May 17, 2004

Mr. Jeff Forbes
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE ARKANSAS NUCLEAR ONE, UNIT 2, LICENSE RENEWAL APPLICATION (TAC NO. MB8402)

Dear Mr. Forbes:

The U.S. Nuclear Regulatory Commission (NRC) is reviewing a license renewal application (LRA) submitted by Entergy Operations Inc., (Entergy or the applicant) dated October 14, 2003 for the renewal of the operating licenses for Arkansas Nuclear One, Unit 2, pursuant to Title 10 *Code of Federal Regulations* Part 54 (10 CFR Part 54). The NRC staff has identified, in the enclosure, areas where additional information is needed to complete the review. Specifically, the requests for additional information (RAIs) are from Section 4.3 Metal Fatigue, Section 4.5 Concrete Containment Tendon Prestress, Section 4.6 Containment Liner Plate and Penetration Fatigue Analyses, and Section 4.7.6 High Energy Line Break Analysis. These RAIs have been discussed with your staff.

Your responses to these RAI's are requested within 30 days from the date of this letter. If you have any questions, please contact me at (301) 415-1124 or e-mail gxs@nrc.gov.

Sincerely,

/RA/

Gregory F. Suber, Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No.: 50-368

Enclosure: As stated

cc w/encl: See next page

May 17, 2004

Mr. Jeff Forbes
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE ARKANSAS NUCLEAR ONE, UNIT 2, LICENSE RENEWAL APPLICATION (TAC NO. MB8402)

Dear Mr. Forbes:

The U.S. Nuclear Regulatory Commission (NRC) is reviewing a license renewal application (LRA) submitted by Entergy Operations Inc., (Entergy or the applicant) dated October 14, 2003 for the renewal of the operating licenses for Arkansas Nuclear One, Unit 2, pursuant to Title 10 *Code of Federal Regulations* Part 54 (10 CFR Part 54). The NRC staff has identified, in the enclosure, areas where additional information is needed to complete the review. Specifically, the requests for additional information (RAIs) are from Section 4.3 Metal Fatigue, Section 4.5 Concrete Containment Tendon Prestress, Section 4.6 Containment Liner Plate and Penetration Fatigue Analyses, and Section 4.7.6 High Energy Line Break Analysis. These RAIs have been discussed with your staff.

Your responses to these RAI's are requested within 30 days from the date of this letter. If you have any questions, please contact me at (301) 415-1124 or e-mail gxs@nrc.gov.

Sincerely,

/RA/

Gregory F. Suber, Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No.: 50-368

Enclosure: As stated

cc w/encl: See next page

DISTRIBUTION: See next page

Accession No: **ML041380284**

Document Name: C:\ORPCheckout\FileNET\ML041380284.wpd

OFFICE:	LA:RLEP	PM:RLEP	SC:RLEP
NAME:	M. Jenkins	G. Suber	S. Lee
DATE:	4/14/04	5/14/04	5/17/04

OFFICIAL RECORD COPY

REQUEST FOR ADDITIONAL INFORMATION
PERTAINING TO THE REVIEW OF THE ARKANSAS
NUCLEAR ONE - UNIT 2 LICENSE RENEWAL APPLICATION
SECTION 4.3 - METAL FATIGUE,
SECTION 4.5 - CONCRETE CONTAINMENT TENDON PRESTRESS,
SECTION 4.6 - CONTAINMENT LINER PLATE
AND PENETRATION FATIGUE ANALYSIS
SECTION 4.7.6 - HIGH ENERGY LINE BREAK
(TAC NO. MB8402)

4.3 Metal Fatigue

4.3.1 Class 1 Fatigue

RAI 4.3.1-1

Provide the edition of the ASME Section III Code that is applicable to the Class 1 fatigue analysis, and indicate if it was reconciled with the initial edition used for construction.

RAI 4.3.1-2

Table 4.3.1, "RCS Design Transients" lists the number of transient cycles logged as of July 11, 2002, for the listed transients. Indicate if the listed transients have been logged since the start of plant operation. If logging of transients was implemented after the start of plant operation, state the basis for estimating the number of cycles prior to the initiation of logging.

RAI 4.3.1-3

If additional transients have been logged since July 11, 2002, provide an updated Table 4.3.1 reflecting the most recently logged transients.

RAI 4.3.1-4

In Table 4.3.1, the number of logged reactor trips from 100% transients are listed as 77.14, and the logged loss of reactor coolant flow at 100% transients are listed as 2.91. Provide an explanation why these are not whole numbers.

4.3.2 Non-Class 1 Fatigue

RAI 4.3.2-1

Provide the ASME Section III Code edition that was used for Non-Class 1 fatigue analyses.

RAI 4.3.2-2

Provide the basis for the temperature screening criteria 220^{o} F for carbon steel and 270^{o} F for austenitic stainless steel in Section 4.3.2, "Non-Class 1 Fatigue."

RAI 4.3.2-3

The applicant states in Section 4.3.2 of the LRA:

"Only the RCS hot leg sampling piping may exceed 7,000 cycles during the period of extended operation. However, a calculation was revised to justify RCS sampling to occur at any reasonable frequency for 60 years of operation without exceeding the allowable number of cycles. Therefore, fatigue analyses for all non-Class 1 components at ANO-2 remain valid for the period of extended operation, in accordance with 10CFR54.21(c)(1)(i)."

10 CFR 54.21(c)(1) states: .....The applicant shall demonstrate that -
(i) The analyses remain valid for the period of extended operation;
(ii) The analyses have been projected to the end of the period of extended operation

From the description in the LRA paragraph above, if the analysis of the RCS hot leg sampling piping was revised and projected to the end of the period of extended operation, it should, therefore, fall in to the (ii) category. To verify whether it does, and to resolve the apparent ambiguity in the LRA paragraph, the staff requests that the applicant clarify this statement by providing the following:

1. The value of the highest expansion stress range Se, the allowable stress range Sa for the piping material, and the number of expected thermal cycles, in the CLB analysis.

2. The projected number of cycles (less than 20,000) for the period of extended operation, and the basis for this number of cycles.

3. The value of the highest expansion stress range Se and the allowable stress range Sa for the period of extended operation. If Se was calculated from a revised thermal expansion analysis, provide the basis for this analysis.

4.3.3.3 NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification

RAI 4.3.3.3-1

The NRC Safety Evaluation Report of CNP-387-P, included in CNP-387-P, Revision 1-P-A, recommends that licensees perform volumetric examination of the pressurizer elbow body and welds, as part of the ASME Section XI inservice inspection program. State whether volumetric examination of the pressurizer elbow is currently included in the ANO, Unit 2, ASME Section XI inservice inspection program, and whether this will be continued during the period of extended operation.

4.5 Concrete Containment Tendon Prestress

RAI 4.5-1

For the discussion of prestressing force losses over the initial 40-year, the LRA Section refers to SAR Section 3.8.1.3.4. This section of the SAR discusses the design approach used in designing the containment to satisfy the load combinations in SAR Section 3.8.1.3.3. There is no discussion of the estimation of projected prestressing forces after 40 years of operation. As the estimated prestressing forces at 40 years and 60 years depend upon the regression analysis of these time dependent attributes (i.e. creep of concrete and relaxation of prestressing steel), please provide the estimated values of these attributes which were used in arriving at the minimum required prestressing forces.

RAI 4.5-2

The use of 10 CFR 54.21(c)(1)(ii) and (iii) is appropriate for concrete containment tendon prestress TLAA. However, the staff needs to assess the plant specific operating experience regarding the residual prestressing forces in the containment. Based on the analysis performed per 10 CFR 54.21(c)(1)(ii), the applicant is requested to provide the following information:

(a) Minimum required prestressing forces for each group of tendons in terms of force per tendon.

(b) Trend lines of the projected prestressing forces for each group of tendons based on the regression analysis of the measured prestressing forces (see NRC Information Notice 99-10 for additional information).

(c) Plots showing comparisons of prestressing forces projected to 40 years and 60 years of operation, with the minimum required prestress for each group of tendons.

The staff requests that the comparison curves be constructed in force per tendon as opposed to force per wire or strand since the acceptance criteria in Subsection IWL of Section XI of the ASME Code uses these units. Furthermore, as stated in NRC Information Notice 99-10 the "Calculation of the average effective wire forces in the tendon from the measured tendon force is made only to ensure that (the measured lift-off force) does not exceed 70 percent of the guaranteed ultimate tensile strength of the wire."

RAI 4.5-3

In Section A.2.2.4 of the UFSAR Supplement to the LRA, the applicant summarizes the results of this TLAA, and states, "Calculation of the acceptability of the effective prestress of the containment building post-tensioning system at 60 years has been performed to show that the containment building tendon elements will be acceptable for the period of extended operation in accordance with 54.21(c)(1)(ii)." The staff is requesting that the applicant enter the target values in the FSAR Supplement. Previous applicants have provided these values in (1) tabular form, (2) descriptive form, or (3) as a amendment to the plant's Technical Specifications (TS) with a reference to the TS in their FSAR Supplement.

4.6 Containment Liner Plate and Penetration Fatigue Analyses

RAI 4.6-1

Provide the loading conditions and corresponding transient cycles used in the fatigue analysis of the containment liner plate and penetrations.

RAI 4.6-2

Provide the ASME Section III cumulative usage factors and locations from the recently completed containment liner plate and containment penetration fatigue analyses, showing that these fatigue TLAA will remain valid for the period of extended operation.

4.7.6 High Energy Line Break Analysis

RAI 4.7.6-1

Provide a discussion indicating that the surge line fatigue TLAA was reevaluated to determine if additional pipe breaks need to be postulated at locations where the ASME Code Section III CUF may exceed the pipe break postulation criterion for Class 1 piping of 0.1 during the period of extended operation.

DISTRIBUTION: Ltr: to: Mr. Jeff Forbes, Re: ANO-2, Dated: May 17, 2004
Accession no.: ML041380284

HARD COPY
RLEP RF
G. Suber (PM)
E-MAIL:
PUBLIC
J. Craig
D. Matthews
F. Gillespie
C. Grimes
RidsNrrDe
E. Imbro
G. Bagchi
K. Manoly
W. Bateman
J. Calvo
R. Jenkins
P. Shemanski
J. Fair
S. Black
B. Boger
D. Thatcher
R. Pettis
G. Galletti
C. Li
M. Itzkowitz
R. Weisman
M. Mayfield
A. Murphy
S. Smith (srs3)
S. Duraiswamy
Y. L. (Renee) Li
RLEP Staff

J. Minns
R. Gramm
A. Howell
T. Alexion
Linda Smith RIV

Arkansas Nuclear One, Unit 2

cc:

Executive Vice President
& Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Director, Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street, Slot 30
Little Rock, AR 72205-3867

Winston & Strawn
1400 L Street, N.W.
Washington, DC 20005-3502

Mr. Mike Schoppman
Framatome ANP, Richland, Inc.
Suite 705
1911 North Fort Myer Drive
Rosslyn, VA 22209

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 310
London, AR 72847

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

County Judge of Pope County
Pope County Courthouse
Russellville, AR 72801

Vice President, Operations Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, MS 39205

Garry Young
1448 SR 333
Russellville, AR 72802

Mr. Fred Emerson
Nuclear Energy Institute
1776 I St., N.W., Suite 400
Washington, DC 20006-3708