NUREG-0308
Suppl. No. 2

Safety Evaluation Report

U. S. Nuclear Regulatory Commission

related to operation of

Office of Nuclear Reactor Regulation

Arkansas Nuclear One, Unit 2

Docket No. 50-368

Arkansas Power and Light Company

September 1978

Supplement No. 2

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Supplement No. 2 to
NUREG-0308

SUPPLEMENT NO. 2

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 2

DOCKET NO. 50-368

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

APPENDICES

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 Introduction

On November 11, 1977 the Nuclear Regulatory Commission (Commission) issued its Safety Evaluation Report regarding the application for a license to operate the Arkansas Nuclear One - Unit 2 (ANO-2) facility. The application was filed by the Arkansas Power and Light Company. Supplement No. 1 to the Safety Evaluation Report was issued on June 10, 1978 and documented the resolution of certain outstanding review items noted in the Safety Evaluation Report and summarized the status of the remaining outstanding issues.

This report, Supplement No. 2 to the Safety Evaluation Report, provides (1) our evaluation of additional information received from the applicant since preparation of Supplement No. 1 regarding previously identified outstanding review items, (2) our responses to comments made by the Advisory Committee on Reactor Safeguards in its report dated April 12, 1978, and (3) our evaluation of additional or revised information related to new issues that have arisen since preparation of Supplement No. 1.

Sections of this supplement carry the same numbers as those of the Safety Evaluation Report and Supplement No. 1 which these sections supplement or modify, and except where specifically noted, do not replace sections of the Safety Evaluation Report.

1.6 Summary of Outstanding Review Items

Items previously identified as outstanding have been resolved since preparation of the Safety Evaluation Report Supplement No. 1 as indicated below to support issuance of the operating license. New issues addressed since the preparation of Supplement No. 1 as noted by an asterisk are also included in the listing below. The resolution of the remaining outstanding item will be required prior to the authorization for operation of the plant in operational mode 2 and 1 conditions.

Resolved Items

Items are identified by the section number of this report in which they are discussed.

3.10 Seismic Qualification of Safety-Related Instrumentation

4.2.1* Burnable Poison Design Verification

4.2.4* CEA Guide Tube Integrity

4.2.4* CEA Rod Worth Surveillance

5.2.1* Fracture Toughness - Appendix G

5.2.2* Reactor Vessel Materials Surveillance Program - Appendix H

5.2.9* Inservice Testing of Pumps and Valves

5.4* Inservice Inspection Program

5.6.2* Steam Generator Tube Integrity

6.2.1 Main Steam Line Break Mass and Energy Releases

6.2.1 Environmental Qualifications for Safety-Related Equipment for Main Steam Line Break Inside Containment

6.2.4* Containment Isolation Systems

6.2.6 Containment Leakage Testing Program

6.3.3 Evaluation of Emergency Core Cooling System Performance

6.3.4 Evaluation of Emergency Core Cooling System Operation in the Recirculation Mode

7.1 Verification of Implementation of Instrumentation and Control Systems Design

7.2.2 Input Fault and Surge Testing for Power Supplies

7.2.3 Core Protection Calculator System

7.5.1 Accident and Post Accident Monitoring

7.6.3 Redundant Valve Position Indication

7.8* Electrical Penetrations

7.9.4 Separation Criteria For Conduits

8.2 Offsite Grid Stability

9.5.1* Fire Protection Review

10.6 Feedwater Hammer in Steam Generators

13.3* Emergency Plan

14.0 Initial Tests and Operations

15.4.2 CESEC Code Verification

15.4.4 Main Steam Line Break Analysis

15.4.6* ECCS Pump Room Leakage

20.0 Financial Qualifications

<u>Outstanding Items</u>

3.11 Environmental Qualification of Safety Related Equipment

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

In Supplement No. 1 we stated that we considered the matter of the size of the low population zone radius to be resolved pending documentation in the operating license application of the 2.6 mile radius low population zone.

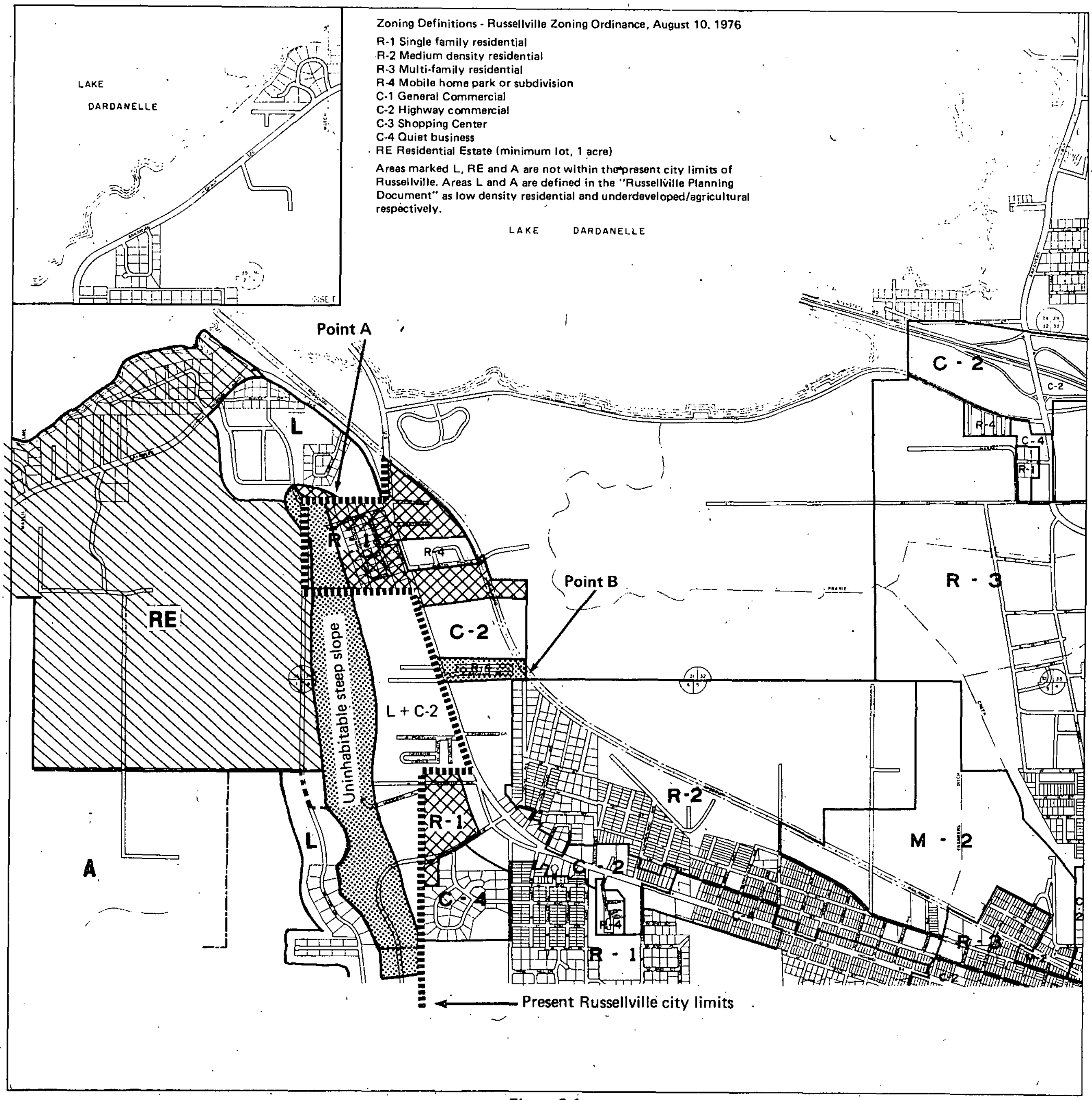
The change to the 2.6 mile low population zone was included in Amendment No. 46 to the Final Safety Analysis Report as related to the accident analyses presented in Section 15.0 and in the facility Technical Specifications.

Since preparation of Supplement No. 1 we have made relative concentration (X/Q) atmospheric diffusion estimates at a distance of 4184 meters (2.6 miles) for various time periods following a postulated short-term release from ANO-2. Our evaluation is based on a directionally independent five percentile model which is described in Standard Review Plan 2.3.4 and is consistent with the 0-2 hour X/Q estimate at the exclusion boundary presented in the Safety Evaluation Report. Meteorological data used in the evaluation were collected onsite from January through December 1975. We assumed a ground level release and used a building wake correction factor, cA, of 1100 square meters. The X/Q estimates follow:

Time Period	X/Q (Seconds/Cubic Meter)
0-8 hrs	5.0×10^{-5}
8-24 hrs	3.6×10^{-5}
1-4 days	1.7×10^{-5}
4-30 days	5.9×10^{-6}

Reference may be made to Tables 15.3, 15.4 and 15.6 of this report which have been revised to reflect the 2.6 mile low population zone.

We have noted that Supplement No. 1 discusses Points A and B on Figure 2.1 but that these points are not marked on the figure. Therefore, Figure 2.1 is included with points A and B properly noted.



Zoning Definitions - Russellville Zoning Ordinance, August 10, 1976
R-1 Single family residential
R-2 Medium density residential
R-3 Multi-family residential
R-4 Mobile home park or subdivision
C-1 General Commercial
C-2 Highway commercial
C-3 Shopping Center
C-4 Quiet business
RE Residential Estate (minimum lot, 1 acre)
Areas marked L, RE and A are not within the present city limits of Russellville. Areas L and A are defined in the "Russellville Planning Document" as low density residential and underdeveloped/agricultural respectively.
LAKE DARDANELLE
LAKE
DARDANELLE
Point A
Point B
L
RE
A
R-1
R-4
C-2
L + C-2
Uninhabitable steep slope
C-4
R-2
R-3
M-2
C-4
Present Russellville city limits

Figure 2.1

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.9 Mechanical Systems and Components

3.9.3 Reactor Vessel Supports and Reactor Internals

In a letter dated June 18, 1976, we requested that the applicant provide additional information required for purposes of making the necessary reassessment of the reactor vessel supports for ANO-2. We requested that the applicant perform a detailed analysis to (1) determine the loads in the reactor vessel support system, (2) evaluate the full restraint capability of the reactor coolant system, (3) evaluate the structural capability of the reactor vessel internals, and (4) evaluate the safety margins for each of the components cited above.

The applicant submitted results of his reactor vessel supports analysis, based on the Combustion Engineering topical report CENPD-42, "Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Conditions," methodology on January 17, 1978. This submittal contained reactor vessel support loads resulting from breaks postulated at the reactor vessel nozzle and reactor coolant piping junction in both the hot and cold legs. It was concluded from the analysis that sufficient margin existed in the vessel supports to accommodate consequences of the postulated breaks in the hot and cold leg piping.

On March 17, 1978, we requested the applicant to summarize results of his pipe break analysis, combined with loads resulting from a safe shutdown earthquake and normal operating loads. The applicant's response to our request demonstrated that the reactor vessel support responses based on CENPD-42 methodology, which result from the combined loadings, are less than allowables specified in Appendix F, Subsection NF of the ASME Boiler and Pressure Vessel Code.

The hydraulic blowdown model used to determine the loads on the reactor coolant system components is the model described in the Combustion Engineering, Inc., Topical Report CENPD-42, "Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Conditions."

As stated in our Safety Evaluation Report we do not plan to continue our review of the CENPD-42 methodology, rather we plan to review the Combustion Engineering methodology for calculating loss-of-coolant blowdown forces to be included by Combustion Engineering, Inc. in a description of the CEFLASH-4B code. We expect to complete our review of the CEFLASH-4B methodology by the end of 1978. It is Combustion Engineering's position that the methodology of CENPD-42 will be shown to be conservative by the use of the CEFLASH-4B methodology. In our Safety Evaluation

Report we stated that the operating license would be conditioned regarding the performance by the applicant of a plant specific comparative analysis to demonstrate that the loads currently calculated for the ANO-2 reactor coolant system are conservative when compared to loads calculated using the CEFLASH-4B methodology. We have since concluded that the staff's requirements for additional actions on the part of the applicant resulting from the resolution of our generic review of this matter will be sufficient. Therefore the ANO-2 operating license will not be conditioned in this regard and any additional requirements will be identified as a result of the resolution of our generic review of this matter.

Based on the results of the current analyses which demonstrate that the reactor vessel support responses are less than the ASME Code allowable values and based on our conclusion that the probability of an instantaneous rupture of the reactor coolant loop piping would approach that of the reactor pressure vessel, we conclude that operation of ANO-2 pending completion of the above mentioned comparative analysis poses no undue risk to the public health and safety and is acceptable.

3.10 Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment

Seismic Qualification - Electrical Aspects

In the Safety Evaluation Report we stated that several items identified in our September 7, 1977, letter to the applicant remained outstanding. In particular, the applicant was requested to provide the basis of acceptability for the seismic qualification of the following Class 1E equipment:

(1) Fisher Porter Pressure Transmitter Model 50EP1000

(2) Foxboro Pressure Transmitter Model E11GM

(3) Fisher Porter Differential Pressure Transmitter Model 2490

The information submitted for these components identified that during the seismic testing, error deviations between 4.4 percent to 12 percent were recorded with no justification provided as to the adequacy for such deviations.

In response to our request for information concerning this matter, the applicant stated in a letter dated January 3, 1978 that the actual frequency and seismic acceleration levels at which the equipment was tested were more severe than the actual conditions that could exist at the installed location of these sensors, and provided the actual deviation values, based on tests that would be detected at their installation location. The actual deviation values for this equipment on the ANO-2 installation location were reported to range between one to 3.7 percent, and have been factored into the plant protection system setpoint calculations. The applicant provided the documentation to support the justification that the actual response spectrum at the equipment location is 0.5 g at 11 to 19 Hertz; and to provide sample calculations which factor these deviations into their setpoint calculations.

Based on our review of the seismic qualification of this equipment and the justification provided by the applicant, we conclude that the design satisfies the Commission's requirements stated in Section 3.10 of the supplement to the Safety Evaluation Report and in our letter referenced above and is, therefore, acceptable.

Seismic Qualification of Mechanical Aspects

We identified specific items of Class 1E equipment and associated unresolved issues pertaining to the seismic qualification of that equipment including the core protection calculator system (CPCS) components in a letter to the applicant dated September 7, 1977. Since that date, the applicant has submitted his qualification procedures and test results for the identified equipment. We have reviewed the applicant's submittals and conclude that the seismic qualification of the CPCS components is acceptable for ANO-2 specific application and also that the applicant's seismic qualification programs for the other identified equipment is acceptable.

We conclude that the applicant's seismic qualification program for seismic Category I instrumentation and electrical equipment provides adequate assurance that such equipment will function when subjected to excitation from a safe shutdown earthquake and during post-accident conditions and is an acceptable basis for satisfying the applicable requirements of General Design Criterion 2.

3.11 Environmental Qualification of Safety-Related Equipment

Subsequent to the preparation of the Safety Evaluation Report, we identified concerns which necessitated a review of additional selected safety-related equipment qualification documentation. Additional information was requested by the staff and submitted by the applicant.

We reviewed the environmental qualification reports, consisting of test procedures and test result summaries submitted by the applicant for the following equipment located inside containment.

(1) Target Rock solenoid valves
(2) Bypass damper motor used in conjunction with the containment cooling fans
(3) Electrical penetrations
(7) Electrical cable
(8) Joy Fan on motor
(9) Foxboro Transmitters
(10) Fisher Porter Transmitters
(11) Hydrogen recombiner

(4) Electrical connections

(5) Limitorque valves

(6) Rotorque valves (used for submergence)

(12) Containment sump level transmitters

(13) Containment radiation monitors

The following evaluation, description and concerns regarding the qualification are provided for each of the items reviewed. The adequacy of the environmental test envelope for the loss-of-coolant accident and the main steam line break (LOCA and MSLB) events is addressed in Section 6.2 of this report.

(1A) <u>Target Rock Control (TRC) Valve Model 75G-003; 1 Inch Y Pattern Solenoid Motor Valve; Ref. Report #18278 dated 11/4/76; Test Procedure Report #1674</u>

Qualification by type test was performed on this equipment during which time the equipment was exercised to demonstrate its functional operability. Prior to the testing under loss-of-coolant (LOCA) conditions the unit was preconditioned by exposing it to 10^8 rads and operated through 1000 cycles in an environment of 175 degrees Fahrenheit, at a relative humidity of 95 percent which was maintained for 10 days. Subsequently, the unit was exposed to a hostile (LOCA) environment of 340 degrees Fahrenheit for three hours twice, followed by reduced temperature and pressure stages which continued for a total of 30 days. Caustic spray with a pH of 10.5 was introduced periodically during the LOCA test sequence. From our review it was not readily apparent that the same piece of equipment was exposed to the sequence described above, and the applicant was requested and verified that the sequence described was indeed correct. We therefore conclude that the qualification of this equipment satisfies the requirements of IEEE 323-1971 and is acceptable.

(1B) <u>TRC Model 72V; 1 Inch Y Pattern Solenoid Motor Valve; Ref. Report 1500 Dated 10/22/77; Test Procedure Report #1383; TRC Model 73E-001; 1 Inch Solenoid Globe Valve; Y Pattern With Position Switch; Report 798-4 Dated 8/3/74; Test Procedure Report #1383</u>

The qualification type tests for this equipment were conducted utilizing the same sequence, LOCA peak temperatures and pressure, and duration as described in (1A), above, for model 75G-003 with minor deviations in preconditioning the equipment and not utilizing caustic spray with a pH of 10.5 as required by the design. The applicant advised us that the material, seals and housing are identical to the equipment identified in (1A), above, which included a caustic spray with a pH of 10.5, and only the internal mechanism which did not come in contact with caustic was different. We therefore conclude that the qualification of this equipment satisfies the requirements of IEEE 323-1971 and is therefore acceptable.

(2) Bypass Damper Motor Qualification

As a result of our review we determined that the bypass damper motor did not have adequate qualification documentation to assure that the equipment will maintain its operability during a LOCA/MSLB accident. In response to our concerns the applicant attempted to justify not qualifying this motor on the basis that it is only required to function for a very short period of time (i.e., within 20 seconds of the accident), and due to its location the ambient conditions would not exceed 200 degrees Fahrenheit. Based upon our evaluation of the applicant's response, we concluded that the justification provided was insufficient, and required that the applicant modify the design and provide a qualified Class 1E motor for this function and provide test documentation to assure that this equipment will function in a LOCA/MSLB environment as committed to in Section 3.11 of the Final Safety Analysis Report.

In response to the staff's requirements via letter from D. Williams to J. Stolz (NRC) dated June 23, 1978, the applicant stated that it had conducted a series of qualification tests on another type motor (e.g., Baldor) and provided a summary of the test. The test consisted of irradiating the motor to 1×10^5 rads, subjecting the irradiated motor to random biaxial seismic testing in conformance with IEEE 344-1975, and then subjecting the unit to a hostile steam environment of 300 degrees Fahrenheit, 60 pounds per square inch gauge (psig), during which time functional operability was verified. The test duration at elevated temperature and pressure was for a short period of time; however, margin was provided to assure that the equipment would maintain its operability for the required design period. The applicant justified omitting a chemical spray environment on the basis that the motor would have completed its function before the containment spray system would be actuated. In addition, the applicant committed to replace the existing motor with the totally enclosed qualified Baldor motor before fuel loading.

Based upon our review of the applicant's response, we conclude that the modified design satisfies the staff's requirements identified above and is, therefore, acceptable.

(3) Electrical Penetrations Qualification

The qualification for the following four types of electrical penetrations used in ANO-2 was conducted by type test.

A. 750 kcmil modules (medium voltage)
B. 3 x 350 kcmil modules (low voltage)
C. Hybrid low voltage module
D. Instrumentation module

The applicant submitted the following qualification documents for our review to demonstrate the adequacy of the design: (1) Test Report #123-1247 dated April 1973, and (2) Test Report #123-2045 (Rev. A) dated March 1975.

For penetration modules B and C, two separate tests were conducted consisting of preirradiating the units to 10^8 rads. One test exposed the units to a LOCA environment of 300°F, 58 psig, for 15 minutes and then at a reduced temperature and pressure (i.e., 255°F, 20 psig) for 24 hours with intermittent chemical spray (pH of 5.0). The other test exposed a preradiated module B to a temperature of 309°F at 63 psig for one hour, followed by reduced temperature and pressure at 259°F, 21 psig, for 24 hours with a caustic spray of pH equal to 10.5. The applicant has indicated that for additional margin, the same (B) module was previously subjected to a LOCA test at a reduced peak temperature before subjecting it to the above test. The second test for module C was conducted by preirradiating the module and exposing the unit to a LOCA environment, initially to a temperature and pressure of 303°F at 55 psig for 15 minutes followed by reduced conditions of 288°F at 42 psig for a total of 24 hours, with caustic spray of pH equal to 10.5. The LOCA qualification test conducted for modules A and D was 305°F at 57 psig for 1/2 hour followed by a reduced temperature of 260°F for the remaining 24 hours for module A, and 303°F at 55 psig for the first 15 minutes, 286°F at 40 psig for 3 hours followed by a reduced temperature of 258°F for the remaining 24 hours for module D. Prior to the LOCA tests, these modules had also been subjected to a radiation dose of 10^8 rads.

During our review, the applicant informed us that the penetrations were not energized during the LOCA environment testing and thus did not fully conform to the requirements of IEEE 317-1972 as previously stated. The applicant justified this exception by analyzing the circuits associated with the penetrations for additional energy content of the circuits which remained continuously energized during the worst case accident event. The applicant's analysis revealed that only four circuits, as a result of being continuously energized, contributed to a higher local temperature environment (above the LOCA temperature). However, the resultant temperature was still below the qualification test envelope for three of the four types of penetrations.

Based upon our review of the test results and the applicant's analysis, we conclude that the qualification test of penetrations type B, C and D sufficiently envelopes with margin the defined worst case environment and is, therefore, acceptable.

The adequacy of the qualification of type A penetration was not fully resolved by the documents referenced above for the following reasons. The energy content for the reactor coolant pump energized circuits associated with penetration A increased the LOCA environment temperature by 33°F above the qualification test envelope. Although the applicant attempted to justify this design by referencing another report that tested similar module material samples to oven tests of 350°F for four hours, we could not conclude that these tests demonstrated that these modules would maintain their integrity if subjected to a LOCA environment, since these samples were not exposed to radiation, pressure and caustic spray conditions.

Subsequently the applicant submitted a letter dated July 13, 1978 which describes additional testing performed on a similar penetration assembly. The tests conducted envelope with substantial margin the worst case environmental conditions that are expected to occur at the plant. These additional tests were conducted on a penetration module that was not identical to the one used at the site; however, the material and seals of the test module are the same as the materials and seals used in the penetration assembly at the site.

Based on our review of this information, coupled with the information reviewed and described as discussed above, we conclude that sufficient qualification information is provided to assure that the medium voltage penetrations will maintain their integrity in the postulated worst case environmental conditions defined for this plant and is, therefore, acceptable.

(4) Electrical Connection Qualification

During our review, the applicant was requested to describe the qualification adequacy of the connections used inside containment. In response, the applicant provided a summary of tests that were conducted on two types of connections used. The samples were qualified by type tests and were subjected to 2×10^8 rads, caustic spray (pH 9.5 to 11.0), and exposed to a LOCA environment of 346°F at 113 psig for a period in excess of three hours (for one test) and 357°F at 70 psig for a period in excess of 10 hours. The test duration at lower temperatures and pressure was 55 hours for one test and 30 days for the other test.

Based upon our review of the applicant's response, we conclude that the qualification test enveloped, with sufficient margin, the plant's defined design criteria and is acceptable.

(5) Limitorque Valve Qualification

Qualification for these valves was conducted by type testing and documented in Test Report FC 3441 which was submitted by the applicant. The units tested were previously exposed to 2×10^8 rads; one of the two units tested was also previously exposed to a steam and chemical spray environment for 12 days and seismically tested before subjecting it to a LOCA environment. The test units were subjected to a temperature pressure environment of 343°F, 0 psig, for the following three hours. The temperature and pressure were again increased to 339°F at 105 psig for an additional three-hour cycle and then reduced to 320°F at 77 psig for three hours followed by 250°F at 15 psig for the next four days. In addition, the applicant referenced Report FC 2232-01 which is available for audit which describes additional tests conducted using borated water of pH equal to 7.7; the duration of these tests was seven days.

Based upon our review of the qualification documentation provided (Report FC 3441 dated September 1972), and the available additional reports available for audit, we conclude that the type test conducted adequately envelopes the design conditions defined by the applicant for a LOCA and satisfies the requirements of IEEE 323-1971 and is, therefore, acceptable.

(6) Rotorque Valve Qualification

During our review, the applicant informed us that certain select valves were required to operate while submerged, and submitted the qualification documentation by type test to demonstrate the adequacy of the design. As a result of our review, we find that the documentation submitted provides adequate assurance that the valves will maintain their functional operability in a LOCA environment of 90 pounds per square inch gauge psig, 340 degrees Fahrenheit (°F), with caustic spray and accumulative dose of 2×10^8 rads. These valves were tested for submerged operation at 54 psig with water temperatures of 300°F for two hours, decaying to 100°F after 30 days. We therefore conclude that the design satisfies the requirements identified in IEEE 323-1971 and is, therefore, acceptable.

(7) Qualification of Electrical Cable

In support of the adequacy of the electrical cable used in the ANO-2 design, the applicant submitted Test Report FC 3341 dated January 1973.

The test was conducted utilizing the following envelope: three hours at 340°F, 105 psig, 100 percent relative humidity, caustic spray (pH 9.5) with an accumulated dose of 10^8 rads followed by a reduced temperature and pressure of 320°F and 75 psig for 3.5 hours, then 250°F, 15 psig for 3.5 days; the test continued for a total of 47 days at incrementally lower temperatures and pressures. During the test sequence the cables were energized and periodically monitored to verify their functional integrity. In addition, in a letter from D. Williams to J. Stolz dated June 23, 1978, the applicant referenced another report which is available for audit (i.e., F-C4350-3), which reported the results of tests of the same type cables to a steam temperature of 346°F and pressure in excess of 100 psig for eight hours. These tests also preirradiated the samples and exposed them to a caustic spray environment with a pH between 9-11. Also, the applicant provided a summary of the qualification tests conducted on the balance of the control cable and all coaxial cable required for post-LOCA conditions, and identified the test reports that are available for staff audit (i.e., Report #EM 517A); the test envelope peak temperature and pressure ranged between 360°F to 370°F at 65 psig to 80 psig for a duration ranging between five hours and 24 hours. Some of the samples were irradiated to 10^8 rads prior to the LOCA test; a caustic spray between a pH of 7 and 8 was used.

Based upon our review of these reports, we conclude that the testing satisfied the requirements of IEEE 323-1971 and is acceptable.

(8) Fan and Motor Qualification

The applicant submitted a fan and motor qualification report dated May 14, 1975, prepared by the Joy Manufacturing Company, to support the qualification adequacy of this equipment.

Qualification testing was conducted by subjecting the equipment to five sequential exposures to a hostile steam environment of 300°F at 90 psig for 2-hour periods, followed by a reduced temperature and pressure (e.g., 200°F, 20 psig) for a period of 7 days. Although the report does not specifically define the radiation and caustic spray environment utilized, information is presented which indicates that the material used in the construction of this equipment is highly resistant to radiation, and that the sodium hydroxide solution is significantly higher than the conditions that may exist during an accident. The values provided are in the range of 10^9 rads at approximately 480°F with caustic spray resistance to a pH of 14.0.

Based upon the information presented, we conclude that the tests conducted for this equipment, and the margins provided, provide adequate assurance that the equipment qualification satisfies the requirements of IEEE 323-1971 and is acceptable.

(9) Qualification of Foxboro Transmitters

The applicant utilizes the following two types of Foxboro Transmitters for safety-related systems actuation and indication: (a) Foxboro Model E11GM, and (b) Model E11AH. In support of the adequacy of the qualification for this equipment, Test Reports T3-1097, dated November 1973, and T3-1013, dated June

1973, were submitted for our review which described the testing conducted on Model E11GM. The applicant stated that Model E11AH was identical to Model E11GM with the exception that one of the sensing ports was sealed in order to sense absolute pressure. Therefore, the report is also applicable to this type of transmitter. The testing consisted of exposing the unit to 300 degrees Fahrenheit (°F) at 60 pounds per square inch gauge (psig) for about two hours, and at reduced temperature and pressure (i.e., 244°F, 20 psig) for the remainder of the test which lasted 24 hours. During the testing, caustic spray of a pH of 10.0 was used and the units were periodically checked to verify the functional operability. The effects of radiation were conducted by separate effects tests conducted on the same type of amplifiers ranging between 8.6×10^7 rads to 2.2×10^8 rads; however, the instruments were not subjected to hostile environments after the radiation test.

Based upon our review of the documentation provided, we find that the qualification methodology utilizing separate effects testing is not acceptable for these sensors. The applicant was advised that these separate effects tests do not provide sufficient assurance that the equipment will be capable of performing its intended function if subjected to radiation and a hostile environment as required in the design.

A more in-depth review of this equipment was conducted on the D. C. Cook Unit 2 plant with similar conclusions. We therefore require that the applicant conduct sequential testing on this equipment which would expose the same piece of equipment to radiation, seismic and environmental effects that are calculated to occur at the plant (with margin), and demonstrate that the equipment will maintain its functional operability under these conditions, or replace this equipment with other transmitters that are qualified to the above requirements. We will require that this matter be resolved to our satisfaction prior to the issuance of authorization for plant operation in operational modes 2 (initial criticality) and 1 (power operation).

(10) <u>Qualification of Fisher Porter Equipment</u>

In support of the adequacy of the qualification of the Fisher Porter equipment used for safety-related equipment, the applicant submitted Test Report #2204-51-B-006 and additional information submitted in a letter from D. Williams to J. Stolz (NRC), dated June 23, 1978, which summarized the type test conducted.

The testing consisted of exposing the units to a temperature and pressure environment of 320°F at 75 psig for one hour followed by reducing temperature and pressure at discrete levels which leveled out at 228°F, five psig, for the remainder of the 24-hour test. During this testing, the equipment was not exposed to any caustic spray environment, nor were the samples exposed to a radiation dose prior to the equipment testing. Separate effects tests were conducted to levels of 1.2 x 10^8 rads; however, the details of this testing were not provided.

Based upon our review of the documentation provided, we find that the qualification methodology utilizing separate effects testing is not acceptable for these sensors. The applicant was advised that these separate effects tests do not provide sufficient assurance that the equipment will be capable of performing its intended function as required in the design. We therefore require that the applicant conduct sequential testing on this equipment which would expose the same piece of equipment to radiation, seismic and environmental effects that are calculated to occur at the plant (with margin), and demonstrate that the equipment maintains its functional operability under these conditions, or replace the equipment with other transmitters that are qualified to the above requirements. We will require that this matter be resolved to our satisfaction prior to the issuance of authorization for plant operation in operational mode 2 (initial criticality) and mode 1 (power operation).

(11) Hydrogen Recombiners Qualification

The applicant referenced Westinghouse Proprietary Report WCAP-7709-L, Supplement Nos. 1, 2, 3 and 4, for the LOCA test qualification documentation to support the adequacy of this equipment. The staff has previously reviewed and accepted this report through Supplement No. 4 and concluded that the design sufficiently satisfies the requirements of IEEE 344-1971 and IEEE 323-1971. We therefore conclude that the applicant's referencing of this report is sufficient and therefore acceptable.

(12) Containment Sump Level Sensor Qualification

In support of the qualification of the GEMS/DELAVAL containment sump level sensors, the applicant identified Test Report FC 3834, which was available for the staff audit, and summarized the qualification testing that was conducted. The type test unit was exposed to two sequential LOCA tests to verify short-term and long-term operability. The unit was previously subjected to a radiation dose of 2 x 10^8 rads before subjecting the units to the LOCA test. The test temperature and pressure were at 282 degrees Fahrenheit (°F) at 59 pounds per square inch gauge (psig) for one hour followed by reduced temperature and pressure of 150°F, 13.5 psig, for an additional 14 days. Subsequently, another environmental exposure was conducted with a peak temperature, transient, of 300°F within 8 minutes of the test, and 4 hours at stabilized conditions of 298°F at 55 psig. Caustic chemical spray of a pH of 10.5 was used. It was not apparent from the applicant's summary description how to determine whether during the test sequence these transmitters were energized and monitored to verify their functional operability. The applicant was requested and verified that the test conducted on these sensors utilized energized components and that functional operability as required by the design had been demonstrated.

Based upon our review of the applicant's summary of the testing conducted on these sensors, and subsequent verification that the equipment was energized and monitored to verify its functional operability during the test sequence, we conclude that the design satisfies the requirements of IEEE 323-1971 and is acceptable.

(13) Qualification of the Containment Radiation Post-Accident Monitoring

During our review, the applicant informed us that the qualification for containment radiation monitors used for post-accident monitoring has not yet been established; the applicant estimates that an approved test plan and test results will be completed by September 1978. The applicant suggested that short-term operation with these nonqualified monitors is acceptable, since portable monitors used outside containment are available as backup. In addition, the applicant stated that the only components that may be exposed to a hostile environment are the sensor itself and its associated cable; the electronics of the monitor is located outside containment. We have included a condition in the operating license which requires that prior to proceeding to operational mode 2 (initial criticality), the licensee shall submit, for our review and approval, a description of and analysis of the use of the portable radiation monitors. The license condition also requires that prior to completion of the initial startup and power ascension testing program, the licensee shall submit, for Commission review and approval, the documentation supporting the environmental qualifications of these monitors for their intended function.

(14) Miscellaneous Balance-of-Plant Equipment Qualification

Subsequent to the preparation of the Safety Evaluation Report, we identified several open items related to balance-of-plant equipment qualification for which additional information was required from the applicant in order to complete our review.

In response to our request, via letters from D. Williams to J. Stolz dated March 30 and April 21, 1978, the applicant submitted the qualification test results of the Fisher Porter transmitters that are required to function in a non-LOCA/MSLB environment. The specific application for these transmitters is to monitor the status of the safety injection tank pressure, level, and flow during normal operation, and also to initiate the out-of-containment recirculation actuation system when required. The design requirement for these sensors is to maintain their operability in an abnormal environment of 140 degrees Fahrenheit and 90 percent relative humidity. Based upon our review of the test results and the information presented in the correspondence identified above, we conclude that the design satisfies the Commission's requirements identified in Section 7.1 of the Safety Evaluation Report regarding this item and is, therefore, acceptable.

In addition, via letters from D. Williams to J. Stolz (NRC) dated June 20, 1978, the applicant submitted the test procedures and a summary of the results that were conducted on the auxiliary relay cabinets that utilize the Potter-Brumfield relays as installed at the plant. The test consisted of exposing the units to discrete temperature and relative humidity conditions varying between 40°F to 122°F and 35 percent to 95 percent relative humidity for durations of four to five hours, at which time functional operability was verified. Prior to these tests, the unit was subjected to seismic testing and the relays were monitored for spurious operation and contact chatter. In addition to the above tests, the applicant stated that prototype tests in accordance with Military Specification MIL-R-1952A, dated December 30, 1966, were also conducted on these relays.

Based upon the information provided, we conclude that the design satisfies the Commission's requirements stated in Section 7.1 of our Safety Evaluation Report regarding this item, and is, therefore, acceptable.

In response to our request regarding the Agastat relay environmental qualification, the applicant, via letter from D. Williams to J. Stolz (NRC) dated June 30, 1977, submitted a summary of the environmental prototype tests that were conducted to demonstrate that this equipment would maintain its functional operability in the limiting environmental conditions defined (i.e., 125°F at 70 percent relative humidity), and referenced the Amerace Corporation Test Report 77-2 dated June 22, 1977, which is available for staff audit. These tests subjected the relays to temperatures ranging between 30°F to 210°F and 95 percent relative humidity at 20°F intervals for a two-hour duration at each level, during which time functional operability was verified at the design maximum and minimum voltage extremes.

Based upon our review of the information submitted, we conclude that the equipment design conforms to the requirements stated in Section 7.1 of the Safety Evaluation Report and is, therefore, acceptable.

4.0 REACTOR

4.2 Mechanical Design

4.2.1 Fuel

Subsequent to the preparation of the Safety Evaluation Report we requested the applicant to provide additional information regarding recent design improvements in the fuel assembly burnable poison rod design. To (a) reduce poison rod growth due to pellet cladding interaction (PCI) and (b) reduce the potential for poison rod failure due to primary hydriding, the fuel vendor, Combustion Engineering, Inc., has made several pertinent design modifications and manufacturing process changes. We requested that the applicant confirm that these revisions were made to the ANO-2 poison rod design and manufacturing processes. The applicant responded that the revisions were incorporated into the ANO-2 Core 1 standard poison rods, and these revisions consist of the following:

(a) increased pellet-to-cladding gap,
(b) tumbled and chamferred pellets,
(c) increased rod pressurization,
(d) reduced plenum spring preload,
(e) reduced pellet moisture limit,
(f) replacement of Al_2O_3 spacers with Zircaloy slugs, and
(g) revised manufacturing processes aimed at reducing mositure ingress to the poison rod.

We have reviewed these revisions and agree that they should significantly reduce PCI and hydride failure of the poison rods.

We have questioned the validity of fission gas release calculations in most fuel performance codes including the FATES code which was used for ANO-2 for burnups greater than 20,000 megawatt days per metric ton uranium. Combustion Engineering, Inc., was informed of this concern and was provided with a method of correcting gas release calculations for high burnups. Other methods of accounting for high burnup releases may be acceptable. Since there has been no question of the adequacy of FATES for burnups below 20,000 megawatt days per metric ton uranium, the Final Safety Analysis Report calculations are acceptable for operation early in life until the peak local burnup reaches this value.

We will require that ANO-2 safety calculations for fuel be redone for burnups greater than 20,000 megawatt days per metric ton uranium with an approved code that accounts for high-burnup releases. This burnup level is roughly equivalent to two reactor operating cycles or on the order of two years. The required calculations should be submitted prior to or with the reload report for that cycle in which the above specific burnup level will be attained.

4.2.4 Reactivity Control System

CEA Guide Tube Integrity

Subsequent to the preparation of the Safety Evaluation Report we learned that there remained another important issue that would require resolution. The problem is that unexpected wear of guide tubes that are under Control Element Assemblies (CEAs) has recently been observed in irradiated fuel assemblies taken from operating Combustion Engineering (CE) reactors. Apparently, coolant flow with accompanying turbulence is responsible for inducing vibratory motions in the normally fully withdrawn control rods, and, when these vibrating rods are in contact with the inner surface of the guide tubes, a wearing of the guide tube wall has taken place. Significant wear has been found to be limited to the relatively soft Zircaloy-4 guide tube because the Inconel-625 cladding on the control rods provides a relatively hard wear surface. The extent of the observed wear has appeared to be plant dependent, but has in some cases extended completely through the guide tube wall.

The Arkansas Nuclear One, Unit 2 (ANO-2) core will be the first CE nuclear power plant to use the 16x16 fuel assembly geometry. It is expected that this new design will be susceptible to CEA guide tube wear as were the older 14x14 fueled plants. Accordingly, the staff required the applicant, Arkansas Power & Light Company (AP&LCo), to submit a safety analysis of the methods by which CEA guide tube wear will be accommodated for in the ANO-2 core. The applicant has provided a topical report, CEN-96(A)-P Revision 1 "ANO-2 Reactor Operation With Modified CEA Guide Tubes and Lengthened Upper Guide Structure Flow Channels," in response to this request.

Summary of Topical Report

The subject topical report describes three methods that will be utilized to reduce the potential for CEA guide tube wear during the first cycle of ANO-2 operation. They are:

(1) the use of chrome-plated, stainless-steel sleeve inserts in selected assemblies,

(2) the use of scupper extensions on the upper guide structure flow channels, and

(3) programmed CEA insertion.

Also described in the topical report is a test program that has been designed to evaluate guide tube wear in four ANO-2 assemblies, which do not use the type of sleeves mentioned in (1) above. These unsleeved assemblies employ 2 types of modifications to the fuel assemblies that are designed to reduce flow-induced vibratory motion.

Sleeve Inserts

The applicant has decided to modify 129 of the 177 fuel assemblies in the first core by the addition of stainless-steel sleeves. No assemblies with unsleeved guide tubes, with the exception of the assemblies containing the test inserts described above, will be positioned under CEAs. The sleeves will serve as an interim method of mitigating the effects of CEA guide tube wear, but will not eliminate CEA vibratory motion.

The sleeves are made of 304 stainless steel, which has been slightly cold worked to provide a yield strength of over 50,000 pounds per square inch. They are chrome plated on the inner diameter and on the upper portion of the outer diameter. The upper end of the sleeves are tapered to match the taper of the upper end fitting post. The sleeve lengths, after insertion in the guide tube, will extend a few inches below the top of the active core. Sleeve attachment is accomplished by mechanically expanding (bulging) both the sleeve and the guide tube in the lower portion of the sleeve. This expansion extends axially for approximately one inch and results in guide tube diametral strains of a few percent.

In addition to this guide tube expansion, the lower axial portion of the sleeves is expanded diametrally outward toward the guide tubes so that the annular gap between the guide tube and the sleeve is eliminated at room temperature. At operating temperature contact stresses develop from differential thermal expansion between the Zircaloy and the stainless steel. The gap in the upper portion of the assembly permits axial and radial differential thermal expansion of the sleeve without imposing significant loads on the assembly.

The hydraulic expansion technique used in attaching the sleeves is essentially the same as the procedure formerly used by CE in spent fuel pool operations that involved the sleeving of fresh as well as worn and irradiated fuel assemblies.

A description is given in the report of full-scale assembly wear tests that have been conducted in the TF-2 loop facility. The results have shown that no appreciable wearing of the chrome-plated sleeves or of the CEA bullet tips have occurred. Metallographic examinations of these sleeved guide tubes are longitudinal "clamshell" sectioning have revealed no evidence of accelerated corrosion in the annular crevice between the sleeve and guide tube. Also a description is provided of the assumptions, calculational methods, and input used in the assessment of the propensity for in-reactor boiling in this annular crevice. These results have been submitted as verification to CE's claim that no boiling will occur in the crevice during reactor operation, and hence the potential for accelerated corrosion is diminished.

Scram time predictions for the 90 percent CEA insertion limit show that sleeved assemblies should have attendant scram time increases of about 0.1 second over a comparable unsleeved assemblies. However, the design basis scram time of 3.0 seconds will still be attainable.

Scupper Extensions

Flow channel scuppers are located at the lower end of the CEA shrouds. They provide a flow path for fluid exiting the top of the fuel assembly to the outlet nozzle plenum. CE has recently conducted flow visualization tests of the ANO-2 reactor design and has observed coolant turbulence in the vicinity of the scuppers. It is postulated that this turbulence may function as a source of CEA vibration in addition to the source of vibration associated with the older 14x14 core designs. Hence, the report describes the addition of scupper extensions to the original ANO-2 reactor design.

The scupper extensions will extend into the fuel alignment plate and thereby serve as a shield for the CEAs thus reducing the potential for CEA vibration due to coolant turbulence. These extensions are to be utilized in all 81 CEA flow shrouds, and the extensions will be attached by seal welding to the scupper and by groove welding to the fuel alignment plate.

Programmed CEA Insertion

Because of the uncertainties associated with predicting the vibration and the resultant degree of wear in the sleeves, a program will be instituted during the first cycle of operation whereby the CEAs will be repositionable. Specifically, the full-out insertion limits for the CEAs will be extended three inches farther into the reactor core. Thus, the magnitude of wear at any one location may be minimized.

Modifications to the Test Assemblies

For demonstration purposes, four rodded Batch A assemblies, which will be discharged at the end of Cycle 1, will contain two types of design modifications to the upper end fitting posts.

One modification of a proprietary nature will be employed in the uppermost portion of all five posts on each test assembly and will result in a slotted upper end fitting post design. Combustion Engineering believes that these devices may alleviate the need for sleeving in future core reloads.

The second modification will consist of an additional sleeve to be inserted into the lowermost portion of the center post on each test assembly. These sleeves will extend a short-axial distance into the center guide tubes and will serve to block the two flow holes in the upper portion of the guide tubes. Thus the thermal-hydraulic aspects of the center guide tube/post structures will be made more similar to those of the corner guide tube/post structures.

Summary of Regulatory Evaluation

We have reviewed the subject report including the engineering analysis, experimental verifications, fundamental assumptions, and references.

Regarding the use of the chrome-plated, stainless-steel sleeves, previous staff reviews of sleeving modifications to guide tubes in operating CE reactors have found this procedure acceptable. Because these previous applications used sleeves on irradiated and worn guide tubes, the current review was less extensive in scope inasmuch as the ANO-2 guide tubes have suffered no prior degradation. Therefore, the ANO-2 safety analysis is in general less restricted in that it contains greater safety margins than that of previously approved safety analyses.

During the review, the staff questioned the fabrication technique used in forming the venting holes and slots in the chrome-plated stainless-steel sleeves. Specifically our concern was whether this punching operation had resulted in the formation of small burrs on the inner diameter of the sleeve, and if such burrs existed would they affect either sleeve/CEA finger wear rates or the flow of coolant through the venting holes and slots.

Upon reinspection of the sleeves, AP&LCo found such burrs and subsequently proceeded to remove the burrs by reaming all sleeves with a constant-diameter reamer.

With respect to the extension of the CEA shroud flow channel scuppers, this design alteration leads to a configuration similar to that of the older 14x14 CE reactor designs. We believe this alteration will result in improved thermal-hydraulic performance of the ANO-2 reactor.

We find that the program to vary the CEA insertion depth at the full-out position will result in a conservative manner of distributing any wear axially along the sleeve inner diameter. Such a plan of apportioning the wear will result in a greater retention of guide tube structural soundness. This benefit offsets the increased axial peaking of four percent which will occur when all the CEAs are inserted up to the new three inch insertion limit.

We have concluded that the four-assembly test program, which is designed to evaluate the performance of the slotted upper end fitting posts, is a prudent method of isolating and evaluating a potential source of the CEA vibration. Complete assurance cannot be given that substantial guide tube wear will not occur in these assemblies. However, if sizeable wearing of the guide tube wall does occur, it will be axially distributed over three inches, and therefore gross degradation that might result in wall penetration of the guide tube will be less likely than if the CEAs were positioned in one location for the entire cycle. Monthly rod exercises will serve to indicate gross wearing that might lead to a stuck rod condition. The potential benefits from testing a possible solution to the problem outweigh the small risk of incurring wear in these four assemblies of the type experienced in earlier plants.

Regulatory Position

The report describes methods which are based upon acceptable engineering principles and practices. The report has demonstrated that CEA guide tube wear can be alleviated and accommodated in a conservative manner for the first cycle of ANO-2 operation. The applicant has agreed to provide a CEA guide tube surveillance program for staff review at least 90 days prior to ANO-2 shutdown for Cycle 1 reload outage. The constitutents of such a program are contingent upon relevant experience gained from other operating CE reactors in the interim; however, the staff has identified observations that may need to be included in the ANO-2 surveillance program. They are observations of:

(1) wear magnitudes in the stainless-steel sleeves and CEA bullet tips,

(2) integrity of the scupper extensions,

(3) looseness of the stainless-steel sleeves,

(4) chipping, pealing, or cracking of the chrome plating,

(5) cracking in the expanded (bulged) portions of the guide tube, and

(6) magnitude of corrosion in the annulus between sleeve and guide tube.

We find the modifications described in the topical report CEN-96(A)-P Revision 1 acceptable for the first cycle of operation of ANO-2. Prior to startup following the first regularly scheduled refueling outage we will review the constituents of the ANO-2 guide tube surveillance program and the results of the application of the program.

CEA Rod Worth Surveillance

Subsequent to the preparation of our Safety Evaluation Report the applicant responded to our request for routine surveillance plans to ensure that reactivity in B_4C-Filled control rods is not lost during the control rod lifetime. The applicant responded with a proposed rod symmetry test program to be conducted at refueling outages. The control element assembly symmetry test was presented in the Final Safety Analysis Report for other purposes and is abstracted on pages 14.1-48, Amendment No. 44. In light of the new information, we have reviewed this program and its capability to monitor changes in the rod nuclear characteristics such as the loss of the boron content.

The symmetry test will involve measurement of the reactivity effects due to interchanging control rods within a control group and between other control groups. Measurements will be taken while the reactor is critical at approximately 545 degrees Fahrenheit and 2250 pounds per square inch absolute. The accuracy of the reactivity measuring method is ± 0.2 cents, and the acceptance criterion will be that relative worths agree to within ± 1.5 cents. We find this method is an acceptable surveillance plan for detecting altered control rod nuclear characteristics.

5.0 REACTOR COOLANT SYSTEM

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.1 Fracture Toughness - Appendix G

In our Safety Evaluation Report we stated in part:

"The method of compliance with 10 CFR Part 50, Appendix G is similar to others we have approved for reactor vessels ordered prior to the publication of Appendix G. We find the method acceptable and conclude that the applicant meets the requirements of 10 CFR Part 50, Appendix G, to the maximum extent practical and, thus, provides reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating fracture can be established for the pressure retaining components of the reactor coolant boundary."

In addition to the above we have determined that an exemption from certain requirements of Appendix G to 10 CFR Part 50 is required and justified and, therefore, will be granted. Our safety evaluation supporting the granting of this exemption will accompany the granting document.

5.2.2 Reactor Vessel Materials Surveillance Program - Appendix H

In our Safety Evaluation Report we stated in part:

"We conclude that the applicant's material surveillance program meets the requirements of Appendix H to 10 CFR Part 50 to the maximum extent practical for a vessel ordered prior to the publication of Appendix H and is, therefore, acceptable."

In addition to the above we have determined that an exemption from certain requirements of Appendix H to 10 CFR Part 50 is required and justified and, therefore, will be granted. Our safety evaluation supporting the granting of this exemption will accompany the granting document.

5.4 Inservice Inspection Program

5.4.1 Inservice Testing of Pumps and Valves Up to Commercial Operation

The applicant has provided a description of his proposed program for inservice testing of ASME Code Class 1, 2 and 3 pumps and valves. The program includes both baseline preservice testing and periodic inservice testing. It provides for both functional testing of components in the operating state and for visual inspection for leaks and other signs of degradation. In accordance with the requirements of

Section 50.55a(g) of 10 CFR Part 50, the applicant proposes the period for which the program is applicable as follows: (1) From the issuance of the Facility Operating License to the start of facility commercial operation, inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be performed in accordance with Section XI, 1974 Edition through Summer 1975 addenda except where specific written relief has been granted in accordance with written Technical Specification 4.0.5.a.1; and (2) Following the start of facility commercial operation, inservice testing of pumps and valves will be performed in accordance with the ASME Section XI Code and applicable addenda as in accordance with the requirements of 10 CFR Part 50, Section 50.55a(g).

The date of the applicants construction permit (Dec. 6, 1972) places this plant under 10 CFR 50.55a(g)(2) which requires compliance with the 1971 edition thru the summer 1971 addenda of Section XI of the ASME Boiler and Pressure Vessel Code. Since inservice testing requirements for pumps and valves were not included in the Code until the Summer 1973 addenda of the 1971 edition, the applicant has chosen to optionally meet the requirements of 1974 Edition, through the Summer 1975 addenda to the extent practical and has requested relief from certain Code requirements. Based upon our review of the program we conclude that it is acceptable for the preservice testing phase of plant operation extending up to the start of commercial operation. Acceptability of the program for the period following commercial operation will be determined prior to the start of facility commercial operation.

5.6 Components and Subsystem Design

5.6.2 Steam Generator Tube Integrity

Steam generator tube denting has been experienced in a number of operating nuclear plants which utilize drilled support plate designs similar to the partial tube support plates in the Arkansas Nuclear One - Unit 2 steam generators. In one of these operating plants, tube denting resulted in the cracking of the solid ligaments between the drilled holes of the support plates. In view of these recent experiences, the steam generators in Arkansas Nuclear One - Unit 2 have been modified. The applicant submitted a report with a letter dated June 9, 1978 which describes the modifications and the analytical considerations of those modifications.

Denting results from a chemical reaction, involving contaminants introduced into the steam generators, which occurs within annuli formed by the heated tubes and the tube holes in the drilled support plates. This reaction forms a non-protective form of magnetite on the surface of the support plate holes. Volumetric expansion of the support plate material, when converted to nonprotective magnetite, causes a squeezing action on the tubes which eventually can deform, or dent, the tube.

If denting occured in an unrestrained support plate, the reaction in the plate to external forces on the tube could cause uniform in-plane expansion of the support plates. This expansion could, if denting were allowed to progress sufficiently, lead to the eventual cracking of the ligaments between the tube holes and flow holes in the support plates. The support plates were, in fact, restrained by attachments to the tube bundle shroud by welded lugs and by a solid rim of metal at the outer periphery of the plates, so the plate expansion cannot occur uniformly. Because of the restraint, denting and plate expansion could lead to stress concentrations between the restraint rim portion and the perforated portion of the support plate containing tube holes and flow holes. Consequently, cracking could occur along the stress concentration lines at smaller tube denting magnitudes than those which would be required in order to cause cracking in an unrestrained support plate.

On May 12, 1978, the applicant and Combustion Engineering met with the staff to present mechanical modifications to the ANO-2 steam generators to mitigate the consequences of the denting phenomenon, should it occur. It was proposed that the peripheral restraints of the support plates and portion of the tube plate along its outer edge be removed. This would allow the tube plate to expand radially without being forced against the shroud. Unrestricted radial expansion could, in turn, prevent cracks in the plate ligaments along the periphery of the plate.

The ANO-2 modifications are similar to those recently implemented on the Millstone, Unit 2 steam generators which have already been reviewed and approved by the staff. An additional motivation for performing the modifications at this time was the fact that the plant had not yet gone into operation. This allowed work to be performed in the steam generators without personnel radiation exposure, thus conforming to the "as low as is reasonably achievable" guidance of Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Will Be As Low As Is Reasonably Achievable." More experienced personnel and better working conditions were utilized on these modifications that would have been possible with contaminated steam generators.

We conclude that ~~these modifications~~ will (1) not diminish the functional performance or structural integrity of ANO-2 steam generators, (2) provide for mitigation of the consequences of tube denting should the phenomenon occur, (3) reduce risk of exposure to radiation by incorporating the modifications on clean steam generators. These proposed modifications and conformance with the applicable codes and standards, staff positions and Regulatory Guides consitutes an acceptable basis for meeting the requirements of General Design Criteria 4, 15 and 31.

6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

In the Safety Eavluation Report, we reported that the applicant had analyzed a spectrum of main steam line break accidents to determine the containment pressure and temperature response. We required that the analysis be performed assuming the worst case single failure of the feedwater system. In a letter dated January 31, 1978, the applicant submitted an analysis assuming various single active failures with and without the availability of off-site power; condensate pump failing to trip; main feedwater regulating valve failing to close; and main feedwater isolation valve failing to close.

Mass and energy releases for a spectrum of steam line breaks were calculated using the SGN-III code that is described in Appendix 6B of CESSAR. The SGN-III code describes the primary and secondary systems of a pressurized water reactor including the core and power excursion which may occur in the core following a main steam line break. The code calculates heat flow from the intact steam generator into the primary system and heat flow from the primary system into the broken steam generator. The primary system heat flow produces additional steam which is added to the containment.

The SGN-III code was found to be acceptable for mass and energy release calculations in the NRC Safety Evaluation Report for CESSAR dated December 1975.

Since ANO-2 is not a CESSAR type plant, the SGN-III analyses were modified with input information specifically for the ANO-2 reactor. The feedwater flow into the affected steam generator is one of the inputs to the SGN-III code. As the feedwater lines depressurize, flashing occurs which drives the fluid into the affected steam generator. The remaining liquid is boiled by heat flow from the primary system. As decompression of the feedwater line continues, essentially all of the unisolated feedwater will flow into the affected steam generator.

The applicant analyzed the feedwater flow assuming single failures in the feedwater isolation and control valves using the RELAP-4 code.

Feedwater flow rates to the affected steam generator were calculated using conservative assumptions for feedwater and condensate pump operation, heat transfer from the feedwater heaters and intact steam generator pressure. No liquid separation was assumed to occur in the feedwater piping.

The most severe single failure was found to be failure of the main feedwater isolation valve to close. Flashing of the unisolated fluid downstream of the feedwater control valve caused 100,510 pounds of feedwater to enter the affected steam generator.

We have concluded that the applicant's calculations for main steam line break mass and energy release are conservative and therefore acceptable.

The applicant's analysis shows that the worst case main steamline break (2770 Megawatts thermal, 85 percent break area) assumes the availability of off-site power concurrent with the failure of the main feedwater isolation valve to close. The calculated peak containment pressure of 57.2 pounds per square inch gauge exceeds the containment design pressure by 3.2 pounds per square inch. Using the applicant's input data, we have performed a confirmatory analysis using the CONTEMPT-LT/026 computer code. Our analysis, which verifies the peak pressure calculated by the applicant, predicts that the containment design pressure will be exceeded for approximately 65 seconds.

In order to reduce the calculated peak containment pressure, the applicant has committed to install a redundant main feedwater isolation valve in each secondary system piping loop. Due to the long lead time necessary to obtain these additional valves, they will not be installed before the first refueling outage. Therefore a condition to the operating license will require that these valves be installed prior to startup following the first regularly scheduled refueling outage. Although these valves will be located in non-seismic piping, they will have seismically qualified, Class 1E operators and they will receive a safety grade signal to close. The redundant feedwater isolation valve which will be located upstream of the present feedwater isolation valve will be capable of terminating the feedwater flow before the containment peak pressure is reached. Since the new valve will be a faster closing valve, the worst single failure will be the failure of the new valve to close. The applicant's analysis shows that this case results in a peak containment pressure of 52.8 pounds per square inch gauge, which is below the design pressure of 54.0 pounds per square inch gauge. Again, our confirmatory analysis is in good agreement with the applicant's analysis.

The maximum calculated containment pressure for the worst case postulated main steamline break accident is, as stated previously, 57.2 pounds per square inch gauge. The containment design pressure, as set forth in the Final Safety Analysis Report, is 54.0 pounds per square inch gauge. The bases for our conclusion that, with the applicant's commitment to install an additional main feedwater isolation valve during the first regularly scheduled refueling outage, it is acceptable to allow the operation of the ANO-2 plant at full power during the first fuel cycle are set forth below.

The applicant states that the containment structural design is identical to the ANO-1 unit, which was designed for 59 pounds per square inch gauge internal pressure. The containment appurtenances (penetration assemblies, etc.) are the limiting items with a qualification pressure of 55.0 pounds per square inch gauge. The containment was successfully pressure tested to a pressure of 62 pounds per square inch gauge and no abnormalities were noted in the records.

The applicant stated that the computer codes used to calculate the containment pressure (57.2 pounds per square inch gauge) provide a conservative pressure calculation and by confirmatory analysis we have verified the acceptability of the applicant's containment response analysis. We have determined that the installation of the additional isolation valve in each of the main feedwater lines will ensure that the originally specified containment design pressure will not be exceeded.

Based on the actual containment structure design pressure of 59 pounds per square inch gauge, the successful containment pressure test of 62 pounds per square inch gauge and the commitment by the applicant to install an additional main feedwater isolation valve in each of the main feedwater lines at the first refueling, the staff finds the containment acceptable.

Environmental Qualification for Main Steam Line Break Analysis

We concluded our discussion of this matter in Section 6.2.1 of our Safety Evaluation Report by noting that we had requested additional information from the applicant and would report our conclusions in a supplement to the report. Since preparation of the Safety Evaluation Report, we have received the additional information and have concluded our review of this matter as stated below.

Analyses of postulated main steam line break accidents inside containment performed by the staff and several applicants have predicted higher calculated containment temperatures, on the order of 400 degrees Fahrenheit, than were used in the environmental qualification testing of certain safety-related equipment. As a result, there is a generic concern regarding the capability of certain safety-related equipment to remain operable in the accident environment which could result from a main steam line break inside containment. However, it has been recognized by the staff that the methods of analyses approved today contain significant conservatisms. Specifically, the staff has required analyses based on an instantaneous double-ended steam line rupture with the assumptions of dry steam blowdown and using conservative assumptions for minimizing containment heat transfer coefficients with a conservative treatment of the thermodynamics of condensate behavior.

The use of component thermal analyses and associated heat transfer coefficients, appropriate containment analytical modeling, and acceptable methods of accident environmental simulation for equipment qualification are under generic review by the staff. We expect our review of these items to be complete within approximately one year. This program will result in the development of a consistent set of environmental qualification requirements which can then be implemented for all plants.

In the meantime, we have developed a best estimate model for the containment response to a main steamline break accident and for the evaluation of component environmental qualification. Using CVTR data, relationships for heat sink condensed mass removal and component heat transfer have been developed. These correlations result in slightly lower containment atmosphere temperature, about 25 degrees Fahrenheit, than previously calculated and a significant increase in the component heat transfer rate. In addition, we have estimated a 25 degree Fahrenheit or more temperature difference (reduction) from that calculated in the containment region where the break is located for those component locations separated by walls, floors, and large distances (i.e., diametrically opposed locations). Based on our best estimate studies, we have developed an Interim Evaluation Model which provides an acceptable method for evaluating main steam line break accidents and the associated environmental qualification requirements for safety equipment.

The applicant has performed a containment response analysis for a spectrum of main steam line break accidents and has calculated a peak vapor temperature of 420 degrees Farenheit. The applicant has used a more conservative assumption regarding heat sink condensate mass removal than that suggested by our interim model. Component thermal analyses were performed using methods which we have found to be acceptable. The results show that all portions of the majority of components needed during a main steam line break accident remain below the qualification temperature limits. However, three components (a sump level detector, a solenoid valve switch, and an electrical cable) had calculated surface temperatures in excess of their qualification temperatures. In the case of the sump level detector and the solenoid valve switch the applicant has provided analyses which show that the internals of these components remain below the qualification temperatures. A thermal response of the electrical cable was performed in which no credit was taken for protection provided by cable grouping in trays or routing in conduits, thus assuring a conservative evaluation. The temperature of the cable insulation was calculated to exceed the qualification temperature by 13 degrees Fahrenheit at the surface and two degrees Fahrenheit at the conductor interface for a duration of approximately 20 seconds. We have reviewed the applicant's analyses and find them acceptable.

Based on the staff's review of the conservatisms provided in the analysis, the short duration of the transient and the qualification testing provided for this equipment, we conclude that the qualification envelope of the equipment for a loss-of-coolant or a main steam line break accident is acceptable.

The applicant has not provided information on the thermal response of the containment cooling system bypass damper motors. Upon receipt of the appropriate signal, these motors will open a damper to allow greater flow to the containment fan coolers during a loss-of-coolant/main steam line break accident and will complete their intended function within approximately 20-25 seconds after the beginning of the accident. Unqualified motors, which were initially installed, have been replaced with motors qualified to 300 degrees Fahrenheit. We will require that the applicant provide an analysis which shows that the component thermal response does not exceed the qualification temperature during a main steam line break accident similar to that performed for the balance of the essential safety components.

Further information on the environmental testing and our assessment of the acceptability of the applicant's equipment qualification is provided in Section 3.11 of this report.

6.2.4 Containment Isolation System

The purge supply and exhaust system consists of 54-inch diameter inlet and outlet lines. In the Safety Evaluation Report, we stated that purge system operation would be limited to one percent of the time per year (about 85 hours), unless the recommendations of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operation" were met.

Our acceptance of the limited use of the containment purge system during normal plant operation is contingent upon the finding that the purge system isolation valves will be capable of closing during a loss-of-coolant accident. At the present time there remain three problems which must be resolved to our satisfaction before we will allow the use of the purge system in even this limited operational mode.

The first problem is that the applicant indicates that a purge valve operability program has not been established as required by Branch Technical Position 6-4.

The second problem is that as stated in a recent letter to the Office of Inspection and Enforcement the applicant has learned that the valves were not adequately qualified for the pressure differential expected as a result of the design basis containment accidents.

The third problem is that for small loss-of-coolant accidents which may be of low mass and energy release rates relative to the containment pressure isolation signal setpoint there must be assurance that the radioactivity released into containment will be sensed and a signal generated and transmitted to close the containment purge system isolation valves. Therefore we have transmitted our position to the applicant that redundant trains of safety grade radiation monitoring systems capable of automatically effecting the isolation of the containment purge lines must be installed.

The applicant's proposal to resolve the second problem discussed above as stated in Final Safety Analysis Report Amendment No. 46 is to limit operation of the containment purge system only to plant Operational Modes 6 (REFUELING) and 5 (COLD SHUTDOWN). Since this measure would ensure that in the event of a loss-of-coolant accident or a main steamline break the purge isolation valves would be fully closed and no radioactivity would be released to the environs we conclude that this is an acceptable measure to resolve the three subject problems. The facility Technical Specifications will restrict operation of the containment purge system to only Operational Modes 6 and 5.

The applicant states that he is continuing to work with the valve vendor to demonstrate the operability of the valves under postulated loss-of-coolant accident conditions. If the applicant proposes at a later time in the plant's life to operate the containment purge system in Operational Modes 4, 3, 2 or 1 he must submit for the staff's review and approval an application for amendment of the Technical Specifications which acceptably resolves the three subject concerns.

6.2.6 Containment Leakage Testing Program

We have reviewed the applicant's containment leak testing program for compliance with the containment leakage testing requirements specified in Appendix J to 10 CFR Part 50, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors. Such compliance provides adequate assurance that the containment leak-tight integrity can be verified throughout service lifetime and that the leakage rates will be periodically checked during service on a timely basis to maintain such leakage within the specified limits. Maintaining containment leakage within such limits provides reasonable assurance that, in the event of any radioactivity release within the containment, the loss of the containment atmosphere through leak paths will not be in excess of the limits specified for the site.

Specifically, we have reviewed the leak testing program to assure that the containment penetrations and system isolation valve arrangements are designed to satisfy the containment integrated leak rate testing requirements and the local leak testing requirements of Appendix J.

Based on our review, we have concluded that the proposed reactor containment leakage testing program complies with the requirements of Appendix J to 10 CFR Part 50 with one exception and is acceptable. Additional efforts on this subject that will lead to a revision of Appendix J are being done in conjunction with generic item A-23, Containment Leak Testing. The outcome of this task will be applicable to all plants depending on their licensing status and design.

We have determined that an exemption from certain requirements of Appendix J to 10 CFR Part 50 is required and justified and, therefore, will be granted. Our safety evaluation supporting the granting of this exemption will accompany the granting document.

6.3 Emergency Core Cooling System

6.3.3 Performance Evaluation

In our Safety Evaluation Report we stated that we were continuing our review of the emergency core cooling system performance analysis and that our evaluation would be reported in a supplement and would include the application of the single failure criterion to a range of pipe breaks, the effects of boron precipitation on long term cooling capability and submerged values within containment.

Large Breaks

The applicant submitted analyses to determine the consequences of a postulated large break loss-of-coolant accident (LOCA) and to assure emergency core cooling system (ECCS) adequacy. These analyses consider a spectrum of break sizes, types and locations. Applicable Combustion Engineering, Inc. studies were referenced which have shown that a pump discharge break is the limiting location for large breaks. The applicant has submitted analyses of a spectrum of seven large breaks which conform to Appendix K of 10 CFR Part 50. The worst break identified by the analyses is a double ended pump discharge guillotine rupture with a discharge coefficient of 1.0 at a peak linear heat generation rate of 14.5 kilowatts perfoot. The peak cladding temperature calculated for the worsat break is 2078 degrees Fahrenheit which is below the 2200 degrees Fahrenheit limit specified by 10 CFR Part 50, Section 50.46. Maximum cladding local oxidation and core wide hydrogen generation calculated for the worst case, 11.82 percent and less than 0.617 percent respectively, are less than the 10 CFR Part 50, Section 50.46 - specified limits of 17 percent and one percent, respectively. The cladding temperature transient is terminated in the analysis at a time when the core is still amenable to cooling. Provisions have also been made to maintain long term cooling for an extended period of time.

Small Breaks From 0.1 - 0.5 Square Feet

The applicant has submitted small break analyses which determine the consequences of a postulated small break LOCA and assess ECCS adequacy. These analyses considered a spectrum of three cold leg discharge break sizes. The worst small break size was calculated to be 0.1 square feet break with a peak cladding temperature of 1460 degrees Fahrenheit, a maximum local cladding oxidation of 0.205 percent and a core wide cladding oxidation of less than 0.027 percent at a peak linear heat generation rate of 16.0 kilowatts per foot. The peak cladding temperature is below 2200 degrees Fahrenheit, the local cladding oxidation is below 17 percent, and the core wide cladding oxidation is below one percent as required by 10 CFR Part 50, Section 50.46.

Small Breaks Less Than 0.1 Square Feet

The applicant has provided information on small break (<0.1 square feet) loss-of-coolant accidents showing adequate core cooling without active depressurization of the primary system. For breaks less than 0.01 square feet, the flow from the high pressure safety injection pumps is adequate to sustain a satisfactory vessel inventory to preclude core uncovery. For breaks between 0.1 square feet and 0.01 square feet, the applicant has referenced topical report CENPD-137P which demonstrates that very small break loss-of-coolant accidents are bounded by large break loss-of-coolant accidents and which shows a trend of decreasing peak clad temperature for break sizes below 0.1 square feet. At break sizes 0.1, 0.3, and 0.5 square feet, ANO-2 peak clad temperature and fuel clad oxidation results were conservatively bounded by those reported in CENPD-137P. A shorter period of core uncovery for ANO-2 is due to a smaller power to safety injection system flow ratio at ANO-2 than that used in CENPD-137P.

As a result of our review of the emergency core cooling system analysis we have concluded that following modifications must be made to the operator emergency procedures for loss-of-coolant accidents. The inspection of the procedures to ensure that these items are included will be performed by the Office of Inspection and Enforcement.

(1) For small break loss-of-coolant accidents, provide a caution that repressurization of the vessel once the reactor coolant system has cooled down should be avoided by maintaining a gas bubble in the pressurizer.

(2) When reducing total high pressure safety injection flow, the operator must close one of the four high pressure safety injection head isolation/throttle valves at a time to reduce delivery to the reactor coolant system. Before closing the last valve in a train, the high pressure safety injection pump in that train must be stopped. For very small breaks, operation of the valves will not change the flow rates significantly. Under these conditions the high pressure safety injection pump(s) must be secured to reduce flow.

(3) Following a recirculation actuation signal (RAS), the operator should verify that the refueling water tank isolation valves shut.

(4) After aligning for hot leg injection following a loss-of-coolant accident, the operator should verify that total high pressure safety injection flow is greater than 500 gallons per minute.

(5) After aligning for hot leg injection following a loss-of-coolant accident, the operator should verify that hot leg and cold leg high pressure safety injection flows are both greater than 250 gallons per minute.

CONCLUSIONS - Large and Small Breaks

We conclude that the emergency core cooling system for ANO-2 is capable of satisfactorily mitigating small break loss-of-coolant accidents while maintaining acceptable values of peak clad temperatures and fuel clad oxidation. The applicant has demonstrated that a large double-ended cold leg break is the most limiting loss-of-coolant accident. We conclude that the submitted large and small break analyses are acceptable and in conformance with 10 CFR Part 50, Section 50.46.

Long-Term Cooling - Boron Precipitation

The applicant has provided proposed procedures for preventing boron precipitation following a loss-of-coolant accident and a description of their boron concentration calculations. Following a loss-of-coolant accident, the high pressure safety injection and low pressure safety injection pumps immediately draw suction from the refueling water tank (RWT). Once the level in the refueling water tank has dropped to approximately six percent of maximum volume, a recirculation actuation signal (RAS) is generated. This signal automatically opens the containment sump isolation valves to safety injection pump suction, shuts off the low pressure safety injection pumps, closes the miniflow isolation valves to the refueling water tank, and closes the refueling water tank suction isolation valves. The realignment to the recirculation mode provides a long-term source of water for the high pressure safety injection pumps.

In order to preclude boron precipitation in the core due to coolant boiloff, the applicant has provided a path for hot leg injection. The hot leg injection isolation valves would be opened and the high pressure safety injection flow to the cold legs is realigned to force cold leg flow through flow-balancing orifices such that equal amounts of water will be injected into the core through the cold legs and the hot legs. This procedure ensures that the sum of hot and cold leg flow matches core boiloff plus more than the minimum required flow rate of 20 gallons per minute flushing the core.

Conservative calculations provided by the applicant show that boron precipitation does not begin prior to eight hours following a loss-of-coolant accident allowing for a four weight percent margin in the boron concentrations. Since the applicant proposes to begin hot leg injection before six hours and since a core flushing is path provided using both the hot and cold legs, the staff finds the equipment and procedures to be adequate to prevent boron precipitation during long-term cooling.

Submerged Valves

The applicant has conducted a review of equipment arrangement to determine if any of the components inside the containment will become submerged following a loss-of-coolant accident and has identified seven valves which may become submerged. We have reviewed the consequences of improper alignment due to submergence of four of these valves and have concluded that these valves would have no safety impact on a loss of coolant accident. The applicant has submitted information which provides adequate assurance that the other three valves (2CV-5647-1, 2CV-5648-2 and 2CV-2060-1) have been acceptably qualified for their expected environmental conditions during a loss-of-coolant accident.

6.3.4 Tests and Inspections

Containment Sump Recirculation Preoperational Tests

Out-of-plant tests were performed on a full-scale model of one of the two plant sumps. Measured pressure losses associated with the trash racks, screens, grids, and pipe entrance were low. There was observed to be little inherent tendency for vortex formation.

Vortices were forced to occur by selective guidance of the approach flow and through blockage of the trash racks. Even with a standing air core vortex, the increase in pressure losses was small as compared to the margin in available net positive suction heat (NPSH). With all vortex suppression devices in place, no vortices involving air entrainment or detectable rotational patterns could be forced to occur.

Only one of the two sumps and none of the containment features (contours, obstructions, and flow sources) were modeled. Due to the lack of an exact duplication of the plant, the tests were not considered to be adequate by themselves to satisfy the preoperational test requirements. However, we agree with the applicant that the testing performed did demonstrate low pressure losses and a lack of tendencies for vortex formation.

The staff, with assistance from consultants at the Institute of Hydraulic Research, University of Iowa, has initiated a program directed toward identifying conservative design guidelines for sump design. While the program is still in its early stages, the approximate range of certain parameters which result in conservative sump design features is known.

Considering some of the Arkansas Unit 2 parameters, the staff notes the low outlet pipe flow velocity of 2.7 feet per second, significant submergence of the outlet pipes below the containment water surface (10-foot minimum), the low average downward velocity in the sump pit (0.19 foot per second), and the low inlet screen velocity (0.1 foot per second). The ratio of submergency to outlet pipe velocity is an important parameter which can be used to identify tendencies for vertex formation and air entrainment. The greater the value of this ratio, the more conservative is the sump design. We conclude that for ANO-2 this ratio (10/2.7) is conservative. Also the screen velocity and the sump downward velocity are low and would, therefore, provide little or no tendency to increase the severity of a vortex if a rotational pattern were to form. The staff, therefore, concludes that based on the parameter study, the sump design is conservative and would not be expected to have vortex, air entrainment, or pressure loss problems.

Based on our observations of a portion of the ANO-2 testing program and our review of the test program we conclude that the tests provided verification of the conclusions stated above. The vortex suppression devices (grids and grid cages) were observed to suppress forced vortices. We have also observed their effectiveness in removing rotational patterns in other plant sump development programs.

The discussion above relates to one of the two ANO-2 sumps for which a full-scale model was built and tested. The untested sump, while slightly smaller and with a slightly different outlet pipe orientation, will operate within essentially the same range of parameters as discussed above. Interactions between the two sump areas would not be expected because of the solid divider plate which divides the sump pit into the two sump areas and the low velocities of the flow approaching the sumps. The applicant has compiled information which demonstrates that the minimum margin (available net positive suction head minus required net positive suction head) for any of the pumps when drawing collectively from the sump will be 1.7 feet. This was based predominantly on experimental information from pump tests before installation, other preoperational tests, and measured losses from the sump test. Based on the above, the staff concludes that there is reasonable assurance that the emergency core cooling system will function in the recirculation mode without excessive pressure losses or limiting vortex formations and, therefore, the design of the ANO-2 containment sumps is acceptable.

6.3.5 Conclusions

We reviewed the emergency core cooling system performance analysis submitted for ANO-2 and conclude that the analysis was performed wholly in accordance with the requirements of Section 50.46 of 10 CFR Part 50. The ANO-2 emergency core cooling system performance assures conformance with (1) the peak cladding temperature limit of 2200 degrees Fahrenheit, (2) the maximum local cladding oxidation limit of 17 percent of total cladding thickness before oxidation, (3) the maximum hydrogen generation core-wide limit corresponding to oxidation of one percent of the total metal in the cladding surrounding the fuel, (4) the core geometry remaining amendable to cooling, and (5) the long-term cooling requirement of maintaining acceptable core temperatures and decay heat removal.

We conclude that the emergency core cooling system for ANO-2 meets all of the criteria of Section 50.46 of 10 CFR Part 50 and the requirements of Appendix K to 10 CFR Part 50 and is, therefore, acceptable.

7.0 INSTRUMENTATION AND CONTROLS

7.1 General

In Supplement No. 1 to the Safety Evaluation Report, we identified several items that as result of our site visit in July 1977 were still unresolved and required additional information to be provided by the applicant.

The applicant's response was included in letters dated January 16, 1978, March 3, 1978, and March 30, 1978, and in Amendment 44 to the Final Safety Analysis Report. We have reviewed the applicant's responses and the modifications described and conclude that design satisfies the staff's requirements identified in Supplement No. 1 and the Safety Evaluation Report and is therefore, acceptable.

7.2 Reactor Trip System

7.2.2 Reactor Trip System - Hardwired Analog Portion

Independence of Redundant Power Supplies

In Supplement No. 1 to the Safety Evaluation Report, we stated that additional justification, analysis and basis was required in order for the staff to support the applicant's claim that the power supplies used in the safety system logic design are valid isolation devices. In response to our concerns, the applicant, via letters from D. Williams to J. Stolz dated March 23 and April 28, 1978, submitted an analysis and a description of modifications made to support the adequacy of the design.

The modifications include replacement of static inverter power supplies, used to power the logic systems, with solid-state control inverters. The cables for the new inverters are routed in separate wireways which preclude the maximum voltages of 508 volts alternating current (VAC) and 140 volts direct current (VDC) previously identified, to be imposed upon the logic circuits. In addition, surge suppression devices were added on the input and output of the inverters to minimize the surge voltages to within acceptable levels determined by analysis that the design may tolerate. As a result of the modification, the redefined worst case fault voltages and surges that can occur on the ANO-2 design were determined to be 132 VAC and 100 volts, respectively.

Although the analysis indicates that the design is capable of maintaining its functional operability if subjected to fault voltages below 400 VAC and instantaneous peak surges of 328 volts, the test data results presented did not sufficiently demonstrate that the integrity of these logic circuits would be maintained at these values. The applicant was advised that we require a type test be performed which simulates the design installation, and demonstrate that the design will maintain its functional integrity when subjected to 132 VAC faults with 100-volt surges. The applicant has conducted a test which envelopes these design conditions and supports the assumptions made in its analysis, and also has provided, in a letter dated July 17, 1978, the results of these tests for our review.

Based on our review of the results, we conclude that the tests envelope the design requirements with margin and support the assumptions made in the analysis.

We, therefore, conclude that the design provides adequate assurance that the equipment will maintain their functional operability if subjected to the limiting fault conditions that are expected to occur and is, therefore, acceptable.

7.2.3 Reactor Trip System - Digital Computer System

Core Protection Calculator System

Introduction

Our Safety Evaluation Report and its Supplement No. 1 included introductory discussions on the core protection calculator system.

The core protection calculator system (CPCS) is designed to provide reactor protection for two conditions: (1) low local departure from nucleate boiling ratio; and (2) high local power density. The remaining 12 of 14 protective functions of the reactor protection system are accomplished by using a conventional analog hard-wired system. The detailed description and our evaluation and conclusions for the hard-wired portions of the protection system are presented in the Safety Evaluation Report and in Appendix D of this report.

Our Safety Evaluation Report and its Supplement No. 1 contained a detailed description of the core protection calculator system and of the staff's review methodology. This report, Supplement No. 2, contains review evaluation results established by the staff subsequent to the preparation of Supplement No. 1 to the Safety Evaluation Report. Our evaluations are presented with respect to the remaining safety positions, which for convenience are defined in Appendix D, Table D.1, of this report.

Summary

The majority of the core protection calculator system review has been completed and the staff has accepted the design and qualification of the system subject to the satisfactory resolution of the remaining safety positions.

There are four safety positions which remain to be resolved. Of these, three safety positions require the applicant to conduct and analyze start-up tests and submit reports for staff review. For the purposes of issuance of an operating license, the staff has completed the review of the appropriate start-up test procedures and finds them acceptable. Additionally, there is one safety position which must be acceptably resolved prior to Mode 2 (Initial Criticality) operations.

Core Protection Calculator System Review Status Summary

The disposition of the 27 safety positions stated in our previous Safety Evaluation Reports are as follows:

(1) The applicant has responded to and fully implemented to the staff's satisfaction 23 of the 27 safety positions generated by the staff. Seventeen issues, designated as items 2, 3, 6, 7, 8, 9, 10, 11, 13, 16, 17, 21, 22, 23, 24, 25, and 27 were evaluated and resolved in the Safety Evaluation Report and supplement number one thereto and are categorized as closed issues. The issues, designated as items 4, 14, 15, 18, 20 and 26 in Table D.1 of Appendix D to this report, have been resolved to the staff's satisfaction as discussed herein and are now categorized as closed issues.

(2) Four of the safety positions defined in Table D.1 remain outstanding. These consist of positions 1, 5, 12 and 19. With respect to positions 1, 5, and 12, the applicant's responses to date have been reviewed and are acceptable. Start-up test data and analyses are required to evaluate the compliance with the remaining concerns. The applicant has committed to conduct the desired start-up tests and provide a test report to the staff. These positions are, therefore, resolved for the purpose of issuance of an operating license. Conditions to the operating license will require that the tests to be performed during start-up be acceptably completed.

The staff requires that safety position 19 be acceptably resolved prior to Mode 2 operation. The operating license will be conditioned accordingly. The staff has reviewed the applicant's partial responses and commitments regarding this position and find them acceptable. The staff has also defined some new information with respect to noise testing that will be required from the applicant. Additionally, we are also requiring that the applicant retain a nuclear computer type consultant for the plant safety committee to evaluate safety implications for proposed software modifications.

A review and discussion of the positions presented in Table D.1 are presented in the following sections of this report:

Position	Section
1	D.3.5
4	D.4.1.4
5	D.4.1.2
12	D.4.1.4, D.4.4.4
14	D.4.2.5
15	D.3.11
18	D.4.4.5
19	D.4.4.6
20	D.4.2.3
26	D.4.1.4

7.5 Safety-Related Display Instrumentation

7.5.1 Accident and Post-Accident Monitoring

We stated in our Safety Evaluation Report that we were continuing our review of the adequacy of the applicant's list of parameters deemed essential for accident and post-accident monitoring.

The applicant has addressed the instrumentation required for accident and post-accident monitoring in Table 7.5-2 of the Final Safety Analysis Report. We have completed our review of this matter and find the parameters to be monitored under accident and post-accident conditions acceptable for providing adequate information to initiate appropriate actions.

7.6 Other Systems Required for Safety

7.6.3 Safety-Related Fluid Systems

In our Safety Evaluation Report, we stated that the applicant was requested to modify the design for the recirculation valve to the refueling water storage tank (valve 2CV-5628-2) and provide redundant valve position indication in the control room which would meet the single failure criterion and, provide the detailed schematic drawings which implement this requirement. In response, included in a letter dated March 3, 1978, the applicant committed to provide redundant Class IE valve position indication in the control room, and has submitted the schematic drawings implementing the design. In addition, the applicant verified that the equipment will be environmentally and seismically qualified to maintain its functional operability as required for this safety system. Although the design has been established, the applicant indicated that the implementation of the redundant position indication could not be accomplished until after fuel load because of component procurement schedules.

Based on our review of the final schematics and the applicant's commitments, we conclude that the design satisfies the Commission's requirements stated in

Section 7.6.3 of the Safety Evaluation Report and is acceptable. A condition in Amendment No. 1 of the operating license will stipulate that this design modification be implemented within six months of issuance of Amendment No. 1 to the operating license.

7.8 Electrical Penetrations

Subsequent to the preparation of our Safety Evaluation Report and as a result of our site visit, the applicant was requested to submit the fault current tests that were conducted on the electrical penetrations. The applicant was also requested to describe the breaker coordination design which demonstrates that the electrical penetrations will maintain their structural integrity in the event maximum credible faults are imposed on these circuits.

In response, the applicant submitted the test documentation and described in Amendment 45 to the Final Safety Analysis Report, the breaker setpoints for the primary and back-up breakers used on the 6900 volt alternating current circuits and on the 480 volt alternating current load center circuits. The setpoints established verify that sufficient margin is provided to assure that the maximum fault currents on these circuits will be terminated in time to preclude electrical penetration damage.

Also, the applicant modified the design of these circuits to assure that a loss of a single power supply to these circuits would not preclude both redundant breakers from performing their function when required. During the review the applicant was requested and agreed to supplement the information in the Final Safety Analysis Report to describe the adequacy of the breaker coordination of the additional six of the existing eight types of circuits that are routed into the containment. In response, the applicant provided in Amendment No. 46 to the Final Safety Analysis Report the description of the breaker coordination and the degree of protection provided for these remaining six circuit types. We have reviewed this information and find that the design criteria are similar to those provided for the previously described circuits identified above and are, therefore, acceptable on the same basis. The applicant has included provisions in the technical specifications for periodic setpoint verification of these breakers that assure that they will perform their intended function.

Based on our review of the testing conducted on the penetrations, the modifications providing independent power supplies to the primary and backup breakers, the setpoints established for the fault current interrupt devices and the provision of periodic surveillance requirements for these devices, we conclude that the design satisfies the Commission's requirements of General Design Criterion 50 and is acceptable.

7.9 Cable Separation Criteria

7.9.4 Separation Criteria Between Redundant Class 1E Circuits in Metal Conduits

In the Safety Evaluation Report, we identified our concerns regarding areas where redundant channel wiring routed in separate and independent metal conduits were routed in close proximity (1 inch apart or less) to each other without provisions for barriers other than the conduit itself. The applicant was requested to review their installation and where events such as heat or missiles may effect the redundant circuits in these conduits, the applicant was requested to provide barriers to assure the integrity of these circuits, or justify their design on some other defined basis.

The applicant's response included in a letter dated October 26, 1977, and in Amendment 45 of the Final Safety Analysis Report submitted additional information defining in more detail the separation criteria of Class 1E circuits routed in metal conduits. In addition the applicant re-evaluated this installation and identified a limited number of circuits (less than ten) where the separation distances between redundant circuits routed in conduit and between redundant circuit routed in conduit and trays was less than the required minimum separation, with no provisions for additional thermal barriers. For a few of these circuits the adequacy of separation could not be justified and therefore, the applicant committed to install additional thermal barriers to ensure the integrity of these safety circuits and provided drawings which implement this commitment. For the remaining circuits the existing separation was justified on the basis that the circuits were low energy instrumentation cables with separation distance equal to or greater than one inch except in areas of the pull boxes. Investigation of these areas indicated that there was sufficient air space between the cable and their associated metal pull boxes to minimize thermal propagation and they were located in areas where damage from external fires were precluded. Based on our review of the applicants response and our review of these circuits during our site visit we conclude that the design satisfies the Commission's requirements stated in Section 7.1 of our Safety Evaluation Report, and the requirement identified during our site visit and is, therefore, acceptable.

8.0 ELECTRIC POWER

8.2 Offsite Power Systems

The Safety Evaluation Report stated that the applicant was requested to evaluate the ANO-2 design of the Class IE electrical distribution system to determine whether the operability of safety related equipment, including associated control circuitry and instrumentation, can be adversely affected by short term or long term degradation in the offsite power system similar to that experienced at the Millstone Nuclear Station on July 5, 1976.

The applicant's response included in a letter dated March 30, 1978, submitted a summary of the results of the voltage degradation study and the details of the implemented modifications which will ensure the functional operability of the Class IE electrical distribution system and their associated instrumentation and control equipment.

The voltage studies assumed the ANO-2 1978 summer peak load levels, with Unit 1 and 2 at the site being off line and receiving offsite power from the transmission system. The most severe condition investigated was the outage of the 500/161/22 kilovolt autotransformer at the switchyard and a coincidental loss of the hydro unit at the Dardanell Dam. The studies identified that; (1) the voltages at the safety related 480 volt busses could degrade below the minimum required values. This was a result of a fast transfer to the other offsite power source coincident with a safety signal during simultaneous starting of the safety related loads, (2) the voltages at the 4160 volt safety busses were marginal and (3) the voltages resulting from a slow transfer to the other off-site sources were not acceptable. As a result the applicant modified the system design to ensure that adequate voltage conditions on the safety busses would exist during all modes of operation.

These modifications include (1) the sequencing of safety loads on the offsite power sources, (2) the tripping of all non-safety loads except the reactor coolant pumps upon receipt of an engineered safety features signal, (3) deleting the slow transfer scheme to the other off-site power source, and (4) modifying various starter control circuits to improve the loading characteristics on the safety busses.

In addition the applicant has provided on each of the safety trains a second level undervoltage protection to trip the incoming offsite power sources at the 4160 volt safety busses in the event the 480 volt safety busses drop below 92 percent of their rated value. This second level protection is provided in addition to the existing undervoltage breakers which trip the incoming offsite power sources to each of the two trains and align the emergency diesel generators onto the safety busses. These breakers activate when the voltage drops below 78 percent of the rated value of the 4160 volt busses. During our review the applicant was requested and agreed to provide a redundant Class 1E trip breaker at the 480 volt busses designed to preclude inadvertant stripping of the safety busses, and amend their technical specifications to include provisions for periodic verification of the trip setpoints of the 78 percent and the 92 percent undervoltage relays at least once every eighteen months.

Although the present design incorporates only one 92 percent relay on each of the two safety trains, the applicant committed to incorporate an additional 92 percent relay in a coincident logic for each safety train during the first shutdown of the unit after receipt of the relays. A condition to the license will stipulate that this coincidence logic be implemented within six months of issuance of the license. We believe that with the existing 92 percent undervoltage relay, although spurious isolation of the offsite source may occur, there exists adequate protection in the design to assure that the health and safety of the public will not be compromized for this period of time.

Based on our review of the applicant's analysis, the details of the implemented modifications and the commitments made to periodically verify the adequacy of the protective devices, we conclude that the design satisfies the Commission's requirements stated in Section 8.1 of the Safety Evaluation Report, and our concerns identified during our review and is, therefore, acceptable.

9.0 AUXILIARY SYSTEMS

9.7 Fire Protection

A brief discussion of the ANO-2 Fire Protection Program was included in Section 9.7 of the Safety Evaluation Report. We stated that our final evaluation and conclusions regarding our review of the Fire Hazards Analysis Report, the applicants' fire protection program analyses and any required modifications to the facility fire protection program would be reported in a future report. Accordingly, upon completion of our review we have published our findings in a document entitled NUREG-0223, "Fire Protection Safety Evaluation Report" by the Office of NRR, USNRC In the Matter of Arkansas Power and Light Co., Arkansas Nuclear One-Unit 2."

Table 3.1 of the report identifies modifications which are to be completed at some time after the report was prepared. We have added a condition to Amendment No. 1 to the Facility Operating License which stipulates the requirement for and the time by which the modifications of Table 3.1, that were not completed at issuance of Amendment No. 1, must be completed.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.6 Water Hammer

We stated in our Safety Evaluation Report that we would require the applicant to perform tests to verify that unacceptable feedwater hammer will not occur using the plant operating procedures for normal and emergency restoration of steam generator water level following uncovering and possible draining of the feedring. We also stated that we wished to review the procedures for conducting these tests and would require that the tests be performed before the plant reaches full power operating conditions.

The ANO-2 feedwater sparger design includes the use of "J" tubes. This modification tends to prevent the water from draining from the sparger, if the sparger were to uncover, thus inhibiting steam-water interaction (water hammer) in the sparger.

The applicant, in a letter dated May 5, 1978, has committed to a test program to show that unacceptable feedwater hammer damage will not result from anticipated transients. These tests will include the uncovering of the feedwater sparger in all steam generators, and then refilling the steam generators using the operating procedures. We conclude that completion of the tests without unacceptable feedwater hammer damage will accomplish the test objective. We find this program to be acceptable.

13.0 CONDUCT OF OPERATIONS

13.3 Emergency Planning

In our Safety Evaluation Report we reported that the applicant's emergency plan met the requirements of Appendix E to 10 CFR Part 50 and provided an adequate basis for an acceptable state of emergency preparedness. We also noted in our Safety Evaluation Report that the details and procedures to implement the Emergency Plan would require inspection and evaluation by our Office of Inspection and Enforcement prior to issuance of an operating license. Subsequent to the formulation of our finding as reported in our Safety Evaluation Report, it was brought to our attention, through the inspection process, that the implementation of the emergency planning program appeared to be seriously deficient in the areas of the scope of drills and exercises and the coordination of these with the offsite support agencies. In view of these inspection findings it was apparent that Arkansas Power and Light Company did not meet what we considered to be the intent of the plans with respect to implementing certain aspects dealing with the scope of drills and exercises and their coordination with offsite support agencies.

The applicant has submitted Amendments Nos. 54, 55 and 56 to the Arkansas Nuclear One plant Emergency Plan in response to our position on this matter. We have reviewed these amendments to the Emergency Plan and find the changes described through Amendment No. 56 acceptable in resolving our concerns as stated above. Therefore we reaffirm our positive conclusions stated in our Safety Evaluation Report regarding the acceptability of the Emergency Plan.

13.6 Industrial Security

This section of this report replaces in its entirety Section 13.6, "Industrial Security," of the Safety Evaluation Report.

The applicant submitted an initial security plan for the Arkansas Nuclear One plant dated September 11, 1972. We have completed our review of the plan and 12 subsequent revisions submitted between February 5, 1973 and June 9, 1978. We conclude that the security plan, entitled "Arkansas Nuclear One Industrial Security Plan," consisting of Revision 9 dated May 23, 1975 (which replaced in its entirety the previously existing plan), Revision 10 dated October 31, 1975, and Revision 12 dated June 9, 1978, is in conformance with existing criteria including Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage," and is acceptable.

The applicant has submitted a further amended physical security plan dated May 25, 1977, in compliance with the requirements of 10 CFR Part 73.55. This amended security plan has been evaluated by the staff and a security plan review team has visited the plant site as part of this overall evaluation. As a result of our evaluation, certain areas have been identified where additional information and upgrading is required before the amended security plan can be found in conformance with 10 CFR Part 73.55. The applicant has made commitments to modify the amended security plan such that the level of protection will be consistent with the performance requirements of Section (a) of Part 73.55. Subsequent to the above, the applicant submitted a revision dated June 11, 1978. We are reviewing this revision to determine that the industrial security plan described by the revision will afford a level of protection consistent with the performance requirements of Section 73.55 of 10 CFR Part 73 when properly implemented.

We will continue to follow the progress of the applicant's implementation of the upgrading measures to assure conformance with the requirements of Section 73.55 of 10 CFR Part 73.

By letter dated June 13, 1978, the applicant requested an exemption from the requirements of Section 73.55 such that the completion of the Arkansas Nuclear One security system is extended from the date required by Section 73.55 of August 24, 1978. In regard to this subject in general, the Commission, on July 5, 1978, has approved for publication in final form, in the Federal Register, of the amendments to 10 CFR Part 73 set forth in Enclosure "A" of SECY-78-210A, which would extend the requirement for full implementation of the physical protection requirements of Section 73.55 until February 23, 1979. This action meets the objectives of the applicant's request dated June 13, 1978.

14.0 INITIAL TESTS AND OPERATION

In Safety Evaluation Report Supplement No. 1 (June 1976), Section 14.0, we stated that the outstanding item regarding control element assembly drop time testing was resolved pending documentation of the applicant's verbal commitment to perform certain tests. The Final Safety Analysis Report was subsequently revised to include this documentation and this item is resolved.

Since the Safety Evaluation Report was published, the applicant has submitted a description of emergency core cooling system sump model tests which were conducted as an alternative to demonstrating in-plant recirculation from the emergency core cooling system sump in accordance with Regulatory Guide 1.79. The staff has reviewed this information and concludes that this item is resolved. See Section 6.3.4 of this supplement for further information resolution.

Subsequent to publication of the Safety Evaluation Report, the following items were identified as outstanding:

(1) The applicant stated that several preoperational tests may not be completed until after fuel loading. The applicant's letter of July 3, 1978 listed the tests, or portions of tests, that will be deferred until after fuel loading and specified when they will be completed. We have reviewed this information and conclude that it is acceptable. Therefore, this item is resolved.

(2) Portions of the applicant's thermal expansion and vibration monitoring programs were not being documented in the tests procedures. The applicant has informed us of corrective action to be taken which we conclude will resolve this matter if correctly implemented. Our Office of Inspection and Enforcement will verify that the appropriate corrective action is properly implemented. Therefore, we consider this item to be resolved.

(3) The method of conducting the loss of offsite power test was modified in Final Safety Analysis Report Amendment 44 such that the objectives of the test could not be satisfied. The applicant described an acceptable method for performing this test and a commitment to perform the test in this manner in its letter of April 11, 1978. Therefore, this item is resolved.

We conclude that the information provided in the application meets the acceptance criteria in Section 14.2 of the Standard Review Plan and describes an acceptable initial test program that will demonstrate the functional adequacy of plant structures, systems, and components.

15.0 ACCIDENT ANALYSIS

15.4 Postulated Accidents

In Section 2.1 of this report, we provided our conclusions regarding the matter of the size of the low population zone radius for the ANO-2 plant. We had previously reported in the Safety Evaluation Report that a low population zone radius of as low as 2 miles (3,200 meters) was acceptable from a radiological dose consequences standpoint. Therefore, the doses we reported in Table 15.6 for Section 15.4, in Table 15.4 for Section 15.4.6, and in Table 15.5 for Section 15.4.7 of our Safety Evaluation Report were based upon a 2-mile distance.

As Section 2.1 of this report states, this matter has been resolved with the choice of a low population zone radius of 2.6 miles (4,183 meters). Accordingly, Tables 15.4, 15.5 and 15.6 of the Safety Evaluation Report are to be replaced in their entirety by Tables 15.4, 15.5 and 15.6 of this report. Since the reported doses are lower in the revised tables, we reaffirm our conclusions of acceptability as stated in our Safety Evaluation Report.

15.4.2 Reactor Coolant Pump Rotor Seizure

In our Safety Evaluation Report we stated that we had completed our review of the analysis of an instantaneous seizure of the reactor coolant pump rotor, as evaluated by the computer codes CESEC and TORC. We concluded that the plant design, in this regard, is acceptable subject to (1) the receipt of a commitment from the applicant to perform confirmatory tests in support of the utilization of the CESEC code for the ANO-2 analyses, (2) receipt of a description of the test program, and (3) submittal to the staff of the actual test data and results obtained with proper instrumentation which confirms that pretest predictions made by the CESEC code.

The licensee submitted (1) a list of the tests to be performed in a letter dated March 13, 1978, (2) the pretest predictions and a copy of the test procedure by letter dated July 13, 1978, and, (3) additional information on startup test results from another operating reactors startup testing program by letter dated April 5, 1978. We have addressed our requirements for the submittal of the test data and results in a condition to the operating license. This information will be obtained by the licensee during the startup and power ascension testing program when the tests are to be conducted.

We conclude that the licensee has acceptably satisfied our requirement for information, as stated in the Safety Evaluation Report, to be provided prior to the

TABLE 15.4

LOSS-OF-COOLANT ACCIDENT ASSUMPTIONS

Power Level, thermal megawatts	2955
Operating Time, years	3.0
Reactor Building Leak Rate (0-24 hours in percent per day)	0.10
(> 24 hours in percent per day)	0.05
Iodine Composition, percent	
Elemental	91
Particulate	5
Organic	4
Relative Concentration (X/Q)	
0-2 hours @ 1045 meters	7.7×10^{-4}
0-8 hours @ 4183 meters	7.0×10^{-5}
8-24 hours @ 4183 meters	4.6×10^{-5}
24-96 hours @ 4183 meters	1.8×10^{-5}
96-120 hours @ 4183 meters	4.9×10^{-6}
Spray Effectiveness	
Maximum Elemental Iodine Decontamination Factor	100
Elemental Iodine Removal Coefficient during the injection phase hours	10
Particulate Iodine Removal Coefficient, per hour	0.5
Organic Iodine Removal Coefficient	0.0
Containment Parameters	
Region 1 - sprayed volume, cubic feet	1.517×10^6
Region 2 - unsprayed volume, cubic feet	1.46×10^5
Region 3 - unsprayed volume, no communication with Regions 1 and 2, cubic feet	1.17 x 105
Transfer rate between Regions 1 and 2, cubic feet per minute	4800
Total Containment - free volume, cubic feet	1.78×10^6

Abbreviations

X/Q = atmospheric dispersion coefficient in seconds per cubic meter

10^x refers to 10 to the x power; for example, 10^{-6} = 0.000001

TABLE 15.5

FUEL HANDLING ACCIDENT ASSUMPTIONS

Shutdown Time, hours	72
Total Number of Fuel Rods in the Core	40,716
Number of Fuel Rods Involved in the Refueling Accident	236
Power Peaking Factor	1.65
Iodine Fractions Released from Pool	
Elemental	0.75
Organic	0.25
Effective Filter Efficiency, percent	
Elemental	90
Organic	70
X/Q Values, seconds per cubic meter	
0-2 hours @ 1260 meters	7.7×10^{-4}
0-2 hours @ 4183 meters	7.0×10^{-5}

TABLE 15.6

POTENTIAL OFFSITE DOSES DUE TO DESIGN BASIS ACCIDENTS

	Two-Hour Exclusion Boundary (1045 Meters)		Course of Accident Low Population Zone (4183 Meters)	
Accident	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
Loss-of-Coolant	236	5	160	2
Fuel Handling (In spent fuel pool area)	35	< 1.0	3	< 1.0
Rod Ejection	51	< 1.0	4	< 1.0

issuance of the operating license. We have addressed our requirements following the issuance of the operating license in a condition to the operating license.

15.4.4 Spectrum of Piping Breaks Inside and Outside of Containment

In the Safety Evaluation Report we stated that we had requested the applicant to provide further details on the two-dimensional and one-dimensional power distribution calculations which were used to synthesize the two-dimensional results for the steam line break accident analysis.

We had requested the applicant to provide additional information regarding a comparison of the top peaked axial power distribution shape used in the one-dimensional/two-dimensional synthesis method with the expected axial power distribution shape obtained by a three-dimensional analysis. We also requested a specific description regarding how the core parameters are input to the departure from nucleate boiling ratio evaluation.

The applicant's response indicates that the departure from nucleate boiling ratio calculations were performed using the Macbeth correlation with the one-dimensional/two-dimensional top peaked power distribution. The three-dimensional analysis predicts a bottom peaked power distribution. Based on our review and evaluation of this information we conclude that the departure from nucleate boiling ratio analysis has been conservatively performed, since it will result in a lower departure from nucleate boiling ratio when compared to calculations for the same channel with a bottom peaked power distribution predicted by the three-dimensional analysis. Therefore, this analysis is acceptable to us and this matter is resolved.

15.4.6 Loss-of-Coolant Accident

In our Safety Evaluation Report we concluded that recirculating emergency core cooling fluid which leaked from systems located outside containment would be adequately treated to limit the doses from this pathway. Subsequent to issuance of the Safety Evaluation Report and in discussions with the applicant on the technical specifications, we learned that the applicant did not believe that it could be conclusively demonstrated that the auxiliary building filter systems would sufficiently collect and filter the iodines released as a result of such leakage. On this basis, it appears that the conclusion contained in the Safety Evaluation Report regarding this matter is not correct. Therefore, the second paragraph of Section 15.4.6 of the Safety Evaluation Report should be considered to be replaced in its entirety by the following evaluation.

As part of the loss-of-coolant accident analysis we and Arkansas Power and Light Company have evaluated the consequences of leakage outside the containment of containment sump water which is circulated by the emergency core cooling system after a postulated loss-of-coolant accident. Portions of this recirculation path consisting of emergency core cooling system piping, valves and pumps are located outside of the containment in the auxiliary building. After the accident the water is circulated through equipment located in the auxiliary building to be cooled. Substantial amounts of leakage such as that resulting from failure of a pump shaft seal are postulated to occur at any time after the initial 24-hour period following a loss-of-coolant accident. The leakage normally expected from operation of the systems is also considered in the analysis. If such leakage should develop a portion of the iodine contained in the coolant could become airborne, either by volatilization or directly with the coolant that flashes to steam on leakage from the seal. If allowed to escape freely and if not filtered before release, the calculated doses from iodine releases caused by such leakage could exceed the guidelines of 10 CFR Part 100.

Arkansas Power and Light Company has submitted an analysis of the dose consequences from an assumed leakage rate of five gallons per minute caused by failure of a pump shaft seal. This analysis states that should a pump seal failure occur at such a time following a loss-of-coolant accident the resultant leakage would be quickly detected by alarms from sump level indicators and simple operator actions such as securing the pumps through handswitches in the control room would serve to control further leakage.

The applicant argues that the pump room with the doors closed has been designed to be water tight (although not air tight) and therefore any leakage would be small in extent. Isolation is an acceptable means of control; however, we understand that the doors to these rooms are not normally closed, so that any airborne material would be free to pass to other regions of the auxiliary building. In this event removal of some or substantially all the airborne material by plateout or filtration by the non-ESF ventilation system would occur before the iodine was subsequently released to the environment, depending upon the operability and effectiveness of the system post-LOCA. The applicant has however stated that he is unable to demonstrate any reliable credit for this ventilation system. Accordingly, we have assumed that any releases with the doors open would be unprocessed and would go directly to the atmosphere.

We have reviewed the applicant's analysis and have also performed an independent analysis of the consequences of such leakage. We have assumed the sump contains a mixture of iodine fission products in agreement with Regulatory Guide 1.7, "Control of Combustible Gas Concentration in Containment Following a Loss of Coolant Accident." The worst case situation would be a leakage following a design basis loss-of-coolant accident, where for the purposes of designing engineered safety feature fission product removal systems, a large source term is postulated (50 percent of the inventory of iodine is assumed to be released and would be largely contained in the recirculating water after a few hours). For the assumed conditions the concentration of iodine-131 in the primary coolant after 24 hours would be 57 curies per gallon. Thus, assuming 10 percent of the released iodine becomes airborne, approximately six curies would be airborne per gallon of coolant leaked.

Considering a conservative assumption that the release takes place during night time inversion conditions, the resultant consequences could be significant. Using the 0- to 8-hour meteorological dispersion coefficent for ANO-2 of 7×10^{-5} seconds per cubic meter and a breathing rate of 347 cubic centimeters per second, the dose at the low population zone would be about one quarter rem per gallon of coolant leaked. Thus a conservative assumption of a 50-gallon per minute leak for a 30-minute period would yield, without filtration or holdup, 370 rem. By using a one- to four-day meteorological dispersion coefficient, the dose would be a factor of four lower. Thus, it is important that means be available to prevent the uncontrolled release of airborne iodine resulting from a failed pump seal.

While there is no reason to believe that a major failure of a pump seal is a likely event, the possibility of such a failure during long-term post-loss-of-coolant accident cooling has been considered as an event that should be accommodated by design. Such failures have, for purposes of analysis, been assumed to occur 24 hours after the accident. Considering the eight-day half-life of iodine-131, the specific time selected is not particularly important to the calculation.

The ventilation ducts that serve these rooms are automatically isolated on receipt of a safety injection signal. The applicant has proposed that the doors to these rooms also be closed in the event of an accident. Although the rooms are not air tight, they do offer substantial isolation of the environment inside. Slow leakage of the contents, even for a day or two, would result in consequences at least a factor of four less than a short-term release as discussed above. Our most conservative analysis would thus show that a dose of 92 rem (370/4) should be added to the loss-of-coolant accident (LOCA) dose for emergency core cooling system (ECCS) leakage if isolation of the doors to these rooms are effected. When added to the low population zone LOCA dose of 160 rem, the overall LOCA dose is within the guideline values of 10 CFR Part 100.

Many factors exist which have been ignored, and which would in fact result in a lower calculated dose. These include use of a conservative leak rate; use of a conservative iodine release from the spill; and no credit for iodine removal mechanisms such as plateout or filtration. Nonetheless, we believe that the applicant should include a requirement in his accident recovery procedures to isolate these rooms within 24 hours of a loss-of-coolant accident, to provide an assured control over any radioactive releases from pump seals. The applicant has, therefore, included this requirement in the applicable post-LOCA procedures.

On the basis of the foregoing analysis and compliance by the applicant with our requirements for isolation, we conclude that the doses resulting from the postulated leakage of post-LOCA recirculation water, when added to the direct leakage LOCA doses, result in total doses that are within the guideline values of 10 CFR Part 100.

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

During its 216th meeting on April 6-7, 1978, the Advisory Committee on Reactor Safeguards (hereinafter referred to as the Committee) completed its review of the Arkansas Nuclear One-Unit No. 2 operating license application. A copy of the Committee's report, dated April 12, 1978, which contains certain comments and recommendations, is included as Appendix C to this report. The actions we have taken or plan to take in response to the Committee's comments and recommendations are described in the following paragraphs.

(1) The Committee stated that it wishes to be kept informed of the results of the surveillance program to be executed on the initial fuel loading as that fuel is removed from the core.

The findings resulting from our participation in this fuel surveillance program which shall be executed in increments at the first, second and third regularly scheduled refueling outages, will be discussed in forthcoming records of meetings or site visits by the staff on this subject.

(2) The Committee stated that we had identified six CPCS and a number of other safety-related items which remain outstanding. The Committee recommended that these matters be resolved in a manner satisfactory to us.

Those outstanding issues, which we require to be satisfactorily resolved, are listed in Section 1.6 of this report. These issues have now been resolved as discussed in the sections indicated in Section 1.6 of this report.

(3) The Committee identified the generic problems listed in the Committee's report, "Status of Generic Items Relating to Light Water Reactors: Report No. 6," dated November 15, 1977 which are relevant to ANO-2. The Committee recommended that these problems be dealt with us and the applicants as solutions are found.

Our Safety Evaluation Report previously reported on the status of the various generic problems discussed in the Committee's report on this subject dated January 31, 1977. Although the completion of the staff's evaluation of these issues may result in additional actions regarding the ANO-2 plant, we have determined that at this time the status for each of the items referenced in the Committee's letter dated April 12, 1978 continues to be as reported in the Safety Evaluation Report for the ANO-2 plant.

20.0 FINANCIAL QUALIFICATIONS

20.1 Introduction

The Nuclear Regulatory Commission's regulations relating to the determination of financial qualifications of applicants for facility operating licenses appear in Section 50.33(f) and Appendix C of 10 CFR Part 50. In accordance with these regulations, Arkansas Power and Light Company, hereinafter APLCo, submitted operating cost estimates with its application as well as providing additional financial information at the Staff's request. The following analysis summarizes the review of the financial information and addresses APLCo's financial qualifications to operate ANO-2 and to permanently shut down the facility and maintain it in a safe shutdown condition, should that become necessary.

APLCo is a wholly owned operating subsidiary of Middle South Utilities, Inc. serving 62 of the 75 counties in the State of Arkansas. The four other principal operating companies of the Middle South system - Mississippi Power and Light Company, Louisiana Power and Light Company, Arkansas - Missouri Power Company, and New Orleans Public Service, Inc., share generation capacity and other power resources with APLCo. Sales of electric energy account entirely for the total gross operating revenues of APLCo, thereby comprising approximately 37 percent of the affiliated group's consolidated revenues. Its customers include residential, commercial, industrial and wholesale users. Operating revenues for the 12 months ended September 30, 1977 were $511.3 million and net income was $63.2 million. Invested capital at July 31, 1977 amounted to $1.189 billion and consisted of 49.7 percent long-term debt, 14.5 percent preferred and preference stock, and 35.8 percent common equity. The first mortgage bonds are rated Baa, medium grade obligations, by Moody's and A by Standard and Poor's.

20.2 Estimated Operating and Shutdown Costs

For the purpose of estimating the unit's annual operating costs, the applicant assumed that ANO-2 would commence commercial operation in January 1979. APLCo's estimate of the total annual cost of operating the units during each of the first five years of commercial operation are presented below. The unit costs (mills per kilowatt hour) are based on a net electrical capacity of 912 Mega watts electric and the projected plant capacity factors as shown below. The five-year average costs were calculated by averaging the estimated data for the years 1979 to 1983 inclusively.

ARKANSAS NUCLEAR ONE - UNIT 2

ESTIMATED OPERATING COSTS

	Total Cost (thousands)	Plant Capacity Factor (percent)	Mills per kilowatt hour
1979	$112,391	58.0	24.26
1980	109,840	71.0	19.31
1981	112,676	80.0	17.63
1982	109,943	80.0	17.20
1983	107,610	80.0	16.84
TOTAL	$552,460		
Five-year average			19.05

The estimates of operating costs consider operating and maintenance expenses (including fuel expense), depreciation, and a return on investment.

In estimating the costs of permanently shutting down the facility, the company assumed that after forty (40) years of operation the plant would be fully decommissioned and no longer used as a commercial nuclear power facility. Expected decommissioning activities include processing, shipping and disposal of removable nuclear waste material, removal of all salvagable components, decontamination of radioactive areas by chemical cleaning and flushing, packaging and shipment of resultant radioactive waste to a storage facility, and the entombment of plant components. Present day decommissioning costs are estimated to total $10.0 million. APLCo estimates the annual cost of maintenance after decommissioning to be $40,000. Included in this estimate are the costs of a security force, surveillance, radiation monitoring and miscellaneous operating expenses. Bases upon currently available information such costs are considered reasonable. Additionally, applicant's decommissioning expenses are nominal in light of applicant's total financial resources, and their ultimate impact in unit costs.

20.3 Source of Funds

APLCo expects to cover all operating costs through the revenues generated from its system-wide sales of electricity. Current operating costs will be paid out of current operating revenues. The estimated unit operating costs shown in Section 20.2, above, compare favorably with APLCo's revenue experience. Its average unit price for electricity sold during the 12 months ending July 31, 1977 was 2.852 cents per kilowatt hour (excluding system sales and sales to Middle South Power Pool), well above the total estimated unit operating costs for the subject facility. Additionally, it is noted that the above unit price data does not reflect possible rate increases to be allowed during the first five years of this facility's operation.

Financial data consisting of gross revenues realized and net income earned by the applicant during the years 1972 to 1976, includively, are as follows:

ARKANSAN POWER AND LIGHT COMPANY
(Millions of Dollars)

Year	Gross Revenues	Net Income
1972	$184.8	$36.7
1973	$209.3	$41.9
1974	$296.8	$55.6
1975	$316.8	$40.7
1976	$396.6	$47.0

Based upon the above, the applicant has consistently demonstrated the ability to achieve revenues sufficient to cover all operating expenses and interest charges.

20.4 Conclusion

In accordance with the regulations cited in Section 20.1 above, there must be reasonable assurance that the applicant can obtain the necessary funds to cover the estimated costs of the activities contemplated under the license. This reasonable assurance standard must be viewed in light of the potentially long period of commercial utilization of the facility. Consequently, the assumption is implicit that there will be rational regulatory policies over this period with respect to the setting of rates. This further implies that rates will be set to at least cover the cost of service, including the cost of capital.

Based on the preceding analysis, we conclude that Arkansas Power and Light Company has satisfied this reasonable assurance standard and is accordingly financially qualified to operate and, if necessary, to permanently shut down Arkansas Nuclear One - Unit 2 and maintain it in a safe shutdown condition. This determination is predicated upon the demonstrated ability of the applicant to achieve revenues sufficient to cover all operating costs and interest charges, the favorable comparison of current energy unit prices with those as projected for this facility, and the regulated status of the applicant.

21.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

21.3 Operating Licenses

The first paragraph in Section 21.3 of the Safety Evaluation Report should be replaced with the following parapraph:

"Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for ANO-2 (which has a rated capacity in excess of 100,000 electrical kilowatts) is the maximum amount available from private sources which is currently \$470 million."

22.0 CONCLUSIONS

Our conclusions as stated in the Safety Evaluation Report are hereby replaced by the following conclusions as stated below.

Based on our evaluation of the application as set forth herein, we have determined that, conditioned upon the favorable resolution of the outstanding matters described herein, we conclude that:

1. The application for facility license filed by the applicant dated September 10, 1970, as amended, complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter I; and

2. Construction of Arkansas Nuclear One - Unit 2 (the facility) has proceeded and there is reasonable assurance that it will be substantially completed, in conformity with Construction Permit No. CPPR-89, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

3. The facility requires exemptions from certain requirements of (a) Section 50.55a of 10 CFR Part 50, and (b) Appendices G and H and Appendix J to 10 CFR Part 50. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. With the granting of these exemptions, the facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission; and

4. There is reasonable assurance (a) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (b) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter I; and

5. The applicant is technically and financially qualified to engage in the activities authorized by these licenses, in accordance with the regulations of the Commission set forth in 10 CFR Chapter I; and

6. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public.

APPENDIX A

SUPPLEMENT TO THE CHRONOLOGY OF THE RADIOLOGICAL SAFETY REVIEW

January 31, 1978	Applicant letter providing seismic qualification information.
February 7, 1978	Applicant letter on loss of offsite power test.
February 9, 1978	Applicant letter on fuel and control element assemblies.
February 10, 1978	Applicant letter on containment leakage testing program.
February 14, 1978	Meeting with applicant on Technical Specifications.
February 21, 1978	Applicant letter on proprietary information.
February 24, 1978	Meeting with applicant on offsite power system degradation.
February 28, 1978	Applicant letter on Fire Protection Administrative Controls and Procedures.
March 3, 1978	Applicant letter on 2.6 mile low population zone.
March 3, 1978	Applicant letter transmitting Amendment 54 to Emergency Plan.
March 6, 1978	Applicant letter on turbine control valve testing.
March 6, 1978	Staff letter on main steam line break mass and energy releases and containment leakage testing program.
March 7, 1978	Staff letter on main steam line break mass and energy and releases and containment leakage testing program.
March 10, 1978	Staff letter on reactor vessel seal ring missile problem.
March 10, 1978	Amendment No. 45.
March 10, 1978	Applicant letter on the November 21, 1977 CPCS meeting.
March 13, 1978	Applicant letter or CESEC verification program. Applicant letter on list of preoperational tests to be conducted during power ascension test program.

March 13, 1978	Meeting with applicant to discuss SER outstanding items.
March 16, 1978	Staff letter on containment purge system radiation monitors.
March 17, 1978	Applicant letter on small break ECCS performance analysis.
March 17, 1978	Staff letter requesting additional information on six subjects.
March 17, 1978	Staff letter on evaluation of reactor vessel supports.
March 20, 1978	Applicant letter on long-term post-LOCA cooling methods to preclude boron precipitation.
March 20, 1978	Applicant letter on steam generator tube denting.
March 20, 1978	Combined meeting of the ACRS Electrical Systems, Control and Instrumentation Subcommittee and the ANO-2 Subcommittee with the applicant and the Regulatory staff.
March 22, 1978	Applicant letter supplementing applicant's March 20, 1978 letter on boron precipitation.
March 24, 1978	Applicant letter on Section XI preservice and inservice inspection program.
March 24, 1978	Applicant letter on RELAP-4 analysis of main steam line break mass and energy releases.
March 23, 1978	Applicant letter on input fault and surge testing of vital PPS power supplies.
March 24, 1978	Applicant letter on piping shock and vibration testing program.
March 24, 1978	Applicant letter on "Primary Reactor Containment Integrated Leakage Rate Test" report.
March 27, 1978	Applicant letter on CPCS report numbers CEN-57(A)-P, CEN-69(A)-P, CEN-72(A)-P, and CEN-73(A)-P.
March 27, 1978	Applicant letter on reports CEN-57, 69, 72 and 73.
March 29, 1978	Staff letter requesting additional information on post-LOCA long-term cooling to preclude boron precipitation.

March 29, 1978	Staff letter on ECCS review.
March 30, 1978	Meeting with applicant on compliance with 10 CFR 73.55.
March 30, 1978	Applicant letter on environmental qualification of Fischer Porter and Potter-Brumfield equipment.
March 30, 1978	Applicant letter on ECCS small break analysis.
March 30, 1978	Applicant letter on offsite power systems.
March 30, 1978	Applicant letter on compliance with Regulatory Guide No. 1.44.
March 31, 1978	Applicant letter on fire protection question responses.
April 4, 1978	Applicant letter on CPCS report numbers CEN-63(A) Supplement 1 and CEN-86(A).
April 4, 1978	Applicant letter on reports CEN-86A Supplement 1 and CEN-86A.
April 5, 1978	Applicant letter on CESEC code verification program.
April 5, 1978	Applicant letter on long-term cooling post-LOCA boron precipitation calculations.
April 6, 1978	Applicant letter on Emergency Plan Amendment No. 55.
April 7, 1978	Meeting with applicant on SER outstanding items.
April 10, 1978	Staff letter on implementation of 10 CFR Part 73.55.
April 10, 1978	Amendment No. 55 to FSAR.
April 11, 1978	Staff letter on environmental qualifications of Potter Brumfield relays.
April 11, 1978	Staff letter on environmental qualification of Potter Brumfield relays.
April 11, 1978	Applicant letter on loss of offsite power test procedure.
April 12, 1978	Advisory Committee on Reactor Safeguard's letter to Chairman Hendrie.

April 12, 1978	Staff letter requesting additional information on five subjects.
April 12, 1978	Meeting with the Advisory Committee on Reactor Safeguards.
April 12, 1978	Applicant letter on fire protection question responses and fire hazards.
April 13, 1978	Staff letter on fire protection positions.
April 14, 1978	Staff letter transmitting ACRS letter report dated April 12, 1978 to applicant.
April 14, 1978	Staff letter on environmental qualifications for the main steam line break inside containment.
April 17, 1978	Applicant letter on fire protection question responses.
April 18, 1978	Staff letter on piping shock and vibration testing program.
April 19, 1978	Staff letter on FSAR Table 7.5-1 instrument classifications.
April 19, 1978	Meeting with applicant on environmental qualifications for the main steam line break accident inside containment.
April 21, 1978	Applicant letter on Fischer Porter environmental qualifications.
April 24, 1978	Applicant on reactor vessel support loads summary table.
April 24, 1978	Staff's trip report on visit to witness sump testing.
April 25, 1978	Applicant letter on FSAR Section 14.0 preoperational test program.
April 26, 1978	Staff letter on evironmental qualification documentation for LOCA/MSLB.
April 26, 1978	Applicant letter on fire protection program question responses.
April 26, 1978	Applicant letter on containment purge isolation valves.
April 28, 1978	Applicant letter on low temperature overpressure protection.
April 28, 1978	Applicant letter on steam generator tube denting.

April 28, 1978	Applicant letter on polyethylene cable coatings.
April 28, 1978	Staff letter on Modified Amended Security Plan Meeting of March 30, 1978.
April 28, 1978	Applicant letter on containment electrical penetration breaker coordination.
April 28, 1978	Applicant letter on input fault and surge testing of vital PPS power supplies.
April 28, 1978	Applicant letter on questionnaire for diesel generator reliability study.
May 1, 1978	Applicant letter on ECCS pump room leakage.
May 5, 1978	Staff letter on NUREG-0219, Draft 2 on nuclear security personnel training program.
May 5, 1978	Applicant letter on feedwater system water hammer test procedure.
May 5, 1978	Applicant letter on deletion of bottled oxygen supplies in control room.
May 10, 1978	Staff letter on proprietary core protection calculator information.
May 11, 1978	Applicant letter on proposed Industrial Security Plan relative to 10 CFR Part 73.55.
May 12, 1978	Meeting with applicant on steam generator tube denting problem.
May 17, 1978	Applicant letter in response to instrumentation and control systems questions on environmental qualifications for the LOCA and the main steam line break accidents inside containments.
May 18, 1978	Applicant letter on containment pump testing.
May 19, 1978	Applicant letter submitting Amendment No. 46 to the License Application.
May 19, 1978	Amendment No. 46.

May 23, 1978	Applicant letter in response to containment systems questions on environmental qualifications for the main steam line break accident inside the containment.
May 24, 1978	Meeting with applicant on fire protection review.
May 24, 1978	Staff letter to Dr. Stephen Lawroski, Chairman, ACRS, on CPCS proprietary information.
May 25, 1978	Meeting with applicant on method of compliance with Appendices G and H to 10 CFR Part 50 and the preservice inspection plan.
May 26, 1978	Staff letter on loss of offsite power test procedures.
June 1-2, 1978	Meeting with applicant on technical specifications.
June 2, 1978	Staff's trip report of the March 28 and 29 ANO-2 plant site to examine the containment sump areas of the plant.
June 5, 1978	Staff letter on fire protection document entitled "Manpower Requirements for Operating Reactors."
June 7, 1978	Applicant letter on CPCS Position 26.
June 7, 1978	Applicant letter to ACRS on anchor bolt material.
June 8, 1978	Applicant letter submitting affadavit on previously docketed letters.
June 8, 1978	Applicant letter on environmental qualifications for the main steam line break inside containment.
June 9, 1978	Meeting with applicant on fire brigade size.
June 9, 1978	Meeting with applicant on CEA guide tubes.
June 9, 178	Applicant letter on Industrial Security Plan.
June 9, 1978	Applicant letter on steam generator to be support plate modifications.
June 12, 1978	Staff letter on two reports by Sandia on physical security subjects.

June 12, 1978	Applicant letter on steam generator modifications.
June 13, 1978	Applicant letter on reactor vessel fracture toughness.
June 13, 1978	Applicant letter on five man fire brigade.
June 13, 1978	Applicant letter on inservice inspection plan.
June 13, 1978	Applicant letter on request for exemption from 10 CFR Part 73.55.
June 14, 1978	Applicant letter on Appendices G and H to 10 CFR Part 50.
June 15, 1978	Applicant letter on steam generator operating history questionnaire.
June 15, 1978	Applicant letter on fire protection question responses.
June 15, 1978	Applicant letter on Section XI inservice pump and valve testing program.
June 16, 1978	Staff letter on scheduling of preoperational tests.
June 16, 1978	Amendment No. 47.
June 16, 1978	Applicant letter on proprietary CPCS information.
June 16, 1978	Applicant letter on containment electrical penetration assemblies.
June 17, 1978	Applicant letter on Agastat relays.
June 17, 1978	Applicant letter on ITE relays.
June 17, 1978	Applicant letter on reactor vessel internals hold down ring clamping force.
June 20, 1978	Applicant letter on ECCS pump room leakage.
June 20, 1978	Meeting with applicant on compliance with 10 CFR Part 50 Appendix J.
June 20, 1978	Applicant letter on Potter Brumfield relays.

June 20, 1978	Applicant letter on fire brigade size.
June 21, 1978	Applicant letter on request for exemption from 10 CFR Part 73.55.
June 22, 1978	Applicant letter on construction permit extension.
June 23, 1978	Applicant letter on post-LOCA environmental qualifications.
June 23, 1978	Applicant letter on low pressurizer pressure trip setpoint.
June 26, 1978	Applicant letter on CEA guide tube modifications.
June 27, 1978	Applicant letter on environmental qualifications for the main steam line break accident inside containment.
June 27, 1978	Applicant letter on seismic qualification of PPS cabinet 2C15.
June 29, 1978	Applicant letter on reactor coolant system flow rate technical specification.
June 29, 1978	Applicant letter on fire protection program commitments.
June 30, 1978	Applicant letter on Emergency Plan Amendment No. 56.
June 30, 1978	Applicant letter on environmental qualification classification of overpressure protection valves.
June 30, 1978	Applcant letter on fire retardant cable coating.
July 3, 1978	Applicant letter on environmental qualification testing of Foxboro equipment.
July 3, 1978	Applicant letter on CPCS documentation.
July 3, 1978	Applicantletter on scheduling of preoperational tests.
July 3, 1978	Applicant letter on reactor vessel seal ring missiles.
July 6, 1978	Applicant letter on high pressurizer pressure and trip response time.
July 6, 1978	Applicant letter on FSAR changes.

July 7, 1978	Applicant letter on ECCS pump room leakage.
July 7, 1978	Applicant letter on fire protection fire brigade size.
July 7, 1978	Meeting with applicant on control element assembly guide tube modifications.
July 7, 1978	Applicant letter on affadavit for previously docketed letters.
July 11, 1978	Applicant letter on construction permit.
July 12, 1978	Staff letter to Dr. Stephen Lawroski, Chairman, ACRS transmitting revised Supplement No. 1 to the Safety Evaluation Report.
July 12, 1978	Staff letter to applicant transmitting revised Supplement No. 1 to the Safety Evaluation Report.
July 13, 1978	Staff letter on NUREG/CR-0181 on physical security system assessment.
July 13, 1978	Applicant letter on secondary financial protection requirements of 10 CFR Part 140.
July 13, 1978	Applicant letter on containment electrical penetration assembly epoxy material.
July 13, 1978	Applicant letter on ECCS pump room leakage.
July 13, 1978	Applicant letter on environmental qualifications of motors, valve operators and cables.
July 13, 1978	Applicant letter on affadavit for previously docketed letters.
July 13, 1978	Applicant letter on CESEC Code verification program.
July 17, 1978	Applicant letter on environmental qualification of Fischer Porter equipment.
July 18, 1978	Issuance of Facility Operating License NPF-6.
July 18, 1978	Applicant letter on control element assembly guide tube modifications.

July 20, 1978	Staff letter on NRC guidance entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications."
July 20, 1978	Staff letter on NRC guidance on radiological environmental monitoring.
July 21, 1978	Staff letter on authorization to proceed to operational Modes 6 and 5.
July 24, 1978	Applicant letter on inservice testing of charging pumps.
July 25, 1978	Applicant letter on input fault and surge testing of vital PPS power supplies.
July 25, 1978	Applicant letter on control element assembly guide tube sleeve deburring.
July 26, 1978	Applicant letter on fire barrier seals in conduit penetrations.
August 2, 1978	Staff letter on environmental qualifications of containment bypass damper fan motor.
August 4, 1978	Meeting with applicant on environmental qualifications of Foxboro and Fischer Porter equipment.

APPENDIX C

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 12, 1978

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON ARKANSAS NUCLEAR ONE, UNIT 2 NUCLEAR POWER PLANT

Dear Dr. Hendrie:

During its 216th meeting, April 6 and 7, 1978, the Advisory Committee on Reactor Safeguards completed its review of the application of Arkansas Power and Light Company (Applicant) for a permit to operate the Arkansas Nuclear One, Unit 2 Nuclear Power Plant (ANO-2). The application was also considered at the 214th ACRS meeting, February 9-11, 1978, and was reviewed at Subcommittee meetings on June 24, 1977 in Russellville, Arkansas and February 2 and March 20, 1978 in Washington, DC. Subcommittee meetings were also held on February 28, 1975 and May 20, 1977 in Windsor, Connecticut and on June 30, 1977 and March 20, 1978 in Washington, DC to review the Combustion Engineering designed Core Protection Calculator System (CPCS) which will be employed on ANO-2. A tour of the ANO-2 facility was made by Subcommittee members on June 24, 1977. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, Combustion Engineering, Inc. (CE), Bechtel Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

ANO-2 is the second nuclear unit constructed on the Arkansas Nuclear One site which is located on the Arkansas River in Pope County, Arkansas about six miles from the city of Russellville. The two units differ in that Unit 1 utilizes a Babcock and Wilcox Nuclear Steam Supply System (NSSS) which was licensed on May 21, 1974 to operate at 2568 MWt, while Unit 2 is a CE NSSS for which a license to operate at 2815 MWt is sought. The Committee reported on the construction permit application for ANO-2 in its letter of February 10, 1972.

The ANO-2 NSSS is similar to the Calvert Cliffs 1 and 2 and St. Lucie 1 nuclear units which are now operating; however, ANO-2 will be the first reactor to use CE 16 x 16 fuel assemblies. The NRC Staff concluded that the Applicant has acceptably established the basis for this new fuel design. The Committee agrees with this conclusion. The NRC Staff will require that the

Applicant conduct a surveillance program on the new fuel as it is removed from the core. The Committee wishes to be kept informed of the results of this program (Generic Item IIB-2 in ACRS Report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977).

The Applicant proposes to make use of the CPCS as part of the reactor protection system. The CPCS consists of four redundant digital computers which acquire data from plant process sensors and from two redundant, computer-based control element assembly calculators which provide control rod position information. This application of the CPCS will mark the first use in a United States power reactor of an online digital computer as part of the reactor protection system. The Applicant has developed an extensive series of tests for determining proper operation of both the hardware and the software that make up the system. The NRC Staff has concluded that, subject to resolution of several issues which appear to have available solutions, the CPCS is acceptable (Generic Item IIB-1 in ACRS Report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977).

The NRC Staff has identified six CPCS and a number of other safety related items which remain outstanding. These matters should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

Various generic problems are discussed in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977. Those problems relevant to the Arkansas Nuclear One, Unit 2 Nuclear Power Plant should be dealt with by the NRC Staff and the Applicant as solutions are found. The relevant items are: II-1, 2, 3, 4, 5B, 6, 7, 10; IIA-2, 3, 4; IIC-1, 3A, 3B, 4, 5, 6; IID-2.

The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the Arkansas Nuclear One, Unit 2 Nuclear Power Plant can be operated at core power levels up to 2815 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

Additional Comments by Member William Kerr

I urge the NRC Staff to reconsider its decision to require the Applicant to disconnect the data links from the Core Protection Calculator System to the Plant Computer following initial startup and subsequent refueling startups. The additional information which can be provided by the use of these links could enhance the reliability of both the protection system and of plant control. I find the Staff's arguments against the use of these links unconvincing.

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Supplement No. 1 to the Safety Evaluation Report (USNRC Report NUREG-0308) by the Office of Nuclear Reactor Regulation in the Matter of Arkansas Power and Light Company Operation of Arkansas Nuclear One, Unit 2," Docket No. 50-368, March 6, 1978.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report by the Office of Nuclear Reactor Regulation Related to the Arkansas Power and Light Company Operation of Arkansas Nuclear One, Unit 2 Nuclear Power Plant, Docket No. 50-368," USNRC Report NUREG-0308, November, 1977.
3. Arkansas Power and Light Company (AP&L Co.), "Arkansas Nuclear One, Unit 2 Nuclear Power Plant Final Safety Analysis Report" with Amendments 1-44.
4. Letter from D. H. Williams, Manager of Licensing, AP&L Co., to J. F. Stolz, Chief, Light Water Reactors Branch No. 1, concerning seismic qualification of a process protective cabinet, dated January 24, 1978.
5. Letter from D. H. Williams, Manager of Licensing, AP&L Co., to E. M. Howard, Director, Office of Inspection and Enforcement (I&E), Region IV, concerning cracking of pump support columns for low pressure safety injection pumps, dated January 16, 1978.
6. Letter from D. A. Rueter, Director of Technical and Environmental Services (TES), AP&L Co., to E. M. Howard, Director, Office of I&E, Region IV, concerning emergency feedwater pump piping, dated November 18, 1977.
7. Letter from D. A. Rueter, Director of TES, AP&L Co., to E. M. Howard, Director, Office of I&E, Region IV, concerning valve motor operators, dated November 7, 1977.
8. Letter from D. A. Rueter, Director of TES, AP&L Co., to E. M. Howard, Director, Office of I&E, Region IV, concerning control room emergency chillers, dated October 17, 1977.
9. Letter from D. A. Rueter, Director of TES, AP&L Co., to E. M. Howard, Director, Office of I&E, Region IV, concerning high pressure safety injection pump flow rates, dated September 30, 1977.

APPENDIX D

TABLE OF CONTENTS

LIST OF TABLES

Supplement No. 2 To
APPENDIX D

Core Protection Calculator System

D.1 General

This Supplement No. 2 to Appendix D of the Safety Evaluation Report presents further details of the staff's review evaluation results which have been established subsequent to the preparation of Supplement No. 1 to the Safety Evaluation Report. Section D.3 of this report is concerned with the evaluation of protection algorithms, D.4 with hardware design and qualification, while D.6 presents references.

Acronyms are extensively employed in certain sections of this appendix. Therefore, a listing of the most frequently used acronyms and their meaning is included below.

CPCS - core protection calculator system.
CPC - core protection calculator.
CEA - control element assembly, i.e., control rod.
CEAC - control element assembly calculator.
RSPT - reed switch position transmitter.
AC - alternating current.
DNBR - departure from nucleate boiling ratio.
LPD - local power density.
COLSS- core operating limit supervisory system.
MACS - multipurpose acquisition control system.
I/O - input/output device.
PUPS - power utility plant simulator.

D.3 Protection Algorithms

D.3.5 Power Distribution Uncertainty

Although the precise value of the power distribution uncertainty cannot be verified prior to the completion of the startup test program, the staff has determined that the current status of the concern expressed by position 1 does not preclude issuance of an Operating License for the following reasons: First, alternative systems (COLSS and INCA) are available during startup testing to calculate required power distribution parameters. Second, the basic methodology and instrumentation used in previous Combustion Engineering plants to calculate excore axial off-set is similar to that employed by the Core Protection Calculators (CPCs); thus, errors in the calculated CPC synthesis uncertainty are not expected to be extreme. Third, the applicant will operate at less than design power level during the startup verification of the CPC power distribution algorithm to accommodate with margin, credible inaccuracies in the CPC power distribution uncertainty. Therefore, position 1 remains outstanding and we will present our evaluation of the startup test results in a forthcoming safety evaluation.

D.3.11 Addressable Constants

In our Safety Evaluation Report, Supplement No. 1, we identified that all addressable constants have acceptable range limit checks with the exception of the shape annealing matrix (SAM) components. The staff required reasonability checks of addressable constants as a method for detecting gross errors upon operator entry. The staff required the applicant to identify acceptable range limits on the shape annealing matrix components based on ANO-2 design calculations.

In evaluating the applicant's response, the staff has determined that the present large range limits for the Shape Annealing Matrix (SAM) components are acceptable and are not required to be more restrictive. In the applicant's response, reference 2-20, it is demonstrated that the deviation of any signal SAM component from its correct value will lead to increased conservatism in the prediction of the power distribution parameters. In addition, the administrative procedure employed when SAM components are entered into the CPCs was reviewed during Phase II testing and was found to be a significant deterrent to the entering of erroneous SAM components. With this information, the staff now considers position 15 to be fully resolved.

D.4 Design and Qualification

D.4.1 Hardware Design

D.4.1.2 Signal Generation and Process Equipment for the CPCS Safety Position 5

Our review of the process instrumentation for the CPCS is presented in the Safety Evaluation Report and Supplement No. 1. The review revealed that all of the analog sensor signal processing for the entire reactor protection system (RPS) is being processed and housed within the Process Protection Cabinet 2C15. This cabinet is 16 feet long and 10 feet high and is physically separated into four redundant channels. During the drawing review an associated circuit problem was identified within the 2C15 cabinet. The concern expressed by the staff was the close proximity of the Class IE and non-Class IE wiring, and the susceptibility of the Class IE circuits to noise or electromagnetic interference (EMI) from the non-Class IE circuits. This concern was formalized as safety position 5.

Also our review of the CEA position cable assembly is presented in the Safety Evaluation Report and Supplement No. 1. Due to the physical constraints in the reactor vessel head area, both Class IE and non-Class IE signals are transmitted within the same cable assembly from the reactor vessel head to a point outside containment. Within this cable assembly, six of the conductors are used for discrete position information (non-Class IE) which is transmitted to the control element drive mechanisms control system (CEDMCS) and three are inputs to the CPCS. For example, Channel "C" CPC has 61 CEAs, therefore, 366 conductors are non-Class IE and 183 are Class IE analog signals that are transmitted to the CPCS. It was noted that all of these conductors are contained in the same raceway, inside containment. Accordingly, the Class IE conductors are dominated by non-Class IE conductors and a concern for noise susceptibility exists. This concern has also been expressed in safety position 5.

D.4.1.4 <u>Control Element Assembly Calculator (CEAC)</u>
<u>Optical Isolators</u>
<u>Safety Position 12</u>

The staff has discussed with the applicant the effects of exposure to the optical isolators to radio frequencies (RF) greater than 100 Megahertz (MHz) upon the response of the CPCS. Our evaluation and concerns regarding this issue are discussed in the Safety Evaluation Report and Supplement No. 1 as safety position 12.

The applicant responded to the position by performing a noise susceptibility test of radio frequencies from 35 MHz through 2GHz on the CPCS. During this test several susceptible frequencies were encountered within this range. However, at each susceptibility point, the CPCS responded in a fail safe manner, i.e., a trip. At the frequency band of the radio transceivers (walkie-talkies) to be used in ANO-2 plant, extended tests were run at power levels up to 17 Volts per meter to verify that the CPCS was not susceptible.

In response to position 5 and position 12, the applicant has performed a noise immunity qualification susceptibility test on the single channel CPC system. This test determined the susceptibility of the system to EMI. A graph of susceptibility field strengths and corresponding frequencies were established as a baseline. We have reviewed the test procedures, reference 21, and test report, reference 22, and conclude that the noise immunity tests are acceptable subject to satisfactory completion of EMI measurements.

The applicant has committed to measure the actual levels and frequencies of EMI onsite to confirm that these measurements fall within the acceptable range of the baseline graph. The results of the onsite measurements will be submitted in the startup test report. It is the staff's opinion that the frequencies and levels of the radiated EMI that the single channel was exposed to during the noise susceptibility test were conservative with respect to the expected onsite measurements, thus justifying the initial operation of the CPCS. We will review the startup test report and address our resolution of this item in a supplement to this report.

Safety Position 26

Our initial review and evaluation of the optical isolators are presented in the Safety Evaluation Report (SER). Our concerns regarding the qualification of the isolators were expressed as position 26. The applicant has responded to the position by proposing a test program for the optical isolators. We received the qualification test procedures, reference 2-4, of the optical isolators and after review of the procedures concluded that they were acceptable. The staff also reviewed the test report, reference 2-5, and concluded that the test results did not satisfy the acceptance criteria stated in reference 2-4 and thus the test report was not acceptable. Refer to SER Supplement No. 1 for more detail on the unacceptability aspects of the test.

The applicant submitted a new test procedure, reference 2-6 and test report reference 2-7, which included the design changes that would combat the failures experienced during the first qualification test. Also an analysis was performed to prove the functional correctness of the modifications made to the CPC system's Digital Input (DI) and Digital Output (DO) data link cards. The staff reviewed the analysis and test procedures and concluded that they were acceptable.

The staff reviewed the test report, reference 2-7, and expressed the following concern to the modifications made to the Digital Input Card:

(1) A fusible resistor was used to prevent the high energy source from initiating a resistor fire upon a fault. The concern was that the fusible resistor would not open fast enough to prevent a fault from exploding the input diode of the isolator. Also the repeatability of the test data for the modified design to perform its function upon application of a fault was a concern.

(2) A diode was installed back to back with the input diode of the isolator to prevent damage from the high reverse voltage that would be seen with a 120 Volt alternating current (AC) fault. The concern here was the need for the diode modification and the potential requirement for periodic test.

The applicant submitted a new test procedure (reference 2-9) for evaluating the repeatability of test data on fault performance and to determine if periodic testing is necessary for the protection diode. This test procedure was reviewed and found acceptable.

On May 18, 1978 the staff met with Arkansas Power and Light Company and Combustion Engineering, Inc., in Windsor, Connecticut, to audit the optical isolator qualification test. The tests were conducted on the Single Channel CPC System in accordance with test procedures in Reference 24. The single channel was configured as a CEAC and a 120 Volt AC was applied across the signal lead and the +12 Volt return. No abnormal behavior was observed and the online diagnostic verified that bit seven did fail. Both the DI card and DO card were pulled from the chassis and the isolation impedance was verified to be greater than 20 Megohm. There was no evidence of damage on adjacent circuit boards. The cards were returned to the chassis and the system configured as a CPC, and a 120 Volt AC was applied across the signal lead and the +12 Volt DC supply. Again there was no evidence of damage to the board or the adjacent circuit boards.

An additional 16 tests were run to verify that the modified data link card could withstand the maximum credible fault. Eight tests were run with the back diode in the circuit, and then eight circuits were faulted with the diode removed. The test results demonstrate that the CPC data link isolation circuitry can successfully withstand the maximum credible fault without progagating the fault. It was also demonstrated that periodic testing of the reverse voltage diode circuit is not necessary. Additional detail of the staff's audit of the test may be found in reference M27.

The staff reviewed the test report, reference 2-8, and found it acceptable. The applicant has addressed all of the concerns in position 26 and the staff concludes that the optical isolators are acceptable to be used as isolation devices in the CPCs.

Safety Position 4

Our initial review of the CEA output data link cards are presented in the Safety Evaluation Report and Supplement No. 1. The staff concern on this issue was how the output of the optical isolator cards within the CEAC would meet the single failure criteria. The staff's evaluation and acceptance of the applicant's response to position 4 (discussed in our previous SERs) was contingent upon the successful resolution of safety position 26. As safety position 26 has been successfully resolved, safety position 4 is also satisfactorily resolved.

D.4.2 Test, Maintenance, Monitoring and Qualification

D.4.2.1 Operational Testing

In our Safety Evaluation Report Supplement No. 1 we stated that our concerns regarding the time interval for periodic testng would be resolved by establishing conservative test intervals and additional CPCS surveillance requirements in the plant technical specifications. Our requirements for periodic test intervals, surveillance requirements, allowed outages and actions to be taken with equipment out of service or in bypass for the CPCS are included in the Arkansas Nuclear One Unit No. 2 Plant Technical Specifications Section 3/4.3.1, Reactor Protective Instrumentation. Based on the periodic testing and surveillance requirements for the CPCS set forth in these technical specifications, we consider that our concerns regarding the intervals for and adequacy of the periodic testings of the CPCS are resolved.

D.4.2.3 Plant Computer System Monitoring

In our Safety Evaluation Report Supplement No. 1 we stated our position regarding the data links between the non-safety plant computer system and the CPCS as follows:

The data links would be allowed to be connected to all six CPCS computers during startup operations for a sufficient period of time to allow for collection of data prior to the end of the startup testing phase of operations. Similar operation would be allowed on subsequent startups after refueling. To evaluate this configuration, we requested that the applicant provide information describing (a) the specific uses and benefits of the PCS during this period which relied upon the data links; (b) the required duration of operation with the links connected; (c) the procedures for disconnecting the links at the end of this period; and (d) the test criteria and test methodology to be employed to ensure that the data links have been correctly implemented.

We also stated that, unless the applicant agreed to the data link operation as described and provide the information requested, we would require that the plant computer data links to the protection computers be removed and that the plant computer service program be deleted from automatic program scheduling in the CPCS.

By letter dated March 10, 1978, the applicant agreed to use and operate the data links in accordance with our position. That is all data links between the protection computers and the plant computers will be connected during start-up and power ascension testing. Within 10 days after completion of the power ascension tests the six data links will be removed.

The data links will be used during the start-up and power ascension tests to obtain the data to be analyzed and used to verify the following calculated CPCS data base constants:

- Shape annealing factor matrix constants - Boundary point power correlation constants
- CEA shadowing factors
- Temperature shadowing factors
- CEA deviation penalty factors

Upon completion of the data collection, the data link cables will be disconnected and stored in accordance with approved procedures for storage and handling of Class 1E electric equipment. After removal of the data link, CPCS operation will be verified by performing a periodic test of each CPC and CEAC.

We have reviewed the information provided by the applicant for using the data links between the CPCS and plant computer system during initial startup and power ascension tests and at each refueling. We have also reviewed the procedures for disconnecting the links at the end of this period. Based on the information provided, we consider that our concerns regarding the data links from the CPCS to the plant computer system are resolved.

D.4.2.5 Seismic Qualification

Our review of the seismic qualification of the CPCS is also reported in the previous Safety Evaluation Report. Our safety concerns regarding seismic qualification are stated as position 14 of Table D.1.

Our review of seismic qualification of seismic Category 1 instrumentation and electrical equipment is also presented in Section 3.10 of this report. In our review, we identified specific items of Class 1E equipment and associated unresolved issues pertaining to the seismic qualification of that equipment including the Core Protection Calculator System (CPCS) components in a letter to the applicant dated September 7, 1977. Since that date, the applicant has submitted his qualification procedures and test results for the identified equipment. We have reviewed the applicants' submittals and conclude that the seismic qualification of the CPCS components is acceptable for ANO-2 specific application and also that the applicant's seismic qualification programs for the other identified equipment is acceptable. We conclude that all aspects of safety position 14 are resolved.

D.4.2.6 Pre-Operational Test

In our Safety Evaluation Report, Supplement No. 1, we discussed our concerns regarding the lack of failure - response type tests. These concerns were expressed by a consultant to the staff who was engaged to review the test procedures. Also, assisted by the consultant, we conducted an audit of the preoperational test, wherein these concerns were restated. The minutes of our audit are presented in reference M25. The consultants' original concerns are presented in reference 1.

During the audit of the Pre-Operational Test, the applicant committed to revise his test procedures and conduct failure-response type test. The revised tests procedures have been reviewed by the staff and the test consultant and found satisfactory (see reference 2-2). The execution of the tests have been monitored by the Office of Inspection and Enforcement. The staff considers this matter resolved.

D.4.4 Software Qualification

D.4.4.4 Phase II Test Results

Process Noise Test Program

In our Safety Evaluation Report Supplement No. 1, the staff expressed a continuing concern about the effects of process noise on CPCS performance. The difficulties encountered in predicting the noise effects on Phase II test results and the sensitivity of power calculations to the dynamic components of the thermal power algorithm were cited as examples pertinent to staff position 12.

The staff has now completed review of the report, "Description of CPC Process Noise Evaluation Program and CPC Heat Flux Sensitivity to T_{HOT} Noise," January 1978, which was docketed with the applicant transmittal letter of March 10, 1978 (reference 2-15).

The noise evaluation program defined by the applicant in response to NRC position 12 includes testing of the CPC response to recorded signals from operational plants manufactured by Combustion Engineering. Also, a sensitivity study will quantify the CPC response to single-input and multi-input synthesized process noise for various operating conditions. Input signals for the sensitivity study will in part be based upon the results of a spectral analysis of process noise data recorded at the St. Lucie Unit 1 plant. These tests are to be performed on the single channel test system at Windsor. Noise analysis results reported to date are limited to the studies performed to obtain the predicted variance of Phase II test results with noise on the signal generator simulator. These results indicated that a noise environment tends to reduce the operating margin to a trip output relative to a noise free environment. The applicant has concluded that the CPCS operation will not be impaired by process noise and has committed to supply additional data to support this conclusion (reference 2-15).

The applicant has also addressed (in reference 2-16) the staff concern with the overall response of the CPC dynamic power compensation. Design characteristics of the CPCS are such that the system response to hot leg noise cannot be readily evaluated by analysis and must be investigated by means of simulation and testing. Large negative DNBR spikes observed during Phase II testing are attributed to the quantization of the test simulator digital-to-analog converters which provided 0.75 degrees Fahrenheit step change in hot leg temperature inputs to the CPCS. The applicant believes that similar CPC behavior will not occur in the plant noise environment. Noise test results are expected to support this position.

The applicant has committed to provide a final test report on the effects of noise on CPCS operation. This report is to present the results of the single channel testing in addition to the evaluation of noise effects during post-core hot functional testing and during power ascension testing. The staff will use this report as the basis for a final evaluation to resolve position 12. In addition, the staff will rely on this report to demonstrate the single channel testing techniques which will be proposed to evaluate possible effects of software changes on the CPCS noise response.

The test program for software changes must include noise response testing for software changes which may affect the system's response to noise. Also in this regard, the testing should be conducted with ANO-2 noise conditions imposed on all CPCS signal inputs. The staff requires that the applicant describe the noise test program for software changes and the capability of the signal channel test system to synthesize noise. The staff will review this information within the scope of position 19, and report on our evaluation in a supplement to this report.

The report transmitted with reference 2-15 is acceptable to the staff as reasonable assurance that a more complete test program will confirm the acceptability of the CPCS in the ANO-2 noise environment. The ANO-2 license will be conditioned on the submittal of a noise evaluation test report which is acceptable to the staff for resolution of position 12.

D.4.4.5 Integrated System Burn-In Test

Staff Position 18 required performance of an integrated system burn-in test as a condition to demonstrate the acceptability of the CPCS. The history of the applicant's response to this requirement and the staff evaluation of earlier test results is summarized in Supplement No. 1 to Appendix D of the Safety Evaluation Report, dated June 1978. At that time, the staff found the results of burn-in testing to be unacceptable.

Additional requirements to resolve position 18 were as follows:

(1) Revision to the test procedures of reference (2-10) as specified in Supplement No. 1 to Appendix D of the SER.

(2) Re-execution of the integrated system test for a minimum two-week specified test procedure.

(3) Submittal of an acceptable test report.

In response to the staff requirements, the applicant has submitted revised test procedures reference (2-11), performed additional integrated system testing at the plant site from 2/11/78 to 2/25/78, and submitted a test report, reference (2-12), summarizing results of the testing at ANO-2 and the testing at Systems Engineering Laboratories during July 27 to August 7, 1977.

The staff has reviewed the cited submittals and the test logs have been audited by a staff consultant. The test procedures, reference (2-11), incorporated the modifications which were described in our SER Supplement No. 1 as conditions for acceptability.

Four software coding errors were identified during the software burn-in period at System Engineering Laboratory. One of these errors involves a non-significant range-limit error for one of the addressable constants in the CPC. The other three errors were in the CEAC software and are indicative of a need for more attention to the scope of CEAC testing for Quality Assurance of software modifications. Necessary software modifications were performed and the affected modules were Phase I retested. The changes were implemented on the single channel system at Windsor and the affected disk tracks were regenerated. The modifications necessitated a repeat of the two week software burn-in test in accordance with the test procedure.

The test configuration at the plant site incorporated the test panel from the Combustion Engineering Signal Channel test facility to provide the input signals to the integrated four channel system. Repeated auto-restarts were experienced on CPC Channel A during testing in a simulated static power condition. The problem was traced to the Interdata Universal Clock Module (UCM) which experienced intermittent loss of interrupts and was replaced on February 20, 1978. Since the failure did not necessitate a design change, test procedures did not require a restart of the test. CPC channels B, C and D and one of the CEA channels operated continuously for the two week period with no auto-restarts or other anomalies.

The staff has noted that the test history of the CPCS has resulted in a number of clock problems. The periodic test program is designed to detect timing errors of the type previously encountered. This periodic test did not detect the most recent failure which was attributed to an intermittent loss of interrupts. The periodic test intervals required by technical specifications has been chosen with due regard to test failures, including the previous clock failures. In addition, technical specifications require demonstration of calculator operability when three or more auto restarts are experienced in a 12-hour interval. The staff believes that these provisions of the technical specifications provide adequate safeguards against the existence of undetected clock failures on more than one CPCS calculator at any given time.

The audit of test data and test logs did not reveal any records which were inconsistent with the results and conclusions of the integrated system burn-in report, references (2-12) and (2-14).

The staff finds the software burn-in test report and test results acceptable. We conclude that all aspects of position 18 are resolved.

D.4.4.6 Qualification of the Single Channel Test System for Testing of Future Software Modifications

Qualification of Software Change Procedures

Requirements for qualification of software change procedures are described in staff position 19. A status summary of this position as described in the Safety Evaluation Report Supplement No. 1 is as follows:

(1) An acceptable test program was required for demonstration of the following single channel test system capabilities:

(a) testing of interfaces between the CEAC, CPC, and operator's module

(b) execution of either option for high power selection, and

(c) testing of multi-variable transients.

(2) Additional analyses, supported by test data, were required to complete our evaluation of the single channel test system as an acceptable test system for final qualification testing of modified CPC software.

(3) A license condition was to prohibit changes to the qualified ANO-2 software until a change procedure has been fully qualified and technical specifications were to address the software change restrictions, including documentation and submittal of information on all changes to the staff.

(4) An acceptable test program for final testing of modified software was required to assure that software changes do not result in errors or unexpected effects on the functional performances of the CPCS.

The applicant in a meeting with the staff on March 29, 1978 and in reference (2-16) has addressed the concern of item (1) above. The single channel test system is to be modified to enable multi-variable transients to be performed. This will be accomplished by a Dynamic Software Verification Test (DSVT) in which portions of the CPC executive and unused core are overlayed in order to process predefined time variant CPC inputs to the CPC protection algorithms and data base. Since the inputs can be synchronized with time in the same manner as the correspondng CPC FORTRAN test cases, test results can be compared to the FORTRAN test results without regard to the uncertainties associated with live inputs. Differences in results are then clearly due to software error or machine processing differences. Selected dynamic test cases for qualification of the DSVT program are defined in reference (2-16) and the applicant has committed to submit the test results for staff review. The applicant proposes to use the DSVT program in lieu of Phase II type multi-variable live input cases for testing of future software modifications. Live input tests in the single variable mode would be retained.

The applicant has committed to re-execute all dynamic test cases described in reference (2-18) with the high power select option of the software in a normal state. The staff will review the results of these tests and report on our evaluation in a supplement to this report.

The single channel system is being modified to include a separate CEAC calculator (and data link) to the CPC. The resulting CPC/CEAC/Operator's Module configuration will permit testing of all interfaces. Test case 21 (CEA drop) of reference (2-18) will be re-executed using the CEAC to generate and transmit the resulting penalty factors. The operator's module interface with the CPC and CEAC are also to be exercised.

The applicant has committed to submit results of the described single channel qualification tests in a supplement to reference (2-18). We will report on our evaluation of test results in a supplement to this report.

The applicant has restated his intent to perform changes to the software design and non-addressable constants in accordance with 10 CFR Part 50.59 (including reporting requirements), i.e., the licensee would be responsible to determine if the software change is an unreviewed safety question which must be submitted to the staff. This is in conflict with the original staff Position 19E and item (4) of the above status summary.

The applicant in his letter of May 17 (reference 2-17) addressed the staff requirement (item 2) for additional testing to demonstrate the dynamic response of the single channel system. The applicant committed to use the DSVT program executed on the single channel test system and on the Louisiana Power and Light CPCS to demonstrate that the dynamic response of the single channel system is identical to that of the CPCS. The Louisiana Power and Light CPCS hardware configured at Systems Engineering Laboratories in Ft. Lauderdale is a duplicate of the ANO-2 hardware and was chosen to minimize impact on the ANO-2 startup schedule. The execution of these tests were audited by a staff consultant and his audit report, reference 2-13, confirmed execution in accordance with test procedures and anticipated results. The staff will present an evaluation of the test report in a supplement to this report.

<u>Summary</u>

The staff has reviewed references (2-16) and (2-17) and finds the proposed test commitments and modifications to the single channel test system to be an acceptable basis for resolution of the staff concerns expressed in items (1) and (2) of the preceding status summary.

The staff has slightly modified the position [item (3)] of our previous Safety Evaluation Report with respect to position 19E. We will not require that all software changes be submitted to the staff and will permit the licensee to make a determination of the safety significance.

However, all aspects of the software program which affect the margin to trip cannot be modified without prior Nuclear Regulatory Commission approval. In addition, we will require that a software consultant who is fully qualified to evaluate the safety significance of proposed software changes be included on the plant safety committee. This requirement will be reflected in a change to the current technical specifications.

The applicant has committed to provide a revision to reference 2-19 to reflect a test program consistent with the upgraded single channel test capability in reference 2-16. This test program is intended to satisfy item (4) of the preceding status summary. Position 19 remains outstanding pending the submittal to and approval by the staff of the following:

(1) Revision of the Software Change Procedure (reference 2-19).

(2) Supplement to the Single Channel Qualification Test Report (reference 2-18) (including results of the DSVT tests).

(3) Designation of a qualified software consultant on the plant safety committee.

(4) A description of a noise test program to be used in the qualification of software changes. Also, a description of the noise synthesis capabilities of the single channel test facility.

With the exception of item 2 above, the applicant has committed to provide the information requested and resolve the concerns prior to mode 2 operation. Because of equipment acquisition and installation delays to the single channel test facility, the test report required in item 2 will be delayed beyond the start of mode 2 operation. The applicant has proposed that the concerns related to item 2 be resolved prior to mode 1 operation rather than mode 2 operation. We have evaluated this proposal and find it acceptable provided that no modifications to the Core Protection Calculator System are defined and/or required during mode 2 operation. Should a modification be required during mode 2 operation, we require that all facets of position 19 be acceptably resolved prior to the execution of the modification.

D.4.4.7 <u>Startup Tests</u>

In our Safety Evaluation Report Supplement No. 1, we presented the status of our review of the startup tests. The review consisted of evaluating the startup test requirements and of evaluating detailed test procedures. Our review of the detailed test procedures has been completed and is presented herein.

We have reviewed the ANO-2 test procedures entitled; Induced Xenon Oscillation, Temperature Decalibration Verification, CEA Shadowing Factor Verification, Radial Peaking Factor Verification, and Shape Annealing Matrix and Boundary Condition measurements. We believe the test methods, prerequisites, and acceptance criteria will provide sufficient experimental data to evaluate the adequacy of the CPC synethisized power distributions.

Information submitted in references 2-3, 2-4, and 2-5 has been previously reviewed and supports the finding that the protection algorithms will conservatively predict the power distribution parameters used in the CPC calculation of margins to the trip set points. The adequacy of these margins will be verified as the power ascension program progresses.

The staff intends to audit the execution of the Startup Tests and we will report on the audit and of our evaluation of the Startup Test Report with respect to Positions 1, 5 and 12 in a supplement to this report.

TABLE D.1

Core Protection Calculator System Positions

A listing of the staff's positions that were outstanding as of the last supplement (Supplement No. 1) to the Safety Evaluation Report is presented below. Each position number and title is followed by a section number of this report in which further detail on the position is presented. The current status of the position is also stated. We will report our further evaluation of the outstanding issues in a supplement to this report.

(1) Uncertainty Associated with the Algorithms, Section D.3.5, Outstanding

We believe that it is necessary to experimentally qualify the adequacy of these uncertainties, specifically those associated with the synthesis of axial power distribution. We will require that confirmatory measurements be performed durng startup to demonstrate the adequacy of the axial power synthesis by comparing to in-core measurements and analysis for various power conditions.

(4) CEAC Separation Criteria at the Output of the Optical Isolator Cards, Section D.4.1.4, Closed

We will require that the applicant identify their design basis events for the control element assembly calculatory (CEAC) and verify that no credible single event either internal or external to the CEAC will result in loss of function.

(5) Cable Separation, Section D.4.1.2, Outstanding

The applicant identified an area where safety-related control rod drive position sensor cables are run together with nonsafety cable. The applicant will reevaluate this design and advise the staff as to its resolution.

(12) Electrical Noise and Isolation Qualification, Sections D.4.1.4 and D.4.4.4, Outstanding

Tests for electrical isolation separation and noise susceptibility will be required. The applicant shall develop and submit for approval test plants and detailed procedures for these tests prior to their undertaking. In addition, due to the CPCS design and packaging, these tests should be performed on the fully configured integrated system or an acceptable analysis clearly establishing the adequacy of component testing is required for staff evaluation. The staff's July 7, 1976 letter provides supplemental details on this concern.

(14) Seismic Qualifications, Section D.4.2.4, Closed

The staff has found the seismic qualification test plan not acceptable. Current criteria for multi-frequency input and sine beat tests for seismic qualification have been provided to the applicant. Submittal date for a satisfactory seismic qualification plan and review completion date have yet to be determined.

(15) Addressable Constants, Section D.3.11, Closed

Any changes in addressable constants must be provided with adequate safeguards to protect against unreasonable entries. The proposed safeguards against unreasonable entries are basically administrative and are subject to human error. To enhance safety by minimizing human error and to utilize capabilities of the computer to audit the input, the staff requires that the computer program be modified to conduct reasonability tests and to reject unreasonable values of addressable constants as they are entered from the Operator's Module. The operator is to be notified upon failure of the reasonability test. Qualification testing of the modification must also be conducted.

(18) Burn-In Test, Section D.4.4.5, Closed

We find the proposed duration of the burn-in test (three to six months) acceptable subject to our review of test ground rules and acceptance criteria which must be submitted in the form of the test plan before the test commences. We will require that the software on the system during the test incorporate all design changes which have been identified by the applicant and the staff prior to a new freeze on the design. The staff will require testing of the total system after installation of the CPCS and associated process instrumentation in the plant protection system cabinet number 2C15. Failure to incorporate this equipment for the burn-in test will necessitate a more extensive field test program for the entire system.

The staff has reviewed the applicant's supplemental response to position 18, which deals with the burn-in test. Based on the new information presented and the additional testing proposed, the execution of the burn-in test with the frozen software is acceptable, subject to the conditions stated herein.

Conditions for Hardware Burn-In Test

(a) A staff review of the test procedures to be used in the hardware burn-in test is in progress. These procedures must be consistent with industrial practice for computer system testing and acceptable to the staff.

(b) Additional tests to demonstrate and evaluate the integrity of software and the integrated system are needed. The staff requires a minimum test period of two weeks, with the system operating continuously on live input signals in addition to satisfactory performance of static and dynamic test cases to demonstrate the integrity of the integrated system. This test must be conducted with the same configuration and the same environment as that used for the hardware burn-in test conducted with the frozen software. This is required to assure that problems encountered after installation of the system in a new environment (the ANO-2 site) do not interfere with evaluation of the final software.

(19) <u>Qualification of Software Change Procedures, Section D.4.4.6, Outstanding</u>

Following are the primary requirements for qualification of software change procedures:

(a) All changes are to be performed strictly in accordance with the documented quality assurance procedures which are to be available for review by the staff. The documentation must accurately reflect the status of the altered program.

(b) The FORTRAN version of the modified program is to be subjected to a complete static and dynamic test program to demonstrate conservatism with respect to trip requirements defined by the ANO-2 accident analysis.

(c) The assembly language version of the qualified FORTRAN is to be subjected to a static and dynamic test program on an acceptable test system. The test program is to include sufficient reactor simulated transient test cases, static test cases and single parameter transient test cases to demonstrate that the program response corresponds to its FORTRAN version. The test program is also to include testing of the man-machine interface.

(d) The software is to be transferred to the plant system in accordance with the applicant's proposed pocedures prior to the burn-in test. All four channels will again be subjected to static and dynamic test cases to demonstrate that the response is identical to that observed on the test bed system. This step is to demonstrate the adequacy of the quality assurance procedures for transfer from the test bed to the plant system.

(e) Step d need not be repeated for future software revisions. All software design changes and revisions to constants in memory (except addressable constant) are subject to documentation, review and approval by the Regulatory staff.

(20) Data Link to Plant Computer, Section D.4.2.3, Closed

The core protection calculator system is designed with a data link and a special program module in each protection computer to service the plant computer. These data links and programs are an addition to the traditional plant computer interconnects in analog, hard-wired protection systems which are also included in the ANO-2 reactor protection system. It is our position that these data links and the plant computer service program do not satisfy the requirements of General Design Criterion 24, "Separation of Protection and Control Systems," and IEEE Standard 279-1971, Section 4.7, "Control and Protection System Interaction," regarding independence of protection systems. Therefore, we will require that the plant computer service data links to the protection computers be removed and that the plant computer service routine be deleted from automatic program scheduling.

(26) Optical Isolator, Section D.4.1.4, Closed

It is the staff's position that as the optical isolator is to be utilized as an electrical isolation device, the applicant must demonstrate that any single credible fault (125 volts alternating current or 125 volts direct current) applied to the device output will not degrade the operation of the circuit connected to the device input. Also, the application of the same credible fault must be applied to the input of the device with no degradation of the circuit connected to the device output. (See Figure 7A.4-23 of the Final Safety Analysis Report.)

TABLE D.2

CPCS REFERENCES AND MEETING MINUTES

REFERENCES

2-1 Letter, January 30, 1978, from K. L. Gimmy, E. I. DuPont De Nemours and Company, to L. Beltracchi, NRC.

2-2 Letter, March 29, 1978, from K. L. Gimmy, E. I. DuPont De Nemours and Company, to L. Beltracchi, NRC.

2-3 CEN-44 (A)-P, "Core Protection Calculator Functional Description," January 7, 1977, ANO-2 Unit One. Supplement - 1(P), May 16, 1977, Supplement - 2(P), May 19, 1977, Supplement - 3(P), September 2, 1977.

2-4 CEN-45 (A)-P, "Core Element Assembly Calculator Functional Description," January 7, 1977, ANO-2 Unit One.

2-5 CEN-53 (A)-P, "CPC and CEAC Data Base Document," May 20, 1977, ANO-2, Unit One. Supplement - 1(P), June 28, 1977, Supplement - 2(P), September 2, 1977.

2-6 CEN-70(A), Revision 1, "Test Procedure for the Core Protection Calculator Data Link Isolation Circuits," January 18, 1978.

2-7 CEN-84(A), "Test Report for the Core Protection Calculator Data Link Isolation Circuits," February 1978.

2-8 CEN-84(A), Supplement 1, "Test Report of the Core Protection Calculator Data Link Reverse Voltage Diode and Repeatability Determination," May 24, 1978.

2-9 CEN-92(A), "Test Procedure for the Core Protection Calculator Data Link Reverse Voltage Diode and Reliability Determination," May 4, 1978.

2-10 CEN-60(A), Supplement 1, "Core Protection Calculator Integrated System Burn-In Test Procedure," issued November 18, 1977.

2-11 CEN-60(A), Revision 03, "Core Protection Calculator Integrated System Burn-In Test Procedure," issued February 6, 1978.

2-12 CEN-86(A)-P, "Core Protection Calculator Integrated System Burn-In Test Report," March 6, 1978.

2-13 Letter, "Audit of CPC Dynamic Software Verification Field Test Procedure," from J. B. Bullock, Oak Ridge National Laboratory, to Leo Beltracchi, NRC, dated July 6, 1978.

2-14 Attachment to Letter from L. C. Oakes, Oak Ridge National Laboratory, to R. Mattson, Director, DSS, "Progress Report for April and May 1978, Letter dated June 23, 1978.

2-15 Letter from Daniel H. Williams, AP&L Co., to Director of Nuclear Reactor Regulation, 2-038-10, Dated March 10, 1978.

2-16 Letter from Daniel H. Williams, AP&L Co., to Director of Nuclear Reactor Regulation, 2-058-11, Dated May 11, 1978.

2-17 Letter from Daniel H. Williams, AP&L Co., to Director of Nuclear Reactor Regulation, 2-058-12, dated May 17, 1978.

2-18 CEN-71(A)-P, "Core Protection Calculator, Single Channel Qualification Test Report," October 19, 1977.

2-19 CEN-39(A)-P, "CPC Protection Algorithm Software Change Procedure," October 1, 1976.

2-20 CEN-63(A) Supplement 1 (NP) "Core Protection Calculator System Supplemental Shape Annealing Matrix Information," March 1, 1978.

MINUTES OF MEETINGS

The following meeting minutes reflect meetings and audits conducted during the period of January 1978 to early July 1978, with previous activity reported in the Safety Evaluation Report.

M25 "Core Protection Calculator System, Site Audit of Pre-Operational Test, January 25, 1978," to T. A. Inppolito, Feb. 27, 1978.

M26 "March 29, 1978 Meeting Minutes Core Protection Calculator System," to T. A. Ippolito, May 9, 1978.

M27 "Trip Report - Core Protection Calculator System (CPCS) Audit of Optical Isolator Qualification Test - May 18, 1978," to C. Miller, July 12, 1978.

M28 "Summary of Meeting on Technical Specifications," Docket No. 50-368, June 16, 1978.

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16. ABSTRACT (200 words or less)

This report, Supplement No. 2 to the Safety Evaluation Report, provides (1) our evaluation of additional information received from the applicant since preparation of Supplement No. 1 regarding previously identified outstanding review items, (2) our responses to comments made by the Advisory Committee on Reactor Safeguards in its report dated April 12, 1978, and (3) our evaluation of additional or revised information related to new issues that have arisen since preparation of Supplement No. 1.

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