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February 7, 2002

Nuclear Regulatory Commission
Document Control Desk
Mail Station OP1-17
Washington, DC 20555

Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Comments Regarding the Draft NRC Safety Evaluation for the Proposed ANO-2 Power Uprate

Dear Sir or Madam:

By letter dated December 19, 2000 (2CAN120001), Entergy Operations, Inc. submitted an "Application for License Amendment to Increase Authorized Power Level." Supplemental letters were submitted based on discussions with the Nuclear Regulatory Commission staff during the course of the staff's review of the application. On January 18, 2002, the staff issued the draft safety evaluation (SE), "Arkansas Nuclear One, Unit 2 (ANO-2) - Draft Safety Evaluation Regarding the Proposed Extended Power Uprate (TAC MB0789)."

The staff requested comments by February 8, 2002, concerning the draft SE in regard to inclusion of proprietary information and factual accuracy. The draft SE contains no information considered to be proprietary. Regarding factual accuracy, Entergy is in general agreement with the wording and conclusions in the document but clarifications and corrections for some sections are provided in the attachments to this letter.

Entergy acknowledges the substantial effort required by the staff to review our license application. We found the staff's review to be thorough and rigorous. The review encompassed a wide variety of design and licensing basis issues and involved many different branches within the NRC. Entergy extends its appreciation to the numerous NRC technical reviewers who provided input into the draft SE. In particular, we would like to thank Mr. Tom Alexion for his efforts in coordinating the staff's review.

This submittal contains no regulatory commitments.

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I declare under penalty of perjury that the foregoing is true and correct. Executed on February 7, 2002.

Sincerely,

Glenn R. Ashley

Glenn R. Ashley
Manager, Licensing

GRA/dwb
Attachments

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Attachment 1

Comments on "Arkansas Nuclear One, Unit 2 (ANO-2) - Draft Safety Evaluation Regarding the Proposed Extended Power Uprate (TAC MB0789)."

Cover Letter, paragraph one - The following three letters should be added to the list of supplemental letters:

1) June 27, 2001, "Error in the CEFLASH-4AS Computer Code"

2) January 31, 2002, "Response to Follow-up Request for Additional Information Concerning SGTR and MHA Dose Assessment Calculations Supporting ANO-2 Power Uprate" (submitted after the draft Safety Evaluation was issued)

3) February 6, 2002, "Response to Request for Additional Information on Vessel Head Penetration Nozzles Regarding the ANO-2 Power Uprate License Application" (submitted after the draft Safety Evaluation was issued)

Page 1 - Section 2.0 <u>Background</u>

Paragraph 1, last sentence: Change "design" to "best estimate" and "1048 MWe" to "1065 MWe." This change was discussed with the staff during a teleconference on February 5, 2002. The new sentence reads, "The proposed change would increase the unit's best estimate gross electrical output from about 958 megawatts electric (MWe) to about 1065 MWe." The basis for the change is explained in Attachment 2.

Page 5 - Section 3.6.1.1 <u>Large Break Loss-of-Coolant Accident (LBLOCA) and</u> ...

Paragraph 3, second sentence: Consistent with our letter dated June 27, 2001, the value 2066°F should be changed to 2090°F, 10.78% should be changed to 12.5% and 0.67% should be changed to 0.73%.

Page 7 - Section 3.6.1.2.2 <u>Post-Loss-of Cooling Accident Long-Term Cooling (LTC)...</u>

Paragraph 2 following the list of 8 items, second sentence: The lower plenum mixing volume is not addressed in the December 19, 2000, application nor in CENPD-254-P-A. We suggest striking these two references from the sentence. The reference to the October 17, 2001, letter is correct and should remain.

Page 11 - Section 3.6.2 <u>Non-Loss-of Coolant Accident Events Analysis</u>

Paragraph 1, second sentence: The accident analysis methodologies are defined in Table 3 of the November 16, 2001, supplement. Although the core operating limits supervisory system (COLSS) and core protection calculators (CPCs) are NRC approved methodologies, they are more appropriately considered programs for ensuring the plant is operated within the analyzed assumptions.

Page 11 - Section 3.6.2.2 Uncontrolled Control Element Assembly Withdrawal ...

Paragraphs 1 and 3 refer to both BOC and EOC kinetics; however, the analyses were performed only with EOC kinetics.

Page 12 - Section 3.6.2.3 Control Element Assembly Misoperation

Paragraph 1, third sentence: The physics code, ROCS, was utilized for the CEA misoperation event. CENTS was not utilized for the transient part of this effort as indicated in Section 7.3.3.4 of the license application.

Paragraph 1, last sentence: Section 7.3.3 of the license application identifies the input parameter changes addressed in the CEA misoperation analysis. Some of these input parameter changes are directly related to power uprate while others are not. The last sentence of paragraph 1 does not appear to acknowledge these other input parameter changes.

Paragraph 2, first sentence: The DNB safety limit was verified not to be exceeded for the CEA Misoperation events. These events are non-limiting primary and secondary pressure events; hence, no peak pressures were explicitly calculated.

Page 12 - Section 3.6.2.5 Uncontrolled Boron Dilution Incident

Paragraph 1, last sentence: Section 7.3.4 of the license application identifies the input parameter changes addressed in the uncontrolled boron dilution incident. The last sentence of paragraph 1 does not appear to acknowledge these input parameter changes.

Page 13 - Section 3.6.2.5 Uncontrolled Boron Dilution Incident

Paragraph 3, first sentence: The first sentence should be reworded to clarify the analyzed condition. The revised sentence would read (inserted words are indicated with bold faced font), "Dilution during cold shutdown with the RCS filled was also analyzed with the results indicating that if the count rate monitors are operable and **no** CEAs are withdrawn, **the count rate monitors will alarm** or if the count rate monitors ..."

Page 15 - Section 3.6.2.15 Inadvertent Loading of a Fuel Assembly into the...

Paragraph 1, first sentence: "CENTS" should be replaced with "ROCS" and the reference to CETOP should be deleted as discussed in Section 7.3.12.4 of the license application. The revised sentence would read, "The licensee analyzed this event using the approved computer code ROCS for calculation of rod worth and power peaking factors."

Page 16 - Section 3.6.2.17 Control Element Assembly Ejection

Paragraph 1, fifth, six and seventh sentences: ANO-2 calculated clad damage based on a peak pellet average enthalpy $\leq$ 200 cal/gm and incipient centerline

melting at a threshold enthalpy $\leq$ 250 cal/gm. Using the more restrictive limit of these two criteria, ejected 3-D peak versus ejected CEA worth was provided. Additionally, as discussed in Section 7.3.14.5 of the license application, radiological consequences associated with 14% of the fuel failing the peak pellet average enthalpy criterion with no fuel exceeding the incipient centerline melting threshold enthalpy criterion were presented. It is recommended that the fifth, sixth and seventh sentences be revised by the following text or in a similar manner:

"For both limiting cases, the ejected 3-D peaks versus ejected worths were generated based on the acceptance criteria for total average enthalpy and incipient centerline melting threshold. No incipient centerline melting fuel failures will occur as long as the cycle-specific data remain within these limits. Up to 14% of the fuel may exceed the acceptance criteria for total average enthalpy. As discussed in Section 7.2.3, the doses associated with up to 14 % fuel damage were found to be less than the acceptance criteria noted in SRP 15.4.8 and, therefore, acceptable."

Page 18 - Section 3.9.1 <u>Thermal-Hydraulic Analyses</u>

Last paragraph, last sentence: "The CEN-161 (B)-P, methodology more appropriately relates to fuel performance characteristics versus currently referenced thermal hydraulic considerations.

Page 20 - Section 4.1.2 <u>Reactor Core Support Structures and Vessel Internals</u>

Paragraph 2, second sentence: For clarification, recommend inserting "stresses" into the sentence. The revised sentence would read as follows: "In the evaluation of RVI components where the revised data was encompassed by the AOR, **the stresses calculated in** the AOR **were** retained as-is..."

Page 23 - Section 4.1.4 <u>Replacement Steam Generators</u>

Fourth paragraph, last sentence: insert the phrase "above the allowable limit" at the end of the sentence. The new sentence would read, "Therefore, the licensee concluded that the proposed power uprate does not increase the potential for flow induced vibration for the RSG tubes **above the allowable limit.**

Page 24 - Section 4.1.7 <u>Nuclear Steam Supply System Piping and Pipe Supports</u>

Paragraph 3, fourth sentence, the word "stresses" should be replaced with the word "loads" in both locations where it is used. The sentence would then read, "The licensee concluded that all piping **loads** due to thermal, deadweight, and seismic loads for the power uprate were found less than the envelope **loads** employed in the CEN-367-A LBB evaluation and are, therefore, acceptable for ANO-2." This comment is based on the text contained in section 5.6 of Enclosure 5 of the application that specifically refers to loads rather than stresses.

Page 25 - Section 4.1.7 <u>Nuclear Steam Supply System Piping and Pipe Supports</u>

Paragraph 3, fifth sentence states: "The licensee also evaluated the response of the surge line to LOCA loads, and found that the effects on the surge line were

enveloped by the effects of the five major BLPBs." The intent of this sentence appears to summarize the response of the surge line to motion of the pressurizer as discussed in the last paragraph of section 5.7.1 of Enclosure 5 of the application. If so, the phrase "LOCA loads" should be replaced with the phrase "pressurizer motion caused by smaller pipe breaks at the top of the pressurizer." If not, this sentence should be clarified.

Paragraph 3, sixth sentence: Recommend adding the words "in the pressurizer surge line nozzle" after the word "stresses." This suggestion is based on the context of the paragraph and the fact that there are other stress values provided in Attachment 2 of the August 23, 2001, letter besides those in the pressurizer surge line nozzle. The new sentence would read, "The calculated stresses **in the pressurizer surge line nozzle** provided in...below the allowable limits."

Paragraph 3, seventh sentence: For clarification, we suggest adding the word "nozzle" after the word "line." The new sentence would read, "The Code and Code Edition used for the pressurizer surge line **nozzle** is the 1968 Edition of the ..."

Page 27 - Section 4.1.8 <u>Balance-of-Plant Piping</u>

Paragraph 2, last sentence: Replace the words "will be" with "were." This action has been completed.

Paragraph 3, eighth sentence: Based on a teleconference with the NRC staff on February 5, 2002, the startup testing program will be modified to utilize hand-held collection of vibration data on the main steam piping inside containment rather than rely on installed vibration monitoring instrumentation. The basis for this change is provided in Attachment 3.

Page 35 - Section 4.9.1 <u>Suitability of Existing Instruments</u>

List of bullet items following paragraph 1: Recommend adding a seventh bullet item, i.e., "The plant protection system pressurizer pressure low setpoint will be adjusted based on power uprated conditions." This information is discussed in the second paragraph of Enclosure 5, Section 7.4 of the license application.

Page 36 - Section 4.9.2 <u>Reactor Protection System/Engineered Safety Features</u>...

Paragraph 1, seventh sentence: Suggest rewording the sentence as follows: "In addition, the licensee stated that care is taken to sort random error components from non-random components, and room temperatures are **typically** based on the worst-case normal and accident conditions to obtain the highest **uncertainty unless less extreme conditions are specifically justified by the calculation**. Also for any **safety-related** instrument that falls outside..."

Paragraph 2, last sentence: Suggest inserting the word "adversely" prior to affected. The licensing basis is affected by some of the setpoint changes, e.g., the low pressurizer pressure setpoint change; however, the change is not adverse. The new sentence would read, "Therefore, the existing licensing basis is not **adversely** affected by the setpoint changes to accommodate the proposed power uprate."

Page 37 - Section 4.10 Testing

Paragraph 4: As delineated in Section 14.1 of the ANO-2 Safety Analysis Report, please note that ANO-2 is committed to Regulatory Guide 1.68 dated November 1973.

Page 39 - Section 5.1, Background

Third sentence: The Operating License for ANO-2 was issued on July 18, 1978. The license limited ANO-2 to loading nuclear fuel and allowed operation in Mode 5. Amendment #1 to the Operating License was issued on September 1, 1978. Amendment 1 authorized operation to 2815 megawatts thermal.

Page 43 - Section 5.8, Steam Dump and Bypass System

Paragraph 1, second sentence is recommended to be changed from, "The licensee...to accommodate the higher steam flow and reactor heat load" to, "The licensee...to accommodate the higher steam flow and change in steam header pressure versus power level." The wording in the first paragraph of section 2.4.4.1 of Enclosure 5 of the license application "...flow and higher reactor heat load" is incorrect. The correct wording is "...pressure versus power level" which makes the sentence consistent with the first sentence of the second paragraph of section 4.2.3 on page 4-9 of Enclosure 5.

Page 46 - Section 5.14 High Energy Line Break

Paragraph 1, fourth bullet item: For clarification, "chemical and volume control letdown" should also include "charging."

Page 47 - Section 5.17 Flow-Accelerated Corrosion

Paragraph 1, first sentence: Recommend deleting the word "slight." Some of the increases are considered to be more than slight. The revised sentence would read, "The power uprate will result in an increase of the flow rates in certain systems of the plant." The increases in flow rates have been evaluated and are acceptable.

Paragraph 1, fourth sentence: Recommend replacing the phrase "which were considered to be most susceptible to FAC" with the phrase "that are in the FAC program." The revised sentence would read, "The study included the following systems that are in the FAC program: main steam, main feedwater...and SG blowdown." Main steam and main feedwater were not considered to be highly susceptible; however, were included in the study because they are part of the FAC program.

Page 47 - Section 5.18 Electrical Systems

Paragraph 1, last sentence: Change "1048" MWe to "1065" MWe. This change was discussed with the staff during a teleconference on February 5, 2002. The basis for the change is explained in Attachment 2.

Page 49 - Section 5.18.2 Main Generator

Paragraph 1, second line: "370 MVAR" should be changed to "362 MVAR" and "1052 MWe" should be changed to "1065 MWe." The basis for the change is explained in Attachment 2.

Page 49 - Section 5.18.3 Main Power Transformer

Paragraph 1, second sentence: The 1092 MVA value is incorrect. The value would be more correctly expressed as 1023.5 MWe (1065 MWe best estimate full power minus 41.5 MWe for house loads).

Page 54 - Section 7.1 Atmospheric Relative Concentration Estimates

Paragraph 1, first sentence: ANO-2 used five years of onsite meteorological data to estimate the atmospheric relative concentrations used in the control room only. The EAB and LPZ dispersion factors were based on original licensing basis considerations.

Paragraph 3, second sentence: The word "current" should be replaced with "original proposed." This sentence would then read, "The LPZ values are slightly higher than those currently in Section 2.3 of the updated SAR because they are calculated for a shorter LPZ distance than the **original proposed** distance of four miles."

Paragraph 3, third sentence: The ANO-2 SAR describes the use of the X/Q equation in Regulatory Guide 1.4, and selected the 5 percentile X/Q values as a function of direction. This is true for the 0-8 hours X/Q, but Section 2.3.4.2 of the SAR describes an equation used for > 8 hours derived from Equation 3.142 on page 112 of Meteorology and Atomic Energy, stating that it has recently become accepted meteorological practice to use it in place of the sector spread equation in Regulatory Guide 1.4.

Page 57 - Section 7.2 Radiological Analysis

The third paragraph from the bottom of the page, last sentence, states, "Until the licensee submits control room analyses demonstrating that GDC19 is met, the facility's ability to meet GDC 19 as a result of this power uprate will be considered as an open item. In a letter dated January 14, 2002, ANO-2 submitted analyses to demonstrate the capability of maintaining the control room in accordance with GDC 19, "Control Room," of Appendix A to 10 CFR Part 50. Compliance with GDC-19 is based on the NRC's acceptance of 61 scfm inleakage as the new design bases for ANO-2.

Second paragraph from bottom of page: This paragraph needs to reflect the SGTR dose information provided in supplemental letters dated October 31 and December 20, 2001, and January 31, 2002.

Page 58 - Section 7.2.1 Large Break Loss-of-Coolant Accident

Paragraph 3, last sentence: In regard to containment mixing, the draft Safety Evaluation notes that "The resolution of this issue has not yet been determined and it may affect future analyses." Based on telephone conversations with the NRC staff, successful resolution has been reached as a result of the information provided to the staff in ANO-2 letter dated January 31, 2002.

Paragraph 4: This paragraph should be revised consistent with the supplemental information submitted in letters dated January 14 and 31, 2002.

Page 60 - Table 7.2-1 Assumptions for Loss-of-Coolant Accident Analysis

The value for Reactor Building Free Volume (ft^3) should be changed from "1.84E6" to "1.778E6," the value for Reactor Building Sump Liquid Volume (ft^3) should be changed from "5.83E4" to "62,898," and the Leakage Rate (%/day) for the 0-24 hour timeframe should be changed from "0.2" to "0.1" and for the > 24 hour timeframe should be changed from "0.1" to "0.05" consistent with the information provided in Section 7.3.10-1 of the license application.

Page 63 -Table 7.2-2 Assumptions for Fuel Handling Accidents

The Number of Assemblies should be changed from "157" to "177" consistent with the information provided in Section 5.1.1 of the license application. Also, the 0-2 hour Control Room Atmospheric Dispersion Factor (sec/m^3) should be changed from "9.77E-4" to "1.2E-3" consistent with Section 7.3.15.4, item 8 of the license application.

Page 79 - Section 8.2.2 Fires

Paragraph 2, fifth sentence: The phrase "...and, unlike fire areas, the zones are separated according to Appendix R to 10 CFR Part 50 requirements." should be deleted because fire area boundaries do meet Appendix R separation requirements.

CLARIFICATIONS IDENTIFIED DURING THE REVIEW OF THE DRAFT SAFETY EVALUATION, BUT NOT PREVIOUSLY INCLUDED IN THE LICENSE APPLICATION OR SUPPLEMENTS

Page 50 - Section 5.18.4 Unit Auxiliary Transformer

Third sentence: "0.56 MVA at 0.9" should be changed to "0.70 MVA at 0.89." The change is necessary because the auxiliary loads that were added as a result of power uprate were determined to be slightly higher than originally evaluated.

Page 77 - Section 8.2.2 Fires

Paragraph 1: The reference to EPRI Report 3385-01 was to a draft document. The report has now been issued as TR-105928.

Page 44 - Section 5.11 Other Balance-of-Plant Evaluations and Page 46 - Section 5.16 Fire Protection Program

Section 5.11 and 5.16 both imply that the evaluation of the impact of power uprate on the fire protection program has been completed. This was based on information provided in the December 19, 2000, license application. This information needs to be clarified. Substantial reviews of the fire protection program regarding modifications and analyses supporting power uprate have been conducted. The reviews of the 2R15 refueling outage modifications for Fire Protection Program impacts are being finalized. No further fire protection program modifications are expected from the remaining reviews.

TYPOGRAPHICAL ERRORS

Page 4 - Section 3.6 Accident Analysis Evaluation

Paragraph 2, last sentence: "Table 7.3.0.1-1" should be changed to "Table 7.3.0-1."

Page 6 - Section 3.6.1.2.2 Post-Loss of-Cooling Accident Long Term Cooling (LTC)

In the above title, the word "Cooling" should be "Coolant"

Page 12 - Section 3.6.2.5 Uncontrolled Boron Dilution Event

Paragraph 1, third sentence: It should be noted that the analyses were performed for Modes 1-6, not Modes 1-5 as stated in the draft Safety Evaluation.

Page 20 - Section 4.1.1 Reactor Vessel

Second paragraph, second sentence: The reference to Table 3-1 of the August 23, 2001, RAI response is not correct. The correct reference should be Table 3-1 of Enclosure 5 of the application.

Page 22 - Section 4.1.4 Replacement Steam Generators

Paragraph 1, fourth sentence: "Table 2-5" should be "Table 2-4."

Page 26 - Section 4.1.8 Balance-of-Plant Piping

Paragraph 1, second sentence: Insert the word "in" after the word "listed."

Page 37 - Section 4.10 Testing

Paragraph 3: "establish" should be "established"

Page 40 - Section 5.3 Service Water System

Second sentence: The phrase "engineering" safety features equipment should be changed to "engineered" safety features equipment.

Page 44 - Section 5.11 Other Balance-of-Plant Evaluations

Paragraph 1, first sentence: The word "license" should be "licensee."

Page 49 - Section 5.18.2 Main Generator

Paragraph 1, third line: The "0.94" power factor should be "0.95." The power factor for Cycle 16 forward that was included in Table 2-3 of Enclosure 5 of the application is 0.95.

Page 59 - Section 7.2.3 Rod Ejection

Paragraph 1, seventh sentence: "15%" should be "14%."

Page 62 - Table 7.2-1 Assumptions for Loss-of-Coolant Accident Analysis (continued)

The first item under Control Room should be changed from "0-28 hours" to "0-2 hours."

Page 73 - Table 8.1 Post-Initiator Operator Actions Affected by Extended Power Uprate

Event Name "EHF2Y1Y2SP" should be "EHF2Y1Y2XP."

Available Time (minutes) for the Pre-Extended Power Uprate condition for Event THF2OTCLRP was stated to be "50"; it should be "52."

Attachment 2

Generation Changes Since Submittal of License Application

The December 19, 2000, license application included a gross generation best estimate output of 1048 MWe. This was based on the rated (i. e., 100% power level at power uprate conditions) heat balance provided by General Electric (G.E.) in its design documentation package. The guaranteed generation output of 1048 MWe was equal to the rated generation. The rated and guaranteed output was supposed to be based on the performance of the new high pressure turbine and four replacement stages in the low pressure turbines.

Data from startup after 2R14 demonstrated that the high pressure turbine was performing considerably better than predicted by the G.E. rated heat balance. In particular, heater drain pump flows were about 8% greater than predicted. ANO engineering reviewed several possible root causes, one of which was greater high pressure turbine efficiency than shown on the G.E. heat balance. ANO engineering calculated that the original high pressure turbine had better isentropic efficiency than the new high pressure turbine. This did not make sense, and performance of the new high pressure turbine indicated that this was not the case. ANO engineering contacted G.E. engineering, which agreed that the heat balance was incorrect.

A revised G.E. documentation package, that included revised heat balances, Mollier diagrams and generator performance curves, was received from G.E. in August 2001. The rated heat balance showed a gross generation output of 1073 MWe. However, ANO engineering has determined that the best estimate rated gross generation is 1065 MWe, a reduction of 8 MWe from the G.E. rated heat balance. This is because the G.E. heat balance is based on pristine feedwater heaters and assumed drain cooler approaches (DCAs) and terminal temperature differences (TTDs). The ANO-2 feedwater heaters, while in very good condition, are not pristine and the DCAs and TTDs are not quite as good as assumed in the G.E. heat balance analyses. Although the rated output increased from 1048 to 1073 MWe, the guaranteed output remains at 1048 MWe.

As mentioned above, the G.E. documentation package included revised generator performance curves. Per the revised Generator Reactive Capability Curve, the maximum reactive power that the generator can provide at 1065 MW is approximately 362 MVARS. This equates to a power factor of approximately 95%.

The generation best estimate vs. guaranteed capability was discussed with the NRC staff during a teleconference on February 5, 2002.

Attachment 3

Justification for Utilization of Hand-Held Vibration Monitoring Instrumentation on the Main Steam Lines Inside Containment

Arkansas Nuclear One, Unit 2 (ANO-2) is planning to change the vibration startup testing inside the containment building for the post power uprate startup from refueling outage 2R15, scheduled for the spring 2002. In a supplemental letter dated August 23, 2001, Entergy responded to NRC Question 11 by stating that the same testing approach as that utilized following refueling outage 2R14 would be followed during startup from 2R15. This approach would involve the use of temporarily installed vibration measurement equipment on the main steam and main feedwater piping inside containment. Temporary vibration measuring equipment was installed during startup testing from 2R14 because at that time the ANO practice was not to allow containment building entries at power for the purpose of collecting vibration data. There has been a shift in philosophy at ANO regarding containment entries during power operations. The option of entering the containment building at power to collect startup testing vibration data is now available.

The installation of the temporary vibration measurement equipment requires a considerably longer time spent in the containment building to remove and reinstall the vibration pickups and associated equipment than it would require to measure the vibration using hand-held instruments. The temporary instruments were originally installed during 2R14, and were used to collect vibration data during the power ascension to the pre-power uprate 100% power level. After the startup testing was completed, the portion of this temporary data collection system that was outside containment was removed; however, due to the lack of access at power, the bulk of the equipment remained in place during the operating cycle between 2R14 and 2R15. Because the equipment was not intended to sustain long term exposure to the temperatures of these piping systems, the equipment inside containment is not expected to be accurate for re-use during the 2R15 startup testing. The existing temporary system inside containment would have to be removed during 2R15, and new temporary equipment would have to be installed if the same methods were used in the 2R15 startup testing. In addition, an entry would have to be made to remove this equipment either at power or during 2R16.

The affected piping is outside the shield walls, and existing pipe clamps can be accessed to collect vibration readings with hand-held instruments during power operations. The hand-held instruments provide accuracy equivalent to the temporarily installed instruments, and the hand-held instruments offer more flexibility and reliability. The vibration monitoring will be conducted in accordance with the guidance of ASME/ANSI OM-3, "Operation and Maintenance Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems." The 0.5 inch per second screening criterion will continue to be utilized as recommended by OM-3.

The option of collecting the vibration data inside the containment building by use of hand-held instruments offers several advantages. Use of the hand-held instruments allows a visual confirmation of the vibration levels. The visual observation of the piping vibration allows monitoring of the entire piping system including elements such as vents, drains, and instrument tubing. The elimination of reinstalling and again removing the temporary instruments inside containment is expected to reduce the radiation exposure required to collect the data. The measurement readings are directly recorded into the hand-held instruments instead of transmitted through temporary cables to containment penetrations and then to data recorders outside of containment. Instrument malfunctions can be more easily corrected with the hand-held instruments than with the temporarily installed instruments. Additionally, more data collection points will be available.

Because of these advantages, and because the option is now available, the startup vibration testing on the main steam and main feedwater piping inside the containment building is planned for completion using hand-held instruments during the startup from the 2R15 refueling outage. Since there is no change expected in the range of operation that was tested during 2R14, the 2R15 data collection will be performed once at the pre-power uprate full power level which will be at approximately 90% of the uprated full power level, and once again at the uprated 100% full power level.