

December 5, 1990

Docket No. 50-313

Mr. Neil S. Carns  
Vice President, Operations ANO  
Entergy Operations, Inc.  
Route 3 Box 137G  
Russellville, Arkansas 72801

Dear Mr. Carns:

SUBJECT: ISSUANCE OF AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE  
NO. DPR-51 - ARKANSAS NUCLEAR ONE, UNIT NO. 1 (TAC NO. 74894)

The Commission has issued the enclosed Amendment No. 140 to Facility Operating License No. DPR-51 for the Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the license and Technical Specifications (TSs) in response to your application dated August 8, 1990, as supplemented by letters dated August 23, October 25, November 5, November 7, and November 14 (2 letters), 1990.

The amendment revises license condition 2.c.(1) to increase the authorized steady-state reactor core power level to a maximum of 2568 megawatts thermal (100% of full power). The amendment also revises the TSs to reflect an increase in the Borated Water Storage Tank (BWST) level and revises the number of high pressure injection motor operated valves referenced in the TSs.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Thomas W. Alexion, Project Manager  
Project Directorate IV-1  
Division of Reactor Projects III, IV, and V  
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 140 to DPR-51
2. Safety Evaluation

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UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

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Sincerely,

A handwritten signature in cursive script that reads "Thomas W. Alexion".

Thomas W. Alexion, Project Manager  
Project Directorate IV-1  
Division of Reactor Projects III, IV, and V  
Office of Nuclear Reactor Regulation

Enclosures:

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2. Safety Evaluation

cc w/enclosures:  
See next page

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Arkansas Nuclear One, Unit 1

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UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

ENERGY OPERATIONS INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 140  
License No. DPR-51

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Entergy Operations, Inc. (the licensee) dated August 8, 1990, as supplemented by letters dated August 23, October 25, November 5, November 7, and November 14 (2 letters), 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, paragraph 2.c.(1) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

- (1) Maximum Power Level

- EOI is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

3. Additionally, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.c.(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

- (2) Technical Specifications

- The Technical Specifications contained in Appendix A, as revised through Amendment No. 140, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

*Theodore R. Quay*

Theodore R. Quay, Director  
Project Directorate IV-1  
Division of Reactor Projects III, IV, and V  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: December 5, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 140

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Revise the following page of the Operating License and pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

REMOVE PAGES

3 (License)

18a

36

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INSERT PAGES

3 (License)

18a

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39

- (5) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;
  - (6) EOI, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.
- c. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:
- (1) Maximum Power Level  
EOI is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.
  - (2) Technical Specifications  
The Technical Specifications contained in Appendix A, as revised through Amendment No. 135 are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications.
  - (3) AF&L\* may proceed with and is required to complete the modifications identified in Paragraphs 3.1 through 3.19 of the NRC's Fire Protection Safety Evaluation (SE) on the facility dated August 22, 1978 and supplements thereto. These modifications shall be completed as specified in Table 3.1 of the Safety Evaluation Report or supplements thereto. In addition, the licensee may proceed with and is required to complete the modifications identified in Supplement 1 to the Fire Protection Safety Evaluation Report, and any future supplements. These modifications shall be completed by the dates identified in the supplement.

\*The original licensee authorized to possess, use, and operate the facility was AP&L. Consequently, certain historical references to AP&L remain in the license conditions.

- 3.1.2.7 Prior to reaching fifteen effective full power years of operation, Figures 3.1.2-1, 3.1.2-2 and 3.1.2-3 shall be updated for the next service period in accordance with 10CFR50, Appendix G, Section V.B. The service period shall be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with Specification 4.2.7. The highest predicted adjusted reference temperature of all the beltline region materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.8. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.
- 3.1.2.8 The updated proposed technical specifications referred to in 3.1.2.7 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specifications submitted in accordance with 10 CFR Part 50, Appendix G, Section V.C.
- 3.1.2.9 With the exception of ASME Section XI testing and when the core flood tank is depressurized, during a plant cooldown the core flood tank discharge valves shall be closed and the circuit breakers for the motor operators opened before depressurizing the reactor coolant system below 600 psig.
- 3.1.2.10 With the exception of ASME Section XI testing, fill and vent of the reactor coolant system, emergency RCS makeup and to allow maintenance of the valves, when the reactor coolant temperature is less than 280°F, the High Pressure Injection motor operated valves shall be closed with their opening control circuits for the motor operators disabled.
- 3.1.2.11 The plant shall not be operated in a water solid condition when the RCS pressure boundary is intact except as allowed by Emergency Operating Procedures and during System Hydrotest.



### 3.3 EMERGENCY CORE COOLING, REACTOR BUILDING COOLING AND REACTOR BUILDING SPRAY SYSTEMS

#### Applicability

Applies to the emergency core cooling, reactor building cooling and reactor building spray systems.

#### Objectivity

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building cooling and reactor building spray systems.

#### Specification

- 3.3.1 The following equipment shall be operable whenever containment integrity is established as required by Specification 3.6.1:
- (A) One reactor building spray pump and its associated spray nozzle header.
  - (B) One reactor building cooling fan and its associated cooling unit.
  - (C) Two out of three service water pumps shall be operable, powered from independent essential buses, to provide redundant and independent flow paths.
  - (D) Two engineered safety feature actuated low pressure injection pumps shall be operable.
  - (E) Both low pressure injection coolers and their cooling water supplies shall be operable.
  - (F) Two BWST level instrument channels shall be operable.
  - (G) The borated water storage tank shall contain a level of  $40.2 \pm 1.8$  ft. ( $387,400 \pm 17,300$  gallons) of water having a concentration of  $2470 \pm 200$  ppm boron at a temperature not less than 40F. The manual valve on the discharge line from the borated water storage tank shall be locked open.
  - (H) The four reactor building emergency sump isolation valves to the LPI system shall be either manually or remote-manually operable.

370,100 gallons of borated water are supplied for emergency core cooling and reactor building spray in the event of a loss-of-coolant accident. This amount fulfills requirements for emergency core cooling. Approximately 16,000 gallons of borated water are required to reach cold shutdown. The original nominal borated water storage tank capacity of 380,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent crystallization and local freezing of the boric acid. The boron concentration is set at a value that will maintain the core at least 1 percent  $\Delta k/k$  subcritical at 70°F without any control rods in the core. The concentration for 1%  $\Delta k/k$  subcriticality is 1609 ppm boron in the core, while the minimum value specified in the borated water storage tank is 2270 ppm boron.

Specification 3.3.2 assures that above 350°F two high pressure injection pumps are also available to provide injection water as the energy of the reactor coolant system is increased.

Specification 3.3.3 assures that above 800 psig both core flooding tanks are operational. Since their design pressure is  $600 \pm 25$  psig, they are not brought into the operational state until 800 psig to prevent spurious injection of borated water. Both core flooding tanks are specified as a single core flood tank has insufficient inventory to reflood the core.<sup>(1)</sup>

Specification 3.3.4 assures that prior to going critical the redundant reactor building cooling unit and spray are operational.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

When the reactor is critical, maintenance is allowed per Specification 3.3.5. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. Exceptions to Specification 3.3.6 permit continued operation for seven days if one of two BWST level instrument channels is operable or if either the pressure or level instrument channel in the CFT instrument channel is operable.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2300°F and the metal-water reaction to that representing less than 1 percent of the clad.

The service water system consists of two independent but interconnected, full capacity, 100% redundant systems, to ensure continuous heat removal.<sup>(4)</sup>

One service water pump is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 140 TO

FACILITY OPERATING LICENSE NO. DPR-51

ENERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated August 8, 1990, as supplemented by letters dated August 23, October 25, November 5, November 7, and November 14 (2 letters), 1990, Entergy Operations, Inc. (the licensee) submitted information supporting a license amendment to return Arkansas Nuclear One - Unit 1 (ANO-1) to full power operation (Ref. 1). An 80 percent power level operation was imposed on ANO-1 on the basis of an Appendix K small break loss-of-coolant accident (LOCA) analysis for a high pressure injection (HPI) line break. In addition, calculational errors were identified affecting the low pressure injection (LPI) and the reactor building spray (RBS) system pump net positive suction head (NPSH) when aligned to take suction from the reactor building sump. However, it was shown that 80 percent power level operation was safe and acceptable.

The supplemental letter dated August 23, 1990, contained an affidavit from B&W Nuclear Technologies requesting that part of the August 8, 1990 application be withheld from public disclosure, as it contains proprietary information. The October 25, November 5, and November 14, 1990 supplements provided additional clarifying information in response to staff questions. The November 7 and November 14, 1990 supplements provided a sample of the detailed calculations that were referenced in the original application.

There are two separate issues associated with the ANO-1 resuming 100 percent power operation: (1) acceptable performance for the HPI system following a small break LOCA for a HPI line break, and (2) the LPI and the RBS NPSH when aligned to take suction from the reactor building sump.

1.1 HPI Line Break

As a result of a review of the HPI system (motivated by a reactor trip on January 20, 1989), Babcock & Wilcox (B&W) discovered (1) that an HPI line break had not been previously analyzed, and (2) such a break was not enveloped by existing analyses. Thus, for an HPI line break the ECCS system might not be able to provide sufficient flow such that the requirements of 10 CFR 50.46 are

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satisfied at 100 percent power operation. Subsequent analysis showed that the plant could be operated safely at 80 percent and a license amendment was issued to that effect. A design modification involving a cavitating venturi in each of the HPI lines was installed in an attempt to resolve the HPI line break problem. However, the system subsequently experienced excessive vibration levels during post modification testing and the plant was therefore returned to its original configuration and to 80 percent power level operation.

This amendment request discusses a different modification of the HPI lines and presents analyses and proposes post-modification testing to support return to 100 percent power level operation.

## 1.2 LPI and RBS Pump NPSH

In December 1989 the licensee discovered calculational errors in the estimation of flow from the LPI and RBS pumps when aligned to take suction from the reactor building sump. Several design basis calculations involve post LOCA emergency operating conditions in which the LPI and RBS pumps take suction from the reactor building sump. Pump suction under these conditions could result in pump cavitation due to inadequate NPSH. The errors involved (1) incorrect accounting for the borated water storage tank (BWST) volume which affected the sump water level, (2) incorrect water density in the BWST, and (3) non-conservative assumptions with respect to the amount of water retained in the RCS.

This amendment discusses a number of modifications and presents analyses to support the conclusion that there exists adequate NPSH. The proposed modifications include: (1) throttling the RBS flow (following a LOCA) to maintain NPSH and high sump water level, (2) modify the Emergency Operating Procedures (EOPs), (3) increase the BWST water level, (4) divert more water under post LOCA conditions to the reactor building sump, (5) perform new analyses to ascertain that the problem has been resolved, and (6) propose post-modification testing to assure adequacy of the new HPI piping.

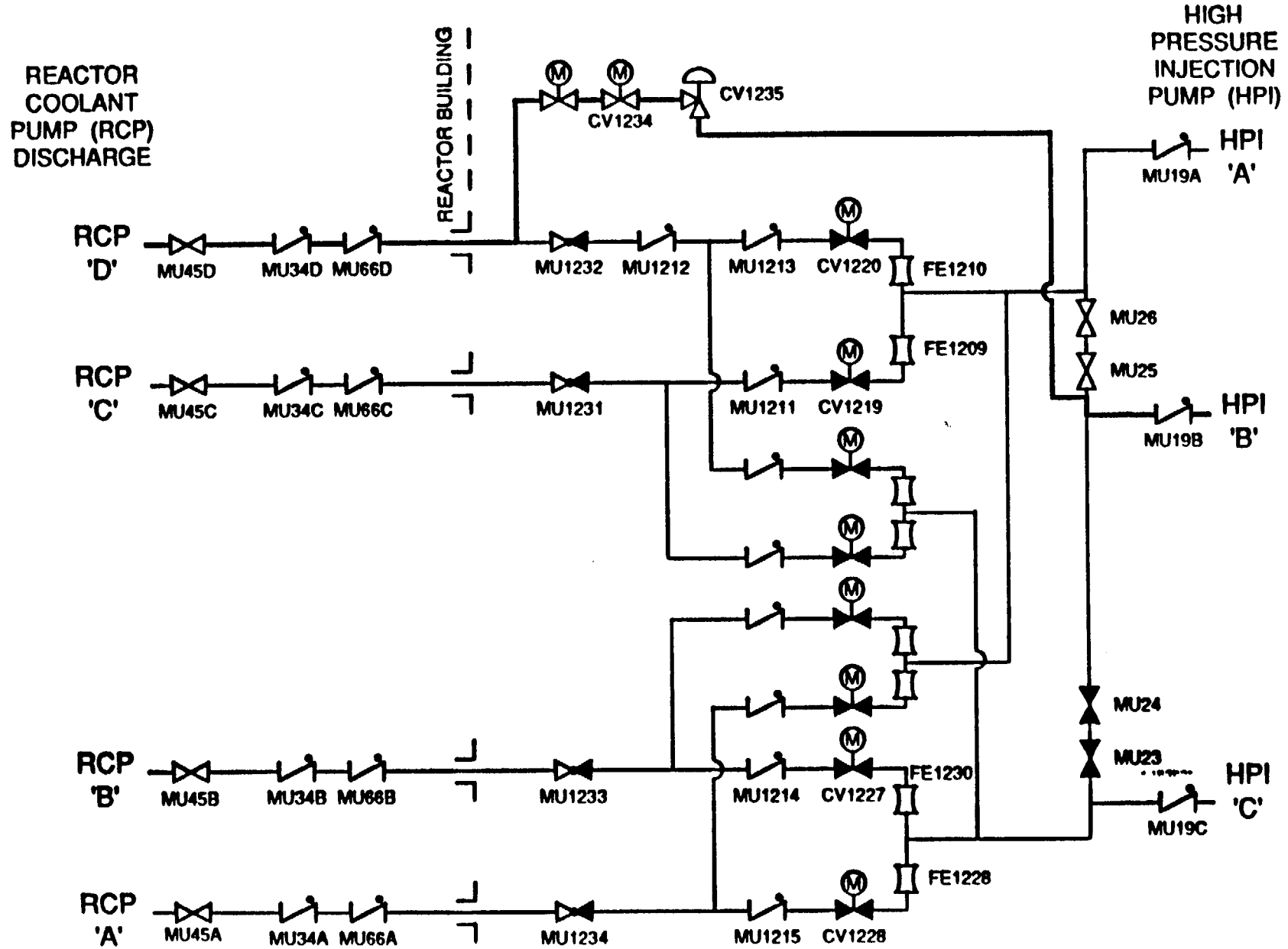
## 2.0 REACTOR SYSTEMS BRANCH EVALUATION

2.1 The events which require HPI flow are; the small break LOCA, a steam line break, and the steam generator tube rupture.

### 2.1.1 HPI Line Modifications

The specific small break LOCA addressed with the HPI system piping modifications analysis and testing is the HPI line break, between the RCS connection and the first HPI line check valve. The proposed arrangement is shown in Figure 2 and includes four additional injection lines with throttling isolation valves and flow instrumentation (flow indicators) powered by the same train of safety grade power as the pump which supplies them. With this modification each injection pump will be able to supply borated water through four lines with individual flow indication and throttling isolation valves. Given any

## FIGURE 2 PLANNED MODIFICATION



single failure, this arrangement allows a broken line to be throttled independently to enable the intact lines to provide adequate HPI flow. This arrangement will incorporate twice as many flow lines as previously with an equal number of flow meters equipped with safety grade chart recorders (Regulatory Guide 1.97, Type A, Category 1). Manual globe valves in the new lines will be pre-throttled to provide a balanced system flow.

In addition:

- ° EOP guidance will direct the operator to throttle the high injection flow line to within 20 gpm from the next highest flow injection line
- ° Each line will be provided with high point vents and low point drain
- ° Post installation testing will confirm the pre-throttled flow
- ° The new valves will be tested to ASME Section XI
- ° The pumps will continue to be tested for degradation to ASME Section XI

#### 2.1.2 HPI Line Analysis and Flow Requirements

The HPI flow rates required in the reactor coolant system (RCS) after this particular small break LOCA have been generically estimated by B&W and are given in tabular form for HPI flow versus RCS pressure. The flows required to be provided to the RCS are different than those provided by the HPI pumps due to the flow through the break. In addition, the actual flow to the RCS is affected by the assumed operator action (throttling of the highest flow HPI line) sometime after the break. However, the pre-throttling by the manual valves will improve the HPI response to an HPI line break. Finally, while pre-throttling improves the HPI line break performance, it changes the flow resistance for other small break LOCAs, therefore, they must be analyzed for the new HPI system.

The proposed system performance analysis is reported in the B&W report 86-1179795-01 (Ref. 2). The following assumptions were used in the evaluation:

- ° The initial flow balancing will be performed using an original HPI pump performance curve, while subsequent flows will assume a 6 percent head or flow degradation,
- ° The required operator action is to throttle the highest flow HPI line to the next highest, within 20 gpm,
- ° The total instrumentation uncertainty for throttling is 15 gpm which in combination with throttling uncertainty results in maximum of 35 gpm flow deviation,
- ° The flow balancing uncertainty is 2.5 percent, and

- ° The HPI system configuration is such that the total flow (four lines) at 600 psig is between 480-500 gpm.

The generic B&W small break LOCA HPI flow requirements analysis was performed for a power level of 2772 MWt. The corresponding ANO-1 flow requirements were appropriately estimated to reflect the fact that 2568 MWt is the ANO-1 full power level. In addition to the HPI line break, the steam line break and the steam generator tube rupture were reexamined to assure adequacy of the new HPI arrangement.

Under the above assumptions and conditions, the analysis showed that for an HPI line break with or without operator action for both 10 min or 20 min after the accident initiation, there is sufficient flow to satisfy the pressure versus flow requirement of the generic B&W analysis. The steam line break could lead to an RCS pressure decrease sufficient to activate the HPI and the core flood tanks (accumulators) assuring reactor shutdown. However, the HPI provides additional cooling to the primary system, but after shutdown there are no requirements on HPI flow rate. In the case of the steam generator tube rupture, the Safety Analysis Report (SAR) stated requirement is for the HPI to offset the tube break flow which is 432 gpm. Including water density changes at a maximum pressure of 1500 psig, the HPI flow should be 312 gpm which corresponds to a higher mass flow than 432 gpm at steam generator pressure. For an intact HPI system this flow can easily be maintained.

The staff concludes that the proposed HPI arrangement provides adequate flow for the limiting HPI line break and meets existing requirements for steam line break and generator tube rupture. Therefore, the staff finds it acceptable subject to confirmation of its performance with post implementation testing.

## 2.2 LPI and RBS Pump NPSH

The proposed changes to assure that the LPI and RBS pump have adequate NPSH include: (1) increasing the water level in the BWST, and (2) revising EOP guidance to reduce RBS and LPI flow prior to taking suction from the sump. However, reducing the RBS flow has the potential of changing the offsite dose, the reactor building pH (for iodine scrubbing) and the pressure-temperature profile. Similarly, the increase of the BWST water affects its seismic analysis. This review is concerned only with the NPSH which is affected by the sump water level and the reactor building pressure profile and the RBS and LPI flows. The related analysis is in an ANO report, attached to the original submittal (Ref. 3). The new analysis corrected calculational errors, inconsistent and erroneous assumptions in accounting for the sump water level and methodology errors (in accounting for the reactor building pressure).

For the items of interest here the approach was (a) to identify the post-LOCA water sources which could increase the sump level and determine that level, (b) assure that the HPI, LPI, and RBS flows are acceptable and consistent with credited operator actions, and (c) determine that for the acceptable flows and water availability there is adequate sump NPSH.

RBS pre-throttling can be used prior to the RBS sump suction lineup. Adequate and satisfactory throttling can also be accomplished from the control room provided that the instrumentation is Category 1 for Type A instruments per Regulatory Guide 1.97. The RBS and the LPI flow indication will be upgraded to the above standards. The throttling tolerance and the instrumentation uncertainty are accounted for to satisfy the required RBS flow range of 1050 to 1200 gpm. For LPI, no manual action is required to assure the allowed flow range due to the cavitating venturis which limit flow to 1910 gpm. Similarly, Reference 4 estimates that for the case of an LPI line break, the expected flow is bounded by 1910 gpm, thus, the remaining lines will provide adequate flow to the core while the total flow cannot exceed 3820 gpm.

A window of concern (in break sizes) was identified for a small break LOCA for which adequate NPSH may not be assured. However, the licensee at the staff's request (Refs. 7 and 8) performed additional analysis which demonstrated that adequate NPSH is available for the entire spectrum of break sizes even though no credit was taken for the reactor building pressure. Therefore, we find that the above NPSH estimate is conservative and acceptable.

### 2.3 Post Implementation Testing

As stated in Section 2.1.2 above, the HPI performance will be tested after completion of the modification. The objectives of this testing will be to demonstrate that the HPI system will perform as expected, i.e., satisfy the small break LOCA requirements. The RCS pressure conditions will be simulated by an orifice plate to be installed in each pump header, simulating a 600 psig RCS back pressure without affecting the individual line balance, thus, simulating the actual conditions of a small break LOCA. Before the test, the system will be balanced by aligning the HPI pumps and adjusting the manual globe valves to obtain a prescribed total system flow. The system flows required by the RCS and to be supplied by the HPI are listed in Tables 3 and 4 of Reference 2.

The post modification testing for the LPI and RBS will be limited to loop functional verification and instrument calibration.

The staff finds the above testing plan adequate for HPI, LPI, and RBS, and thus, acceptable.

### 2.4 Summary

The staff has reviewed the ANO-1 submittal requesting technical specification changes to resume 100 percent power operation. This review was limited to HPI performance for an HPI-line small break LOCA and the adequacy of the NPSH for the LPI and the RBS systems. The licensee modified the HPI system piping, added new HPI and LPI flow instrumentation, and increased the BWST level. In addition, the HPI line break, small break LOCA, and the NPSH for LPI and RBS system sump suction were reanalyzed. The staff finds that adequate flow will be provided to the reactor during the transient, before and after realignment



to the reactor building sump and that there is adequate NPSH for LPI and RBS operation. For the entire range of break sizes adequate NPSH was demonstrated. In addition, no credit was given for containment pressurization in the NPSH analyses. The system modifications will be subjected to adequate post implementation testing. For the above reasons the staff finds the modifications adequate and acceptable.

### 3.0 PLANT SYSTEMS BRANCH EVALUATION

#### 3.1 Reactor Building Temperature and Pressure Profiles and Effect on Equipment Qualification (EQ)

The licensee is proposing to reduce RBS flow during recirculation as a means to increase pump NPSH. The operators will be directed by EOPs to throttle back RBS flow to between 1050 gpm and 1200 gpm (from approximately 1500 gpm) at any time after the beginning of the accident but before switchover to sump recirculation. For the reanalysis of containment (reactor building) temperature and pressure profiles, the licensee used a conservative RBS flow rate of 1000 gpm.

Several reanalyses were performed by Betchel using the COPATTA computer program, using varied assumptions for the input parameters. For the revised design basis accident (DBA) LOCA analysis, the following changes have been made in various parameters, when compared to the original SAR analysis:

	<u>Original</u>	<u>Revised</u>
1. Service water flow to decay heat removal heat exchangers	3000gpm	1600gpm
2. Service water temperature	85°F	95°F
3. Reactor building coolers operating	2	1
4. Initial containment air temperature	110°F	140°F
5. Containment net free volume	$1.865 \times 10^6 \text{ ft}^3$	$1.83 \times 10^6 \text{ ft}^3$
6. BWST temperature	85°F	110°F
7. BWST volume	291,463 gal.	312,210 gal.
8. Time to recirculation	3800s	4257s
9. RBS flow before/after recirculation	1500/1500gpm	1000/1000gpm

In addition, hydrogen recombiner heat loads were added, instrument errors in the BWST level accounted for, containment cooler performance data inaccuracies accounted for, and decay heat generation rate corrected (reduced).

The results of the reanalysis are that containment peak temperature and pressure increase from 280°F and 53.1 psig (original FSAR) to 283°F and 53.4 psig (revised case). Since the revised peak pressure is still less than the containment design pressure of 59 psig, the staff finds it to be acceptable. The revised peak temperature does not exceed the maximum EQ temperature; therefore, the staff finds this to be acceptable.

While revised peak pressures and temperatures are within acceptable limits, the revised time versus pressure and temperature profiles are not completely encompassed within the previous EQ profiles. In the case of the pressure profile, the staff finds that the slightly higher profile will not have a significant effect on EQ; therefore, the staff finds the revised pressure profile acceptable.

Concerning the revised temperature profile, it briefly exceeds the EQ temperature profile. An analysis of the new profile was conducted to evaluate the discrepancies. The new temperature profile crosses the EQ profile at several points within the first hour, then is bounded by the EQ profile for the rest of the transient. However, the licensee concluded that if some of the conservatisms in the analysis were removed, the new temperature profile would be maintained within the EQ profile. In addition, the licensee provided the results of the analysis for staff review. Based on a review of the information provided by the licensee, the staff agrees with the licensee's conclusions that the new temperature profile can be maintained within the EQ profiles, and further concludes that there is no apparent impact on the environmental qualification of equipment within the scope of 10 CFR 50.49.

The staff has reviewed the licensee's revised DBA LOCA analysis of peak containment temperature and pressure and has found the assumptions, methods of analysis, and results to be acceptable. Also, as stated above, the staff finds the revised temperature and pressure profiles to be acceptable in terms of their effect on the environmental qualification of equipment inside containment.

### 3.2 Reactor Building Sump and Spray pH as Related to EQ

The additional volume of borated water assumed in the BWST and the reduced flow rate of the RBS, which would affect the drawdown rate of the sodium hydroxide (NaOH) tank, could affect reactor building spray and sump pH during a LOCA. The licensee has performed calculations which resulted in a minimum and maximum pH for the spray and sump water solutions of 8.8 and 10.4, which are within the 8.5 to 10.5 range specified by Standard Review Plan (SRP) Section 6.5.2, Revision 1 (Revision 2 allows any pH greater than 7.0), and the original design and licensing specification of 8.5 to 11.0. Therefore, the staff finds that the revised sump and spray pH is within acceptable limits and would not have an adverse effect on environmental qualification of equipment (EQ) that may be wetted by the spray or submerged by the sump.

### 3.3 Reactor Building Sump Vortexing

Vortex development in the reactor building (RB) sump can entrain air in the pumped recirculation fluid and impede the adequate performance of the ECCS and RBS systems. Therefore, the revised flows of the RBS and LPI systems have been conservatively assessed by the licensee to determine their acceptability with respect to RB sump vortexing.

A considerable amount of research has been performed to evaluate and quantify the vortex phenomena (NUREG-0897, Rev. 1 and NUREG/CR-2758). This data was used to determine if acceptable conditions will exist in the sump. The licensee evaluated the data provided in the referenced reports to determine applicability to ANO-1 based upon geometric considerations. Based upon the applicable data points in these references and the anticipated sump levels and suction flow rates, the licensee has determined that air entrainment due to vortex formation would not impede pump performance during RB sump recirculation. The staff finds this to be acceptable.

### 3.4 Reactor Building Sump Water Level as Related to EQ

Since the BWST water level limit is being increased, the licensee has reanalyzed the RB sump maximum water level and found it to be 9.18 feet (above elevation 336'-6") for a large break LOCA and 8.88 feet for a small break LOCA. This does not result in the submergence of any additional EQ components not qualified for submergence, with the following exception.

The analysis found that, during a large break LOCA, the potential exists for some of the Emergency Feedwater Initiation and Control (EFIC) steam generator (SG) water level transmitters to be slightly submerged. However, SG cooling during a large LOCA does not occur due to RCS voiding, so no required functions would be lost. The licensee would secure the EFW pumps prior to depleting the BWST (time of maximum sump level) so if the transmitters were to fail low if submerged, SG overflow would not occur. Further, the licensee provided (letter dated November 14, 1990), the results of a failure modes and effects analysis for these transmitters. They found that failure of the transmitters would not cause failure of components outside of the EFIC system, but could cause the EFIC SG pressure transmitter to fail. However, because no operator actions are required that are based on SG conditions, the licensee stated that the operators should not be misled by incorrect SG level or pressure readings. Also, during a small break LOCA, when SG cooling is potentially possible, the EFIC SG level transmitters would not be submerged. Therefore, the staff finds that the revised sump water level transient would not have an adverse effect on EQ.

### 3.5 Reactor Building Sump Temperature as Related to EQ

The licensee has found that maximum sump temperature at the time of recirculation has increased from 250°F to 255°F. The licensee has provided information which shows that the ECCS and RBS sump suction piping would not be adversely affected by this increase. Further, by letter dated November 14, 1990, the licensee stated that submerged equipment is qualified to 282°F. Therefore, the staff finds the RB sump temperature to be acceptable as related to EQ.

### 3.6 Summary

The staff has reviewed the five topics discussed in Sections 3.1 through 3.5 above and finds them to be acceptable. Therefore, the staff finds that the requested license amendment is acceptable.

## 4.0 STRUCTURAL AND GEOSCIENCES BRANCH EVALUATION

### 4.1 Seismic Adequacy of the BWST

The previously discussed system modifications and reanalysis involves increasing the BWST maximum water level Technical Specification limit from 40.0 ft. to 42.5 ft. The licensee, however, has evaluated the BWST considering the maximum level at 43.0 ft. This evaluation addresses the seismic adequacy of the BWST, when the maximum water level in the tank is 43.0 ft.

The BWST (Refs. 7 and 8) is located on the southwest side of the Unit 1 containment structure, and is located on the roof of the Unit 1 tank vault which bears directly on sound rock. The tank shell diameter is 40 ft. 9 in., and the height to the spring line is 40 ft. 0 in. above the bottom of the tank. The tank is fabricated from ASTM A-131, Grade C material with the nominal thickness of the ellipsoidal dome to be 1/4 in., and cylinder thickness varying from 5/16 in. for top course to 9/16 in. for the bottom course. The thickness of the bottom plate is 5/16 in. A corrosion allowance of 1/16 in. is included in the tank shell thickness. The tank is sitting on a concrete ring-wall with the annular space filled with compacted oiled sand.

The tank was originally designed (Ref. 1) using the standard ANSI/AWWA D-100-84 developed by the American Water Works Association (Ref. 9). In the original design the maximum water head was considered as 40.0 ft. The licensee revised the original calculations to reflect the stress increases due to the increased head of water in the shell, roof-weld, anchor-bolt, and anchor-bolt chair. All the revised stresses were found to be well within the allowables.

As the tank is located on a rigid vault, the seismic input for the tank was considered as zero period acceleration (ZPA) of the design ground response spectra. Also, the referenced standard (Ref. 9) utilized TID 7024, in formulating the design forces, i.e., assuming the tank to be rigid. The staff accepts the first assumption as the vault is described as a low level rigid concrete structure (having natural frequencies in excess of 25 Hz). However, for this tank the second assumption is not correct when the natural frequencies corresponding to fluid-tank interaction are considered. As the stress calculations indicated substantial margins above the allowables, the staff, by comparison with other similar tanks reviewed, finds that the tank as designed should be able to withstand the maximum postulated earthquake (ME) without failure with an additional 3 ft. head of water.

This is an interim staff position and the staff requires the following actions by the licensee, prior to reaching a complete resolution of the issue.

1. Walkdown of the tank to assess the "as is" condition of the tank (i.e., identify thickness reduction due to corrosion (if any), weld conditions, the anchor-bolt characteristics such as size, tightness and embedment). Also, a check should be performed to ensure that there are no cracks in the concrete running in the vicinity of the anchor bolts.
2. Perform tank design adequacy calculations considering the tank flexibility and incorporating the results of the walkdown.

By letter dated November 5, 1990, the licensee has committed to complete these actions by the end of January 1991.

#### 4.2 Summary

On the basis of the review, of the licensee submitted documents, and teleconferences, the staff has reached an interim finding that the design of the BWST is adequate to withstand the loads imposed by an additional 3 ft. head of water. As committed by the licensee, the licensee is to check the adequacy of the tank considering the flexibility of the tank, after a walkdown, to assess the "as is" condition of the tank. The staff will review the adequacy check prior to the completion of the next refueling outage.

#### 5.0 HUMAN FACTORS ASSESSMENT BRANCH EVALUATION

##### 5.1 HPI Flow Indications, EOP Guidance and Operator Training

This license amendment includes modifications to the High Pressure Injection (HPI) flow indications, Emergency Operating Procedure (EOP) guidance, and operator training. The staff conducted its review of this license amendment request as it relates to conformance to the human factor issues of the Standard Review Plan, NUREG-0800 Section 13.2.1, Reactor Operator Training; and Section 13.5.2, Operating and Maintenance Procedures.

The staff has reviewed the revised technical content for Engineered Safeguards Actuation System (ESAS) EOP 1202.01 and finds that the changes to be made will provide adequate information and guidance for operator use of the modified portion of the HPI system. The staff also finds that the new flow indications will be appropriately located, and have appropriate scale units (GPM) and scale graduations (2 GPM) for the operator to conduct EOP actions.

By letters dated October 25 and November 5, 1990, the licensee submitted additional discussions of the plant and simulator hardware and software modifications, the technical content of the revision to ESAS EOP 1202.01, and the planned training on the design change. The training will commence with 2 to 4 hours of classroom training including the following topics:

1. Changes to design bases.
2. Panel equipment changes.
3. System flow paths.
4. Automatic actuations.
5. Equipment power supplies.
6. Required operator actions.

Simulator modifications will not be completed prior to plant startup, but within the time limits prescribed by ANS 3.5 as endorsed by Regulatory Guide 1.149. In the interim, simulator training will take place with the exception of simulator training on HPI line ruptures that requires throttling and flow balancing with equipment not installed on the simulator. For training where HPI System/Operator interface is possible or required, trainers will describe and discuss differences between plant and simulator equipment before the training starts, to prevent negative training. When the hardware and software changes have been made to the simulator to model the HPI modifications, the licensee will conduct simulator training in the use of the system in all relevant operating modes. Additional classroom training on the modification will be coordinated to coincide with the simulator training.

The license has committed to have a final ESAS EOP 1202.01 that will be written in accordance with their EOP Writers Guide. The guide was revised in response to the EOP Inspection conducted in April 1990. The final EOP 1202.01 will be in place by December 31, 1991. The staff finds that this is an acceptable method to write the procedure.

## 5.2 Summary

Based on the staff's review of the licensee's commitments and licensee's submittals and teleconferences with the licensee, the staff finds that the proposed EOP guidance, operator training, and the human factor aspects of the change to the control room instrumentation with respect to the modification to be acceptable in accordance with the Standard Review Plan, NUREG-0800.

## 6.0 INSTRUMENTATION AND CONTROL SYSTEMS BRANCH EVALUATION

### 6.1 Modification of the HPI, RBS and LPI Flow Instrumentation

Regulatory Guide (R.G.) 1.97 recommends the use of Type D, Category 2 HPI, RBS and LPI flow instrumentation to monitor the operation of the safety injection systems. R.G. 1.97 defines Type D variables as, "those variables that provide information to indicate the operation of individual safety systems and other systems important to safety." The Category 2 criteria includes environmental qualification, control room display, and a high-reliability power source. In a letter dated June 25, 1984, the licensee committed to conform to the Type D and Category 2 criteria of R.G. 1.97.

In the August 8, 1990 letter, the licensee declared that this modification upgrades the HPI, RBS and LPI flow instrumentation to a Type A variable. R.G. 1.97 defines Type A variables as, "those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events." R.G. 1.97 also states that Type A variables should meet the Category 1 criteria. The Category 1 criteria includes environmental qualification, seismic qualification, redundancy, continuous read-time control room display, and Class 1E power sources. The licensee has committed to meet the Category 1 criteria for HPI, RBS, and LPI flow instrumentation.

## 6.2 Summary

Based on the staff's review of the application, the staff finds that the HPI, RBS, and LPI flow instrumentation at ANO-1 is acceptable with respect to conformance with R.G. 1.97, Revision 3.

## 7.0 RADIATION PROTECTION BRANCH EVALUATION

### 7.1 Post-LOCA Offsite Doses

As stated previously, the amendment involves two separate license issues; (1) High Pressure Injection line break, and (2) Reactor Building Spray (RBS) pump net positive suction head (NPSH). This review is limited to the potential changes to the previously analyzed post-LOCA offsite doses due to the throttling of the RBS flow during and following a LOCA (item 2 only).

The RBS system at Arkansas Nuclear One, Unit No. 1 provides (1) reactor building (RB) atmosphere cooling to reduce the RB pressure to near pre-accident conditions, and (2) the removal of the fission products from the RB atmosphere to reduce the airborne fission product inventory available for leakage to the environment. This license amendment among other things, involves revision of guidance in the Arkansas Nuclear One, Unit No. 1 emergency operating procedures (EOPs) to throttle the RBS flow following a LOCA to provide adequate net positive suction head for the RB sump pump and to avoid RB sump vortexing.

The throttled (reduced) RBS spray flow will affect the removal efficiency of elemental and particulate iodines in the RB atmosphere. The effectiveness of the spray against iodine vapor in the RB is primarily determined by the rate at which the spray solution is introduced into the RB atmosphere. Therefore, as the spray flow increases, so does the removal efficiency. The licensee's proposed spray flow is 1000 gpm compared to the current design flow rate of 1500 gpm.

The staff has reevaluated the following iodine removal efficiencies using the reduced flow rate and the revised methodology provided in the SRP Section 6.5.2, Revision 2.

### Removal Efficiencies (hr<sup>-1</sup>)

	USAR (1)	SSER (2)
Elemental	10	11.2
Particulate	0.72	2.4
Organic	0	0

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(1) Arkansas Nuclear One, Unit No. 1 USAR, Amendment No. 6, Table 14-51.

(2) Staff's calculated values.

Using the above recalculated values for iodine removal efficiencies, the staff calculated the following post-LOCA offsite doses. All other assumptions and parameters including the atmospheric dispersion factors (X/Qs) are the same as shown in Table 15-2 of ANO-1 Safety Evaluation dated June 6, 1973.

Offsite Post-LOCA Doses (Rem)

<u>Exclusion Area Boundary</u>	SER (3)	ANO-1 (4)	SSER (5)
Thyroid	158	148	190
Whole Body	13	5	13
<u>Low Population Zone</u>			
Thyroid	62	52	70
Whole Body	5	2	5

(3) Safety Evaluation for ANO-1, dated June 6, 1973 (Table 15-1).

(4) ANO-1 submittal dated August 8, 1990 (Section 4.2.3).

(5) Staff's calculated values.

## 7.2 Summary

On the basis of this safety evaluation, the staff finds that the proposed license amendment to reduce the RBS system flow rate is acceptable. The bases for our acceptance are that the offsite post-LOCA doses with the reduced RBS system flow rate remain within the dose reference values specified in 10 CFR Part 100 and that this proposed change does not alter our conclusions stated in Section 15 of the ANO-1 Safety Evaluation, dated June 6, 1973.

## 8.0 OVERALL SUMMARY

On the basis of the entire safety evaluation, the staff finds that the proposed amendment to increase the authorized steady-state reactor core power level to a maximum of 2568 megawatts thermal (100% of full power), and to increase the BWST level and revise the number of HPI motor operated valves referenced in the TSs, is acceptable.

## 9.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposures. The Commission has previously



issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

#### 10.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 5, 1990

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## 11.0 REFERENCES

1. Letter N. S. Carns, Entergy Operations, Inc., to USNRC, "Revised Request for License Amendment to Increase Reactor Power to a Level of 100 Percent and Borated Water Storage Tank Technical Specification Change," dated August 8, 1990.
2. B&W Report: 86-1179795-01, "ANO-1 HPI Flow Rate Requirements," by R. N. Ellison and J. C. Seals, dated August 1990.
3. ANO Engineering Report 89R-1006-02, "Resolution of DCD Identified ECCS and RBS NPSH Inadequacies," by B. Daiber, et al., dated August 3, 1990.
4. B&W Calculation 32-1102665-00, Rev. 0, "LPI Flow Split for CF Line Break (NSS-8), dated May 27, 1979.
5. ANO Calculation 89E-0164-01, Rev. 1, "Post Accident Water Level in Containment," dated May 31, 1990.
6. ANO Calculation 89E-0164-05, Rev. 2, "Maximum RB Sump Water Level Post LOCA," dated April 13, 1990.
7. Letter J. J. Fisicaro, Entergy Operations, Inc., to USNRC, "Response to Request for Additional Information on 100% Power Technical Specification Change," dated October 25, 1990.
8. Letter from J. J. Fisicaro, Entergy Operations, Inc., to USNRC, "Response to Request for Additional Information on 100% Power Technical Specification Change," dated November 5, 1990.
9. ANSI/AWWA D-100-84, "AWWA Standard for Welded Steel Tanks for Water Storage."