March 26, 2008

Vice President, Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION 6.6.5, "CORE OPERATING LIMITS REPORT (COLR)" (TAC NOS. MD6220 AND MD6268)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 276 to Renewed Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2 (ANO-2). The amendment consists of changes to the Technical Specifications (TS) in response to your application dated July 31, 2007, as supplemented by letters dated July 31, 2007, and March 11, 2008.

The amendment modifies TS 6.6.5, "Core Operating Limits Report (COLR)," which would add new analytical methods to support the implementation of Next Generation Fuel (NGF). The licensee also provided the revised Emergency Core Cooling performance re-analyses in support of the implementation of Combustion Engineering (CE) 16x16 NGF as described in WCAP-16500-P-A, "CE 16x16 Next Generation Fuel Core Reference Report." In addition, the NRC staff approves a one-time application of a 3.5 percent partial credit for NGF thermal margin gain to be applied to both the Core Operating Limit Supervisory System and Core Protection Calculator System departure from nucleate boiling calculations (in combination with the 2 percent wide-range penalty) for ANO-2 Cycle 20.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Alan B. Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures: 1. Amendment No. 276 to NPF-6
2. Safety Evaluation

cc w/encls: See next page

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Arkansas Nuclear One
Entergy Operations, Inc.
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Alan B. Wang, Project Manager
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Division of Operating Reactor Licensing
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2. Safety Evaluation

cc w/encls: See next page

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GHill, OIS (2)

(**) See previous concurrence

ADAMS Accession Nos.: Pkg ML080840014, Amdt. ML080840015, License/TS Pgs ML080840023) (*) SE input memo

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	DSS/SNPB/BC	DSS/SRXB/BC	OGC – NLO w/comments	NRR/LPL4/BC
NAME	AWang (**)	JBurkhardt (**)	AMendiola (*)	GCranston (*)	MSmith (**)	THiltz
DATE	3/25/08	3/25/08	3/6/08	3/26/08	3/26/08	3/26/08

OFFICIAL RECORD COPY

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ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 276
Renewed License No. NPF-6

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated July 31, 2007, as supplemented by letters dated July 31, 2007, and March 11, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-6 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 276, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications

3. The license amendment is effective as of its date of issuance and shall be implemented prior to startup following the spring 2008 refueling outage. Further, Facility Operating License No. NPF-6 is hereby amended to authorize a change to the Final Safety Analysis Report (FSAR) to reflect the revised loss-of-coolant accident analyses. The FSAR changes constitute a change in the analysis of record and will be a baseline for which future changes will be measured against in accordance with 10 CFR 50.46(a)(3). This action is required for the implementation of Next Generation Fuel as set forth in the license amendment application dated July 31, 2007, and evaluated in the safety evaluation dated March 26, 2008. The licensee shall update the FSAR by adding a description of this change, as authorized by this amendment, and in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas G. Hiltz, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Renewed Facility
Operating License No. NPF-6
Technical Specifications

Date of Issuance: March 26, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 276

RENEWED FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the Renewed Facility Operating License No. NPF-6 and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Operating License

REMOVE	INSERT
-3-	-3-

Technical Specifications

REMOVE	INSERT
6-19	6-19
6-20	6-20
6-21	6-21

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 276 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By application dated July 31, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072200258) (Reference 1), as supplemented by letters dated July 31, 2007, and March 11, 2008 (ADAMS Accession Nos. ML072200528 and ML080710408, respectively) (References 4 and 3), Entergy Operations, Inc. (Entergy, the licensee), requested changes to the Technical Specifications (TS) for Arkansas Nuclear One, Unit No. 2 (ANO-2). The supplemental letters dated July 31, 2007, and March 11, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 28, 2007 (72 FR 49576).

The proposed changes would revise TS 6.6.5, "Core Operating Limits Report (COLR)," which would add new analytical methods to support the implementation of Next Generation Fuel (NGF). The Combustion Engineering (CE) 16x16 NGF design utilizes Optimized ZIRLO™, an advanced cladding alloy. The emergency core cooling system (ECCS) performance analysis computer codes have been updated to included the Optimized ZIRLO™ cladding property changes.

2.0 REGULATORY EVALUATION

The license amendment request involves adding new analytical methods to TS 6.6.5. As required by paragraph 50.46(a)(1)(i) of Title 10 of the *Code of Federal Regulations* (10 CFR), the ECCS performance analysis must conform to the ECCS acceptance criteria identified in 10 CFR 50.46(b) which are: Criterion 1, Peak Cladding Temperature - ≤ 2200 °F; Criterion 2, Maximum Cladding Oxidation - ≤ 0.17 times the total cladding thickness before oxidation; Criterion 3, Maximum Hydrogen Generation - ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; Criterion 4, Coolable Geometry - the core remains amenable to cooling even calculated changes in core geometry; and Criterion 5, Long-Term Cooling - to maintain an acceptably low calculated core temperature after any calculated successful initial operation of the ECCS and to remove decay heat for an extended period of time required by the long-lived radioactivity remaining in the core. Additionally, the ECCS performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents (LOCAs) of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. The evaluation model may be either a realistic evaluation model as described in 10 CFR 50.46(a)(1)(i) or must conform to the required and acceptable features of Appendix K ECCS Evaluation Models.

Also, regulatory guidance for the review of fuel rod cladding materials and fuel system designs and adherence to applicable General Design Criteria (GDC) is provided in NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 4.2, “Fuel System Design.” In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences,

- Fuel system damage is never so severe as to prevent control rod insertion when it is required,

- The number of fuel rod failures is not underestimated for postulated accidents, and

- Coolability is always maintained.

In addition to licensed reload methodologies, an approved mechanical design methodology is utilized to demonstrate compliance with SRP 4.2 fuel design criteria. The NRC staff has previously reviewed and approved the CE 16x16 NGF assembly design for application in CE plant designs (Reference 2).

In Section 50.36, “Technical specifications,” of Title 10 of the *Code of Federal Regulations* (10 CFR), the Commission established its regulatory requirements related to the content of TS. Pursuant to 10 CFR 50.36(d), TS are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The license amendment request involves adding new analytical methods to TS 6.6.5. The proposed TS changes will be evaluated to ensure continued compliance with requirements of 10 CFR 50.36(d). Compliance with this regulation requires a licensee to maintain a list of approved analytical methods (used to establish potentially cycle-specific core operating limits, per NRC Generic Letter (GL) 88-16). The NRC staff’s review will verify that the new analytical methods are applicable to the licensee and will be used in accordance with established conditions and limitations. This amendment addresses the requirements for safety limits.

3.0 TECHNICAL EVALUATION

3.1 Proposed Change to Technical Specification 6.6.5

The proposed change to TS 6.6.5, "Core Operating Limits Report (COLR)," involves adding new analytical methods which will be used to determine the core operating limits related to the implementation of NGF. The proposed changes are provided in Attachment 2 of Reference 1 with the justification provided in Attachment 1 of Reference 1. The new analytical methods being added to the COLR, listed below, have been previously reviewed and approved by the NRC:

- WCAP-16500-P-A, CE 16x16 Next Generation Fuel Core Reference Report

- Addendum 1-P-A to WCAP-12610-P-A and CENPD-404-P-A, Optimized ZIRLO™

- WCAP-16523-P-A, Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux [CHF] in Rod Bundles with Side-Supported Mixing Vanes

- CENPD-387-P-A, ABB Critical Heat Flux Correlation for PWR [Pressurized-Water Reactor] Fuel

- Addendum 1-P-A to CENPD-132 Supplement 4-P-A, Calculative Methods for the CE Nuclear Power Large Break LOCA [Loss-of-Coolant Accident] Evaluation Model – Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood

Attachment 1 of Reference 1 identifies and discusses all of the safety evaluation (SE) conditions and limitations within each of the licensed topical reports (LTRs) being added to COLR. Part of the disposition includes regulatory commitments stated in Section 4.0 (Attachment 3 of Reference 1). The NRC staff reviewed the disposition of each SE conditions and limitations and found, with the exception of WCAP-16500-P-A SE conditions and limitations #5, #6, and #7, that the licensee adequately addressed each one of them.

3.2 WCAP-16500-P-A SE Conditions and Limitations #5 and #6

WCAP-16500-P-A SE conditions and limitations #5 and #6 deal with the Core Operating Limit Supervisory System (COLSS) and Core Protection Calculator System (CPCS) setpoint methodology and the effects of a mixed core of NGF and non-NGF assemblies. Within condition and limitation #6, the licensee states that NGF's improved thermal performance offsets any mixed core effects. However, in condition and limitation #5, the licensee states that a portion of this available thermal margin will be credited by reducing the COLSS and CPCS addressable constants. Any credit for the NGF assembly thermal characteristics (e.g., mixing vane CHF correlations) in a mixed core configuration may represent a deviation from approved methodologies. Sufficient information related to the implementation of this partial credit is not presented within this amendment request for the staff to determine whether the licensee complies with approved methods.

In order to resolve these issues, the NRC staff conducted an audit of the Westinghouse engineering calculations supporting the ANO-2 Cycle 20. This audit was conducted on February 28, 2008, at the Westinghouse/CE offices in Windsor, Connecticut. During the audit, the NRC staff reviewed the ANO-2 Cycle 20 Core Thermal Hydraulic Reload Analysis (CN-ANO220-010) and the NGF DNB [Departure from Nucleate Boiling] Partial Credit Analysis (CN-TAS-08-6). In addition, the Westinghouse staff presented the methodology used in the development of a partial thermal margin credit. Based upon the audit, the NRC staff finds that the partial credit methodology deviates from both the original Modified Statistical Combination of Uncertainties (MSCU) methodology (CEN-356(V)-P-A) cited in ANO-2 COLR (Reference 3 in TS 6.6.5) and the NGF setpoints methodology (WCAP-16500-P-A) cited in the ANO-2 COLR (Reference 11 in TS 6.6.5).

Recognizing the NGF mixed-core partial credit as a deviation from previously approved methods, the NRC staff expanded the scope of the audit to review and approve a one-time methodology deviation for ANO-2 Cycle 20. Using the approved TORC model, the partial credit is based on the ratio of iterated heat flux at any given core location over a wide range of operating conditions.

DNB Adjustment = [(Minimum Ratio – 1.0 – 0.06)/2 + 1.0]

- Minimum Ratio = [$(\text{Heat Flux})_{\text{Iterated – NGF}}$ / $(\text{Heat Flux})_{\text{Iterated – CE1}}$]
- 5 limiting assemblies
- 38 axial shapes
- Rodded and unrodded core operating conditions
- Nominal and off nominal conditions (e.g., Press., Temp., Flow)
- Heat flux iterated to 1.25, 1.0, and 0.8 DNBR [DNB ratio]

The wide range of operating conditions considered in the TORC cases ensures a minimal NGF thermal margin credit. During the audit, the NRC staff questioned the selection of limiting assemblies. The five limiting assemblies identified in the Core Thermal-Hydraulics analysis were all first cycle NGF bundles. The NRC staff had a concern with the application of an NGF partial credit when the limiting assembly in the core was a non-NGF design. Even though the current Core Thermal-Hydraulics limiting assembly selection process is not geared toward minimizing NGF/CE-1 differences, Westinghouse was convinced that the limiting assembly would be an NGF design. Examination of the Cycle 20 loading patterns revealed adjacent first cycle NGF bundles located within low-flow core locations. The higher bundle power, flat radial power profile, and low-inlet flow combine to yield limiting thermal margin locations. Hence, the NRC staff concluded it is reasonable to accept Westinghouse's assertion regarding the limiting assemblies.

Furthermore, the calculated minimum ratio using the partial credit methodology was similar in magnitude to NGF thermal margin gains gathered from the ANO-2 Cycle 20 CETOP/TORC multipliers (NGF mixed versus CE-1) and in the sample 1/64 hypercube setpoints calculation within WCAP-16500-P-A (Enclosure 3). This is further proof that the calculated minimum ratio is reasonable.

The above DNB adjustment equation includes the 6 percent interim margin penalty (WCAP-16500-P-A SE condition #5). Although this margin penalty is not directly applicable to the partial credit methodology (6 percent based upon the 1/64 hypercube methods), dividing the net margin gain by a factor of 2.0 yields a conservative partial credit.

For ANO-2 Cycle 20, the minimum heat flux ratios and resulting DNB adjustments are listed below:

Minimum Ratio (Narrow Range Axial Shape Index (ASI)) = 1.131
Minimum Ratio (Wide Range ASI) = 1.095

DNB Adjustment (Narrow Range ASI) = 1.035
DNB Adjustment (Wide Range ASI) = 1.017

The CPCS algorithms allow for an independent wide-range ASI penalty factor. The current ANO-2 Reload Data Block constants include a 2 percent wide-range penalty (beyond $\pm$ 0.30 ASI). By crediting the wide-range penalty factors (which are not being credited for any other purpose), a 3.5 percent partial NGF thermal margin gain may be applied to both the COLSS and CPCS DNB calculations. Although not widely used, credit for the CPCS wide-range ASI penalty factors is not a methodology change.

In addition to core thermal-hydraulics and COLSS/CPCS setpoints, the NRC staff's audit also investigated the impact of the mixed core on transient analyses. Westinghouse stated that all transient thermal margin requirements and accident analyses were performed assuming an entire core of non-NGF assemblies. CETOP-D and TORC calculations were performed with the standard assembly dimensions and CE-1 DNB statistics. No credit was taken for the improved thermal performance of the NGF design.

Ignoring the mixed core may be conservative for many transient analyses (due to ROPM calculations and inherent thermal margin gains with NGF which offset flow starvation). However, transients dealing with absolute minimum DNBRs (for fuel failure calculations) or fuel temperature calculations may be adversely affected by specific fuel design characteristics. During the audit, the NRC staff questioned input and assumptions for the CEA [Control Element Assembly] Ejection Analysis. It was determined that the analysis correctly included both current and NGF fuel rod dimensions in the fuel enthalpy calculations. In addition, fuel failure calculations employed inputs which bounded both current and NGF designs.

COLSS and CPCS addressable constants will be calculated following the approved MSCU methodology and standard process. CETOP-D calculations were performed with the standard assembly dimensions and CE-1 DNB statistics. CETOP-to-TORC multipliers were based on a full core of standard assemblies. Within the statistical analyses, no credit was taken for the improved thermal performance of the NGF design. The partial credit DNB adjustment (discussed above) will be applied directly to the final BERR1 and EPOL2/4 addressable constants.

Based upon the information reviewed during the audit, the NRC staff finds the one-time application of the 3.5 percent partial credit (in combination with the 2 percent wide-range penalty) acceptable for ANO-2 Cycle 20.

3.3 WCAP-16500-P-A SE Condition and Limitation #7

WCAP-16500-P-A SE condition and limitation #7 states that "[i]mplementation of CE 16x16 NGF assemblies necessitate re-analysis of the plant-specific LOCA [Loss of Coolant Accident] analyses. Licensees are required to submit a license amendment containing the revised LOCA analyses for NRC review. Upon approval, the revised LOCA analyses constitute the analysis of-record and baseline for which future changes will be measured against in accordance with 10 CFR 50.46(a)(3)." The licensee provided this reanalysis by letter dated July 31, 2007 (Reference 4). The NRC staff review of this reanalysis is provided below.

3.3.1 Large Break LOCA (LBLOCA)

The Westinghouse ECCS Performance Appendix K Evaluation Model for CE plants is the 1999 Evaluation Model (1999 EM) for LBLOCA. The 1999 EM for LBLOCA is augmented by CENPD-404-P-A for analysis of ZIRLO™ cladding and by Addendum 1 to CENPD-404-P-A for analysis of Optimized ZIRLO™ cladding. Also, the 1999 EM is supplemented by WCAP-16072-P-A for implementation of ZrB_2 integral fuel burnable absorber (IFBA) fuel assembly designs.

The 1999EM for LBLOCA includes the following computer codes. The CEFLASH-4A computer code is used to perform the blowdown hydraulic analysis of the reactor coolant system (RCS) and the COMPERC-II computer code is used to perform the RCS refill/reflood hydraulic analysis and to calculate the containment minimum pressure. It is also used in conjunction with the methodology described in CENPD-213-P to calculate the FLECHT-based reflood heat transfer coefficients used in the hot-rod heatup analysis. The HCROSS and PARCH computer codes are used to calculate steam cooling heat transfer coefficients. The STRIKIN-II computer code is used for the hot-rod heatup analysis to calculate the peak cladding temperature and maximum cladding oxidation. Core-wide cladding oxidation is calculated using the COMZIRC computer code. The initial steady-state fuel rod conditions used in the analysis are determined using the FATES3B computer code.

The Appendix K steam cooling heat transfer component model for less than 1 inch per second core reflood in the 1999 EM has been modified to include spacer grid-heat transfer effects. For ANO-2, the LBLOCA analysis does not credit the use of the modified model including spacer grid-heat transfer effects.

In performing the LBLOCA calculations, conservative assumptions are made concerning the availability of safety injection flow. It is assumed that offsite power is lost and all pumps must await diesel startup before they can begin to deliver flow. Also, it is assumed that all safety injection flow delivered to the broken cold leg is lost directly to the containment.

Entergy performed a study to determine the most limiting single failure of ECCS equipment. The study analyzed no failure, failure of an emergency diesel generator, failure of a high-pressure safety injection (HPSI) pump, and a failure of a low-pressure safety injection (LPSI) pump consistent with approved topical reports. Maximum safety injection pump flow rates were used in the no failure case and minimum safety injection pump flow rates were used in the emergency diesel generator, HPSI ,or LPSI pump failure cases. The pumps were actuated on a safety injection actuation signal (SIAS) generated by low-pressurizer pressure with appropriate startup delay. Minimum refueling water storage pool temperature was used in all four cases as a result of a sensitivity study of the refueling water storage pool water temperature. The study also investigated the impact of variation in safety injection tank (SIT) pressure, water temperature and water volume on peak cladding temperature, and peak local cladding oxidation. A spectrum of guillotine breaks in the reactor coolant pump discharge leg was analyzed and the results show that the discharge leg is the most limiting break location and a guillotine break is more limiting than a slot break.

Important core, RCS, ECCS, and containment design data used in the LBLOCA analysis are listed in Tables 5-1 and 5-2 of Reference 1. The listed fuel rod conditions are for rod average burnup of the hot rod that produced the highest calculated peak cladding temperature. Table 5-3 lists the peak cladding temperature and oxidation percentage for the spectrum of LBLOCAs and times of interest are listed in Table 5-4 of Reference 1. The results of the full-core implementation of NGF demonstrate conformance to the ECCS acceptance criteria as stated in Section 2.0 of this evaluation. These results support a peak linear generation rate of 13.7 kiloWatts per foot.

The NRC staff has reviewed the assumptions, plant design data, and the results of the revised ECCS performance analysis provided by Entergy and found them acceptable because conservative assumptions are used and the results (Reference 5) meet the ECCS acceptance criteria of 10 CFR 50.46(b).

3.3.2 Small Break LOCA (SBLOCA)

The SBLOCA ECCS performance analysis used the Supplement 2 version (referred to as S2M or Supplement 2 Model of CENPD-137, Supplement 2-P-A, “Calculative Methods for the ABB CE Small Break LOCA Evaluation Model”). The S2M for SBLOCA is augmented by CENPD-404-P-A for analysis of ZIRLO™ cladding, and by Addendum 1 to CENPD-404-P-A for analysis of Optimized ZIRLO™ cladding. Also, S2M is supplemented by WCAP-16072-P-A for implementation of ZrB_2 IFBA fuel assembly designs.

The S2M for SBLOCA uses the following computer codes. The CEFLASH-4AS computer program is used to perform the hydraulic analysis of the RCS until the time the SITs begin to inject. After injection from the SITs begins, the COMPERC-II computer program is used to perform the hydraulic analysis. COMPERC-II is only used in the SBLOCA evaluation model for large-break sizes that exhibit prolonged period of SIT flow and significant core voiding. The hot-rod cladding temperature and maximum cladding oxidation are calculated by the STRIKIN-II computer program during the initial period of forced convection heat transfer and by the PARCH computer program during the subsequent period of pool boiling heat transfer. Core-wide cladding oxidation is conservatively represented as the rod-average cladding oxidation of the hot rod. The initial steady-state fuel rod conditions used in the analysis are determined using the FATES3B computer program.

The SBLOCA analysis was performed for the fuel rod conditions that result in the maximum initial stored energy in the fuel. The calculations included the analysis of both UO_2 and ZrB_2 burnable absorber fuel rods in both the NGF and standard fuel rod designs. For ANO-2, the analysis was performed using the failure of an emergency diesel generator as the most limiting single failure of the ECCS. The emergency diesel generator failure causes the loss of a HPSI pump and LPSI pump, and results in a minimum of safety injection water being available to cool the core. The LPSI pumps are not explicitly credited in the SBLOCA analysis since the RCS pressure never decreases below the LPSI pump shutoff head during the portion of the transient that is analyzed.

A spectrum of three break sizes in the reactor coolant pump discharge (PD) leg was analyzed to bracket the limiting break size, which for ANO-2 was 0.04 square feet per PD break. The reactor coolant PD leg is the limiting break location because it maximizes the amount of spillage from the ECCS. The limiting SBLOCA is the largest small break for which the hot-rod cladding heatup transient is terminated solely by injection from a HPSI pump.

Important core, RCS, and ECCS design data used in the SBLOCA analysis are listed in Tables 5-7 and 5-8 of Reference 1. Table 5-9 lists the peak cladding temperature and oxidation percentages for the spectrum of SBLOCAs and times of interest are listed in Table 5-20 of Reference 1. The results for the 0.04 square feet per PD break, the limiting SBLOCA, demonstrate conformance to the ECCS acceptance criteria.

The NRC staff has reviewed the assumption, plant design data and the results of the analysis provided by Entergy and found them acceptable because the assumption is conservative and the results (Reference 5) meet the ECCS acceptance criteria of 10 CFR 50.46(b). The NRC staff also agrees with the licensee's conclusion that no SBLOCA mixed-core analysis is necessary during transition core cycles due to the negligible effect of variation in core hydraulic losses on SBLOCA analysis results.

3.3.3 Post-LOCA Long-Term Cooling

Entergy stated that the analyses performed with the Westinghouse post-LOCA long-term cooling evaluation model for CE plants (CENPD-254-P-A) are not sensitive to the fuel assembly changes being introduced for the CE 16x16 NGF design. The NRC staff agrees with the licensee's conclusion that no plant-specific post-LOCA long-term cooling analyses were required to support the introduction of the CE 16x16 NGF assembly based on the result of the analyses performed.

3.3.4 Transition Mixed Core

A transition mixed core assessment was performed for NGF in order to address the impact of co-resident hydraulically dissimilar fuel assemblies on ECCS performance. The NGF core hydraulic resistance is greater than the standard fuel assembly due to the addition of mixing grids. Therefore, adjacent NGF and standard assemblies will experience a net redistribution of flow from the higher resistant NGF assembly to the lower resistant standard assembly.

The flow redistribution in the NGF mixed transition cores produces a slight penalty on the NGF assembly ECCS performance during the LBLOCA. However, a smaller cross-sectional core area for coolant flow (relative to a full core of NGF assemblies) is credited in the transition core assessment to improve the core hydraulics behavior during the blowdown period. Also, the smaller cross-sectional core area increases the core reflooding rates during the reflood period relative to the bounding full core of NGF analysis. The net impact on ECCS performance is a slight reduction in the peak cladding temperature, peak cladding oxidation, and core-wide cladding oxidation percentage.

For ANO-2, one mixed core configuration was examined to address core loading differences that are expected in the coming cycles of operation assuming a half core loading pattern for NGF assemblies. The transition mixed core ECCS performance assessment indicted that the results were bounded by the results of the full core NGF implementation analysis.

The NRC staff has reviewed Entergy's description for the transition mixed core assessment and found it acceptable because of the lower impact on ECCS performance.

3.4 WCAP-16500

The NRC staff notes that for any particular cycle-specific core operating limit there are many approved analytical methods which can be used. According to GL 88-16 guidance, TS 6.6.5.b should list the main approved methods used to support the cycle-specific core operating limit. Therefore, the NRC staff requested the licensee to identify the main methods to reflect the GL 88-16 guidance to minimize the number of the approved methods entitled to be listed in TS 6.6.5.b. In response to the NRC staff request, the licensee committed (Reference 3) to submit an amendment to minimize the number of references consistent with the guidance specified in GL 88-16 within 12 months following NRC issuance of the approved amendment for the current requested changes to TS 6.6.5. This commitment is acceptable because the TS 6.6.5.b will only list the main methods supporting plant- and cycle-specific operating limits listed in TS 6.6.5.a. In addition, the NRC staff recommends that the licensee maintain within the COLR only the current methods being used to determine core operating limits. Superseded methods should be removed from TS 6.6.5. For example, the COLR lists two NSSS simulation codes, CESEC-III and CENTS. If the CENTS code has replaced CESEC-III for the purpose of determining core operating limits, then Reference #6 should be deleted.

The NRC staff has reviewed the request by Entergy to approve the revised ECCS analysis to support the license amendment for the implementation of CE 16x16 Next Generation Fuel (NGF) described in WCAP-16500 and concludes that the revised ECCS analysis is acceptable and meets limitation and condition #7 because the analyses used approved methodologies and results meet the ECCS acceptance criteria of 10 CFR 50.46(b). The NRC staff verified that each of the LTRs being added to TS 6.6.5 are applicable to the CE-designed ANO-2 reactor core. Entergy procedure NF-105, “Reload Process,” Revision 3 (dated January 22, 2004) provides guidance with regard to development of cycle-specific groundrules. The groundrules document is one of the processes used to assure that LOCA analysis input values bound as-operated plant values (Reference 6). Based upon compliance with SE limitations and conditions, the groundrules document, and considering the regulatory commitments identified in Section 4.0, the NRC staff finds the proposed changes to TS 6.6.5 acceptable. The NRC staff may audit the licensee in the future to ensure that the licensee's commitments to meet the requirements specified in the conditions and limitations of each approved methodology is maintained.

In addition to the changes to TS 6.6.5, the licensee identified administrative changes involving relocating text within TS pages 6-19 through 6-21. The NRC staff agrees that these changes are administrative in nature and therefore are acceptable.

4.0 LIST OF REGULATORY COMMITMENTS

The licensee made the following list of regulatory commitments with respect to is its licensing amendment request. These commitments were identified in Attachment 3 to its application.

1. Additional growth data will be obtained from future LTA [lead test assembly] exams ahead of the exposure achieved by batch implementation. This data will be provided to the NRC as it becomes available. This is expected sometime after July 2009 after LTA programs have burnups that bound current ANO-2 burnup limits.

2. If required to maintain acceptable COLSS and CPC DNB operating margin throughout the transition cycle (Cycle 20), a portion of the potential DNB margin gain may be credited to reduce the DNB uncertainty addressable constants for COLSS and CPC. In this case even after applying a conservative 6% margin penalty, no more than one half of the net margin gain, will be credited to reduce the COLSS and CPC DNB uncertainty addressable constants.

3. The 6% interim margin penalty described in the Limitations and Conditions of the NRC Safety Evaluation of WCAP-16500-P-A will be applied to the resultant addressable constants until its removal has been approved by the NRC.

4. For the transition cycle, the COLSS on-line monitoring system and the CPC system will continue to utilize the current models and the CE-1 CHF correlation.

5. If the optional steam cooling model described in CENPD-1 32, Supplement 4-P-A, Addendum 1-P and Final Safety Evaluation were to be used for ANO-2 ECCS [Emergency Core Cooling System] Performance Analyses at some time in the future, then a license amendment request would be submitted including the analyses and comparison graphical results needed to confirm the acceptability of the use of the optional steam cooling model.

6. Entergy commits to evaluate other similar plants' TS methodology references that reflect NRC-approved methods used in establishing the COLR parameter limits. Based on this evaluation, Entergy will propose a change to TS 6.6.5 to

minimize the number of references consistent with the guidance provided in GL 88-16, “Removal of Cycle-Specific Parameter Limits from Technical Specifications.” This proposed TS change will be submitted within 12 months following NRC issuance of the approved amendment for the current requested changes to TS 6.6.5.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on August 28, 2007 (72 FR 49576). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Letter from Entergy to U.S. Nuclear Regulatory Commission, “License Amendment Request to Revise Technical Specification 6.6.5, Core Operating Limits Report,” Docket No. 50-368, 2CAN070701, July 31, 2007 (ADAMS Accession No. ML072200258).

2. WCAP-16500-NP-A, “CE 16x16 Next Generation Fuel Core Reference Report” (ADAMS Accession No. ML060670514).

3. Letter from Entergy to U.S. Nuclear Regulatory Commission, “Response to Request for Additional Information License Amendment Request to Revise Technical Specification 6.6.5, Core Operating Limits Report,” Docket No. 50-368, 2CAN030801, March 11, 2008 (ADAMS Accession No. ML080710408).

4. Letter from Entergy to U.S. Nuclear Regulatory Commission, “Emergency Core Cooling System Performance Analysis,” Docket No. 50-368, 2CAN070702, July 31, 2007 (ADAMS Accession No. ML072200528).

5. Letter (2CAN070702) from TGM to USNRC, “Emergency Core Cooling System Performance Analysis,” Arkansas Nuclear One, Unit 2, Docket No. 50-368, License No. NPF-6, July 31, 2007.

5. Letter (2CAN030804) from DEJ to USNRC, ‘Response to Request for Additional Information, Emergency Core Cooling System Performance Analysis,” Arkansas Nuclear One, Unit 2, Docket No. 50-368, License No. NPF-6, March 20, 2008.

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Date: March 26, 2008