April 2, 2002

Mr. Craig G. Anderson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 2 - ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT RELATED TO EXEMPTION FROM THE REQUIREMENTS OF 10 CFR PART 50, APPENDIX G (TAC NO. MB3301)

Dear Mr. Anderson:

Enclosed is a copy of the Environmental Assessment and Finding of No Significant Impact related to your application for an exemption from the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.60, and 10 CFR Part 50, Appendix G, for Arkansas Nuclear One, Unit 2 (ANO-2). The proposed exemption would allow application of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Case N-641 in establishing the reactor vessel pressure limits at low temperatures for ANO-2. This action is in response to your letter dated October 30, 2001, as supplemented by letters dated February 25 and March 13, 2002, that submitted new pressure temperature (P-T) limits, and low temperature overpressure protection (LTOP) system technical specification (TS) restrictions for ANO-2. The new P-T limits and the LTOP TS restrictions were developed using the methodologies in ASME Code Case N-641.

The assessment is being forwarded to the Office of the Federal Register for publication.

Sincerely,

/RA/

S. Patrick Sekerak, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosure: Environmental Assessment

cc w/encl: See next page

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Docket No. 50-368

Enclosure:
Environmental Assessment

cc w/encl: See next page

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UNITED STATES NUCLEAR REGULATORY COMMISSION

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

ENVIRONMENTAL ASSESSMENT AND FINDING OF

NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (NRC) is considering issuance of an exemption from Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.60 and 10 CFR Part 50, Appendix G, for Facility Operating License No. NPF-6, issued to Entergy Operations, Inc. (the licensee), for operation of the Arkansas Nuclear One, Unit 2 (ANO-2), nuclear power plant, located in Pope County, Arkansas. Therefore, as required by 10 CFR 51.21, the NRC is issuing this environmental assessment and finding of no significant impact.

ENVIRONMENTAL ASSESSMENT

Identification of the Proposed Action:

The proposed action would allow a one-time exemption from 10 CFR Part 50, Appendix G requirements that pressure-temperature (P-T) limits be established for reactor pressure vessels (RPVs) during normal operating and hydrostatic or leak testing conditions. Specifically, 10 CFR Part 50, Appendix G, states that “[t]he appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions.” Appendix G of 10 CFR Part 50 specifies that the requirements for these limits are contained in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Appendix G.

To address provisions of an amendment to the Technical Specifications (TSs) P-T limits and low-temperature overpressure protection (LTOP) system TS restrictions, the licensee requested in its submittal dated October 30, 2001, as supplemented by letters dated February 25 and March 13, 2002, that the NRC staff exempt the ANO-2 nuclear power plant from the requirements of 10 CFR Part 50, Appendix G. The exemption requested would allow the use of ASME Code Case N-641 in establishing the reactor vessel pressure limits at low temperatures.

Code Case N-641 permits the use of an alternate reference fracture toughness (K_{IC} fracture toughness curve instead of the K_{IA} fracture toughness curve) for reactor vessel materials in determining the P-T limits, LTOP system setpoints, and LTOP system effective temperature (also known as the LTOP system enable temperature, T_{enable}), and provides for plant-specific evaluation of T_{enable}. Since the K_{IC} fracture toughness curve shown in ASME Section XI, Appendix A, Figure A-2200-1 (the K_{IC} fracture toughness curve) provides greater allowable fracture toughness than the corresponding K_{IA} fracture toughness curve of ASME Section XI, Appendix G, Figure G-2210-1 (the K_{IA} fracture toughness curve), and a plant-specific evaluation of T_{enable} would give lower values of T_{enable} than use of a generic bounding evaluation for T_{enable}, use of Code Case N-641 for establishing the P-T limits, LTOP system setpoints, and T_{enable} would be less conservative than the methodology currently endorsed by 10 CFR Part 50, Appendix G. Although the use of the K_{IC} fracture toughness curve in ASME Code Case N-641 was recently incorporated into Appendix G to Section XI of the ASME Code, an exemption is still needed because 10 CFR Part 50, Appendix G requires a licensee's analysis to use an edition and addenda of Section XI of the ASME Code incorporated by reference into 10 CFR Part 50, Section 50.55a, i.e., the editions through 1995 and addenda through the 1996 addenda (which do not include the provisions of Code Case N-641). Therefore, an exemption to apply the Code case is required by 10 CFR Part 50, Section 50.60.

The proposed action is in accordance with the licensee's application for exemption dated October 30, 2001, as supplemented by letters dated February 25 and March 13, 2002.

The Need for the Proposed Action:

ASME Code Case N-641 is needed to revise the method used to determine the reactor coolant system (RCS) P-T limits, LTOP setpoints, and T_{enable}.

The purpose of 10 CFR Part 50, Section 50.60(a), and 10 CFR Part 50, Appendix G, is to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. This is accomplished through these regulations that, in part, specify fracture toughness requirements for ferritic materials of the RCPB. Pursuant to 10 CFR Part 50, Appendix G, it is required that P-T limits for the RCS be at least as conservative as those obtained by applying the methodology of the ASME Code, Section XI, Appendix G.

Current overpressure protection system (OPPS) setpoints produce operational constraints by limiting the P-T range available to the operator to heat up or cool down the plant. The operating window through which the operator heats up and cools down the RCS becomes more restrictive with continued reactor vessel service. Reducing this operating window could potentially have an adverse safety impact by increasing the possibility of inadvertent OPPS actuation due to pressure surges associated with normal plant evolutions, such as reactor coolant pump start and swapping operating charging pumps with the RCS in a water-solid condition. The impact on the P-T limits and OPPS setpoints has been evaluated for an increased service period for operation to 32 effective full-power years for ANO-2, based on ASME Code, Section XI, Appendix G requirements. The results indicate that these OPPS setpoints would significantly restrict the ability to perform plant heatup and cooldown, create an unnecessary burden to plant operations, and challenge control of plant evolutions required with OPPS enabled. Continued operation of ANO-2 with P-T curves developed to satisfy ASME Code, Section XI, Appendix G, requirements without the relief provided by ASME Code

Case N-641 would unnecessarily restrict the P-T operating window, especially at low temperature conditions.

Use of the K_{IC} curve in determining the lower bound fracture toughness of RPV steels is more technically correct than use of the K_{IA} curve, since the rate of loading during a heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The staff has required use of the conservatism of the K_{IA} curve since 1974, when the curve was adopted by the ASME Code. This conservatism was initially necessary due to the limited knowledge of the fracture toughness of RPV materials at that time. Since 1974, additional knowledge has been gained about RPV materials, which demonstrates that the lower bound on fracture toughness provided by the K_{IA} curve greatly exceeds the margin of safety required, and that the K_{IC} curve is sufficiently conservative to protect the public health and safety from potential RPV failure. Application of ASME Code Case N-641 will provide results that are sufficiently conservative to ensure the integrity of the RCPB, while providing P-T curves that are not overly restrictive. Implementation of the proposed P-T curves, as allowed by ASME Code Case N-641, does not significantly reduce the margin of safety.

In the associated exemption, the NRC staff has determined that, pursuant to 10 CFR Part 50, Section 50.12(a)(2)(ii), the underlying purpose of the regulation will continue to be served by the implementation of ASME Code Case N-641.

Environmental Impacts of the Proposed Action:

The NRC has completed its evaluation of the proposed action and concludes, as set forth below, that there are no significant environmental impacts associated with the use of the alternative analysis method to support the revision of the RCS P-T limits, LTOP setpoints and proposed T_{enable}.

The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released off site, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential nonradiological impacts, the proposed action does not involve any historic sites. It does not affect nonradiological plant effluents and has no other environmental impact. Therefore, there are no significant nonradiological environmental impacts associated with the proposed action.

Accordingly, the NRC concludes that there are no significant environmental impacts associated with the proposed action.

<u>Environmental Impacts of the Alternatives to the Proposed Action:</u>

As an alternative to the proposed action, the staff considered denial of the proposed action (i.e., the “no-action” alternative). Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

<u>Alternative Use of Resources:</u>

The action does not involve the use of any different resource than those previously considered in the Final Environmental Statement for the ANO-2 nuclear power plant, dated June 1977 (NUREG-0254).

<u>Agencies and Persons Consulted:</u>

On March 18, 2002, the staff consulted with the Arkansas State official, Mr. B. Bevill of the Division of Radiation Control and Emergency Management of the Arkansas Department of Health, regarding the environmental impact of the proposed action. The State official had no comments.

FINDING OF NO SIGNIFICANT IMPACT

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated October 30, 2001, as supplemented by letters dated February 25 and March 13, 2002. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1-800-397-4209 or 301-415-4737, or by e-mail to pdr@nrc.gov.

Dated at Rockville, Maryland, this 2nd day of April 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert A. Gramm, Chief, Section I
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Arkansas Nuclear One

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