Entergy

Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802
Tel 479-858-4619

Dale E. James
Manager, Licensing

0CAN060703

June 4, 2007

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

Subject: Errors or Changes in the Emergency Core Cooling System Evaluation Model: Annual Report for 2006
Arkansas Nuclear One – Units 1 and 2
Docket Nos. 50-313 and 50-368
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 from January 1 through December 31, 2006, have on the limiting ECCS analysis.

This report fulfills the reporting requirements referenced above.

This submittal contains no commitments. If you have any questions regarding the data, please contact David Bice at (479) 858-5338.

Sincerely,

Dale E. James

DEJ/dbb

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

A002
NRR

cc: Dr. Bruce S. Mallett
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Arkansas Nuclear One
P.O. Box 310
London, AR 72847

U.S. Nuclear Regulatory Commission
Attn: Mr. Alan Wang
Mail Stop 0-7 D1
Washington, DC 20555-0001

Attachment 1 to

0CAN060703

ANO-1 Emergency Core Cooling System Report

ANO-1 Emergency Core Cooling System Report

1.0 AREVA Generic Model Changes and Error Corrections

This section describes the generic Evaluation Model (EM) error corrections or changes that were made in 2006.

1.1 CRAFT2 LOCA Evaluation Model Error Corrections or Changes

No errors or changes were reported in the CRAFT2-based Babcock & Wilcox (B&W) Emergency Core Cooling System (ECCS) EM, BAW-10104P-A, Rev. 5 for large break loss of coolant accident (LBLOCA) (Reference 1.1) and BAW-10154-A, Rev. 0 for small break loss of coolant accident (SBLOCA) (Reference 1.2), during 2006. Currently, all B&W plants rely on the BWNT loss of coolant accident (LOCA) EM (Section 1.2) to support the cycle-specific LOCA linear heat rate (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

1.2 BWNT LOCA EM 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 1.3).

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 1.4),

2. BAW-10171P-A, REFLOD3B to calculate the time for refill of the lower plenum and core reflood rate (Reference 1.5),

3. BAW-10095-A, CONTEMPT to compute the containment pressure response (Reference 1.7), 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 1.6).

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 1.4), and

2. BAW-10095-A, CONTEMPT to compute the containment pressure response (Reference 1.7), if needed.

An NRC-approved fuel code (currently BAW-10162P-A, TACO3 (Reference 1.9) or BAW-10184P-A, GDTACO (Reference 1.10)) 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 1.8).

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. A summary of the intermediate EM changes and EM error corrections made to BAW-10192P-A (Reference 1.3) is discussed in Section 3.1 and Table 1. No changes or errors were identified in calendar year 2006.

2.0 BWNT LOCA Evaluation Model Generic Analyses

Three generic B&W evaluations were completed in 2006. These topics do not constitute EM changes or error corrections, and are provided as information.

2.1 BAW-2374

AREVA has completed Revision 2 of BAW-2374 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 (SG) tubes, tube repair products, and tube-to-tubesheet joints induced by a LOCA in the large-bore piping of the reactor coolant system (RCS). This revision was prepared to address compliance to 10 CFR 50.46 and 10 CFR 100 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 (SGTR) of degraded tubes due to the high SG tensile loads during the refill of the RCS. The revised topical report was submitted to the NRC for review and approval (Reference 1.11).

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 regards to 10 CFR 100 dose, a generic LOCA analysis was performed specifically for this scenario that could result in consequential SGTR. 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.

Various analyses and evaluations were performed to support the release of the topical report, which are documented in References 1.12, 1.13, 1.14 and 1.15.

- Other – Revision to BAW-2374
- There are no peak clad temperature (PCT) changes associated with this topic.

2.2 Control Rod Temperature Operability Evaluation

A review of the updated safety analysis reports (USARs) for most of the B&W plants has indicated that the control rod temperature analysis has not been revised since the original analyses in the early 1970's. This issue is identified in AREVA Condition Report (CR) 2006-439. A deviation review was completed identifying and assessing potential safety concerns. This issue was determined to be a deviation, as documented in the CR attachment titled "CR 2006-439 Deviation Determination."

An initial operability assessment for the control rods based on a review of the current control rod temperature evaluations described in some of the B&W plant USARs was performed in Reference 1.16. It was determined that the control rod temperature evaluations are not based on the most current configuration of the control and fuel assemblies, nor on the current LOCA 10 CFR 50.46 cold leg pump discharge (CLPD) licensing case results. The preliminary temperature predictions have exceeded the silver-indium-cadmium (AIC) melt temperature and the eutectic temperature between the iron and nickel in contact with zircaloy or M5 guide tubes.

The results of the preliminary evaluations performed in Reference 1.16 concluded that there is a reasonable expectation that the control rods will continue to control reactivity and that no expulsion of control absorber material that causes a loss of coolable geometry will result. The expectation of continued long-term cooling and a coolable geometry is maintained. Therefore, the initial operability evaluation concluded that the control rod maintains its safety function during a LOCA. However, it was deemed necessary to perform additional evaluations to confirm the conclusions of the initial operability assessment. Since the evaluation of this issue was expected to take more than 60 days for completion, AREVA submitted an interim Part 21 notification to the NRC to inform the staff of this issue (Reference 1.17).

AREVA participated in a joint task with Westinghouse at the direction of the Pressurized Water Reactor Owner's Group (PWROG) to evaluate the question of control rod survivability in more detail. AREVA's role in the original task was to review Westinghouse's evaluation, come to agreement on a common approach, and present this approach to the NRC. A follow-on task was proposed to document any analyses that were identified as necessary to close-out the issue. The results of that work were informally communicated to the NRC and documented in a White Paper that was transmitted to the NRC (Reference 1.18). The conclusions in the White Paper are based partly on preliminary LOCA analyses performed by AREVA that explicitly included a model of the control rod in the RELAP5/MOD2-B&W. These preliminary AREVA control rod temperature predictions for a generically limiting B&W plant limited the peak control rod temperature to a value well below 2200 °F in order to address the mechanical aspects of survivability (based on control rod hoop stress). The original AREVA control rod temperature predictions reported in Reference 1.16 were found to be overly conservative and new results were obtained. These results were generically provided in the White Paper to set the maximum control rod temperature in the range of 1700 to 1800 °F for determination of the eutectic reaction rates and the mechanical performance (hoop stress).

Based on the conclusions stated in the White Paper, AREVA closed the Part 21 evaluation by letter to the NRC (Reference 1.19) since this issue is not considered to be a defect.

- Other – Control Rod survivability
- This evaluation does not deal with PCT effects.

2.3 Boron Precipitation Analysis Review for B&W Plants

A review of the current post-LOCA boric acid precipitation methods and calculations for the B&W-designed plants (excluding Oconee units) with respect to recent NRC concerns was performed in Reference 1.20. Each plant was found to have an overall conservative method to assure that the boric acid concentration in the core would not exceed the solubility limit following a LOCA ensuring that the requirements of 10 CFR 50.46 are met.

- Other – Boron Precipitation Management
- This review does not deal with PCT effects.

3.0 Generic EM 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 that do not relate directly to PCT predictions are discussed in this section. These topics do not constitute EM changes or error corrections, and are provided as information.

3.1 Revision 06 to RELAP5 Topical BAW-10164

A revision to the RELAP5 topical report (Reference 1.21) has been prepared that includes typographical corrections and adds BHTP critical heat flux (CHF) option, user-input BHTP CHF option and the user-input BWU EM CHF option. The new revision to the RELAP5 topical has been submitted to the NRC for review and approval (Reference 1.22). This report formalizes the 10 CFR 50.46 changes associated with items R0.8 and R0.9 in Table 1.

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

3.2 Outline of Evolution of BAW-10192P-A Changes

The general outline for application of the EM is described in BAW-10192P-A. The code topical report revisions identified in BAW-10192P-A, Rev. 0 for use in LOCA analyses are:

BAW-10164P-A, Revision 3, RELAP5/MOD2-B&W	Reference 1.23
BAW-10171P-A, Revision 3, REFLOD3B	Reference 1.5
BAW-10166P-A, Revision 4, BEACH	Reference 1.24
BAW-10095-A, Revision 1, CONTEMPT	Reference 1.7
BAW-10162P-A, Revision 0, TACO3	Reference 1.9

Since the approval of BAW-10192P-A, the codes and methods have evolved through approved code revisions, identification of specific codes not identified in the EM, and the addition of new methods and error corrections made under 10 CFR 50.46. The following NRC-approved topical reports have been added as part of the EM. However, these revisions are not explicitly identified in BAW-10192P-A, Rev. 0.

BAW-10164P-A, Revision 4, RELAP5/MOD2-B&W	Reference 1.4
BAW-10166P-A, Revision 5, BEACH	Reference 1.6
BAW-10227P-A, Revision 1, M5 Cladding (Revision 1 is related to non-B&W plant fuel and is not necessary for B&W plant licensing. However, the most recent revision of a topical report is referenced.)	Reference 1.8
BAW-10184P-A, Revision 0, GDTACO	Reference 1.10

The addition of new methods and error corrections made under 10 CFR 50.46 are reported to the utilities at least annually for inclusion in plant-specific 50.46 reports to the NRC.

In the case of new methods, acceptable licensing cases are typically not reanalyzed or reevaluated in order to take advantage of the new methods. (Error corrections are applied to current licensing basis cases as necessary.) Therefore, the last acceptable licensing cases for a specific plant may not have been performed with the most recent EM changes (approved or made via 10 CFR 50.46). In order to assist the utility in identifying which EM and 10 CFR 50.46 changes were utilized in their licensing cases, Table 1 was created. The different stages of models available via reference to BAW-10192P-A are identified in Table 1 as versions denoted by "RN.n", where "R" stands for revision, "N" represents the revision of BAW-10192P-A (in all cases, 0) and "n" is the subrevision identifier of the modeling change not explicitly described in the NRC-approved revision of BAW-10192P-A. These informal revision levels are for internal tracking purposes in order to identify those EM and 10 CFR 50.46 changes that are applied in the analyses. The revision level is reported in the plant-specific PCT summary tables alongside the final licensing PCT.

- Other – Outline of evolution of BAW-10192P-A.
- No 10 CFR 50.46 changes or corrections are discussed.

Table 1: Evolution of Changes to BAW-10192P-A [1]

Date First Introduced	BAW-10192P-A Revision 0	Description of Change		Method of Change and/or Approval
		LBLOCA	SBLOCA	
1993	R0.0	--	--	Original (approved 1997)
1994	R0.1	Core Flood Tank (CFT) Inventory (PSC 5-94)	CFT Inventory (PSC 5-94)	50.46 Change (Ref. 1.25)
1999	R0.2	M3-Modified RCP degradation multiplier (PSC 1-99)	*Not Applicable*	50.46 Change (Ref. 1.26)
1999	R0.3	Extension of BEACH acceptable range of application for initial cladding temperature	*Not Applicable*	50.46 Change (Ref. 1.26) Approved BAW-10166P-A Rev. 5 (Ref. 1.6, 2003)
1999	R0.4	M5 Cladding (if used)	M5 Cladding (if used)	Approved Change BAW-10227P-A (Ref. 1.8, 2000)
2000	R0.5	*Not Applicable*	M3-Modified Reactor Coolant Pump (RCP) degradation multiplier (via PSC 2-00)	50.46 Change (Ref. 1.27) Approved via Reference 1.28 and 1.29 (2003 and 2005)
		Not Applicable	Void-dependent cross-flow (via PSC 2-00)	50.46 Change (Ref. 1.27) Approved via BAW-10164P-A Rev. 4 (Ref. 1.4, 2002)
2002	R0.6	Appendix K Power Uprate (if applicable)	Appendix K Power Uprate (if applicable)	50.46 Change (Ref. 1.30)
2002	R0.7	Hot pin modeling for LBLOCA	*Not Applicable*	Approved Change BAW-10164P-A Rev. 4 (Ref. 1.4, 2002)
2003	R0.8	BHTP CHF correlation for analysis of Mark-B-HTP fuel (if used)	BHTP CHF correlation for analysis of Mark-B-HTP fuel (if used)	50.46 Change (Ref. 1.31)
2005	R0.9	*Not Applicable*	Application of BHTP CHF correlation for analysis of Mark-B-HTP fuel (if used)	50.46 Change (Ref. 1.32 and 1.33)

[1] Higher revision levels implicitly include the changes contained in the previous revisions. Some changes are specific to a fuel assembly design or plant change, and are not applicable to all analyses (marked as "if used" or "if applicable").

4.0 References

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

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

1.3 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.

1.4 AREVA NP Proprietary Topical Report BAW-10164P-A, Rev. 4, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," November 2002.

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

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

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

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

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

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

1.11 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).

1.12 AREVA NP Document 51-5040700-00, "AIS for SG Tube Loads," April 2006

1.13 AREVA NP Document 32-5054876-01, "GOTHIC Containment Analysis for BAW-2374," May 2006.

1.14 AREVA NP Document 32-9011327-000, "BAW-2374 HL SBLOCA Analysis," may 2006.

1.15 AREVA NP Document 86-9019830-001, "Summary of Results for Plant Analyses to Support BAW-2374," July 2006.

1.16 AREVA NP Document 51-9016313-001, "B&W Plant Control Rod Temperature Operability Evaluation," March 2006.

1.17 Letter NRC:06:022 dated April 13, 2006 from R. L. Gardner (AREVA) to Document Control Desk (US NRC), Subject: Interim Report of an Evaluation of a Deviation Pursuant to 10 CFR 21.21(a)(2).

1.18 Letter OG-07-15 dated January 9, 2007 from F. P. Schiffley (PWR Owners Group) to Document Control Desk (US NRC), Subject: PWR Owners Group Assessment of Post LOCA Control Rod Survivability, PA-ASC-0313.

1.19 Letter NRC:07:003 dated January 12, 2007 from R. L. Gardner (AREVA) to Document Control Desk (US NRC), Subject: Status of the Evaluation of a Deviation Pursuant to 10 CFR 21.21(a)(2).

1.20 AREVA NP Document 51-9013299-000, "Boron Precipitation Analysis Review for B&W Plants," June 2006.

1.21 AREVA NP Proprietary Topical Report BAW-10164P, Rev. 6, "RELAP/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," January 2006.

1.22 Letter NRC:06:017 dated March 31, 2006 from R. L. Gardner (AREVA) to Document Control Desk (US NRC), Subject: Request for Review and Approval of BAW-10164(P), Revision 6, "RELAP/MOD2 – B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

1.23 AREVA NP Proprietary Topical Report BAW-10164P-A, Rev. 3, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," July 1996.

1.24 AREVA NP Proprietary Topical BAW-10166P-A, Rev. 4, "BEACH – A Computer Program for Reflood Heat Transfer During LOCA," February 1996.

1.25 AREVA NP Document 47-5022659-00, "Draft 1994 NRC Annual Report," 12/3/2002.

1.26 AREVA NP Document 47-5007106-00, "1999 DRAFT ECCS Annual Letter," 2/16/00.

1.27 AREVA NP Document 47-5011843-00, "2000 Draft ECCS Annual Letter," 4/19/01.

1.28 Letter dated April 10, 2003 from H. N. Berkow USNRC to J. Mallay (AREVA), Subject: Evaluation of Framatome ANP Preliminary Safety Concern (PSC) 2-00 Relating to Core Flood Line Break and Operator Action Time (TAC No. MA9973), Project No. 728, ADAMS Accession Number ML031010143.

1.29 Letter dated January 10, 2005 from R. A. Gramm USNRC to J. S. Holm (AREVA), Subject: Request for Amendment of Safety Evaluation for "Report of Preliminary Safety Concern (PSC) 2-00 Related to Core Flood Line Break with 2-Minute Operator Action Time" (TAC No. MA9973), Project No. 693, ADAMS Accession Number ML043550355.

1.30 AREVA NP Document 47-5026040-00, "2002 Draft Annual BWOG 50.46 Letter," 5/14/03.

1.31 AREVA NP Document 47-5039949-00, "2003 Draft Annual BWOG 50.46 Letter," 3/30/04.

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

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

5.0 ANO – 1 Specific Information

In addition to the information identified above, the following licensing analyses or evaluations were performed in 2006 and are applicable only to ANO-1. Limiting ECCS analyses are tracked for the Mark-B9 and Mark-B-HTP fuel rod designs for the RELAP5-based EM. Each activity and the effect on the limiting ECCS analyses are summarized in Table 2.

5.1 Cycle 21 Reload Licensing

Cycle-specific licensing documentation was prepared for ANO-1 Cycle 21. An evaluation of LHR limit applicability to the final plant and fuel design was performed in Reference 2.1. The evaluation concluded that the Mark-B-HTP LOCA models were bounding and applicable to Cycle 21.

The following additional documents were prepared in support of Cycle 21 operation.

Reference 2.2, Input to the Reload Report

Reference 2.3, Input to the Reload Technical Document (RTD)

- Other – Cycle-specific LHR limits reported.
- Evaluations were performed for Mark-B-HTP fuel which determined that the previously reported PCTs remain bounding. No evaluations for the Mark-B9 fuel were performed to support Cycle 21 reload licensing.

5.2 Debris Evaluation of Fuel Assembly (GSI 191)

The evaluation in Reference 2.4 estimated the effect on core cooling from debris that may enter the reactor coolant system from containment when the ECCS suction is switched to the containment sump. In particular, an assessment of potential core blockage was performed that considered all RCS break locations, the potential for particulates to be transported to the core inlet, and the effect of fibrious debris and Calcium Silicate. The document provides the inputs required for the subsequent analysis. The document was revised subsequent to its release to update the particle lift calculation guidance.

- Other – Study on coolability of the fuel assembly
- This study does not deal with PCT effects.

5.3 Energy Deposition Factor (EDF) Disposition

The effect of the correction in the gamma heating fractions on the current LOCA analyses of record at ANO-1 was evaluated in Reference 2.5. The evaluation considered the Mark-B9 and Mark-B-HTP analyses and determined that a PCT penalty for the end-of-life (EOL) Mark-B9 LBLOCA cases was necessary to accommodate the penalty associated with the correction of the gamma heating fractions. However, the effect on the reported licensing PCT for the Mark-

B9 fuel was +0 °F since the limiting LBLOCA PCT was observed at beginning-of-life (BOL) conditions and was not affected. Mark-B-HTP LBLOCA analyses and SBLOCA analyses were determined to be unaffected.

- Application Error Correction – An error was discovered in the gamma energy fractions which were used to calculate the EDF for LOCA analyses.

- The effect on the reported licensing PCT for the Mark-B9 LBLOCA was +0 °F. The Mark-B-HTP LBLOCA and SBLOCA reported licensing PCTs were to be unaffected.

6.0 References

2.1 AREVA NP Document 51-9032903-000, "ANO-1 CY21 LHR Limits," October 2006.

2.2 AREVA NP Document 51-9037943-000, "ANO-1 CY21 Task 4 RR Input," December 2006.

2.3 AREVA NP Proprietary Document 51-9030944-000, "ANO-1 CY21 Task 4 RTD Input," November 2006.

2.4 AREVA NP Proprietary Document 51-5070055-01, "GSI-191 Downstream Effects Evaluation for ANO-1," April 2006.

2.5 AREVA NP Proprietary Document 51-9035953-000, "EDF Disposition for ANO-1," November 2006.

Table 2: 2006 LOCA Licensing Activity for ANO Unit 1

Plant Name:		Arkansas Nuclear One – Unit 1	***LOCA Spectrum***		
Utility Name:		Entergy	*Mark-B-HTP LBLOCA*	*Mark-B9 LBLOCA*	*SBLOCA*
Item #	***Reporting Category***	***Description***	***PCT or (PCT Change)***		
Licensing Basis at End of 2005			**2008.1 °F [1] Analyzed EM R0.9**	**2000 °F Estimate EM R0.3**	**1179.7 °F Analyzed EM R0.9**
2006 Licensing Activity					
5.1	Other	Cycle 21 Reload Licensing	Previous Bounding	N/A	Previous Bounding
5.2	Other	Debris Evaluation of Fuel Assembly (GSI 191)	N/A	N/A	N/A
5.3	Application Error Correction	EDF Disposition	N/A	(+0 °F)	N/A
Licensing Basis at End of 2006			**2008.1 °F [1] Analyzed EM R0.9**	**2000 °F Estimate EM R0.3**	**1179.7 °F Analyzed EM R0.9**

[1] The reported PCT of 2008.1 °F is based on a full-core configuration. The mixed–core PCT of 1981.4 °F may be reported until a full-core configuration is implemented.

Attachment 2 to

0CAN060703

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 models for 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

- ROD-TO-ROD RADIATION ENCLOSURE SELECTION PROCESS IMPROVEMENT FOR THE 1999 EM

  (Enhancements / Forward-Fit Discretionary Changes)

  Background

  The Appendix K ECCS Performance Analysis for LBLOCA for CE plants is performed with the 1999 EM. The hot rod heat-up portion of this analysis contains a component model for rod-to-rod radiation, which utilizes an enclosure of fuel rods. In the EM topical report, the rod-to-rod radiation methodology and a related Safety Evaluation Report (SER) limitation / constraint require that a bounding radiation enclosure be used in the analysis. Search criteria are specified in the NRC-accepted topical report for ensuring that these conditions are met. The process for identifying candidate limiting enclosures for the rod-to-rod radiation model includes the use of an automated survey of the core on a pin-by-pin basis. The REX Code is the utility code that executes the surveying process for identifying potentially limiting radiation enclosures for evaluation in the LBLOCA Performance Analysis.

  In 2005, a problem developed with the REX Code, in that inappropriate radiation enclosures for the rod-to-rod radiation model were being identified. This had the potential for adding considerable inefficiency to the reload analysis process, since all identified candidates must be dispositioned for the analysis. This problem coincided with the introduction of ZrB_2 Integral Fuel Burnable Absorber (IFBA) bearing cores, which have flatter power distributions. It was found that some candidate enclosures contained target hot rods operating below the power of the average rod of the hot

assembly. This result produced candidate enclosures that fall outside the range of applicability of the rod-to-rod radiation methodology and, therefore, are inappropriate for the analysis. The REX utility code was modified to eliminate inappropriate enclosures derived from the survey process. This modification has no impact on the final limiting enclosure used in determining peak clad temperature (PCT).

Estimated Effect

This process improvement has no impact on the licensed methodology or on the NRC-accepted search criteria and does not conflict with the SER limitation / constraint imposed on the radiation model. There is no impact on PCT.

2.2 Appendix K Small Break – S2M Related Items

There are no issues to report for 2006.

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 2006 that have an impact on the Arkansas Nuclear One – Unit 2 (ANO-2) LBLOCA, SBLOCA and LTC Analyses of Record which support the operation for the calendar year 2006 reporting period.

4.0 ANO-2 Specific Information

There is no plant specific LBLOCA or SBLOCA effect due to the modifications performed to the REX code.

A complete reanalysis of both LBLOCA and SBLOCA ECCS performance was revised in 2006 for ANO-2 in order to implement ZIRLO clad, ZrB_2 IFBA and UO_2 fuel rod designs with 2 x 8 inch low-enriched axial blankets. The 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. There is no plant specific LBLOCA or SBLOCA effect due to the reanalysis performed in 2006.