July 21, 2000

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 - DENIAL OF AMENDMENT RE: LICENSE CHANGE FOR CYCLE 14 OPERATION BASED ON A RISK-INFORMED DEMONSTRATION OF PREDICTED STEAM GENERATOR TUBE INTEGRITY (TAC NO. MA8418)

Dear Mr. Anderson:

By your application dated March 9, 2000, as supplemented by letters dated April 11 and 28, May 30, June 20, 22, 23 (two letters), and 30, and July 7, 8, and 11, 2000, you requested a license amendment that would permit operation of the reactor based on a risk-informed demonstration that predicted steam generator tube integrity, with consideration of eggcrate axial flaws, is adequate to meet Regulatory Guide 1.174 numerical acceptance criteria.

After careful review, the Nuclear Regulatory Commission has concluded that your request cannot be approved. The basis for this finding is documented in the enclosed Executive Summary and Safety Evaluation.

A copy of the Notice of Denial of Amendment to be published in the *Federal Register* is enclosed for your information.

Sincerely,

/RA/

Stuart A. Richards, Director
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures: 1. Executive Summary
2. Safety Evaluation
3. Notice of Denial

cc w/encls: See next page

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EXECUTIVE SUMMARY

On March 9, 2000, Entergy Operations, Inc. (the licensee) filed an application to amend the technical bases for its license to operate the Arkansas Nuclear One, Unit No. 2 (ANO-2) nuclear reactor. The application addressed the principles for integrated decision-making, outlined in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (RG 1.174). The requested change would allow the licensee to operate the reactor until a planned refueling outage in September 2000, without shutting down to perform an inspection of the steam generator tubes. ANO-2 has been experiencing steam generator tube cracking at the "eggcrate" support structures in its steam generators, and the period between inspections necessary to detect and plug tubes as they fall below the strength requirements in the plant's current licensing basis have become increasingly shorter.

The U.S. Nuclear Regulatory Commission's staff found the original application to be comprehensive and well organized. However, the staff identified several deficiencies in the application during its review of the licensee's data, logic, and computational processes used to quantify the risk and assess the remaining safety margins.

The most serious deficiency is related to the probability of detection (POD) of significant flaws by the steam generator tube eddy current inspection conducted in January 1999 (2R13 outage). As part of its risk-informed review process, the staff concluded that the conditional probability of a tube rupture under steam-side depressurization conditions would need to be less than 0.05 to meet the guidelines in RG 1.174. This conditional probability is equivalent to a 95 percent probability that the last inspection has detected all defects that, during the current period of operation, could grow to the size that would rupture if a design-basis event were to occur. Based on the staff's review of the licensee's operational data for POD, the licensee did not demonstrate that its eddy current examination provided this level of effectiveness.

The process the licensee used to project the sizes of flaws in the tubes during the current period of operation (second half of Cycle 14), when applied to the previous period of operation (first half of Cycle 14), under-predicted the results of the November 1999 (2P99 outage) inspection. In response to this conflict, the licensee stated that the inspection process that was used in 2P99 was substantially improved in several ways over the process used in 2R13. Based on a review of the licensee's methods and data, the staff concluded that the significant factors, including, among other things, the signal-to-noise ratio associated with the eddy current examinations, were not improved, and the licensee did not demonstrate a significant improvement in the inspection process.

In addition, a significant change in the 2P99 inspection results for steam generator "B," in the absence of similar changes in steam generator "A," suggests to the staff that some change in the degradation process may be taking place in steam generator "B." Although the results of the licensee's recent in-situ pressure tests (on the most severely degraded tubes detected at each outage) have shown burst pressures at least 1.4 times that needed to survive design basis accidents (the design basis acceptance criterion), the change in the inspection results create doubt that burst pressures will remain acceptably high in the future. The staff's assessment of the licensee's inspection process is that it could not be demonstrated to provide the high probability for detecting flaws necessary to meet the guidelines of RG 1.174.

Other deficiencies were found with the thermal-hydraulic analyses used by the licensee to predict the physical conditions that would be experienced by the tubes in the event of a core damage accident. This is important to the assessment of the performance of the tubes as part of the containment boundary for these types of accidents. (Although such accident sequences are beyond the design basis of current reactors, they have been shown to be important contributors to public risk and are, therefore, evaluated for risk-informed applications.)

In an effort to address both the thermal-hydraulic and tube flaw size issues for core damage accidents, the licensee amended its application. The amendment took credit for (1) recent modifications to the plant, and (2) its operating procedures for depressurizing the reactor in the event that such an accident progressed to the point of overheating the reactor core. The new procedures are designed to prevent conditions that would cause flawed tubes to fail during core damage accidents. The licensee assessed the human error probabilities for the revised procedures and performed thermal-hydraulic simulations to demonstrate that the new strategy would be reliable and effective enough to preclude significant concern about the performance of flawed tubes during core damage accidents.

The staff reviewed this information and performed confirmatory thermal-hydraulic analyses. The staff concluded that the licensee's procedure is sufficiently reliable, and effectively diminishes the thermal-hydraulic challenge to the tubes. Therefore, the staff believes that implementation of the new procedure provides a significant reduction in the estimated probability of a large release of radioactive material from ANO-2 due to a core damage accident, even if the reactor has severely degraded steam generator tubes. However, these modifications did not resolve the staff's concerns regarding the POD and the margin for steam system depressurization events. Accordingly, the requested amendment is denied.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO APPLICATION FOR AMENDMENT TO

FACILITY OPERATION LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By letter dated March 9, 2000, as supplemented by letters dated April 11 and 28, May 30, June 20, 22, 23 (two letters), and 30, and July 7, 8, and 11, 2000, Entergy Operations, Inc. (Entergy Operations or the licensee), submitted a request for changes to the Arkansas Nuclear One, Unit No. 2 (ANO-2), license. The requested changes would revise the license as follows:

> For Cycle 14 only, Entergy Operations shall be permitted to operate the reactor based on a risk-informed demonstration that predicted steam generator tube integrity, with consideration of eggcrate axial flaws, is adequate to meet Regulatory Guide 1.174 numerical acceptance criteria. In accordance with Principle 5 in Regulatory Guide 1.174 concerning monitoring operational experience to ensure that performance is consistent with risk predictions, if Entergy Operations plugs or repairs steam generator tubes during Cycle 14, then the steam generators shall be reinspected to the extent necessary to verify that they have been returned to a condition consistent with the risk assessment.

2.0 BACKGROUND

2.1 Plant Condition

The ANO-2 steam generator (SG) tubes have experienced many degradation mechanisms. The active degradation mechanism that is limiting ANO-2's operating cycle length is axial outside diameter stress corrosion cracking (ODSCC) at the hot leg eggcrates. During the November 1999 (2P99) mid-cycle outage, the licensee performed a bobbin probe inspection of 100% of the SG tubes from the tube end on the hot leg to the seventh hot leg eggcrate for the purpose of detecting axial ODSCC at the eggcrates. All eddy current distorted signals that were identified with the bobbin probe, regardless of their location, were further inspected with a rotating pancake coil (RPC) probe. The RPC probe is more effective at characterizing eddy current signals. Decisions on whether the tube was flawed, and therefore required plugging or repair, were based on the RPC data. In addition to the bobbin inspections and follow-up RPC inspections, the licensee performed a limited RPC inspection in the “A” SG, at the top-of-tubesheet, to evaluate the potential for leakage due to circumferential stress corrosion cracking.

Prior to the 2P99 mid-cycle outage, the licensee determined that these circumferential flaws were not expected to be an issue with respect to the three times normal operating pressure (3ΔP) structural integrity criterion set forth in the design basis.

The licensee identified a total of 234 axial ODSCC indications in SG tubes at the hot leg eggcrate intersections, all of which were plugged prior to restart from the 2P99 mid-cycle outage. This was a significant increase from the number of axial eggcrate indications identified in the previous inspection in January 1999 (2R13), and significantly more than predicted by the licensee's operational assessment performed to justify operation until the 2P99 outage. Table 1 provides a historical list of axial eggcrate indications going back to the March 1998 (2P98) mid-cycle inspection. The table clearly shows that a significant increase in axial ODSCC indications occurred in SG "B" during 2P99. A limited number of flaw indications were also identified and tubes plugged at other locations within the SG. The licensee determined that these were not a challenge to the 3ΔP structural integrity criterion and are not the subject of the licensee's risk-informed license amendment application.

TABLE 1

INDICATION HISTORY OF AXIAL ODSCC AT EGGCRATES			
YEAR	**OUTAGE**	**SG "A"**	**SG "B"**
1998	2P98	45	74
1999	2R13	38	71
1999	2P99	49	185

The licensee identified tubes with significant degradation due to axial ODSCC at the eggcrates, and performed in-situ pressure testing on six SG tubes to assess whether (1) they would have leaked at main steam line break (MSLB) pressure differentials, and (2) they could meet the 3ΔP structural integrity criterion.

The licensee concluded from its eddy current data that the flaws in these six tubes were the most significant flaws identified during the 2P99 mid-cycle outage. None of the six tubes leaked at MSLB pressure differentials. Five of the tubes withstood 3ΔP differentials without bursting. In-situ pressure testing on the sixth tube, tube R72C72 in SG "B", was not completed to 3ΔP due to leakage in excess of the testing equipment capability. The licensee reviewed the in-situ pressure test results to determine the impact it would have on the acceptable operating cycle length following restart from the 2P99 mid-cycle outage. The licensee's conclusions in this regard are discussed in the next section of this report.

2.2 Operational Assessment for Current Operating Period

The licensee submitted a report dated December 21, 1999, that documented, in part, its preliminary evaluation, which predicted the period of operation allowable following restart from the 2P99 mid-cycle outage. The licensee concluded that the plant could operate for seven months before the next inspection (i.e., until June 25, 2000). This conclusion was based, in part, on the licensee's assumption at the time that tube R72C72 burst during the 2P99 outage at the highest in-situ test pressure reached.

Following the December 1999 submission, the licensee continued to investigate the circumstances related to the SG tube degradation and the results of the 2P99 mid-cycle outage inspection. Its findings from this investigation were documented in an operational assessment and submitted to the Nuclear Regulatory Commission (NRC or the Commission) in a letter dated February 11, 2000. The operational assessment documented refinements in the licensee's ability to predict flaw growth rates and the inspection's detection capability, as well as refinements in the estimates of the burst strength of tube R72C72. The licensee performed an analysis to demonstrate that tube R72C72 did satisfy the 3ΔP criterion, despite the difficulties encountered during the in-situ pressure test. As documented in the operational assessment, the licensee concluded that ANO-2 could operate within the 3ΔP criterion for the full cycle.

The staff does not agree with this conclusion. Although the staff did not believe ANO-2 could operate within the 3ΔP criterion for the full cycle, the staff did accept the licensee's February 11, 2000, operational assessment for the near term. A summary of the staff's assessment and position on the February 11, 2000, submission is contained in two NRC letters to Entergy Operations dated May 2 and June 23, 2000 (accession numbers ML003710343 and ML003726321, respectively).

2.3 Risk-Informed Application

As an alternative to the deterministic demonstration that the tubes in ANO-2 SGs will continue to meet design margins until the scheduled shutdown in September 2000 (2R14), the licensee submitted a risk-informed application to change the basis for continued operation. The application conforms to the guidance in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (RG 1.174). It addresses the five principles in that guide for integrated decision-making in a risk-informed manner. Those principles are:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored using performance measurement strategies.

Much of the effort in the licensee's application and the staff's review was directed at Principle 4. In order to address that principle, the licensee performed extensive analyses of the difference in risk between two options. The first option evaluated was a plant shutdown in the spring of 2000 (2P00) to perform a tube inspection, and to repair any defective tubes before returning to power and completing the fuel cycle in September 2000. The second option is to run until the September shutdown without any further tube inspections.

The risk equation has several distinct parts, of which three typically dominate the result:

1. Spontaneous tube ruptures that are not properly mitigated and result in core damage with bypass of the containment by the radioactive materials released from the reactor core.

2. Steam system depressurization events (such as the MSLB accident in the plant's design basis), which induce tube rupture by increasing the pressure loading on the tube. Failure to properly mitigate these events also leads to core damage with bypass of the containment by the radioactive materials released from the reactor core.

3. Core damage accidents that occur with the reactor coolant system (RCS) at high pressure and the SGs dry, which can induce tube rupture by increasing the temperature of the tubes, which decreases their strength.

Of these, the first two increase core damage frequency (CDF) and large early release frequency (LERF) for radioactive materials. The third is already a part of the CDF calculated by the plant's probabilistic risk assessment (PRA), but it increases LERF by converting a contained core damage accident to one that is not fully contained. As tubes degrade during service, they first become susceptible to the third type of sequence, then to the second, and finally to the first: spontaneous rupture during operation. Usually, the degree of degradation in the SG tubes is not severe enough to make the CDF increases approach the acceptance guidelines as closely as the LERF approaches its guidelines.

Section 3.0 of this Safety Evaluation (SE) addresses the quantitative evaluation of the risk increase; the other principles are addressed in Section 4.0.

2.4 Amended Application

During the review process for the original application, it became apparent that difficulties in resolving deficiencies with the licensee's thermal-hydraulic analyses and tube flaw detection and growth rate analyses could not be resolved in the time available. The licensee modified the plant and its procedures to allow operator action to depressurize the RCS during core damage accidents. These actions changed the thermal-hydraulic sequences and flaw sizes that needed to be considered, which made it easier to complete the analysis for severe accidents. The staff reviewed the amended application to determine if that action would be successful in preserving tube integrity for those accidents and would have a sufficiently high probability of being completed in a timely manner to reduce the risk increment to acceptable levels, in accordance with Principle 4. The staff also reviewed the licensee's analyses of flaw detection probability and growth rates to ensure that sufficient strength margin and leak tightness would be available for design basis accidents (DBAs) throughout the remainder of the operating period, without requiring an inspection at 2P00.

3.0 TECHNICAL EVALUATION

3.1 Overview and Scope of Review

The licensee's application is a comprehensive and well-organized document that addresses all of the risk-informed decision-making principles specified in RG 1.174. In addressing Principle 4, the size of the risk-increase, the application provides a very complete accounting of the various accident sequences that could create increases in CDF and LERF due to SG tube ruptures. Most of these sequences were originally modeled in the licensee's PRA without consideration of tube ruptures, but were reexamined for this application to include the potential effects due to tubes weakened by flaws. The application appropriately presents this information in a manner that directly addresses the difference in risk between the two potential courses of action (to reinspect at 2P00 or to not inspect before September 2000). The staff finds the scope and overall logic of the application to be appropriate to the request being made. The staff, therefore, focused its review on the assumptions and analyses that were used in quantifying the risk increment.

An important aspect of this application is that it predicts a small change in risk by predicting a high probability of tube failure, whether the SGs are reinspected at 2P00 or not, for some sequences that were found to have low tube failure probabilities in plants previously analyzed by the staff. This makes the review more difficult for the staff because assumptions that previously were "conservative" (i.e., increased failure probability) are potentially "non-conservative" for this review. For example, flaws that propagate completely through the tube wall and leak, but are too short to burst during severe accidents, may still affect the LERF by several phenomena. We currently do not have sufficient information to accurately evaluate these phenomena, which include the effect of leakage on the temperature of gases reaching the tubes, and the effect of jets of hot gas and particles on intact tubes adjacent to the leak. So, through-wall (TW) flaws that are over 0.25 inches long have previously been treated as if they were tube ruptures for purposes of severe accident analyses. This assumption was considered to conservatively bound the TW flaw length that would actually create the effects of concern, based on the available expert judgment that 0.25 inch long flaws actually were too short to produce these effects. However, if this assumption erroneously predicts that tubes will always fail during a particular sequence when actual bursts would only be predicted near the end of the operating period, then this assumption would be non-conservative with respect to the analysis for this application, i.e., the change in burst probability at the end of the cycle would be underpredicted.

The licensee's analysis also results in a small risk increment due to applying a slow growth rate for the flaws that are in the SGs. This is an important factor for accident sequences where the probability of failure is not already predicted to be near maximum at the beginning of the operating period.

Another important parameter resulting in the licensee's estimate of a small risk increment is a high probability of detection (POD) for flaws of significant size, so that flaws left in service at the beginning of the operating period cannot grow large enough to be susceptible to steam system depressurization accidents or rupture spontaneously.

The staff focused its review efforts on these factors to determine whether the methods and data used to quantify the risk increment were adequate to demonstrate that the risk increase would be small.

3.2 Review of Application Revision

As discussed in detail in later sections of this SE, the staff's review of the original application revealed deficiencies in the licensee's thermal-hydraulic simulations of some of the severe accident sequences. The staff's review also revealed discrepancies in the licensee's characterization and quantification of flaw sizes, growth rates, and POD during the most recent inspection. These issues are difficult to resolve for the flaw sizes that are susceptible only to severe accidents. In order to simplify the process for resolving these difficult issues to confirm that the risk increment is small, the licensee adopted a revised approach to minimize the risk from severe accidents. That approach uses an operator action to depressurize the RCS in the event that a high-pressure core damage accident reaches the stage when core damage begins.

In order to take credit for operator action to depressurize the reactor, the licensee had to consider the conditions that were responsible for creating the accident sequence. The severe accident sequences of concern for tube integrity are those where core damage occurs when the reactor is at high pressure and the SGs are dry and depressurized, called high/dry sequences. In typical PRAs for pressurized water reactors (PWRs), these sequences are most likely created by loss of off-site power and failure of the emergency diesel generators, resulting in a station blackout (SBO) type sequence. However, the ANO-2 PRA indicates that the dominant sequence for that plant is failure of one of its two safety-grade direct current (DC) electrical buses. This failure, plus a variety of consequential and independent failures, leads to loss of some alternating current (AC) power and high/dry conditions at the time of core damage. So, this type of sequence differs from the typical SBO sequence in that one of the DC buses is not functional.

ANO-2 needs power from both DC buses in order to depressurize the reactor. Unlike most PWRs, ANO-2 does not have power operated relief valves (PORVs) on its pressurizer. It does have a vent path that is normally blocked by two DC-powered motor operated valves (MOVs). One valve is powered from each DC bus, so both need to be available to depressurize the RCS. The licensee had not originally sought credit for operator action to depressurize because the most probable high/dry core damage sequences did not have power available on both DC buses.

In order to allow credit for operator action to depressurize the RCS, the licensee has modified the plant to allow temporary jumpers to supply power from the functional DC bus to the vent valve on the non-functional bus. The licensee has also modified the plant procedures to ensure that a trained person is always on shift, to identify when that person should go to the location for jumper installation, and remain there until instructed to install the jumper.

Staff review of the revised application focused on two areas. The first was the effectiveness of the depressurization operation in reducing risk from high/dry core damage sequences. This focused on the probability that a human will actually perform the action in a timely manner when required, and on the effectiveness of the depressurization in protecting the tubes from failure during and subsequent to the depressurization. The second area of focus was on the level of confidence that the last tube inspection can provide that flaws will not degrade to a level of unacceptable margin for DBAs such as MSLB. This addresses the conditional probability of tube rupture under MSLB conditions. It also addresses the amount of leakage that would occur from tubes that develop TW cracks but do not rupture under MSLB conditions.

In the sections that follow, the thermal-hydraulic, tube integrity, and probabilistic evaluations are discussed. These sections first address the original application to document findings and deficiencies encountered. They then address the revised application review and findings. This provides perspective on the issues involved and the benefit of the revised approach to resolving them. In the concluding sections addressing the five principles for making risk-informed regulatory decisions and overall staff conclusions, only the revised application approach is assessed for adequacy.

3.3 Thermal-Hydraulic Analyses for Original Application

The application identified a variety of thermal-hydraulic cases to represent the conditions in the RCS during the severe accident sequences that were identified from the plant's PRA as potentially challenging to the integrity of flawed SG tubes. The application provided results of computer simulations of each case, using the Modular Accident Analysis Program (MAAP), Version 4.0.3. These results were combined with the licensee's projections of tube flaw size and growth rates to produce their estimates of the probability that a tube would be the first part of the RCS pressure boundary to fail during each type of accident.

Staff review of the thermal-hydraulic analyses was conducted with the aid of independent thermal-hydraulic simulations performed at the Idaho National Engineering and Environmental Laboratory, using the SCDAP/RELAP computer code. Initially, cases for the staff analyses were configured to mimic the MAAP analyses in the application. However, as described below, the staff found it necessary to investigate the cases with medium RCS pressure in a more realistic manner than was used for those cases in the application.

Two classes of thermal-hydraulic cases are discussed below: those for which the RCS pressure remains at the pressurizer safety valve setpoint until the RCS pressure boundary fails due to creep rupture, and those during which RCS pressure slowly decreases due to a small opening such as a safety valve that fails to close completely. Due to the design of the reactor coolant pump seals in ANO-2, the probability of them developing large leak rates was considered low enough to be insignificant to the results.

3.3.1 High RCS Pressure Cases

The licensee's analyses addressed several cases with the RCS pressure at the pressurizer safety valve setpoint pressure. These included cases with the steam-side depressurized in one and both SGs and with both clear and intact "loop-seals," which are created by water trapped in the intake to the reactor coolant pumps. Without loop seals, steam can circulate around the full SG loop paths in the RCS, which leads to much more rapid heating of SG tubes than would result with the partial-loop path that occurs when the loop seals block full loop circulation.

SCDAP/RELAP Code analysis of this type of sequence indicates that loop seals in ANO-2 are not likely to clear out before the surge line fails by creep. Therefore, the staff assumed that the licensee's frequency for high RCS pressure cases with cleared loop seals should be treated as additions to its frequency for sequences with intact loop seals.

The results of the staff's analysis of tube failure potential for these sequences were generally similar to the licensee's results, but had some important differences. As often experienced in previous comparisons of MAAP results to SCDAP/RELAP results, MAAP indicates that an RCS hot leg pipe will be the first part to creep rupture, while SCDAP/RELAP predicts that the surge line will be first. (Based on detailed staff review of MAAP analyses submitted by the licensee to support the amended application, it appears that MAAP does calculate surge line failure, but the licensee does not credit it or report it due to concerns about the adequacy of the failure model for the surge line.) When SG tube flaws are considered, both codes agree that a tube would fail first if it contains a large enough flaw. However, the codes differ with respect to the size of the flaw that is necessary to cause a tube to fail first. The licensee's calculations indicate that sequences with depressurized SGs and intact loop seals would have a high probability of failing tubes with flaws that are less than 40% deep. The staff's analyses indicate a low probability of failing such flaws, although the ANO-2 calculations did indicate a greater challenge to the tubes than similar calculations had indicated for the Westinghouse-type reactors previously analyzed by the staff. There are several differences between the assumptions and techniques used by the licensee and staff that create these different results. They include assumptions about steam mixing in the SG inlet plenum and tube-to-tube temperature variations, which are not clearly resolvable on the basis of existing information. These differences become inconsequential due to the plant modifications described in the amended application.

3.3.2 Medium RCS Pressure Sequences

The licensee's MAAP analyses for sequences that have a slowly decreasing RCS pressure were simplified by artificially decreasing the pressurizer safety valve setpoints to 1,400 pounds per square inch (psi) from 2,500 psi. The staff chose to simulate these sequences more realistically, by introducing a fixed-size hole in the pressurizer, such as would result from a safety valve that stuck partially open during the sequence.

This created substantial differences in the results, due to the behavior of the loop seals. In the licensee's approach, the loop seals are created with water that has equilibrated to the lower RCS pressure, which is maintained throughout the remainder of the sequence. In the staff's analysis, the loop seals are established at the normal RCS pressure, and must cool by evaporation to accommodate decreasing RCS pressure. This creates a mechanism for clearing the loop seals. Because the MAAP code does not model the physical phenomena necessary to predict loop seal clearing, it is not capable of adequately evaluating these depressurization sequences.

The staff analyses included four cases that simulated the pressurizer safety valves sticking open three different times during the sequence and with two different effective discharge area areas for one of those times. All four cases indicated that the loop seals in both loops would clear during the transient, but which loop cleared first cannot now be predicted. In the staff's analyses for these sequences, the unflawed SG tubes did not fail as a result of loop seal clearing. If a SG remained pressurized on the steam side, this appeared to be adequate to protect flawed tubes. For cases with depressurized SGs, the size of the flaw that would cause a tube to be the first RCS pressure boundary failure varied substantially from case to case. Tube failure probability appears to vary substantially over the range of uncertainty about the size of the flaws that are in the ANO-2 SGs at various points in the current operating period.

3.3.3 Conclusions of Thermal-Hydraulic Analyses for Original Application

The staff concludes that the licensee's thermal-hydraulic analyses are insufficient to demonstrate that there is little change possible in the conditional probability of flawed tube failure during severe accident sequences over the current operating period. This conclusion is, in part, due to the unrealistic nature of the licensee's analyses for the medium RCS pressure sequences, and partly due to the difference in the sizes of the flaws that the licensee's and the staff's analyses found to be susceptible to the high RCS pressure sequences. In addition, the staff notes that there is a large difference between the number and sizes of flaws that the licensee found in the "B" SG during the 2P99 inspection and the licensee's characterization of the flaws that were returned to service at the beginning of the current operating period. If the flaws that the licensee found during the 2P99 inspection are indicative of what the flaws will be like at the end of the current operating period, as the staff expects, that would mean that the strengths of the flawed tubes might change substantially over the current period of operation. Accordingly, it is difficult to settle on a small range of flaw sizes as the only ones for which the thermal-hydraulic analyses must be appropriate. Because the failure probability result is very sensitive to both the sizes of the flaws that are present and the tube temperatures estimated by the thermal-hydraulic analyses, the high degree of uncertainty in both of these parameters for ANO-2 makes it very difficult to reach a technically defensible conclusion.

3.3.4 Thermal-Hydraulic Analyses for Amended Application

Due to the uncertainties in the flaw size distribution and the conclusions of the thermal-hydraulic analyses for the original application, the licensee amended its application to credit modifications to the plant and its procedures to allow for operators to depressurize the reactor during severe accidents. This has the benefit of reducing the severity of the challenge to the SG tubes.

3.3.4.1 Licensee's Analysis Using MAAP

The licensee performed a number of severe accident analyses with the MAAP code to assess the effectiveness of RCS depressurization to avert a temperature-induced SG tube rupture. The purpose of these analyses was to evaluate the time available for plant personnel to start RCS depressurization following the onset of conditions or cues that would indicate that RCS depressurization was to be performed, i.e., dry SGs coupled with core exit temperatures above 800 °F. The limiting severe accident event analyzed was the high/dry/low scenario: <u>high</u> RCS pressure (prior to RCS depressurization), <u>dry</u> SGs, and one SG depressurized (at <u>low</u> pressure). Four analyses were performed: (1) no operator action to depressurize the RCS, (2) RCS depressurization 15 minutes after the cues, (3) depressurization at 30 minutes after the cues, and (4) depressurization at 45 minutes after the cues.

The RCS was depressurized in the simulation by opening the emergency core cooling system (ECCS) vent path on the pressurizer. The effective area of the vent path was set to be equivalent to a 1.87-inch diameter opening (an area of 0.0191 feet^2). This is about one-half of the nominal vent path area. It was modeled this way to include the effects of the additional loss factors in the vent path from the pressurizer to the quench tank (piping lengths and sizes, elbows, reducers, etc.). Loop seal clearing was not predicted to occur in the MAAP analyses. MAAP models cold leg loop seal clearing by specifying a minimum volume of water in the cold leg loop seal, about 71.2 feet^3. If the volume decreases below this value, then gas flow through the loop seal region is established. At the end of blow down, MAAP predicted that about

95.4 feet^3 of water remained in each cold leg. Without loop seal clearing, hot gases exiting the reactor core are drawn to the surge line from both hot legs and not through the SG tubes. MAAP computes a creep rupture index for the surge line, the hot legs (broken loop, intact loops), and the tubes in each SG. Surge line creep rupture, while computed, was not credited by the licensee.

The RCS depressurization cues were calculated to occur about 90 minutes after the onset of the severe accident event. Without RCS depressurization, creep rupture of a SG tube was predicted to occur 132 minutes after the onset of the severe accident event, with surge line failure at 138 minutes and hot leg failure at 151 minutes. With RCS depressurization at 15 minutes following the cues, surge line failure was predicted at 148 minutes and hot leg failure at 192 minutes. No SG tube failures were predicted at the end of the MAAP run, which was 212 minutes into the sequence. With RCS depressurization at 30 minutes following the cues, surge line failure was predicted at 153 minutes and hot leg failure at 181 minutes. No SG tube failures were predicted at the end of the MAAP run, which was 212 minutes into the sequence. With RCS depressurization at 45 minutes following the cues, creep rupture of a SG tube was predicted to occur 132 minutes after the onset of the severe accident event, with surge line failure at 140 minutes. No hot leg failures were predicted at the end of the MAAP run, which was 156 minutes into the sequence. Based on the MAAP analyses, the operator needs to start RCS depressurization within 30 minutes following the cues to avert a temperature-induced SG tube rupture.

3.3.4.2 Staff Analysis Using SCDAP/RELAP

The staff has conducted thermal-hydraulic analyses to confirm that the operator actions are effective in preserving the tubes in the event of a core damage accident with the RCS at high pressure. The beginning of this sequence is modeled the same as the high RCS pressure sequence in the original application, except that the SG tubes are modeled as 17.97% plugged in SG “A” and 17.43% plugged in SG “B”. After the reactor trips, both SGs reach their safety valve setpoints and begin to depressurize because these valves are modeled to remain open after their first lifts. This cools the RCS until the SGs dry out. Then the RCS reheats until its pressure reaches the pressurizer safety valve setpoint. The RCS loses inventory through the pressurizer relief valves as they cycle to maintain pressure. When the core is uncovered due to coolant loss, it begins to heat-up further. However, when it has heated to the point that the core exit thermocouples (CET) read 800 °F, the new procedure instructs the operators to open the ECCS vent valves to depressurize the RCS. The licensee’s analyses of human error probabilities indicates that a period of 27 minutes is required before the probability of this action not being taken has fallen below 0.25. So, the SCDAP/RELAP model was modified to open the valve 27 minutes after the central core exit node reached 800 °F (700 K). For this simulation, creep damage to the tubes was tracked assuming 14 different levels of degradation; from none to almost enough to cause spontaneous rupture at normal operating conditions.

Based on insights from the licensee’s MAAP analyses, the staff investigated the effect of the ECCS vent path piping between the pressurizer and the pressurizer relief tank. Due to the small pipe sizes and long path length, the flow friction in the piping was more restrictive than the ECCS vent valve throat areas. A separate SCDAP/RELAP model of the vent path was used to determine an effective flow area of that path to be used in the severe accident analysis. The result was equivalent to throttling both pressurizer relief valves at about 63% of their full-open area.

The results of the staff's SCDAP/RELAP simulation indicate that the RCS depressurization begins at 7,822 seconds (2 hours, 10 minutes) after the reactor trips and reaches the safety injection tank (SIT) pressure by 8,494 seconds (about 11 minutes later). At that time, the SITs begin to discharge very cool water (about 120 °F, 322 K) into the RCS, causing pressure fluctuations by condensing steam at the interface with the cool water, and then creating more steam as the water contacts core surfaces. When more steam is created, the RCS pressure rises and the SIT injection ceases until enough of the steam is released through the ECCS vent path to allow the RCS pressure to drop below the remaining pressure in the SITs. The SITs then inject more water, repeating the cycle at a reduced pressure level. This continues for the remainder of the analysis (to 49,330 seconds), by which time the SITs were still not completely emptied.

Flow of hot steam and gases from the core through the hot leg and surge line to the ECCS vent path heats the piping. The surge line is predicted to creep rupture at 20,790 seconds. However, this was not simulated in the model in order to determine when other failures would occur if the surge line failure did not relieve RCS pressure. The hot leg pipe between the reactor and the surge line is predicted to creep rupture at 31,100 seconds. Similarly, this hot leg failure was not simulated in the model. The hottest tubes in the SG on the pressurizer loop with stress magnification factors of 7.5 were predicted to creep rupture at 33,420 seconds. However, it should be noted that these tubes would have already failed due to the increased pressure differential caused by the initial depressurization of the SGs, as modeled in this analysis. They were included because different causes of SG depressurization may occur later than those modeled, and some of these mechanisms may result in a smaller pressure differential.

By the end of the analysis, the hottest tubes with stress multipliers of 6.0 and 6.5 had failed in the pressurizer and the other RCS loop, respectively. None of the tubes that would have been able to withstand the full RCS pressure with a depressurized SG at normal operating temperatures (i.e., those with a stress magnification factor less than 4.5) were predicted to fail by creep before the end of the analysis. Prior to the time of the first SG tube failure, the surge line was heated to temperatures near 1,700 K. Given that stainless steel melts between solidus and liquidus temperatures of 1,675 and 1,727 K, the possibility of surge line survival (from 20,790 seconds) to the time of the first SG tube failure (at 33,420 seconds) seems very unlikely. Furthermore, if a break in the RCS was (more realistically) introduced at the time of the predicted surge line failure, one would expect quick depletion of remaining SIT inventory with minimal re-pressurization (because a break as large as the surge line should readily vent most of the generated steam). A gradual boil-off would then be required before significant heating of any RCS piping could resume.

It is expected that the effects of any subsequent heating of the RCS piping would be focused on the surge line and hot leg, consistent with the break flow path. Therefore, it seems clear that operator actions to open ECCS vent valves (as modeled in this case) will result in surge line failure well before the first possible SG tube failure. In addition, any challenge to SG tube integrity appears to be minimal if the expected surge line failure occurs, indicating that the operator action considered in this case will be successful.

3.3.5 Conclusions of Thermal-Hydraulic Analyses for Amended Application

On the basis of our confirmatory thermal-hydraulic analysis, the staff concludes that the procedure to open the ECCS vent valves at ANO-2 would be effective in preserving the integrity of flawed tubes during high temperature challenges associated with severe accidents, if the action is taken within the time frame specified. However, if one or both SGs become depressurized before the RCS is depressurized by operator action, the tubes could still be ruptured by the higher differential pressure, as described for steam-side depressurization events in Section 2.3. The risk increment associated with pressure-induced tube ruptures during severe accident sequences is discussed in Section 3.5.2.

3.4 Tube Integrity Analyses

The staff has concluded that the ANO-2 SG tubes will not meet the structural integrity requirements of being able to withstand a pressure of 3ΔP (the current licensing basis) at the end of the current operating cycle. To address this, the licensee submitted a risk-informed license amendment application in accordance with the guidance in RG 1.174. The purpose of the proposed risk-informed amendment would be to modify the licensing basis for the plant for the remainder of the operating cycle. The criteria for evaluating the acceptability of the proposed change include demonstrating a minimal change in risk and maintenance of sufficient safety margin and defense-in-depth consistent with RG 1.174.

If the NRC had concluded that the license amendment application could be approved, a second mid-cycle inspection would not have been necessary. Entergy Operation's license amendment application identified an initial flaw size distribution (i.e., flaws not identified during the 2P99 SG inspection), growth rate distribution, and POD as inputs to the assessment process described above. Based on these inputs, combined with a thermal-hydraulic analysis, the licensee concluded that there was minimal change in risk over the operating period. The staff encountered several difficulties (discussed in more detail in Sections 3.4.2 and 3.4.4) in verifying the accuracy of the initial flaw size distribution and the POD distribution function, and, therefore, the change in risk over the operating period. These difficulties were expressed to the licensee through requests for additional information.

Initially, the licensee modified some assumptions and analyses in an attempt to address the staff's concerns. Eventually, the licensee chose to modify the plant and operating procedures in order to enable depressurization of the RCS during severe accidents. As set forth in Section 3.5.5 of this SE, depressurization of the RCS successfully protects the SG tubes during severe accident sequences with sufficient probability that the change in risk due to severe accidents will be small enough so as to be acceptable for the remainder of the operating cycle.

The licensee amended the license amendment application to credit operator actions for the RCS depressurization. This provided an approach to address severe accident concerns, leaving for evaluation the spontaneous tube rupture and MSLB. That is, the staff assessed the projected degradation to determine if the SG tube leakage integrity and structural integrity at the end of the cycle would be sufficient such that the risk contribution from spontaneous tube rupture and MSLB events is not significantly different than that expected under licensing basis conditions (3ΔP criterion).

The following sections summarize the staff's review and findings regarding the licensee's growth rate analysis, inspection capability, and leakage and structural integrity assessments.

3.4.1 Growth Rate Analysis

The growth rate of SG tube flaws plays an important part in a licensee's operational assessment. The licensee performed a growth rate evaluation for ANO-2 SG tubes based on a bobbin coil depth sizing methodology. The licensee sized flaws identified during the 2P99 mid-cycle outage using bobbin coil data to obtain maximum depths. The licensee then retrieved maximum flaw depths for those flaws from bobbin coil data from the 2P98 mid-cycle inspection. The depth from the 2P98 mid-cycle inspection was subtracted from the 2P99 depth to obtain the growth rate for that flaw. (At the time that the 2P98 bobbin coil inspection data was originally analyzed, the licensee did not believe there were flaws or distorted signals present with respect to certain tubes. However, with the knowledge of the presence of flaws in those tubes, the data from the 2P98 inspection was reevaluated at the identified locations in each tube and, within the inherent uncertainties of this approach, each flaw was identified and sized.)

The licensee presented a histogram of the growth rates of all analyzed 2P99 flaws. A wide range of growth rates was identified, ranging from approximately -32.5% TW per effective full power year (EFPY) to 42.5% TW per EFPY. Although the mean growth rate was estimated to be about 7% TW per EFPY, prior to the 2P99 inspection, the licensee assumed a growth rate of 28% TW per EFPY in its deterministic operational assessment, which justified operation until 2P99.

Following the 2P99 inspection, the licensee utilized a probabilistic approach for determining the growth rate of a flaw. The licensee used this approach because it believed the large amount of negative growth (seen in the growth rate histogram) is symptomatic of eddy current uncertainties mixed with actual physical growth. The licensee used a statistical optimization process to estimate actual growth, which was then used to develop a growth rate distribution. The licensee calculated the 95/95 value (95% probability with 95% confidence) to be 22.1% TW per EFPY. The licensee calculated the 95/50 value (95% probability with 50% confidence) to be 15% TW per EFPY.

The staff did not review the licensee's probabilistic growth rate approach in detail. The staff's confidence level in the growth rate estimate is limited because it is based on data from the eddy current bobbin coil probe, which is not qualified for sizing of cracks and produces depth estimates which can be unreliable for axial cracks. Instead, for the structural integrity analysis discussed in Section 3.4.6.2, the staff selected a more conservative growth rate estimate than the licensee's 95/95 calculation. Although the growth rate assumed by the staff does not bound the licensee's histogram of growth rates, the staff considers it to be conservative based on comparison with prior operating experience at ANO-2 and experience at other Combustion Engineering plants. Also, the staff assessed the sensitivity of the end of cycle (EOC) flaw depths to a range of growth rates and assessed a combination of detection thresholds and growth rates that bound the conditions found at the end of the last cycle of inspection. These calculations are discussed in more detail in Section 3.4.6.2.

3.4.2 Inspection Capability

The main purpose of the planned ANO-2 2P99 mid-cycle outage was to perform SG tube inspections due to the ongoing tube degradation at the hot leg eggcrates. To inspect for this degradation, the licensee performed a 100% bobbin probe inspection from the tube end on the hot leg to the seventh hot leg eggcrate. All bobbin coil signal distortions were reinspected with a RPC probe, which is more effective at characterizing eddy current signals. Decisions on whether the tube was flawed, and therefore required plugging or repair, were based on the RPC data.

To a large degree, the sensitivity of the bobbin coil probe determines the POD of the inspection. To resolve questions raised by the staff (discussed in Section 3.4.4) during its review of flaw distributions and related POD distributions supplied by the licensee for use in the risk assessment, the licensee stated that the 2P99 bobbin coil inspection was more sensitive than the 2R13 inspection, and thus had a higher POD. The licensee indicated that a different calibration standard was used in the 2P99 inspection, and use of this standard resulted in an improvement to the signal characteristics. More specifically, the licensee indicated that the bobbin signal amplitude, measured in volts, was on average larger than the signal amplitude experienced in previous outages, which resulted in a more sensitive inspection. The licensee identified several other lesser factors (e.g., training of analysts, 2P99 localized testing, etc.) that it believes may have also contributed to a more sensitive bobbin coil inspection during 2P99.

The staff performed an independent analysis to assess the validity of the increase in bobbin probe inspection sensitivity due to the new calibration standard. This analysis was performed through review of 2P99 bobbin probe eddy current data from ANO-2 SG tubes. The staff concluded that, in addition to determining whether the flaw signal amplitudes had increased, it was necessary to determine whether the signal-to-noise (S/N) ratio at the eggcrates had improved. If the S/N ratio did not improve, then the flaw signal amplitude increase has no effect on the sensitivity (POD) of the inspection. The staff assessed the difference in signal amplitudes (i.e., volts) when using the 2R13 and 2P99 calibration standards, and made the same observations as the licensee regarding the 2P99 increase in signal amplitude. However, the S/N ratio remained the same (i.e., the noise amplitude increased proportionally to the signal amplitude). The staff concluded that the new calibration standard used during the 2P99 inspection did not improve the bobbin coil's sensitivity to flaws as compared to the 2R13 inspection.

The staff reviewed the other factors identified by the licensee as contributing to a more sensitive bobbin coil inspection during 2P99 (e.g., training of analysts, 2P99 localized testing, etc.). Although training of analysts and other factors can have a positive effect on the sensitivity of eddy current inspections, it was not borne out in this case based on independent NRC review of eddy current data and the evaluation of the inspection results (i.e., comparison of distorted support signals versus RPC confirmation rates). For these reasons, as well as those set forth below, the staff concluded that these factors would not be expected to have a significant, if any, impact on the sensitivity of the bobbin coil inspection.

The staff compared the 2R13 and 2P99 bobbin coil distorted support plate indication (DSI) calls and RPC confirmation calls to determine whether this was indicative of an increase in the sensitivity of the bobbin coil inspection. A significant increase in the number of DSIs would potentially indicate an increase in the sensitivity of the bobbin coil inspection. However, this would need to be coupled with minimal change in the RPC confirmation rate. A significant increase in the RPC confirmation rate is more indicative of either an improved RPC inspection or the presence of a higher number of deep flaws that are more readily detected by RPC.

The licensee has stated that the RPC inspections in 2R13 and 2P99 are essentially the same. Therefore, if the sensitivity of the bobbin coil inspection had increased significantly in 2P99, the staff would have expected to see an increase in the number of DSIs, as well as minimal change in the RPC confirmation rate in 2P99 as compared to 2R13. As can be seen in Table 2, SG “A” inspection results may have been due to a slight increase in the sensitivity of the bobbin coil inspection. However, the SG “B” inspection results did not meet the above criterion, as there was a modest increase in the number of DSIs but a significant increase in the RPC confirmation rate.

TABLE 2

INSPECTION RESULTS

	2R13		**2P99**	
	SG “A”	**SG “B”**	**SG “A”**	**SG “B”**
DSI	163	158	265	202
RPC Confirmation	38	71	49	185
RPC Confirmation Rate	23.3%	44.9%	18.5%	91.6%

Based on the discussions above, the staff concluded there was no evidence of an increase in the sensitivity of the bobbin coil inspection during 2P99.

3.4.3 Possible Inspection Improvements

The previous section contains a discussion of licensee-indicated 2P99 inspection improvements. The staff concluded that there was no evidence of an increase in the sensitivity of the bobbin coil inspection. However, the staff does believe there are actions the licensee could have taken to increase the sensitivity and quality of its SG tube inspections.

The licensee’s report on its site-specific performance demonstration states that the bobbin coil mix channel residual and the presence of deposits can strongly affect detection. The staff believes the reduction of this eddy current mix residual at the eggcrates is a key to improving the sensitivity of the inspection. This could have been accomplished by the licensee through the creation of an improved calibration standard by adding a better eggcrate simulation (better than the drilled tube support plate currently used) and simulated deposits, and thereby significantly reducing the size of the eddy current mix residual at the eggcrates. This could have had a significant effect on the ability of the bobbin coil inspection to detect degradation. Currently, the eddy current mix residual is large enough that it can potentially mask significantly sized flaws at the ANO-2 eggcrates.

3.4.4 Staff Difficulties with Flaw Sizes, POD, and Growth Rates Used in the Risk Assessment

The licensee provided estimates of flaw growth rates, beginning-of-cycle (BOC) flaw distributions, and POD as a function of flaw size. These parameters are based on eddy current data from SG tube inspections, and were used by the licensee to conclude that flaws left in-service (i.e., not detected) following the 2P99 inspection were small and would grow slowly. Consequently, the licensee's risk assessment concluded that there was minimal change in risk over the operating period.

The flaw size data originally used by the licensee in its risk assessment was based on its "field analysis" of the RPC probe eddy current signals. The staff noted that assessment of flaw size data based on a "profile analysis" of the same RPC data resulted in deeper, and thus weaker, flaws. (Both "field analysis" data and "profile analysis" data for a flaw consists of an assessment of the length and depth of that flaw, and are collected from the same eddy current data. However, the "profile analysis" data is a more refined and precise analysis of the flaw's length and depth.)

At the staff's request, "profile analysis" data and subsequently revised PODs were provided for all SG "B" eggcrate flaws. The revised POD estimates were higher and resulted in a significantly reduced estimate of the number and size of flaws that were left in-service (i.e., present but not detected) following the 2P99 inspection. This is the BOC flaw distribution that was used to project a 2R14 EOC flaw distribution when combined with a growth rate distribution. The 2R14 EOC flaw distribution predicted significantly fewer and shallower flaws than those found during the 2P99 inspection. However, since the inspection process used during 2P99 was essentially the same as that used during 2R13, the 2R14 predictions would have been expected to match the 2P99 inspection findings.

In order to investigate the reason for this conflict, the staff requested that the licensee perform an analysis using its current flaw growth rate estimates to (1) determine the distribution of flaws that were left in-service following the 2R13 inspection (BOC 2P99 flaw distribution) based on the 2P99 as-found flaw distribution, and (2) use this estimated BOC 2P99 flaw distribution to estimate the actual 2R13 POD. This analysis (the "2R13 Lookback") resulted in a distribution of PODs as a function of flaw depth that had a maximum 2R13 POD of less than 0.4 for the largest flaws. This result is in conflict with the 2P99 PODs, which were estimated by the licensee as greater than 0.9 for similarly sized flaws.

To resolve this conflict, the licensee asserted that the 2P99 bobbin coil inspection was much more sensitive than the 2R13 inspection due to the use of a new calibration standard and several other factors, and that this accounted for the difference in PODs. The staff's assessment of these assertions can be found in Section 3.4.2; the staff concluded there was no evidence of an increase in the sensitivity of the bobbin coil inspection during 2P99.

The staff has identified significant conflicts that could not be resolved with the data the licensee provided to be used in the risk assessment. The staff considered other means of assessing the change in risk due to SG tube degradation. The staff's efforts in this regard are discussed in Section 3.4.6.

3.4.5 Leakage Integrity Assessment

The licensee calculated a projected EOC SG tube leak rate under postulated MSLB conditions in order to determine whether ANO-2 would be within the current licensing basis assumption of one gallon-per-minute (gpm).

For axial cracks at eggcrates, the licensee developed a BOC flaw distribution and assigned a probabilistic growth rate and material properties to each flaw to project an EOC distribution of flaws. It used the EOC population of flaws to calculate a leak rate under postulated MSLB conditions. Version 3.0 of the “Pipe Crack Evaluation Program” was used to compute flow rates through cracks as a function of pressure differential, temperature, crack opening area, and total TW crack length. EOC leakage for eggcrate flaws at postulated MSLB conditions, based on a 95/95 level of confidence, was calculated by the licensee to be slightly less than 0.05 gpm through one SG.

The licensee also calculated a projected EOC leak rate under MSLB conditions for all degradation modes (excluding eggcrates) for a full cycle of operation. The leak rate projections for these degradation modes was calculated to be 0.11 gpm through one SG. Therefore, the combined EOC leak rate for all degradation modes under postulated MSLB conditions calculated by the licensee would be 0.16 gpm, which is well within the one gpm assumption.

At the request of NRC staff, the licensee calculated the leak rate under MSLB conditions of one flaw (“B” SG, tube R8C134) identified during the 2P99 mid-cycle outage, which was not in-situ pressure tested. Based on the eddy current profile analysis data, the staff believed it was likely this tube would have leaked under postulated MSLB conditions. The licensee concluded that one quarter-inch of the flaw would have been 100% TW and leaking at a rate of 0.044 gpm at room temperatures. This is bounded by the licensee’s 0.05 gpm assumed leak rate for all eggcrate flaws.

The staff performed an independent calculation to determine the leak rate of tube R8C134 under postulated MSLB conditions. Assuming a quarter-inch flaw TW, the staff concluded the leak rate would have been 0.19 gpm. The flaw in tube R8C134 was approximately 0.53 inches long. The licensee assumed the flaw would only penetrate 100% TW for a quarter-inch. Based on flaw characteristics, and potential propagation of the flaw under high pressures, the staff believes it is possible the flaw may have penetrated 100% TW for up to 0.45 inches. Therefore, the staff performed other calculations for flaws up to 0.45 inches and estimated a leak rate of about 3.5 gpm. Although tube R8C134 was plugged during the 2P99 mid-cycle inspection, it is reasonable to assume that a similarly sized flaw with a similar leak rate could be present by the EOC 14, because there have been no improvements to the inspection process.

Discussions regarding the difference in leak rate calculations were held between the staff and the licensee. In order to address this difference, the licensee made a regulatory commitment to administratively lower the technical specification reactor coolant and secondary coolant specific activity limits to the values specified in its letter dated June 30, 2000. The licensee determined that these activity limits would allow SG accident leakage of up to 10 gpm without exceeding 10 CFR Part 100 guidelines. The calculations, performed to determine the activity limits that correspond to accident leakage, contain conservatisms and higher leakage could be accommodated by using more realistic calculations. The staff concluded that a projected 10 gpm EOC leak rate was reasonable.

Typically, this increase in postulated accident leakage would have to be analyzed under severe accident sequences. However, due to the licensee's actions to modify the plant and operating procedures in order to enable depressurization of the RCS during severe accidents, this analysis does not have to be performed for ANO-2.

Based on the staff's assessment and the licensee's regulatory commitment to administratively control the reactor coolant activity, the staff concluded there was reasonable assurance that SG tube leakage integrity would be maintained to meet the dose guidelines provided in 10 CFR Part 100 for the remainder of the current operating cycle.

The staff also concluded there was reasonable assurance that SG tube leakage integrity would be maintained to meet the control room operator dose criterion specified in General Design Criteria (GDC) 19 of 10 CFR Part 50 for the remainder of the current operating cycle. This conclusion is based on information provided in the ANO-2 Updated Final Safety Analysis Report for control room design and the licensee's regulatory commitment to administratively control reactor coolant activity.

3.4.6 Structural Integrity Assessment

3.4.6.1 Introduction

The licensee submitted a report dated December 21, 1999, that documented, in part, its preliminary evaluation, which predicted the period of operation allowable following restart from the 2P99 mid-cycle outage. The licensee concluded that the plant could operate for seven months before the next inspection. This conclusion was based, in part, on the licensee's assumption at the time that tube R72C72 burst during the 2P99 outage at the highest in-situ pressure reached during the test.

Following the December 1999 submission, the licensee continued to investigate the circumstances related to the SG tube degradation and the results of the 2P99 mid-cycle outage inspection. The licensee's findings from this investigation were documented in an operational assessment and submitted to the NRC in a letter dated February 11, 2000. The operational assessment documented revisions in the licensee's approach to predict flaw growth rates and the inspection's detection capability, as well as in the estimates of the burst strength of tube R72C72. The licensee performed an analysis to demonstrate that tube R72C72 did satisfy the $3\Delta P$ criterion, despite the difficulties encountered during the in-situ pressure test. The operational assessment concluded that ANO-2 could operate within the $3\Delta P$ criterion for the full cycle. The staff does not agree with this conclusion. A summary of the staff's assessment and position on the February 11, 2000, submittal is contained in two NRC letters to Entergy Operations dated May 2 and June 23, 2000.

The staff assessed ANO-2 SG tube structural integrity against two key principles in RG 1.174: (1) the proposed change must maintain sufficient safety margins, and (2) when a proposed change in the licensing basis results in an increase in CDF or LERF, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement. The staff performed two separate assessments of the first key principle (i.e., maintenance of sufficient safety margins). These assessments consist of a deterministic analysis of the SG tube operational assessment and a separate assessment of burst pressure predictions under MSLB conditions. These assessments are summarized in Sections 3.4.6.2 and 3.4.6.3 below.

The staff’s assessment of the second key principle (i.e., change in CDF or LERF) is based on a probabilistic analysis and is summarized in Sections 3.5.2.2 and 3.5.2.1.2.

3.4.6.2 Operational Assessment

The staff’s assessment of SG tube structural integrity included a deterministic assessment of margins in order to address Principle 3 in RG 1.174 regarding maintaining sufficient safety margins. In addition, sensitivity studies were performed to assist in evaluating the random variables in the probabilistic assessment and the licensee’s ability to accurately quantify them. These variables include growth rate distributions, eddy current uncertainties, and the POD distributions.

Both the December 21, 1999, and February 11, 2000, licensee submissions contained deterministic projections for the most significant flaw, in terms of flaw depth, at the end of the current operating cycle. These projections were developed based on assumed initial flaw depths and growth rates. The licensee correlates flaw depths to burst pressure, the parameter of interest in the analysis, through the use of the Framatome Burst Equation. The predicted tube burst pressures are then compared against pressure limits of interest (e.g., 3ΔP and 1.4 times MSLB pressure differentials).

In the December 1999 submission, the licensee determined R72C72's structural depth, based on its in-situ burst pressure. It then calculated a start of Cycle 14 flaw depth for R72C72 using the same assumed growth rate that was used to justify operation until the 2P99 mid-cycle outage. The licensee assumed this same initial flaw depth and growth rate to justify seven months of operation, at which time 3ΔP would be exceeded.

The staff reviewed these analyses and performed an independent assessment of the data. Using the licensee’s initial flaw depth and growth rate, which justified operation for seven months following restart from the 2P99 mid-cycle outage, the staff projected the tube flaw depth (burst pressure) for the EOC 14 (September 2000). Since this calculation is consistent with approximately seven months of operation within the 3ΔP criterion, the projected burst pressure at EOC is less than 3ΔP and less than 1.4 times MSLB pressure. The staff used this as a best estimate (“base case”) for the most structurally significant flaw expected to exist in the SG at the EOC 14, because this combination of initial flaw depth and growth rate corresponds with the condition of the tube (R72C72) at the end of the prior cycle of operation (2P99).

The staff also performed additional calculations to determine the sensitivity of the EOC burst pressure to different input assumptions. The additional calculations were performed using different growth rates and initial flaw depths from those used in the base case. While more conservative combinations of assumptions could have been used in these calculations, the staff believes that its assumptions for growth rates and initial flaw depths are appropriate to develop an understanding of the possible EOC conditions, given the uncertainties that arise from using plant inspection data.

For the first case, the staff increased the base case initial flaw depth by approximately 5% of the wall thickness and assumed the same growth rate as the base case. For the second case, the staff increased the base case initial flaw depth by approximately 5% of the wall thickness and decreased the growth rate to the licensee’s 95/95 value. For the third case, the staff increased the base case initial flaw depth by approximately 15% of the wall thickness and decreased the growth rate to the licensee's mean value. Cases 2 and 3 would have predicted tube R72C72 bursting below 3ΔP and near 1.4 times MSLB pressure if these cases had been applied in the half cycle leading up to 2P99, and these cases would have predicted EOC burst pressures below 1.4 times MSLB pressure when applied to the current half cycle through September 2000. Case 1 is the limiting case assumed, which bounds the 2P99 inspection findings, and resulted in predicted EOC 14 burst pressures substantially less than 1.4 times MSLB pressure.

The staff has concluded that if the plant were to continue operating until late September 2000 without another inspection, the projected EOC structural integrity margin against burst may be substantially less than the current licensing basis of 1.4 times MSLB pressure. As previously discussed, the staff's review indicates that the uncertainties in the POD and growth rate are large and must be considered when performing tube burst pressure predictions. The ability of the reduced factors of safety suggested by the staff's sensitivity studies to provide reasonable assurance that a tube would not fail under MSLB conditions is assessed in the next section.

3.4.6.3 Tube Burst Pressure Predictions

The Framatome Burst Equation was used by the licensee and the staff to estimate the burst pressure of a tube, based on the estimates of a flaw's length and depth. The estimates of the structurally significant length and depth values are derived from the licensee's profile analyses of the RPC eddy current data. The equation also requires an estimate of the tube material strength properties and has a constant that is used for fitting the measured data with some degree of confidence. The equation is

$$P_B = 0.58\ (\sigma_y + \sigma_u)\ (t / R_i)\ [\ K - L\ h / (L + 2\ t)]$$

where:

P_B = tube burst pressure
$(\sigma_y + \sigma_u)$ = yield plus ultimate material stress = S
t = tube wall thickness (0.048 inch)
R_i = inside tube radius (0.327 inch)
L = flaw structural length
h = ratio of flaw structural depth to tube thickness

The value of K has been derived for best estimate (1.104), 90% lower bound (1.014), and 95% lower bound (0.988). (This information was submitted by the licensee in Attachment 1 to a letter dated June 30, 2000.) Because the material stress properties are not precisely known for the tubes tested in-situ, that parameter is also sometimes used for fitting specific sets of data. The licensee has used a material property value for stress of 134,000 psi for the operating temperature calculations and a value of 144,500 psi for the room temperature in-situ test calculations. In order to get a close fit to the data, the licensee uses a value of K = 1.0 and then adds 1,080 psi to the value from the equation to best match the results to its data.

The staff has found that using a material stress value of 134,000 psi for the room temperature in-situ pressure test results provides a reasonable approximation to the best estimate for nine ANO-2 pressure test data with K = 1.104, just bounds that data with K = 1.014, and appears to be consistent with the 95% bounding expectation when K = 0.988. When the staff uses a material stress value of 144,500 psi, the results appear to generally overestimate burst pressure compared to the ANO-2 burst data. Therefore, the following staff analyses assume material stress of 134,000 psi and use the relationship of K to the fraction of data bounded as described above. The staff notes that its best estimate equation still produces an over-prediction of the measured burst pressure by 1,000 psi for one of the nine data points. This is most probably due to the uncertainties in the estimation of flaw size from eddy current data. It illustrates the need for proper consideration of the uncertainty in flawed tube strength calculations when making probabilistic evaluations. The comparison of calculated and measured burst pressures is provided in Table 3.

TABLE 3

COMPARISON OF CALCULATED TO MEASURED BURST PRESSURES

		S = 144500	S = 134000		
Tube	Measured	K = 1.104	K = 1.104	K = 1.014	K = 0.988
B 72 - 72	4147	4038	3744	2718	2420
B 85 - 87	3958	4933	4575	3548	3251
A 70 - 98	3250	3432	3183	2156	1860
A 16 - 56	3200	4498	4171	3145	2848
B 16 - 60	3975	3976	3687	2661	2364
B 37 - 67	4650	5002	4638	3611	3315
A 82 - 118	5000	5492	5094	4067	3770
B 96 - 116	8123	8553	7932	6906	6609
B 19 - 55	9810	10213	9471	8444	8148

In Section 3.5.2.2, the staff estimated that using a conditional burst probability of 0.05 in the steam system depressurization sequences would result in meeting the RG 1.174 ΔLERF guideline for the licensee's application. Using the 95% bounding value for K and 134,000 psi in the Framatome Burst Equation, the staff calculated that a flaw with a structural length of 1.25 inches and a structural depth of 81% of the tube wall thickness would have a burst pressure of 2,690 at the in-situ test temperature and 2,500 psi at operating temperature. This reflects a factor of 0.927 relating the material strength properties at operating temperature to the properties at the in-situ test temperature. The length was picked on the basis of the long flaws found in the ANO-2 "B" SG during the last inspection. These had overall lengths that exceed that value and structurally significant lengths that approach it.

Based on the above calculations, the staff estimates that the conditional probability of tube rupture at MSLB conditions would be less than 0.05 if none of the long flaws in the SG grow to more than 81% TW during the current operating period. Reasonable variations in the structural length of the flaw or the pressure differential to which it is exposed make only small changes in the structural depth of concern.

However, the staff is not able to conclude that the 2P99 inspection can be credited to the extent of ensuring that tubes with flaws large enough to grow to that effective depth (81% TW) during this operating period were all detected and plugged. The “2R13 Lookback Evaluation” submitted to the NRC on May 30, 2000, indicates that the deepest flaws found in the 2P99 inspection probably grew at higher than average rates since they were missed by the 2R13 inspection. These rates were estimated to be in the range of 12% to 16% of the tube wall thickness per year, for a period of 0.72 year. Thus, during the 2P99 inspection, the POD for the flaws with initial structurally-effective depths in the range of 70% would need to be very good to achieve this result. The licensee’s POD values for such flaws are in the range of 0.90 to nearly 1.00.

However, the “2R13 Lookback Evaluation” indicates that the actual POD for flaws of this depth was less than 0.4 during 2R13. As discussed in Section 3.4.2 of this SE, the staff has determined that there is no evidence of an improvement in the 2P99 inspection as compared to the 2R13 inspection. On this basis, the staff concludes that the information provided by the licensee is insufficient to credit the 2P99 inspection with high confidence for detecting the flaws that are significant to this SE.

Although the experience to date is that ANO-2 tube flaws that were in-situ pressure tested have successfully held more than the MSLB pressure without bursting, that information is not sufficient to conclude that this will be the case through the end of the current operating cycle. The information on POD and growth rates is not adequate to dismiss with reasonable confidence the possibility of a combination of a few large flaws going undetected and growing at the bounding growth rates estimated by the licensee.

For example, one short flaw (about 0.5 inches long) apparently had propagated completely through the tube wall at one point and was estimated to have a structurally significant depth of 94% at the time of the 2P99 inspection. This flaw was too short to burst at MSLB pressures, but is calculated to pop open and leak at a pressure differential well below MSLB conditions. (The licensee did not in-situ test this flaw, although it appears to meet the Electrical Power Research Institute (EPRI) guidelines for testing to establish the leakage at MSLB.) At its apparent growth rate, continued operation without the mid-cycle inspection in November 1999 would be expected to have allowed this short flaw to develop into a leak during normal operation. During the 2R13 inspection, this flaw produced a 0.63 volt signal on the bobbin coil, but was missed by the analysts. Because some other flaws found in 2P99 that were long enough (but not yet deep enough) to be able to burst also produced bobbin signals in the 0.6 to 0.7 volt range, the staff is not convinced that longer flaws that can degrade to a similar depth (e.g., > 90% structurally significant depth) were more reliably detected during the recent inspections.

In addition, there is evidence of a change in either POD, growth rates, or flaw initiation rate at ANO-2, based on comparison of the results of the last two inspections. During the 2R13 inspection, 71 axial eggcrate ODSCC indications were identified in the “B” SG. During the 2P99 inspection, 185 axial eggcrate ODSCC indications were identified in the “B” SG. This significant increase was not predicted by the licensee’s SG tube flaw prediction model. Although the licensee claims that this increase was due to improvements in the inspection process, not changes in flaw initiation or growth rates, the staff’s review did not identify evidence of inspection improvements to support this claim (see Section 3.4.2.).

3.4.7 Conclusions on Tube Integrity Analysis

The staff has concluded that the projected EOC SG tube structural integrity margin against burst may be substantially less than the current licensing basis of 1.4 times MSLB pressure based on its review of the licensee's deterministic operational assessment. In addition, the burst pressure cannot be reliably predicted to provide adequate assurance of integrity during MSLB conditions on the basis of past in-situ pressure test results and the 2P99 inspection results. Therefore, the staff cannot conclude with reasonable assurance that the intent of RG 1.174, which states that sufficient safety margins should be maintained, will be satisfied for the remainder of the current operating cycle. Furthermore, based on the above evaluations, the staff could not conclude that a conditional failure probability of less than 0.05 could be demonstrated in order to support meeting RG 1.174 guidelines as discussed in Section 3.5.2.2.

Based on the licensee's regulatory commitment to administratively control the reactor coolant and secondary coolant specific activities, the staff has concluded that SG tube leakage integrity would be maintained to meet the dose guidelines provided in 10 CFR Part 100 and the control room operator dose criterion specified in GDC 19 of 10 CFR Part 50, for the remainder of the current operating cycle.

3.5 Probabilistic Analysis

3.5.1 Review of the Original Application

The licensee's original assessment of the risk-increment associated with not performing an additional inspection at 2P00 begins with an evaluation of the frequency of the various sequences that might be affected by SG tube degradation. Spontaneous tube rupture was estimated to produce a CDF and LERF contribution of 1.4×10^{-7}/reactor year (RY) and to be unchanged by the tube degradation in the eggcrate regions. So, the licensee estimated that the risk increment for the spontaneous rupture sequences was zero. Pressure-induced ruptures during steam system depressurization accidents were estimated to contribute CDF and LERF values on the order of 10^{-10}/RY, with smaller contributions for the ΔCDF and ΔLERF risk increments associated with the application. Pressure-induced ruptures due to anticipated-transient-without-scram sequences were projected to contribute CDF and LERF values in the 10^{-9}/RY range, with ΔCDF and ΔLERF contributions of about 1×10^{-9}/RY over the four month period in question. Thermally-induced ruptures during high/dry core damage sequences were estimated to contribute about 3×10^{-6}/RY to total LERF and a ΔLERF of about 4×10^{-7}/RY (but no ΔCDF increment). Thus, the major contributors to ΔLERF were estimated by the licensee to come from the high/dry core damage sequences. The licensee's estimate of total CDF, including internal and external event-initiated sequences not influenced by SG tube degradation, was about 2×10^{-5}/RY, with about 1.3×10^{-5}/RY (65%) of that being high/dry sequences potentially vulnerable to thermally-induced tube rupture. The licensee's estimate of total LERF was about 5×10^{-6}/RY, comprised of about 3.2 to 3.9×10^{-6}/RY primarily from thermally-induced tube failures, plus 1.6×10^{-6}/RY that is not related to tube degradation. Thus, the licensee projected that the requested amendment would meet RG 1.174 criteria for total and incremental values of both CDF and LERF.

The licensee broke down the thermally-induced ΔLERF contributions by severe accident sequence as indicated in Table 4. The licensee's estimates are highly dependent on three parameters: (1) the SG tube flaws have slow growth rates, (2) the tube inspections find virtually all flaws that are subject to pressure-induced rupture, and (3) the thermal challenge to tube integrity from high/dry core damage sequences is substantial. Staff review was therefore initially focused on verifying those parameters, as discussed in the previous sections.

TABLE 4

LICENSEE'S ESTIMATES OF LERF AND ΔLERF FOR HIGH/DRY SEQUENCE

Sequence	Total	Thermally Induced LERF		
Designation	Frequency	@ 2R14 w/o repair	@ 2R14 w/ repair	ΔLERF
H/D/Hi	2.763 E -6	3.396 E -8	1.699 E -8	1.697 E -8
H/D/Hc	1.464 E -7	1.797 E -9	9.229 E-10	8.741 E-10
H/D/Li	2.494 E -6	2.398 E -6	2.306 E -6	9.2 E -8
H/D/Lc	1.321 E -7	1.269 E -7	1.252 E -7	1.7 E -9
M/D/Hi	3.662 E -6	1.748 E -8	7.027 E -9	1.045 E -8
M/D/Hc	1.941 E -7	9.247 E-10	3.816 E-10	5.431 E-10
M/D/Li	3.306 E -6	1.234 E -6	9.534 E -7	2.806 E -7
M/D/Lc	1.752 E -7	6.530 E -8	5.177 E -8	1.353 E -8
totals	1.287 E -5	3.879 E -6	3.462 E -6	4.17 E -7

where: the first letter is the RCS pressure condition (H = normal, M = 1,400 psi)
the second letter indicates that the SGs are dry (D = dry)
the third letter indicates the SG steam system pressure (H = normal, L = atmospheric)
the lower case letter indicates the condition of the reactor coolant loop seal (i = intact, c = clear)

As can be seen from the table, the licensee estimates that the ΔLERF is dominated by the M/D/Li. These sequences have a relatively high frequency and a modest delta due to tube plugging assumed to occur in a spring outage. The H/D/Hi, H/D/Li, and M/D/Hi sequences all have similar frequencies, but lower deltas due to repair. In particular, the H/D/Li sequence shows a high conditional tube rupture probability even with repair.

As discussed in the previous sections, the parameters used by the licensee to derive its ΔLERF estimates were not confirmed by the staff's independent analyses of the thermal-hydraulic characteristics of these sequences and the staff's review of the expected flaw sizes as a function of time. Specifically, the staff found that the licensee's thermal-hydraulic simulations of the medium RCS pressure sequences was too unrealistic to capture the necessary phenomena. This was due to a combined effect of an over-simplification of the depressurization mechanism and the inability of the MAAP computer code to predict loop seal clearing phenomena. The staff's thermal-hydraulic analyses showed that the loop seals would always clear for the slow RCS depressurization sequences and that the behavior of the tubes during such events was sensitive to the degree of degradation in the range for which the licensee's analysis was found to have substantial discrepancies when compared to known information. In addition, the H/D/Li sequences, as modeled by the staff, were not as likely to cause tubes with small flaws to fail as was predicted by the licensee's analyses. These sequences were also found to not exhibit loop seal clearing. Taken together, these findings call into question the licensee's analyses for about 6×10^{-6}/RY of sequences that might contribute substantially to the ΔLERF.

The sequences with high SG steam-side pressures would not be significantly affected according to the staff's modeling. However, what fraction of the high/dry sequences would have SGs at high pressure is very uncertain, too. The licensee took from NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," dated March 1998, an estimate of 0.5 as the probability that one or more SGs would blow down once they became dry. This value was picked by the authors of the NUREG for a sensitivity study, with some support that these events occur in operating experience.

However, the staff has no basis for estimating the industry-average probability of SG blow-down, much less the ability to predict plant specific values. Licensees with PWRs have long been exempted from the requirements for leak rate testing of their steam system isolation valves. The ANO-2 licensee has never checked its SG isolation valves (main steam isolation, main steam isolation bypass, feedwater, and blow-down valves) to assess their leakage rates since the original tests following construction. Effective leakage areas somewhat smaller than a square inch can and have gone unnoticed during normal operations, but are still sufficient to blow down a dry SG within the two or more hours available during a core damage accident.

All of these issues together make the potential ΔLERF difficult to bound at a value less than 1×10^{-6}/RY, even for the four month period considered in the application. Thus, the staff is unable to confirm that the ΔLERF will not exceed RG 1.174 acceptance guidelines for the amendment as originally proposed.

3.5.2 Review of Probabilistic Aspects of the Revised Application

In order to make progress toward the approval of its application, the licensee has revised its approach and application to take credit for an operator action to depressurize the RCS at the onset of core damage. The licensee's goal is to obtain a reduction in the high/dry frequency by a factor that is less than 0.25, producing a new frequency of about 3.2×10^{-6}/RY. The licensee's amended application also rescheduled the postulated spring outage from May 15 to June 16, 2000, so the ΔLERF was evaluated by the staff for 0.25 RY of operation. Therefore, if the depressurization strategy were successful, the potential contribution to ΔLERF from SG tube rupture induced by the thermal challenges during core damage accidents would be bounded at a value of 8×10^{-7}/RY. The potential for pressure-induced tube ruptures during severe accident sequences must still be considered for those cases that have steam-side depressurizations before the operator takes the action to depressurize the RCS. (Contributions from spontaneous ruptures and ruptures induced by steam-side depressurization initiating events must still be considered separately, because they are not affected by the depressurization procedure.)

3.5.2.1 High/Dry Sequences

The ECCS vent valve arrangement using DC-powered MOVs at ANO-2 provides some advantages over a system that uses PORVs. In particular, once opened they will remain open unless action is taken to close them. PORVs depend upon DC power and usually on a reserve air supply to open and to remain open. It is not clear whether a PORV that has been repeatedly cycling automatically to relieve excess RCS pressure would retain the capacity to be opened manually late in a severe accident sequence. If it does open, there is the potential for it to reclose at an unfortunate point in time if the DC batteries become depleted or the air supply is exhausted. Due to the ANO-2 configuration, none of these concerns needed to be addressed for this application.

The licensee provided an analysis using the MAAP code to support the efficacy of its depressurization strategy. It showed that operator action to depressurize the RCS would protect the tubes if it is accomplished within 30 minutes of the occurrence of the appropriate cues, but not if the action is delayed for 45 minutes.

The staff's confirmatory analysis used the SCDAP/RELAP code as described in Section 3.3.3 of this SE, and concluded that depressurization was effective even if the operator action was delayed by 27 minutes after the first occurrence of the indications described in the procedure for initiation of the process.

3.5.2.1.1 Human Error Probability for RCS Depressurization Procedure

A staff review was performed to determine whether there was a significant likelihood of successful reactor depressurization for the high/dry scenarios. For an operator action to be successful, the operators must first have a clear indication that there is a problem, second they must be capable of formulating a solution to the problem, and third they must be able to execute the required actions successfully. In order to support its case for the effectiveness of the procedural modification, the licensee provided the following:

1.01 The top 100 core damage cutsets

1.02 Annunciator response procedures 2K01(to deal with the loss of a DC bus) and 2K07 (to deal with high Core Exit Thermocouple (CET) alarm)

1.03 Severe Accident Management Guidelines (SAMG) Developed Strategy 02 - Emergency Power for Unit 2 ECCS Vent Valves

1.04 Dedicated Cross-Tie Watch Study Guide and Dedicated Cross-Tie Watch Qualification Guide, Revision 1

1.05 A human reliability analysis to establish the time at which the probability of failure to perform the depressurization is less than 0.25

1.06 Extract from Procedure/Work Plan 1015.001 - Conduct of Operations

The likelihood of success is affected by the plant conditions during the development of the scenario. In the context of the PRA model, this implies that the probability of failure of the mitigating action may vary from cutset to cutset, according to the conditions each cutset implies, which includes whether there have been other human failures contributing to the scenario.

There are two classes of cutsets that are of concern: the first involves a loss of a DC bus (2D01 or 2D02) as an initiating event or a consequential event (e.g., following a reactor trip and loss of an AC bus) with subsequent loss of all heat removal; the second involves a loss of all heat removal with DC buses available. The first class contributes most to the high/dry frequency in the current licensee model. The plant procedures have been modified to direct that the dedicated crosstie operator (DXO) report to the control room on either a plant trip or a loss of a DC bus. The operator's function on arrival in the control room is to monitor the CETs and the SG wide-range monitors, and alert the control room operators when the cues to begin depressurization of the RCS occur. The indications of a reactor trip or loss of a DC bus are sufficiently clear that there would be a very low likelihood of them not being observed. The time taken to report to the control room is short compared to the time to reach the conditions that would require initiating the depressurization.

Loss of DC bus Case: Upon the annunciation of the loss of a DC bus, the annunciator response procedure 2K01 directs the DXO to go to the control room and obtain SDS-02, Emergency Power for Unit 2 ECCS valves. Once in the control room, the DXO is directed by Attachment 1 to SDS-02 to monitor the CETs and SG wide-range monitors until such time as the cues for depressurizing the RCS are obtained. When either SG level is below 70 inches (wide-range) and at least five available CETs are above 800 ^{o}F, the DXO is directed to begin the cross-tie, using SDS-02. From discussions with the licensee, the indications required to direct the appropriate response will be available for the case of one DC bus failed. Even though this second cue occurs late in the scenario, when conditions in the plant have been deteriorating significantly, the stationing of the DXO to monitor the CETs and wide-range monitors provides reasonable assurance that the cues will not be missed.

The cross-tie is achieved using a cable that is kept in a locked cabinet in the vicinity of the buses to be cross-tied, and which is inserted into permanently mounted plugs. A walkdown of the procedure by the licensee's human reliability consultant confirmed that the connection could be readily made following the procedure. The walk-through of the process showed that the cross-tie and subsequent opening of valves would take approximately 10 minutes, including the travel time from the control room to the local station. Given that there is at least twice that much time expected to be available, even if the failures in the core damage cutset were all to occur at time zero, the staff concludes that the likelihood of success should be relatively high.

The highest frequency cutsets typically include one or more human failure events, but those are restoration activities associated with cross-ties between buses to restore power to operable equipment. Because of the clear indications and the focus on the need to depressurize, there is likely to be little dependency between these failures and the failure to perform the depressurization.

No Loss of DC bus Case: In this case, the response to a reactor trip is to require the DXO to go to the control room to monitor the CETs and SG wide-range level monitors. In this case, there is also an alarm (2K07, D-8) that would give warning to the operators (alarm set at 700 ^{o}F). However, this alarm would occur late in the sequence. The operators should have already depressurized the RCS in order to initiate once through cooling for this sequence. To get to this point, the operators already would have had to either not attempted to depressurize or been unsuccessful in doing so. In this case, it would be difficult to argue that, having failed to depressurize to prevent core damage, the operators would be very likely to depressurize in response to the SAMG. However, cutsets that would represent such scenarios were not identified in the top 100 cutsets and, therefore, have very low frequency (below 1.7×10^{-8}/RY).

The evaluation of the time at which the probability of failure to successfully depressurize the RCS is performed by generating a time-reliability curve using methods developed by the EPRI that have been used extensively in individual plant examination submittals, and the Accident Sequence Evaluation Program procedure developed for NRC. The method includes an assessment of the failure to detect the plant condition and failure in response planning, as well as failure to implement the strategy. Estimates of times required as input to the evaluation were obtained from observations made at the plant simulator or from job performance measurements. There appear to be some minor errors in application of the model. However, they are conservative errors and correction of these errors would result in lower failure probabilities.

In conclusion, the staff concurs that there is a relatively high likelihood that the operators would be able to successfully depressurize the reactor vessel for the scenarios of interest.

3.5.2.1.2 Pressure-Induced Ruptures During High/Dry Sequences

There are several mechanisms that could depressurize the steam-side of one or both SGs during the various high/dry core damage sequences. These include (1) steam line relief valves that stick-open when actuated to control pressure; (2) inoperable SG isolation valves in the steam system, feedwater system, blow-down lines, and the steam supply to the turbine for the emergency feedwater system; and (3) leakage through one or several valves that are closed. Stuck-open valves will depressurize the steam-side of a generator long before the operator initiates the depressurization procedure discussed above. High total leak rates through one or several valves can also substantially depressurize a SG in the period of hours before the operator would initiate depressurization of the RCS.

If substantial SG depressurization does occur before the RCS is depressurized, there is a potential for flawed tubes to rupture, just as they would in the MSLB DBA analyses. If a tube does rupture and the operator still depressurizes the RCS when the CETs read 800 $^\circ$F, it is not clear what the total release of radioactive materials would be without further detailed analysis. Accordingly, the depressurization action might or might not prevent a large early release if a pressure-induced tube rupture occurs during a core damage accident sequence. However, as discussed in Section 3.5.2.2, the conditional probability of tube rupture under MSLB conditions should be less than 0.05 in order to satisfy the ΔLERF criterion for the sequences with steam system depressurization as the initiating event. Therefore, even if all high/dry sequences had rapidly depressurized SGs, if that conditional rupture probability is achieved, then the frequency of those with pressure-induced tube rupture could only be 6.4×10^{-7}/RY and contribute about 1.6×10^{-7} to the ΔLERF. This would be further reduced by considering a realistic probability for SG depressurization. Accordingly, pressure-induced tube rupture during high/dry sequences is not a significant contributor to ΔLERF for ANO-2, provided the conditional probability of the tube rupture is low enough to satisfy the ΔLERF criterion for sequences initiated by steam system depressurization events.

3.5.2.2 Steam System Depressurization Sequences

The licensee's application states that "[t]ransients with a stuck open secondary relief valve are assumed to be dominated by the risk associated with the temperature induced SGTR [steam generator tube rupture] risk." So, its application uses only the frequency of steam and feedwater pipe breaks for the initiating event frequency of the sequence that begins with depressurization of a SG and consequently induces rupture of a flawed tube. The staff does not agree that consideration of thermally-induced rupture in the high/dry sequences is a rational surrogate for considering pressure-induced rupture in sequences initiated by steam system depressurization events (nor, necessarily, for consideration of pressure-induced rupture in the high/dry sequences). Stuck-open steam line safety valves have occurred during operational transients and have depressurized SGs (although no flawed tubes were induced to rupture). Stuck valve events dominate this initiating event frequency.

The staff based its review of these sequences on the information in NUREG-1570. The staff estimates that a conditional probability of tube rupture of about 0.05 would produce a ΔLERF contribution of 1×10^{-6} for this application. This is based on an operational period of three months, a depressurization event frequency of 7×10^{-3}/RY, and a mitigation failure probability of 10^{-2}, which is dominated by human error. The 0.05 conditional rupture probability is a less difficult criterion than the value of 0.01 that was proposed in Draft Regulatory Guide 1074 (DG-1074). However, DG-1074 was intended to address a different situation with a longer duration.

Because the licensee’s estimated flaw growth rates and POD distribution do not properly predict past inspection results, the staff had to consider that either one or both are in error. This makes it complicated to logically deduce a conditional probability of tube rupture, given a steam-side depressurization in a SG.

As explained in Section 3.4.6.3, the staff has concluded that the licensee's inspection at 2P99 was not adequate to demonstrate that the conditional burst probability would remain below 0.05 for the operating period. Therefore, the staff is not able to conclude that the ΔLERF contribution from sequences initiated by steam-side depressurizations will be below the numerical guidance in RG 1.174 for the three month period considered.

3.5.2.3 Spontaneous Rupture Sequences

The staff has not attempted to estimate the increase in spontaneous rupture probability. Coefficients for the Framatome Burst Equation are not available to estimate the high level of confidence that would be required (at least 0.99), and the uncertainties regarding the flaw POD and growth rates make it infeasible to produce such an estimate. Thus, no estimate was produced for the ΔLERF contribution from spontaneous rupture sequences.

3.5.2.4 Conclusions for ΔLERF for Revised Application

The staff was able to conclude that the ΔLERF contribution from high/dry accident sequences was within RG 1.174 numerical guidance, provided that the depressurization procedure is put in place. However, the licensee's technical analyses were not adequate to demonstrate that steam-side depressurization events were within that guidance. The additional assessments conducted by the staff raised additional concerns that the licensee's latest inspection may not be adequate to provide reasonable assurance that the tubes will not become susceptible to steam-side depressurization before the end of the current operating period. The staff did not attempt to estimate the ΔLERF associated with spontaneous rupture of eggcrate flaws.

3.6 Technical Evaluation Conclusions

The licensee's original application is comprehensive and well organized. However, the staff's technical evaluation has found significant deficiencies with the licensee's quantification of the risk increment associated with not inspecting the SG tubes before the end of the current fuel cycle. The most problematic deficiency is that the licensee's estimates for flaw growth rates and probability of flaw detection substantially under-predict the number and size of the flaws that are found during subsequent inspections. This results in a lack of reliable projections for the size of flaws as a function of time. Another deficiency involves unrealistic thermal-hydraulic analyses for some of the high/dry core damage sequences. In an effort to simplify the analyses, the licensee modified the plant and its operating procedures to provide for operator depressurization of the RCS during high/dry accident sequences. The staff evaluation found that this is effective in mitigating the risk from the high temperatures associated with those sequences. These mitigation measures, however, do not affect sequences initiated by steam system depressurization events. The staff concludes that the licensee did not demonstrate that sufficient tube strength margins will remain for the MSLB DBA. The staff also concludes that the licensee's last inspection was inadequate to keep the conditional probability of pressure-induced tube ruptures low enough for such sequences to satisfy the RG 1.174 guidance.

4.0 INTEGRATED DECISION PROCESS

4.1 Principles of Integrated Decision-Making

RG 1.174 provides five principles for risk-informed decision-making. Each of these is addressed below in the context of this proposed license amendment.

Principle 1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change. The staff is unable to conclude that the mid-cycle inspection is not necessary and that operation in accordance with this license amendment meets the regulations, including GDC 14. Although the staff notes that the previous inspections have discovered no flaws that were close to the strength limit needed to withstand a MSLB DBA, the staff's assessment notes that the change in the inspection results for SG "B" may indicate that conditions in that generator can no longer be presumed to remain as demonstrated by the previous in-situ tube testing. Accordingly, the staff has determined that the licensee's inspection process does not provide assurance that the SG tubes will continue to meet the requirement for a low probability of burst during the MSLB DBA.

Principle 2. The proposed change is consistent with the defense-in-depth philosophy. Because the requested change involves the integrity of the SG tubes, it affects two of the three physical barriers provided by the regulations to prevent the release of radioactive material to the environment. The LERF criterion addressed by Principle 4 is therefore a primary consideration when considering the adequacy of these barriers for this request. From a probabilistic perspective, defense-in-depth is provided by a combination of low challenge frequency and low conditional probability of failure due to the challenge. The staff's consideration of the likely condition of the tubes at the end of the current cycle indicates that the tubes may not retain sufficient structural integrity to survive DBAs (i.e., steam line break), but will very likely have adequate margin for normal operation. Consequently, tube degradation is not expected to initiate an accident sequence, but has a higher than normal potential for transforming a low-frequency design basis transient into a beyond DBA precursor. However, the staff notes that mitigation capability would remain for protecting the remaining single barrier, the fuel cladding, if a tube rupture was induced by steam-side depressurization. For the more severe challenges from high/dry core damage sequences, there is defense-in-depth due to the low frequency of those challenges and the availability of an effective mitigation strategy: manual RCS depressurization. On this basis, the staff finds that the principle of defense-in-depth would be maintained without benefit of a mid-cycle inspection of the SGs.

Principle 3. The proposed change maintains sufficient safety margins. Due to the failure to resolve inconsistencies in the licensee's estimates of flaw growth rates and POD, the staff is not able to reliably quantify the margins in tube strength that are expected at the end of this operating period. Based on the probabilistic discussions in Section 3.5.2.2, the staff concludes that the uncertainty in the burst pressure predictions is too large to conclude that margins will remain sufficient for the duration of this period of operation.

Principle 4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement. RG 1.174 provides a set of numerical guidelines for combinations of the total and incremental CDF, and another set of numerical criteria for total and incremental LERF. As discussed in Section 3.5 of this SE, the CDF is not as significantly affected by the licensee's request as is the LERF. Therefore, the ΔLERF has been the focus of the staff's review. There are three potentially dominant types of sequences that involve tube degradation that can increase LERF. The staff concludes that the licensee's new RCS depressurization strategy is effective and sufficiently reliable to mitigate the high temperature challenges associated with high/dry core damage accident sequences. However, the licensee did not demonstrate that the ΔLERF associated with steam-side depressurization transients that induce tube ruptures would meet the numerical guidance for this principle. (Sequences involving spontaneous tube ruptures were not fully evaluated by the staff.)

Principle 5. The impact of the proposed change should be monitored using performance measurement strategies. This principle cannot be directly addressed for the amendment requested. If the 2P00 inspection is not performed, no further data on the adequacy of the licensee's inspection will be produced, because the SGs will be replaced without being reinspected at the end of the current fuel cycle. However, the purpose of this principle is to prevent repetitive occurrence of undesirable conditions by detecting and correcting conditions that are not in accordance with the assumptions in the risk assessment and other analyses used to support a change. In this regard, the staff notes that recurrence after the end of the current cycle is precluded by replacement of the degrading SGs. Therefore, the staff concludes that this principle is not a primary factor in this case. In addition, the application was deficient in that it did not address the possibility of unexpected tube leakage or rupture during the remainder of the operating period. While this deficiency would not require denial of the amendment, it would need to be addressed by a license condition.

4.2 Uncertainties

RG 1.174 guidance on risk-informed decision-making also involves consideration of the uncertainties and their potential effects on the decision. There are a large number of variables in the foregoing analysis with varying degrees of uncertainty.

Sensitivity to the thermal-hydraulic analysis for the high/dry core damage sequences has been substantially reduced by introducing an operator action to depressurize the RCS. This has produced a much wider margin, between first projected RCS pressure boundary component failure and degraded tube failure, than has been found in previous analyses where the RCS remains pressurized to some degree.

Human reliability analysis, particularly the estimation of human error probabilities, is recognized as being an uncertain process. However, the approach used has focused on those factors considered to be important in the determination of reliable human performance and, given the preparation and specific training for this condition, should it arise, the likelihood of success claimed is considered to be reasonable.

As in previous analyses, the estimation of the flaw sizes and growth rates is highly uncertain. However, for this analysis, the sensitivity to that uncertainty is greatly reduced by the use of the depressurization process. That effectively removes the sensitivity to the flaw sizes that are the most difficult to detect. The uncertainty for flaw sizes that are susceptible to steam-side depressurization events (DBA-type accidents) is considered in Section 3.5.2.2. These uncertainties, related to flaw size as a function of time, are the primary source of deficiencies in the licensee’s evaluation.

4.3 Integrated Technical Conclusions

In summary, the staff finds that the licensee has not adequately demonstrated that the risk increment associated with the requested amendment would be small enough to meet the numerical guidance in RG 1.174 for Principle 4, nor that the remaining margins would be adequate, as specified by Principle 3 and involved in Principle 1. The fundamental deficiency in the licensee’s technical support for the application is that the uncertainties in the detection of flawed tubes by the latest inspection and estimates of growth of flaws during the current period of operation are too large to support the findings necessary for approval.

5.0 CONCLUSION

Based on the staff’s conclusions as discussed in Sections 3.6 and 4.3 of this SE, the application is denied.

Principal Contributors: C. Beardslee Kahn
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E. Throm

Date: July 21, 2000

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

NOTICE OF DENIAL OF AMENDMENT TO FACILITY OPERATING LICENSE

AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) has denied a request by Entergy Operations, Inc., (Entergy Operations or the licensee) for an amendment to Facility Operating License No. NPF-6 issued to the licensee for operation of the Arkansas Nuclear One, Unit No. 2, nuclear reactor located in Pope County, Arkansas. Notice of Consideration of Issuance of this amendment was published in the FEDERAL REGISTER on April 5, 2000 (65 FR 17914).

The purpose of the licensee's amendment request was to revise the license to permit operation of the reactor based on a risk-informed demonstration that predicted steam generator tube integrity, with consideration of eggcrate axial flaws, is adequate to meet Regulatory Guide 1.174 numerical acceptance criteria.

The NRC staff has concluded that the licensee's request cannot be granted. The licensee was notified of the Commission's denial of the proposed change by a letter dated .

By [30-day date], the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene pursuant to 10 CFR 2.714.

A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001 Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public

Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date.

A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005-3502, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendment dated March 9, 2000, as supplemented by letters dated April 11 and 28, May 30, June 20, 22, 23 (two letters), and 30, and July 7, 8, and 11, 2000, and (2) the Commission's letter to the licensee dated .

These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and accessible electronically through the ADAMS Public Electronic Reading Room link at the NRC Web site (http://www.nrc.gov).

Dated at Rockville, Maryland, this 21st day of July 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stuart A. Richards, Director
Project Directorate IV and Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation