional Miscellaneous Process Improvements

The following additional process improvements were made. These improvements have no impact on the licensed methodology of the 1999 EM and do not conflict with the SER limitations/constraints imposed on the methodology by the NRC. There is no impact on PCT for 10 CFR 50.46 reporting purposes.

- Modifications to the source file notation and output edit headers for the major computer codes of the 1999 EM have been made to indicate the copyright status and proprietary class in conformance with Westinghouse policy, WCAP-7211.

- The editing logic in CEFLASH-4A, the blowdown thermal hydraulics systems code of the 1999 EM, has been modified to prevent an incorrect error message from causing the utility script, which manages the operation of the 1999 EM, from an improper termination. In CEFLASH-4A, the error message should have been only a debug warning message and should not cause the utility script to stop execution.

- Computer code base and case decks provide the inputs to the operation of the 1999 EM. One of the advancements of the 1999 EM has been the auto-generation of base decks that can be created using inputs provided elsewhere in the 1999 EM code system. This eliminates some documentation and quality assurance and improves quality control by eliminating sources of error. The hot rod heatup computer code of the 1999 EM is STRIKIN-II. A process improvement has been implemented to automatically generate the STRIKIN-II base and case decks in a manner consistent with the fuel performance data set from the FATES3B computer code, which provides initial conditions for fuel rod stored energy and rod internal pressure. For licensing applications, an update has been made to the auto-generation of these input decks for LBLOCA. These updates are categorized as general code maintenance changes to improve the operation and flexibility of this feature of the code.

- The core-wide cladding oxidation is calculated using the COMZIRC computer code. COMZIRC represents the core fuel pin census with a table of values of fuel rod power versus fraction of rods. The current version allows a maximum of only 12 radial intervals to represent the core. The purpose of this change is to increase the radial detail from 12 to 120 intervals, thus allowing enhanced control of the discretionary conservatism needed to bound the core pin census with the COMZIRC input table.

- COMZIRC initializes the radial core regions using information provided by the CEFLASH-4A computer code for its radial distribution in the core. The assignment of the CEFLASH-4A radial regions to the radial pin census in COMZIRC was previously a manual operation performed by the user. Computer logic has been implemented in COMZIRC to automatically confirm and/or provide the assignment of the CEFLASH-4A radial regions to the pin census.

2.2 Appendix K Small Break – S2M Related Items

There are no issues to report for 2007.

3.0 Conclusions

There were no EM changes to, or errors in, the LBLOCA 1999 EM, SBLOCA S2M, or Long-Term Cooling (LTC) EM to report for calendar year 2007 that have an impact on the Arkansas Nuclear One – Unit 2 (ANO-2) LBLOCA, SBLOCA and LTC Analyses of Record that support operation for the calendar year 2007 reporting period.

4.0 ANO-2 Specific Information

There is no plant specific LBLOCA or SBLOCA effect due to the changes made to the 1999 EM.

A complete reanalysis of both LBLOCA and SBLOCA ECCS performance was completed in 2007 for ANO-2 in order to implement Optimized ZIRLO clad and the Next Generation Fuel (NGF) design. These analyses used the Westinghouse ECCS performance evaluation models for Combustion Engineering designed PWRs (i.e., the 1999 EM for LBLOCA and the S2M EM for SBLOCA). The LBLOCA and SBLOCA analyses were performed with bounding physics and fuel performance data in order to be applicable to future cycles of ANO-2. Since ANO-2 had not yet implemented NGF at the end of 2007, those analyses were not effective for the reporting period of this document.