Entergy

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

2CAN080305

August 12, 2003

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

SUBJECT: Supplement to Amendment Request to Revise the Spent Fuel Pool Loading Pattern
Arkansas Nuclear One, Unit 2
Docket No. 50-368
License No. NPF-6

REFERENCES:
1. Entergy letter dated June 30, 2003 to the NRC, License Amendment Request to Revise the Spent Fuel Pool Loading Pattern (2CAN060306)
2. Entergy letter dated August 1, 2003 to the NRC, Supplement to Amendment Request to Revise the Spent Fuel Pool Loading Pattern (2CAN080301)

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Arkansas Nuclear One, Unit 2 (ANO-2) Technical Specifications (TSs) to revise the spent fuel pool (SFP) loading pattern.

On July 23, 2003 and July 24, 2003, Entergy received requests for additional information from the Plant Systems Branch and Mechanical and Civil Engineering Branch, respectively, which were determined to need formal response. Entergy's responses are contained in Attachments 1 and 2. The response to a request for additional information from the Reactor Systems Branch was previously submitted by letter (Reference 2).

The original no significant hazards consideration included in Reference 1 is not affected by any information contained in the supplemental letter. There are no new commitments contained in this letter.

If you have any questions or require additional information, please contact Dana Millar at 601-368-5445.

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 12, 2003.

Sincerely,

Vice President, Operations
Arkansas Nuclear One

CGA/dm

Attachments:
1. Response to Request for Additional Information - Plant Systems Branch
2. Response to Request for Additional Information – Mechanical and Civil Engineering Branch

cc: Mr. Thomas P. Gwynn
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

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

U. S. Nuclear Regulatory Commission
Attn: Mr. Thomas W. Alexion MS O-7D1
Washington, DC 20555-0001

Mr. Bernard R. Bevill
Director Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street
Little Rock, AR 72205

Attachment 1

To

2CAN080305

Response to Request for Additional Information

Plant Systems Branch

Request for Additional Information - Plant Systems Branch
Amendment Request to Revise Spent Fuel Pool Loading Pattern
Arkansas Nuclear One, Unit 2

Question 1:

In the staff's safety evaluation for WCAP-15516-P, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," the staff stated that all licensees proposing to credit soluble boron should identity potential events which could dilute the spent fuel pool soluble boron to the concentration required to maintain the 0.95 k-eff limit and should quantify the time span of these dilution events to show that sufficient time is available to enable adequate detection and suppression of any dilution event. The staff also stated that the effects of incomplete boron mixing, such as boron stratification, should be considered, and that the boron dilution analysis should also be used to justify the surveillance interval used for verification of technical specification minimum pool boron concentration. In order to complete our review, the NRC staff requests that the licensee provides the following information regarding Section 7.0, "Soluble Boron Dilution Evaluation," of Attachment 5, of their application dated June 30, 2003:

The fuel pool heat exchanger provides a boundary between unborated service water on the shell side and borated pool water on the tube side. The operation of the heat exchanger is such that the shell side is at a higher pressure than the tube side. In the event of a tube rupture in the spent fuel pool heat exchanger, unborated water can be introduced into the pool through the ruptured tube. Please describe how this event, and the potential dilution resulting from it, would be detected and mitigated. The response should include expected alarms and specific steps from alarm response or off-normal procedures that would lead to corrective actions.

Response 1:

If there were a tube leak in the spent fuel pool (SFP) heat exchanger (HX), service water (SW) will leak into the SFP water. A SFP high level alarm will annunciate in the control room as level increases. In response to the annunciator, the control room staff enters an abnormal operating procedure. Steps in the procedure require identification and subsequent isolation of the source of in-leakage. In addition to the alarm in the control room, the level is monitored and recorded once per 12 hours by the waste control operator. Actions to identify in-leakage or lowering level would be initiated if any unexpected increase or decrease were noted.

A conservative bounding evaluation, which assumed the worst case differential pressure and a double ended rupture of a single tube in the SFP HX, was performed to determine SW in-leakage and the associated dilution rate. Based on these assumptions leakage was determined to be approximately 250 gallons per minute (gpm). At this dilution rate, it will take about 7.6 hours to dilute the SFP boron concentration to the assumed accident boron concentration of 825 parts per million (ppm) and about 18.1 hours to reach the minimum required boron concentration of 240 ppm for normal conditions. Assuming the fuel transfer gates are closed and an in-leakage of approximately 250 gpm, the SFP high level alarm will annunciate in the control room between two and eight minutes into the event based on the initial SFP level. If for some reason the high level alarm malfunctioned, the level in drain collect tanks would begin to increase about 1.2 hours into the event. Drain tank level is monitored by the control room staff and locally by the waste control operator. In addition, SFP level is monitored locally every 12 hours. If a SFP tube rupture of this size were to occur, the control room staff has adequate time to identify and isolate the source of the in-leakage and restore the boron concentration to the proposed Technical Specification value of 2000 ppm before reaching the limiting boron concentrations of 825 ppm or 240 ppm.

Question 2:

Section 9.1.2 of the ANO-2 Safety Analysis Report states that damage to the spent fuel pool floor resulting from light load drops would not cause a loss of coolant inventory in excess of the capacity of the normal makeup systems. Describe how this event and the potential dilution resulting from use of unborated makeup water sources would be detected and mitigated. The response should include expected alarms and specific steps from alarm response or off-normal procedures that would lead to corrective actions.

Response 2:

Typically the control room will be notified in the event of any type of load drop in the SFP. If, however, the control room were not notified and the SFP level were decreasing as a result of the drop, the lowering level would be detected either by the annunciation of the SFP low level alarm in the control room or by local observation during operator rounds. (The SFP level is monitored and recorded locally every 12 hours and is maintained within a maximum and minimum value.) If a SFP low level alarm is received, procedural steps require SFP makeup as well as various other actions which assist in determining and securing the cause of the low level. The structural integrity of the SFP is procedurally confirmed by locally checking for leakage of the SFP stainless steel liner drains.

SFP makeup is controlled from the control room. Procedural requirements include determining the amount of boric acid and/or water needed to achieve the desired SFP water level and/or boron concentration. If SFP makeup is from any source other than the Refueling Water Tank (RWT), which has a TS minimum boron concentration requirement of 2500 ppm (TS 3.5.4), a chemistry sample of the boron concentration is required after makeup is secured.

Question 3:

Section 9.1.3 of the ANO-2 Safety Analysis Report states that service water makeup is piped to the pool, and the valves are located such that minimal operator action is required to initiate makeup from either or both service water headers. Describe how the boron dilution resulting from both a slow, steady makeup of 8 gpm or makeup at the maximum rate from the source would be detected and mitigated. Again, this response should include expected alarms and specific steps from alarm response or off-normal procedures that would lead to corrective actions.

Response 3:

Seismic class I SW makeup is available to the SFP and is used only as an emergency means of makeup. SW is isolated from the SFP by a minimum of two manual isolation valves, one of which is locked closed with the key under the control of operations personnel. Operator actions, therefore, are required to initiate SW makeup. Prior to commencing makeup, procedural steps require stationing an operator to monitor SFP level. When the desired level is achieved the SW valves are closed. This is a procedurally controlled evolution and should not result in excessive dilution. However, if the SW system passively leaked unborated water into the SFP, assuming the fuel transfer gates are closed and an in-leakage of approximately 8 gallons per minute (gpm), the SFP high level alarm will annunciate in the control room about one hour into the event. As level increases, the same alarms and similar actions as discussed in response to question 1 will be taken.

Attachment 2

To

2CAN080305

Response to Request for Additional Information

Mechanical and Civil Engineering Branch

Request For Additional Information - Mechanical and Civil Engineering Branch
Amendment Request to Revise the Spent Fuel Pool Loading Pattern
Arkansas Nuclear One, Unit 2

Question 1:

In the amendment request, you are proposing to reduce the fresh fuel assembly initial enrichment limit to less than or equal to 4.55±0.05 weight percent. Indicate whether the total weight of the fuel assemblies supported by the spent fuel racks will increase as a result of the proposed request. If the total weight of the fuel assemblies is increased, what are the affects on the spent fuel racks and the spent fuel pool structure due to the increased weight?

Response 1:

The total weight of a fuel assembly will not increase as a result of the proposed change to lower the fuel enrichment. Small variations in uranium density depending on enrichment exist, however the weight difference is negligible. The following example illustrates the weight differences:

Assume a normal uranium loading of 430 kilograms (kg) in each assembly. A 5.0 weight percent (wt%) fuel assembly would have 21.5 kg of ^{235}U and a fuel assembly with an enrichment of 4.55 wt% would have 19.565 kg of ^{235}U. The 4.55 wt% fuel assembly would have roughly 2 kg less ^{235}U. Assuming the ^{235}U was replaced with natural occurring uranium the weight difference would be approximately 25 grams or 0.055 lbs.

Question 2:

Lateral motion of the storage racks under postulated seismic conditions could potentially alter the spacing between racks. Indicate whether you have performed a structural rack dynamic analysis to calculate the required spacing between racks for your criticality calculations. If this analysis has been performed, discuss the methodologies and assumptions used for the analysis, and provide a summary of the analysis results. If you have not performed a structural rack dynamic analysis, provide justification as to why the spacing between the racks need not be verified in the revised configuration.

Response 2:

The interface criticality calculations used to support this application bound rack to rack interfaces as discussed in Section 5.4 of Holtec License Amendment Report (Attachment 5 of the letter dated June 30, 2003 to the NRC, License Amendment Request to Revise the Spent Fuel Pool Loading Pattern).

A new structural rack dynamic analysis was not performed in conjunction with this amendment request; however an analysis was previously performed by Westinghouse. The Westinghouse seismic analysis was performed for the SFP re-rack and subsequently approved by the NRC (TS Amendment 43 dated April 15, 1983). The analysis shows that the maximum displacement of a rack combining the maximum sliding distance with the maximum structural deflection is 0.37 inches. The minimum rack-to-rack gap is 2 5/16 inches which is measured cell wall to cell wall. If the two racks placed closest to each other were out of phase, rack to rack impacts would not occur.