September 5, 2000

MEMORANDUM TO: Stuart A. Richards, Director
Project Directorate 4
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Mark Reinhart, Chief ***/RA/***
Licensing Section
Probabilistic Safety Assessment Branch
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

SUBJECT: SAFETY EVALUATION INPUT FOR RESOLUTION OF UNREVIEWED SAFETY QUESTION ASSOCIATED WITH STEAM GENERATOR REPLACEMENT AT ARKANSAS NUCLEAR ONE UNIT NO. 2 (TAC NO. MA7299)

By letter dated November 29, 1999, Entergy Operations, Inc., requested review and approval of proposed changes to the Arkansas Nuclear One - Unit-2 (ANO-2) technical specifications associated with the installation of the replacement steam generators. The letter also requested NRC review of an unreviewed safety question involving the radiological consequence analyses related to the steam generator replacement.

The Licensing Section of the Probabilistic Safety Assessment Branch (SPSB) has reviewed the amendment request with regard to the unreviewed safety issue and the radiological consequences of the steam generator replacement. Our input to the safety evaluation is attached.

Entergy's original submittal did not evaluate the potential impact of the steam generator replacement in the control room habitability. The staff asked the licensee to provide an evaluation and to substantiate the assumption of 10 cfm unfiltered inleakage to the control room. The licensee's first response to this request was not acceptable. The licensee subsequently re-analyzed the radiological consequences assuming an increased unfiltered inleakage rate intended to bound the leakage rates observed at other facilities that have performed inleakage testing. Based on these re-analysis results, SPSB was able to conclude, with reasonable assurance, that the control room would remain habitable in the event of the main steam line break, feedwater line break, and main coolant pump seized rotor accidents. The remaining design-basis accidents are not affected by the steam generator replacement.

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Entergy has indicated that they will be requesting a power uprate in the 4th quarter 2000. During teleconferences related to this action, the SPSB staff informed Entergy that its use of a bounding in-leakage value, while acceptable for the present action, might not be found acceptable with regard to the power uprate amendment. The staff would expect Entergy to quantify the control room in-leakage with appropriate testing.

Docket No. 50-368

Attachment: As stated

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Attachment: As stated

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SAFETY EVALUATION INPUT BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO REQUEST FOR RESOLUTION OF UNREVIEWED SAFETY QUESTION
ASSOCIATED WITH UFSAR CHANGES SUPPORTING
STEAM GENERATOR REPLACEMENT BY
ENTERGY OPERATIONS, INC.
ARKANSAS NUCLEAR ONE, UNIT NO. 2
DOCKET NO. 50-368

1 INTRODUCTION

By letter dated November 29, 1999, Entergy Operations, Inc., requested review and approval of proposed changes to the Arkansas Nuclear One, Unit-2 (ANO-2) technical specifications associated the installation of the replacement steam generators. The letter also requested NRC review of an unreviewed safety question involving the radiological consequence analyses related to the steam generator replacement. By letters dated May 17, 2000, and August 4, 2000, Entergy provide additional information in response to staff requests.

The replacement steam generators (RSG) are of a different design than the original steam generators (OSG). Entergy proposed several technical specification changes intended to maintain consistency with the transient and accident analyses performed to evaluate the impact of the RSG installation. The installation of the RSGs results in the following changes that were accommodated in the re-analyses performed by Entergy:

- The RSGs are larger than the OSGs, resulting in a larger secondary side mass.
- The larger RSGs contain more tubes that the OSGs, resulting in increased reactor coolant system (RCS) mass.
- The larger RSGs have increased RCS flow and primary-to-secondary heat transfer area, resulting in higher steam generator operating pressures.
- The RSG design incorporates an internal steam flow-limiting nozzle; resulting in a reduction in the effective cross-sectional area of the steam line.

In addition to these changes in analyses parameters, Entergy incorporated the following analysis changes:

- A core power of 3087 MWt (3026 MWt with 1.02 factor) in the radiological analyses of the seized rotor accident and the main steam line break. This is conservatively greater than the current licensed power and is consistent with a planned future power uprate.
- Several changes were made to the methodologies used in assessing the radiological consequences of the affected accidents.

Attachment

2 EVALUATION

In evaluating the consequences of the replacement of the steam generators, Entergy performed an engineering review of the various accident analyses described in the ANO-2 FSAR to determine the effect(s) of the steam generator replacements. Where effects were identified, Entergy performed further evaluations and, where needed, re-analyses. This safety evaluation input only addresses the radiological consequences of these effects and whether these consequences continue to meet dose guidelines and criteria. Entergy identified five affected radiological analyses:

1. Steam Generator Tube Rupture
2. Loss of AC Power Auxiliaries
3. Main Coolant Pump Seized (locked) Rotor
4. Feedwater Line Break
5. Main Steam Line Break

The staff reviewed the results of Entergy's categorization of analysis impacts and, based on its experience with similar applications, finds this listing adequately encompassing. The staff did note, however, that Entergy did not address the potential consequences on post-accident control room operator doses in the docketed material. This issue was discussed with Entergy during teleconferences on April 3, 2000, and April 20, 2000. Entergy responded via a letter dated May 17, 2000. The staff did not find that response acceptable. By letter dated August 4, 2000, Entergy provided additional information. This issue will be further discussed later in the safety evaluation input.

2.1 <u>Steam Generator Tube Rupture (SGTR)</u>

A SGTR accident involves the rupture of a steam generator tube, resulting in the transfer of reactor coolant to the secondary side. A loss of offsite power is conservatively assumed. Because of the loss of AC power, radioactivity is released from the steam generators via steam relief valves and atmospheric dump valves to the environment.

Entergy evaluated the potential impact on the prior SGTR analysis and concluded that no re-analysis was necessary. Entergy stated that the smaller U-tube diameter, higher secondary side pressure, and more restrictive tube flow resistances of the RSGs would result in a lower primary-to-secondary mass transfer than previously estimated. Entergy concluded that the prior analyses with the higher postulated mass transfer would bound the consequences of a SGTR occurring in an RSG, and that no further re-analysis was necessary. The staff agreed with the licensee's conclusion.

2.2 <u>Loss of AC Power Auxiliaries (LOAC)</u>

A LOAC accident results from an event in which offsite AC power is lost. From a radiological standpoint, this event results in a protracted plant cooldown effected by the dumping of steam to the environment via steam generator atmospheric dump valves. It is conservatively assumed that reactor coolant and steam generator leakage and specific activity are at their maximum technical specification values.

Entergy evaluated the potential impact on the prior LOAC analysis and concluded that no re-analysis was necessary. Entergy stated that the radiological consequences of this event are bounded by the analyses for the MSLB and FWLB accidents and no further re-analysis was necessary. The staff agreed with the licensee's conclusion.

2.3 Main Coolant Pump Seized Rotor (LRA)

A LRA results from a mechanical defect that causes the impeller of a main coolant pump in a reactor coolant loop to suddenly stop rotating. This substantially reduces reactor coolant flow in the affected loop resulting in a sharp reduction in reactor coolant pressure in localized regions of the core. Fuel damage is postulated.

Entergy re-analyzed the LRA using the increased reactor coolant and steam generator liquid masses, the increased reactor power, and the changes in dose calculational methodology. Since a specific fuel damage estimate was not available for the cycle 15 fuel load, Entergy performed a re-iterative analysis to determine the amount of fuel damage that could occur with doses still remaining within dose acceptance criteria. From this evaluation, Entergy postulated a fuel damage of 14 percent. In their letter of May 17, Entergy stated that the preliminary results of cycle 15 thermo-dynamic analyses indicate that the estimated fuel damage would be less than 3 percent. Entergy stated that the results of their analyses indicated that offsite doses would remain within a small fraction (10 percent) of the Part 100 accident dose guidelines for estimated fuel damage up to 14 percent. In their letter dated August 4, 2000, Entergy stated that the re-analysis of the control room doses from a LRA would be within the dose criteria of 10 CFR 50, Appendix A, General Design Criterion (GDC)-19. See the discussion in Section 2.6 below.

The staff reviewed the description of the LRA accident analysis and performed an independent analysis to confirm the licensee's conclusions. The staff agreed with the licensee's conclusion that the postulated doses will remain within the applicable acceptance criteria with fuel damage less than 14 percent.

2.4 Feedwater Line Break (FWLB)

A FWLB results from a break in a feedwater line leading to one of the steam generators. This break causes a depressurization of the affected steam generator and a blowdown of its liquid contents through the break. Radioactivity is released to the environment via the break and from steam dumping associated with the plant cooldown.

Entergy re-analyzed the FWLB using the increased reactor coolant and steam generator liquid masses, and the changes in dose calculational methodology. Entergy did not use the increased reactor power in the radiological consequence analyses associated with this event. Entergy stated that the limiting break would occur inside containment since check valves located on the feedwater line would prevent the steam generators from blowing down outside. Entergy did not credit holdup in the containment in assessing the releases. Entergy did not assume an iodine spike as this assumption is not consistent with its licensing basis. Entergy stated that the results of their analyses indicated that offsite doses would remain within a small fraction (10 percent) of the Part 100 accident dose guidelines. In their letter dated August 4, 2000, Entergy stated that the re-analysis of the control room doses from a FWLB would be bounded by the doses postulated for the MSLB. See the discussion in Section 2.6 below.

The staff reviewed the description of the FWLB accident analysis and performed an independent analysis to confirm the licensee's conclusions. The staff agreed with the licensee's conclusion that the postulated doses will remain within the applicable acceptance criteria.

2.5 Main Steam Line Break (MSLB)

A MSLB results from a break in a main steam line leading from one of the steam generators. This break causes a depressurization of the affected steam generator and a blowdown of its contents through the break. Radioactivity is released to the environment via the break and from steam dumping associated with the plant cooldown.

Entergy re-analyzed the LRA using the increased reactor coolant and steam generator liquid masses, the increased reactor power, and the changes in dose calculational methodology. For the MSLB break, Entergy considered pre-incident and accident-initiated iodine spikes and assumed that the break would occur outside of containment but upstream of the isolation valve and that the affected steam generator would boil dry. Plant cooldown would be achieved by dumping steam from the unaffected steam generator. Entergy stated that the results of their analyses indicated that offsite doses would remain within a small fraction (10 percent) of the Part 100 accident dose guidelines for the accident-initiated iodine spike case and would remain within the Part 100 accident dose guidelines for the pre-incident iodine spike case. In their letter dated August 4, 2000, Entergy stated that the re-analysis of the control room doses from a LRA would be within the dose criteria of 10 CFR 50, Appendix A, General Design Criterion (GDC)-19. See the discussion in Section 2.6 below.

In early 1999, another power reactor licensee submitted a licensee event report (LER) regarding non-conservatisms in the assessment of the iodine spike appearance rate that is used in evaluating the accident-initiated iodine spike case. This LER identified that the original vendor assessments assumed a letdown system flow rate less than the maximum possible system flow rate. The staff identified that prior analyses for ANO-2 assumed a flow rate of 40 gpm, about one-third of the maximum system flow of 128 gpm. At ANO-2, the letdown flow rate is a function of the number of charging pumps operating. The staff requested that Entergy address the applicability of the LER issues to ANO-2. In its response of August 4, 2000, Entergy stated that while plant procedures do not prohibit operation of additional charging pumps, operation with more than one charging pump during steady-state operations is rare. Since operation at the higher flow rate is an infrequent situation, the staff agreed with Entergy's position that an iodine appearance rate assessment based on the normal letdown flow rate is appropriate.

The staff reviewed the description of the MSLB accident analysis and performed an independent analysis to confirm the licensee's conclusions. The staff agreed with the licensee's conclusion that the postulated doses will remain within the applicable acceptance criteria.

2.6 Control Room Operator Doses

Entergy did not address the impact of the steam generator replacement on the post-accident doses to the control room operators in its original submittal. In conference calls held on April 3, 2000, and April 20, 2000, the staff requested Entergy to (1) address the impact of the proposed changes on control room habitability, and (2) provide an explanation supporting the conclusion that the control room unfiltered in-leakage is limited to only 10 cfm as stated in the FSAR.

In their letter dated May 17, 2000, Entergy responded to both of these information requests. With regard to the first issue, Entergy provided a discussion on their evaluation that establishes the loss of coolant accident (LOCA) analysis as the limiting accident (the radiological consequences of a LOCA are not affected by the steam generator replacement). Entergy addressed accident-specific differences in atmospheric dispersion, radioactivity release rate, and control room isolation actuation.

Entergy's response to the issue of control room infiltration did not provide a basis for understanding why 10 cfm was an appropriate assumption. The ANO in-leakage assumption of 10 cfm is questionable in light of the industry experience with recent in-leakage testing. In-leakage testing at 20 percent of the current operating plants has shown measured infiltration rates exceeding the values assumed in the design basis analyses. In its response, Entergy took the position that LOCA consequences were bounding and that they were pursuing a generic industry resolution to the in-leakage issue. Since the staff can not make a current finding of acceptability on the basis of future compliance with a voluntary industry initiative, this response was not accepted.

By letter dated August 4, 2000, Entergy submitted a description of the revised control room dose calculations for the MSLB, FWLB, and LRA. These analyses assumed an unfiltered in-leakage flow rate of 5000 cfm. Although not based on the results of in-leakage testing, Entergy stated their belief that the 5000 cfm value was a bounding number. This conclusion was based on the relative volume of the ANO control room in relation to the assumed in-leakage. A control room air exchange of once every 8 minutes is implied by this assumption. Entergy noted that ASHRAE guidance states that typical control room envelopes have air exchanges in the order of once per hour, or about 667 cfm for ANO. Entergy reported the limiting dose consequence (MSLB) to be 21.4 rem with the iodine protection factor (IPF) of 1.38. An IPF of 1.0 (i.e., no credit for control room envelope) would have resulted in a dose of 29.6 rem which is less than the 30 rem thyroid acceptance criteria for control room habitability.

In their response, Entergy noted that they (1) were actively participating in the industry initiative, (2) have initiated a condition report to track and resolve the generic issue, (3) will be giving further consideration to control room infiltration as part of a power uprate submittal scheduled to be presented this year and, (4) have implemented corrective and preventative programs for control room integrity components. Based on its re-analyses with the assumed 5000 cfm in-leakage and the other information, Entergy believes that the setpoint changes associated with the SG replacement may be approved without imposing significant risk to the public or control room personnel.

Based on its review of the additional information docketed by Entergy, the staff has determined that there is reasonable assurance that the ANO control room would be habitable during the MSLB, FWLB, and seized rotor accidents and that this amendment may be approved before the resolution of the generic issues. The staff bases this determination on the results of Entergy analyses that assumed an infiltration of 5000 cfm, and the expectation that 5000 cfm is bounding. The approval of this amendment does not exempt Entergy from regulatory actions that may be imposed in the future as this generic issue is resolved.

2.7 <u>Revision to Control Room Atmospheric Dispersion Values</u>

In re-analyzing the control room doses, Entergy used atmospheric dispersion values, χ/Q, that were generated as part of this effort. Entergy used the NRC-sponsored computer code

ARCON96 to generate these data. These new values were determined based on five years of site-specific meteorological data collected between January 1995 and December 1999 from the site meteorological tower. All releases were considered to be ground level. An attachment to the August 4, 2000 letter provides a description and derivation for each of the computer code inputs. The χ/Q values (sec/m^3) determined are:

	Atmos. Dump Valve		MS Safety Valves		MSL Break	
	Intake 1	Intake 2	Intake 1	Intake 2	Intake 1	Intake 2
0 to 2 hrs	6.31E-4	4.78E-4	8.05E-4	5.91E-4	5.48E-4	4.22E-4
2 to 8 hrs	3.65E-4	2.75E-4	4.64E-4	3.37E-4	3.23E-4	2.51E-4

The staff reviewed the use of the ARCON96 computer code and the code input information provided by Entergy and found the code use and code inputs to be consistent with current staff positions on the use of ARCON96. The staff performed a qualitative review of the code results and deemed them to be reasonable. The staff finds the revised χ/Q values acceptable.

3 LICENSEE COMMITMENTS

The staff did not rely upon any licensee commitments in finding the proposed setpoint acceptable with regard to radiological consequences.

4 CONCLUSION

Based on the information provided by Entergy related to the proposed changes in setpoints associated with a replacement of the steam generators, SPSB finds reasonable assurance that the radiological consequences of anticipated accidents at ANO-2 will continue to be less than the dose guidelines of 10 CFR Part 100 and the criteria of 10 CFR Part 50, Appendix A, GDC-19 and Section 6.4 of NUREG-0800.

Principal Contributor: S. F. LaVie

Date: August 2000