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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

November 1977

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NUREG-0308
November 11, 1977

SAFETY EVALUATION REPORT

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

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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Arkansas Power and Light Company (hereinafter referred to as the applicant) filed with the Atomic Energy Commission, now the Nuclear Regulatory Commission (NRC or Commission), an application dated September 10, 1970 and as subsequently amended, for a license to construct and operate a pressurized water reactor, identified as the Arkansas Nuclear One - Unit 2 plant (hereinafter referred to as ANO-2). The plant is being constructed under Construction Permit CPPR-89 issued on December 6, 1972. On March 1, 1974, the applicant filed as Amendment No. 20 the Final Safety Analysis Report required by 10 CFR 50.34(b) in support of its application for an operating license for Arkansas Nuclear One - Unit 2. The site is located on a peninsula in the Dardanelle Reservoir on the Arkansas River in Pope County, Arkansas. ANO-2 is the second nuclear unit proposed to be operated at the Arkansas Nuclear One plant. A license for operation of Unit 1 was issued on May 21, 1974 (Docket Number 50-313).

The application is for a core power level of 2815 megawatts thermal which with the coolant pump heat of ten megawatts thermal corresponds to a nuclear steam supply system output of 2825 megawatts thermal and is equivalent to a net electrical output of approximately 912 electrical megawatts.

The radiological safety review with respect to a decision concerning issuance of an operating license for ANO-2 has been based on the applicant's Final Safety Analysis Report (Amendment 20) and subsequent Amendments 21 through 43, all of which are available for review at the Nuclear Regulatory Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. and at the Arkansas Polytechnic College, Russellville, Arkansas 72801.

In the course of the safety review of the material submitted, we held a number of meetings with representatives of the applicant, Combustion Engineering, Inc. and Bechtel Corporation, to discuss the plant design, construction, proposed operation and performance under operating, transient and postulated accident conditions. During our review, we requested the applicant to provide additional information that we needed for our evaluation. This additional information was provided in amendments to the application. As a result of our review, a number of changes were made in the facility design and proposed operating practices; these changes are described in the applicant's amendments and are discussed in appropriate sections of this report. A chronology of the principal actions relating to the processing of the application is attached as Appendix A to this Safety Evaluation Report.

The Safety Evaluation Report summarizes the results of the radiological safety review of ANO-2 performed by the Commission's staff. The review and evaluation of the facility for an operating license is only one stage in the continuing review by the staff of the design, construction and operating features of ANO-2. The proposed design of the facility was reviewed before a construction permit was issued. Construction of the facility was monitored in accordance with the inspection program of the Commission's staff. At this, the operating license application phase, we have reviewed the final design to determine that all of the Commission's safety requirements have been met. If an operating license is granted, the facility will be operated only in accordance with the terms of the operating license and the Commission's regulations and subject to the continuing inspection program of the staff.

In addition to our review, the Advisory Committee on Reactor Safeguards is reviewing the application and will meet with both the applicant and the staff to discuss the facility. The Advisory Committee on Reactor Safeguards Report to the Commission on the ANO-2 facility will be provided in a supplement to this Safety Evaluation Report.

The conclusions reached as a result of our evaluation of the applicant's application to operate Unit 2 are presented in Section 22.0 of this Safety Evaluation Report.

The environmental impact from proposed operation, considered in the review of the facility in accordance with 10 CFR Part 51, is discussed in the staff's Final Environmental Statement dated June 1977.

1.2 General Plant Description

The ANO-2 nuclear steam supply system uses a pressurized water reactor in a two-loop reactor coolant system. The reactor will be fueled with uranium dioxide pellets enclosed in Zircaloy fuel tubes with welded end plugs. The fuel rods are grouped and supported in assemblies. The reactor core will initially contain three regions of slightly different enrichments of uranium-235. Light water will serve as both the core moderator and coolant. The reactor coolant system consists of two separate loops, each having a steam generator and two pumps.

An electrically-heated pressurizer will establish and maintain reactor coolant pressure and provide a surge chamber and a water reserve to accommodate reactor coolant volume changes during operation. Heat generated by the reactor, which is rated at 2815 thermal megawatts, will be transported by the reactor coolant to the steam generators where it will be transferred to the secondary (steam) system. The steam thereby produced will be transported to the turbine generator where about one-third of the energy will be converted to electrical energy - approximately 912 net electrical megawatts. The remaining heat energy will be transferred in the steam condenser to the circulating cooling water system which utilizes a cooling tower installation for heat dissipation.

The nuclear steam supply system is housed in a steel-lined, prestressed, reinforced concrete containment structure which consists of a right circular cylinder, with a sphere-torus dome and a flat circular base slab. The containment is designed to safely confine radioactive material that could be released in the event of an accident. The auxiliary building houses the fuel handling system, the radioactive waste treatment facilities, engineered safety feature components, various related auxiliary systems for the reactor, heating and ventilating system components, switchgear, diesel generators and the control room. Another major structure is the turbine building.

The reactor will be controlled by control element movement and regulation of the boric acid concentration in the reactor coolant. The control element assemblies will be moved vertically within the core by control element drives. A reactor protection system will automatically initiate appropriate action whenever a plant condition monitored by the system approaches preestablished limits. This reactor protection system will act to shut down the reactor. Appropriate instrumentation circuitry is provided to initiate closure of isolation valves, and to initiate operation of the engineered safety feature systems should any or all of these actions be required.

Redundant and independent emergency cooling systems are provided to maintain reactor cooling and to provide containment cooling in the unlikely event of an accident. Engineered safety features for this plant include an emergency core cooling system which consists of a core flooding system and both high and low pressure injection systems with provisions for recirculating the borated water after the injection phase. Combinations of these systems will assure core cooling for the complete range of postulated reactor coolant pipe breaks. Other engineered safety features include the containment, containment isolation valves, reactor building spray system, reactor building air cooling system, and combustible gas control systems.

Various secondary systems and components are shared with the Arkansas Nuclear One -Unit 1 plant. These include the electrical switchyard, yard firefighting, fuel handling auxiliary building crane, portions of the solid waste treatment system, station security system, and other auxiliary systems. Shared systems are discussed further in Section 9.1 of this report. A site layout is shown in Figure 1.1.

The plant is capable of being supplied with electrical power from offsite sources via three independent 500 kilovolt and two 161 kilovolt transmission lines and is provided with an independent and redundant onsite emergency power supply capable of supplying power to the engineered safety feature systems.

1.3 Comparison with Similar Facility Designs

Many features of the design of ANO-2 are similar to those we have evaluated and approved previously for other nuclear power plants now under construction or in operation, especially Calvert Cliffs 1 and 2 and St. Lucie 1 and 2. To the extent feasible and appropriate, we have made use of these previous evaluations in

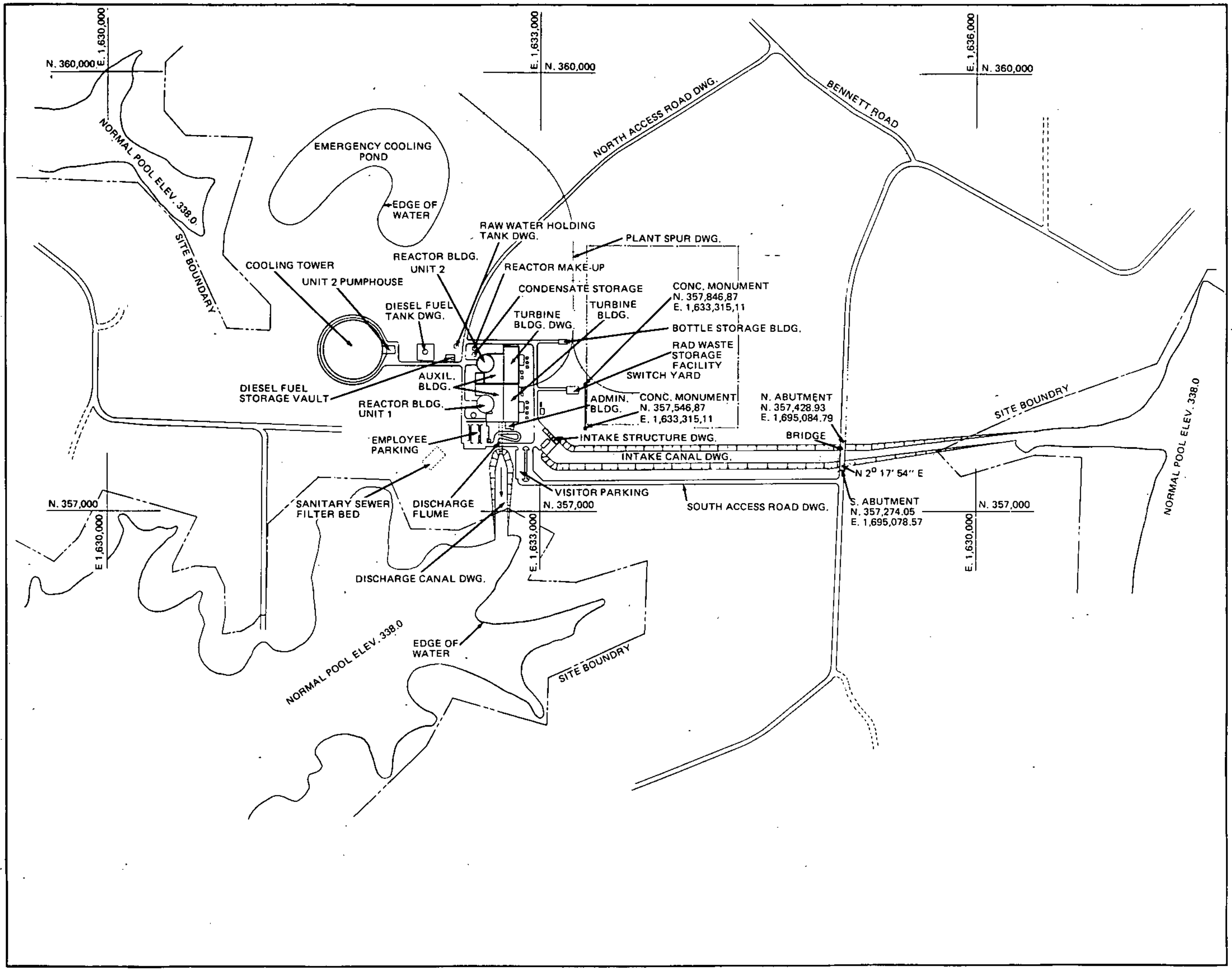

Figure 1.1
Site Layout

conducting our review of ANO-2. Where this has been done, the appropriate sections of this report identify the other facilities involved. Our Safety Evaluation Reports for these other facilities also have been published and are available for public inspection at the Nuclear Regulatory Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C.

1.4 Identification of Agents and Contractors

The Arkansas Power and Light Company is the applicant for the operating license for Arkansas Nuclear One - Unit 2 and is responsible for the design, construction and operation of the plant. The applicant engaged Combustion Engineering to design and manufacture the nuclear steam supply system for a core power of 2815 megawatts thermal and the nuclear fuel for the first core. Bechtel Power Corporation was engaged as the major contractor to provide engineering, management of construction and procurement services. The General Electric Company is the supplier of the turbine generator and its auxiliaries.

1.5 Summary of Principal Review Matters

Our evaluation included a technical review of the information submitted by the applicant particularly with regard to the following principal matters:

(1) We evaluated the population density and land use characteristics of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology and hydrology to establish that these characteristics have been determined adequately and have been given appropriate consideration in the final design of the plant, and that the site characteristics are in accordance with the Commission's siting criteria (10 CFR Part 100), taking into consideration the design of the facility including the engineered safety features provided.

(2) We evaluated the design, fabrication, construction and testing and performance characteristics of the plant structures, systems, and components important to safety to determine that they are in accord with the Commission's General Design Criteria, Quality Assurance Criteria, Regulatory Guides and other appropriate rules, codes and standards, and that any departures from these criteria, codes and standards have been identified and justified.

(3) We evaluated the expected response of the facility to various anticipated operating transients and to a broad spectrum of accidents, and determined that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered. We performed conservative analyses of these design basis accidents to determine that the calculated potential offsite radiation doses that might result in the very unlikely event of their occurrence would not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100.

(4) We evaluated the applicant's engineering and construction organizations, plans for the conduct of plant operation, including the proposed organization, staffing and training program, the plans for industrial security, and the plans for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that the applicant is technically qualified to safely operate the plant.

(5) We evaluated the design of the systems provided for control of the radiological effluents from the plant to determine that these systems are capable of controlling the release of radioactive wastes from the facility within the limits of the Commission's regulations, and that the equipment provided is capable of being operated by the applicant in such a manner as to reduce radioactive releases to as low as reasonably achievable levels in accordance with 10 CFR Part 20 and 10 CFR Part 50.34.

(6) Our evaluation of the applicant's financial qualifications to operate the ANO-2 plant will be reported in a supplement to this Safety Evaluation Report.

1.6 Summary of Outstanding Review Items

At this time, we have not completed our review of a number of items, either because additional information is required from the applicant or because we have not yet completed our review of information submitted by the applicant. These items will be addressed in a supplement to this Safety Evaluation Report. The current status of each item, and the section of this report in which the items are discussed, are identified below.

(1) Low population zone radius (Section 2.1).

(2) Seismic qualification of safety-related instrumentation (Section 3.10).

(3) Environmental qualification of safety-related instrumentation (Section 3.11).

(4) Short-term measures to protect against reactor vessel overpressurization events (Section 5.7).

(5) Main steam line break accident calculated mass and energy releases (Section 6.2).

(6) Containment leakage testing program (Section 6.2.6).

(7) Environmental qualifications of safety-related instrumentation for the main steam line break accident inside containment (Section 6.2).

(8) Evaluation of emergency core cooling system performance considering large and small break analysis, effects of boron precipitation and single failure criteria (Sections 6.3.3 and 15.0).

(9) Preoperational tests to demonstrate the capability of the emergency core cooling system to operate in the recirculation mode (Section 6.3.4).

(10) Review of the safety-related electrical logic and schematic diagrams and the verification of the implementation of the design (Section 7.1).

(11) Evaluation of input fault and surge testing of power supplies (Sections 7.2.2, 7.3.3 and 7.3.6).

(12) Evaluation of adequacy of applicant's list of parameters deemed essential for accident and post-accident monitoring (Section 7.5.1).

(13) Evaluation of redundant valve position indication to meet the single failure criterion and the qualification requirements (Section 7.6.3).

(14) Evaluation of reactor coolant pump coastdown due to grid underfrequency event (Section 7.6.4).

(15) Review of separation criteria for conduits (Section 7.9.4).

(16) The twenty-two outstanding core protection calculator system staff positions, as discussed in Section 7.2.3 and listed in Table 7.1 of this report, are listed in accordance with the position's number which has been used to identify the position in previous documentation. Indicated section numbers refer to the applicable section of Appendix D to this report.

    (1) Uncertainty associated with the CPCS algorithms (Section 3.5).

    (4) CEAC separation criteria (Section 4.1.4).

    (5) Control rod position sensor cable separation from nonsafety-related cable (Section 4.1.2).

    (8) Periodic testing time interval (Section 4.2.1).

    (9) Periodic functional testing of trip functions from sensor inputs to trip outputs (Section 4.2.1).

(10) Verification of validity of calculated results after changes to addressable constants (Section 4.2.1).

(11) Environmental performance qualification test of the integrated computer system (Section 4.2.4).

(12) Electrical, isolation, separation and noise susceptability qualification tests (Section 4.2.4).

(13) Reactor coolant pump speed and control element assembly position sensor qualification (Section 4.2.4).

(14) Seismic qualifications. This is redundant to item 3.10 of this outstanding item list (Section 4.2.5).

(15) Limit magnitude of change allowed for addressable constants (Section 3.11).

(16) Quality assurance plan for development of digital computer protection system (Section 4.3).

(18) Integrated system burn in qualification test (Section 4.1.4).

(19) Qualification of software change procedures (Section 4.4).

(20) Data links to plant computer system (Section 4.2.3).

(21) Use of consistent values for checksums in all protection channels (Section 4.2.1).

(22) Requirement for CEAC penalty factor transmission to CPC to have an input/output error response routine for timeout (Section 4.3).

(23) Requirement for automatic trip of channel upon timeout of the watchdog timer (Section 4.3).

(24) Phase II Test and Test Report (Section 4.4).

(25) Maintainability of the core protection calculator system (Section 4.2.1).

(26) Qualification of optical isolator device (Section 4.1.4).

(27) Periodic tests to verify the isolation characteristics of isolation devices (Section 4.2.1).

(17) Initial startup testing program (Section 14.0).

(18) Evaluation of reactor coolant pump seizure analysis using the CESEC code (Section 15.3.1).

(19) Review of main steam line break analysis (Section 15.3.3).

(20) Financial qualifications (Section 20.0).

1.7 Generic Issues

The following issues are generic in nature and are being pursued primarily with the vendor in question, and through the applicant where appropriate. We will require that any design or procedural change resulting from these generic reviews be incorporated in Arkansas Nuclear One, Unit 2 as appropriate.

(1) Evaluation of reactor vessel supports under loss-of-coolant loadings (Section 3.9.3).

(2) Long-term measures to protect against reactor vessel overpressurization events (Section 5.4.1).

(3) Reactor coolant pump flywheel integrity (Section 5.6.1).

(4) Offsite grid stability (Section 8.2).

(5) Anticipated transients without scram (Section 15.5).

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

The Arkansas Nuclear One - Unit 2 is located adjacent to Arkansas Nuclear One - Unit 1 on a 1,100 acre tract of land on a peninsula in the Dardanelle Reservoir on the Arkansas River in Clark Township, Pope County, Arkansas. The plant is about six miles west-northwest of Russellville, Arkansas, and about two miles southeast of the village of London, Arkansas. Figures 2.1 and 2.2 show the Arkansas Nuclear One-Unit 2 site with respect to the surrounding centers of population.

The applicant has designated an exclusion area which is the area enclosed within a circle of 1046 meters (3432 feet) from the centerline of the reactor as shown in Figure 2.3. The exclusion area includes certain portions of the bed and banks of the Dardenelle Reservoir which are owned by the United States Government. The applicant has negotiated an easement with the U.S. Army Corps of Engineers which will provide the applicant the authority to determine all activities within those portions owned by the United States Government including authority for the exclusion and removal of persons and property.

The applicant has adequately demonstrated by its legal counsel sufficient control provided by ownership over the surface and mineral rights associated with the exclusion area. Therefore, we conclude that the present degree of control demonstrated by the applicant meets the requirements of 10 CFR Part 100. A question of a mineral deed conveyance by a previous grantor has been determined by applicant's counsel to be invalid due to untimely recordation. In the event that there is any attempt by others to exploit the subject mineral rights, applicant will be required to take any steps necessary to obtain or retain legal rights to determine all activities within the exclusion area.

There are 66 residents within one mile of the site and approximately 600 within a two mile radius. Figure 2.4 shows the present and projected cumulative population surrounding the Arkansas Nuclear One - Unit 2 site. The 1970 population within 50 miles was 164,688. The applicant projects that the population in this area will increase by 30 percent in 40 years. We have determined that this projection is in substantial agreement with the population projections of the Bureau of Economic Analysis (BEA) economic areas numbers 116, 117 and 118 as shown in Figure 2.5.

The nearest community with a 1970 census population of 25,000 or more is stated by the applicant to be Hot Springs, Arkansas (1970 population of about 40,000), located

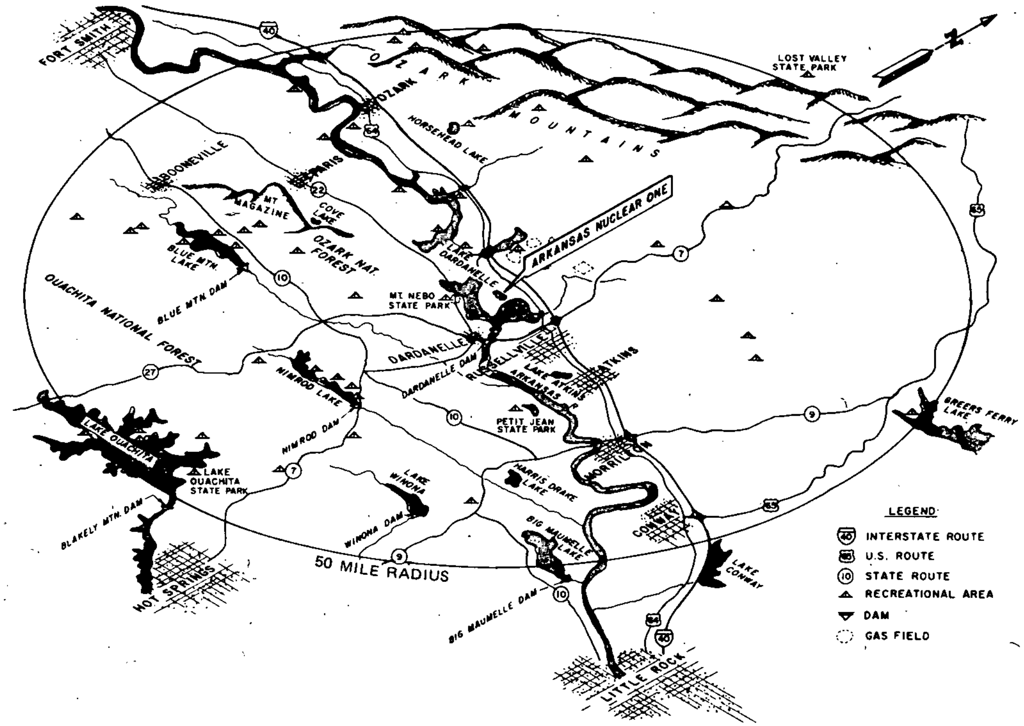

Figure 2.1 General Area Within Fifty Miles

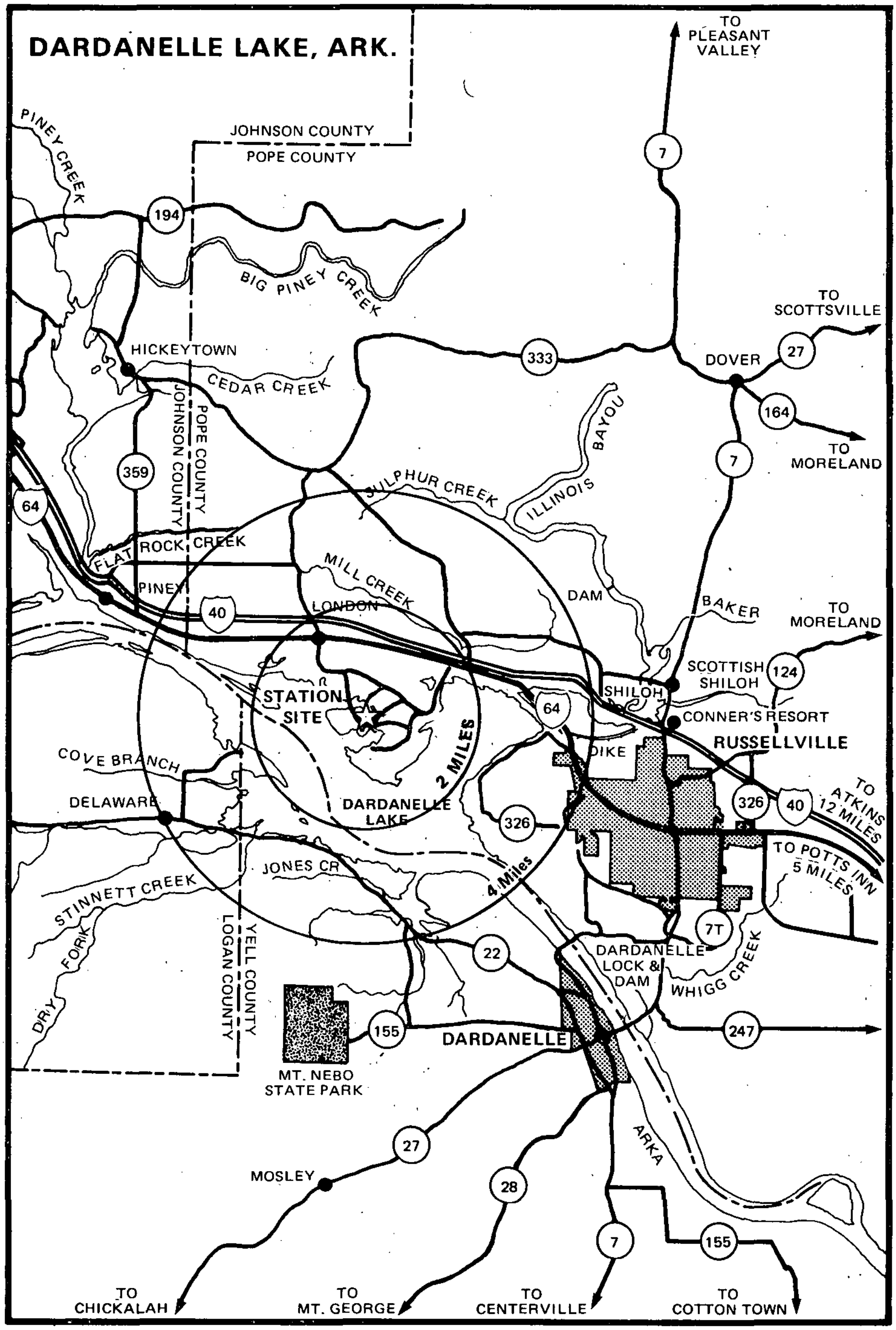

Figure 2.2 Site Location and Low Population Zone

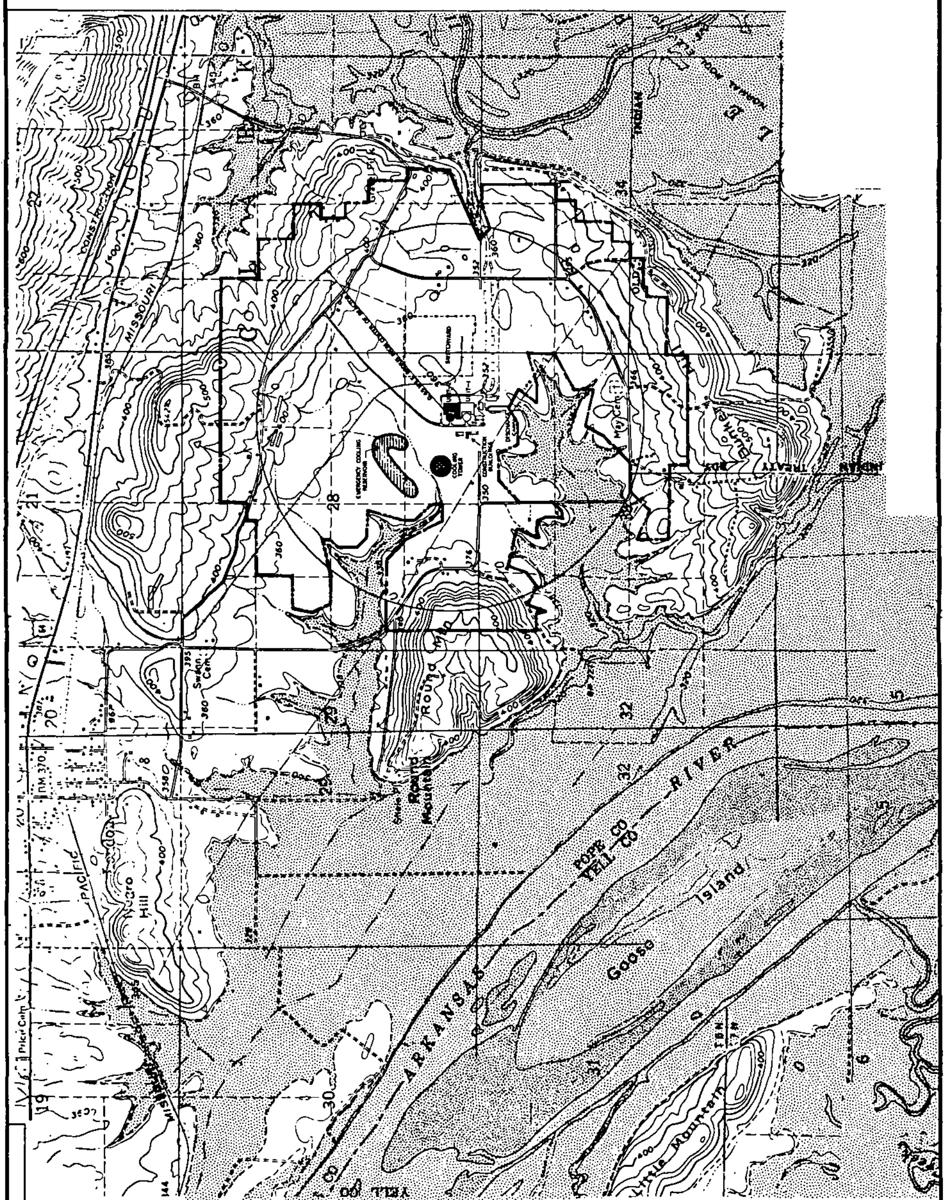

Figure 2.3 Plot Plan and Site Boundary

FIGURE 2.5

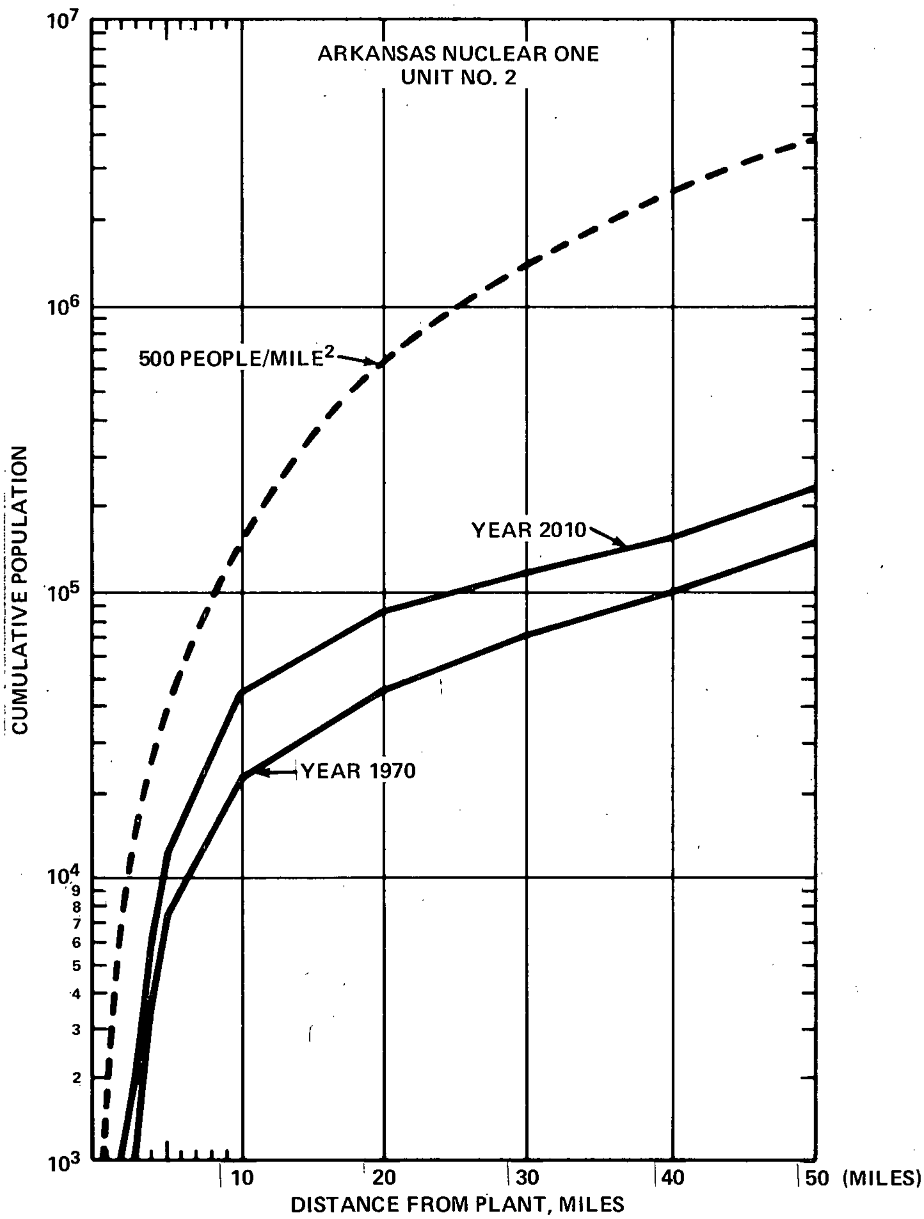

Figure 2.4 Cumulative Population Distribution

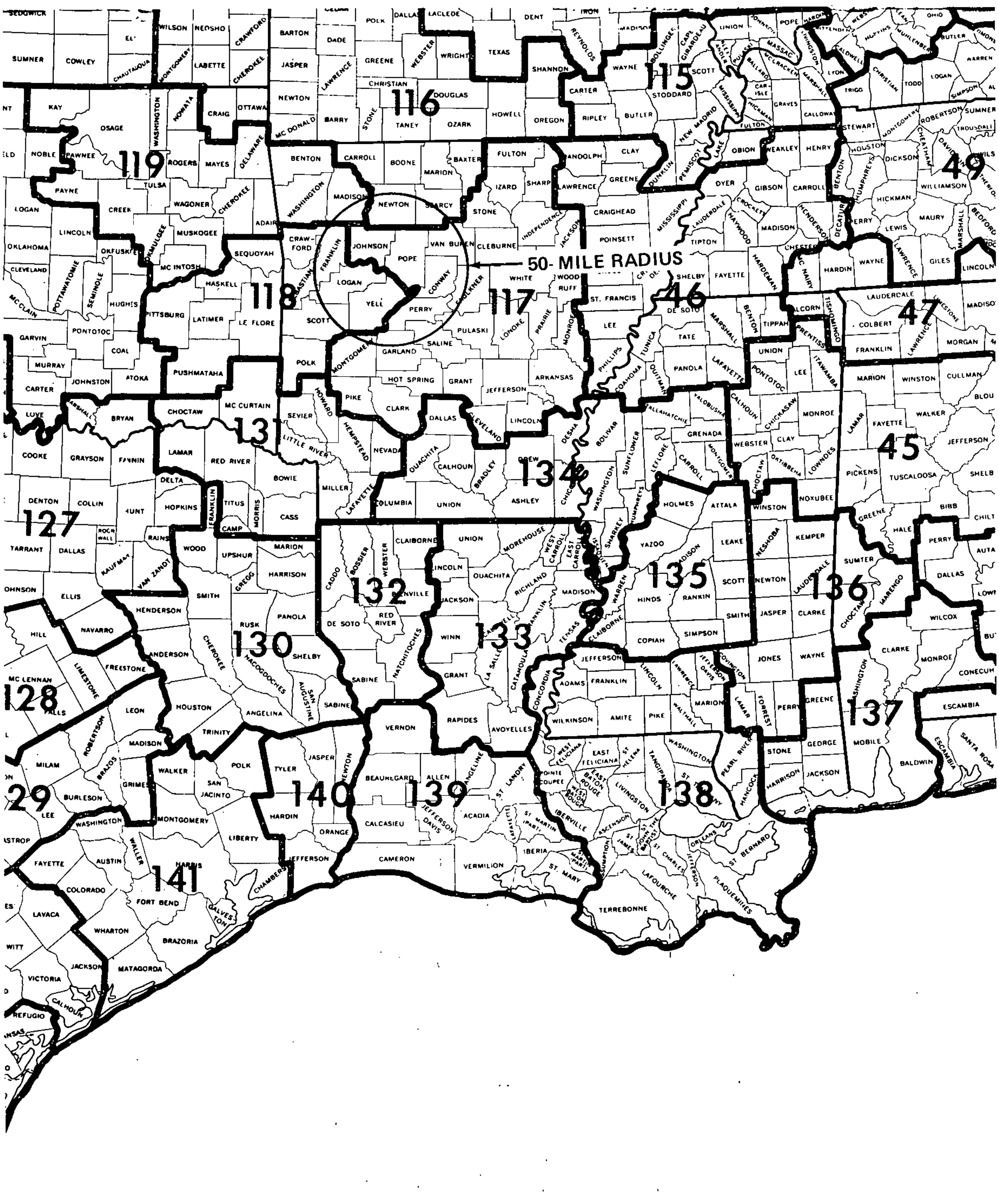

Figure 2.5 Bureau of Economic
Analysis Areas

55 miles south-southeast of the site. The applicant specified Hot Springs as the population center (as defined by 10 CFR Part 100) and also has proposed a low population zone of four miles. The applicant's selection of the city of Hot Springs, Arkansas as the population center was questioned during the staff review based on a document entitled "Preliminary Report Regional Growth and Change, 1975-2000, Demographic and Economic Aspects" and published by the West Central Arkansas Planning and Development District, Inc. This report indicated that by the year 1990 the population of the city of Russellville would be 28,731, and thus become a closer population center than Hot Springs.

Based on our evaluation of this information and further additional information from the applicant and from the city of Russellville, we believe that the city of Russellville and its immediate surroundings should be considered the population center as defined by 10 CFR Part 100 which requires the population center distance to be at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. On that basis, the applicant's proposed low population zone of four miles is not acceptable since the boundary of Russellville is within a 5.3 mile distance of the ANO-2 plant. We believe a low population zone radius of two miles (3218 meters) would be appropriate for this site and have evaluated the site against the exposure guidelines of 10 CFR Part 100 accordingly. As discussed in Section 15 of this report, the consequences of design basis accidents at a low population zone distance of two miles are within the guidelines of 10 CFR Part 100.

According to the applicant's data, the 1970 census shows about 600 residents within a two mile radius of the plant. The Dardanelle Reservoir is a major contributing factor to the part-time use or transient population within the plant area. During summer weekends, the periods of highest utilization, an average of approximately 6,000 visitors are located within four miles (the initially proposed low population zone). The applicant's emergency plan (see Section 13.0 of this report) provides reasonable assurance that adequate protective measures can be taken for the populace including the transient population mentioned above within the low population zone of four miles or less radius in the event of an accident.

The applicant does not agree with the staff's position, as stated above, that a low population zone radius of two miles is appropriate for the ANO-2 site. We are continuing our review of recent additional information provided by the applicant and will report our final evaluation of this matter in a supplement to this report.

2.2 Nearby Industrial, Transportation and Military Facilities

The site is about six miles upstream from the Dardanelle Dam. A Missouri Pacific Railroad line, U. S. Highway 64, and Interstate Highway 40 pass north of the site at distances of 1.1, 1.2, and 1.3 miles, respectively. The Arkansas River shipping canal is about 1.4 miles south of the reactor buildings.

The closest airport is the Russellville Municipal Airport eight miles from the site, a small airport, without control tower, which handles light planes. There is no major airport within 50 miles of the plant site.

Stone quarries exist at Midway and Altus and near the Dardanelle Dam. The nearest quarry, near Dardanelle is approximately five miles to the south, a sufficient distance so that any blasting conducted there will not adversely affect the site.

Sand and gravel deposits of commercial value are near Seranton and the Arkansas River at Dardanelle. Natural gas is produced at a number of locations within ten miles of the site.

A natural gas transmission line owned by the Arkansas-Louisiana Gas Company crosses the site. The safety aspects of this 10.75-inch line which operates at 500 pounds per square inch gauge pressure were analyzed during the construction permit review of Arkansas Nuclear One - Unit One. To meet our requirements during the construction permit review, the line was rebuilt with ASA Code B31.8 pipe for 1200 feet of its length nearest the reactor building and rerouted under the discharge canal with four feet of earth cover. In its present path the line comes no closer than 700 feet from the ANO-2 facility. The applicant has drawn up the Emergency Plan to arrange for prompt closure of nearby isolation valves (south of London and on the west side of Russellville) if the line should leak.

We have examined the probabilities and consequences of several types of natural gas release from the Arkansas Louisiana Gas Company pipeline passing south of the plant site, and have been unable to identify credible accidents which could lead to worse consequences than superficial surface damage from combustion of gas to safety-related plant structures. We conclude that the existence of the pipeline at its present location will not interfere with the safe operation of the plant and need not be considered in its design.

2.3 Meteorology

Information concerning the atmospheric diffusion characteristics of a proposed nuclear power plant site is required in order that a determination may be made that postulated accidental, as well as routine operational, releases of radioactive materials are within Nuclear Regulatory Commission guidelines. Further, regional and local climatological information, including extremes of climate and severe weather occurrences which may affect the safe design and siting of a nuclear plant, is required to ensure that safety-related plant design and operating bases are within Commission guidelines. The meteorological characteristics of a proposed site are determined by the staff's evaluation of meteorological information in accordance with the procedures presented in Sections 2.3.1 through 2.3.5 of the Standard Review Plan.

2.3.1 Regional Climatology

Experiencing the effects of all North American airmass types, the modified continental climate of central Arkansas is typified in summer by extended periods of warm humid weather, while winters are generally mild with occasional arctic and polar airmass outbreaks. Precipitation is well distributed throughout the year with summer precipitation almost exclusively resulting from airmass and convective processes. Late summer and early fall is usually the driest part of the year, while winter and early spring are rainiest because of frontal activity coupled with an abundant moisture supply from the Gulf of Mexico. Freezing precipitation may occur when a shallow cold airmass flows under warm moist Gulf air. Resulting glaze and ice storms may at times be severe, but are relatively infrequent. On the average, snowfall is light with accumulations of less than an inch occurring in one out of every four or five winters.

On about 70 days a year, temperatures may be expected to reach 90 degrees Fahrenheit or higher, on 65 days, 32 degrees Fahrenheit or lower, and on less than one-half of a day per year, equal to or less than zero degrees Fahrenheit. Average annual precipitation is about 49 inches per year. Annual average relative humidity is 70 percent.

Severe weather occurrences in the vicinity of the site are usually associated with thunderstorms or tropical storm and hurricane activity.

During the period from 1871 through 1974, 17 hurricanes and tropical storms passed within 50 miles of the site. Between 1955 and 1967, twenty-three tornadoes were reported within the one degree latitude-longitude square containing the site. The resulting calculated annual tornado frequency is 1.8 and recurrence interval for a tornado at the site is 760 years. During the time interval (1955-1967), hail three-quarters of an inch in diameter or larger was reported on 13 days within the one degree square and storms with winds 50 knots (58 miles per hour) or greater were reported on 11 days. The maximum "fastest mile" wind speed at Little Rock, Arkansas was 65 miles per hour. On an annual average, thunderstorms may be expected to occur on approximately 60 days. Freezing precipitation (ice storms) may be expected to occur about once per year with storms resulting in accumulations of one-half inch or more expected one year in five. Between 1936 and 1970 there were ten cases of air stagnation within the site area lasting four or more days.

The design basis tornado used for the plant design consisted of a maximum wind speed of 360 miles per hour, with a maximum rotational wind speed of 300 miles per hour and translational wind speed of 60 miles per hour, a maximum pressure drop of 3.0 pounds per square inch and a maximum pressure drop rate of 3.0 pounds per square inch in three seconds. The acceptability of the design capability for tornadoes up to this magnitude is discussed in Section 3.3 of this report. The operating basis sustained (fastest mile) wind speed of 80 miles per hour at a height of 45 feet with a return period of 100 years was used in the ANO-2 design.

2.3.2 Local Meteorology

The applicant has provided sufficient information for us to make an evaluation of the local meteorological conditions of importance to the safe design and siting of this plant.

Long-term weather records from Fort Smith, Arkansas, approximately 70 miles west of the site and Little Rock, Arkansas, 65 miles southeast of the site, show that extreme maximum temperatures of 110 degrees Fahrenheit and 108 degrees Fahrenheit, respectively, occurred in August 1964. At other localities in the Fort Smith area, record high and low temperatures of 113 degrees Fahrenheit occurred in 1936 and -15 degrees Fahrenheit in February 1899. Extreme minimum temperatures recorded at Fort Smith and Little Rock were zero degrees Fahrenheit in January 1966 and minus four degrees Fahrenheit in January 1962, respectively. Maximum 24-hour precipitation amounting to 7.96 inches in April 1974 occurred at Little Rock and 7.13 inches in July 1960 at Fort Smith. The maximum 24-hour snowfall at Little Rock occurred in January 1960 and totalled 11.3 inches. The annual average number of days with heavy fog (visibility reduced to one-fourth mile or less) is 16 days for Little Rock and 15 days at Fort Smith. Thunderstorms may be expected to occur approximately 55 times on an annual average.

Wind data collected at the 40-foot level onsite during the period February 1972 to February 1973 show the predominant wind flow to be from the east with a frequency of 16.0 percent and secondary flow from the east-northeast 13.9 percent of the time. Flow from the north-northwest occurred least often with a two percent frequency.

2.3.3 Onsite Meteorological Measurements Program

The onsite meteorological program was begun in September 1967 when wind speed and directional variability measurements were recorded on a 30-foot mast. A 190-foot tower was erected in June 1969 with wind data collected at the 20- and 190-foot levels and temperature differences observed between the 5- and 85-foot, and 85- and 190-foot levels. Elevations of some of the instruments were subsequently changed in July 1971 to reflect changes in accepted practices. After that date, temperature differences were measured from 30 feet to the higher levels and winds at 40 and 190 feet.

The applicant has provided a second full year period of onsite meteorological data collected from February 1972 to February 1973 which were used to verify previous dispersion estimates based upon onsite data collected from June 1969 through May 1970. The dispersion estimates were made using the joint wind speed and direction frequency distributions at the 40-foot level and atmospheric stability based on the vertical temperature difference between the 30- and 190-foot levels. The joint recovery rate for these data was 96 percent.

The onsite meteorological measurements program has been compared with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs." We conclude that the meteorological measurements program has produced data which, in turn, have been summarized to provide a sufficient meteorological description of the site and its vicinity for the purpose of making atmospheric diffusion estimates for accidental and routine airborne releases of effluents from the nuclear facility.

The new operational onsite meteorological system will measure wind speed and direction at the 40- and 190-foot levels and temperature and dew point sensors will be located at the 33- and 190-foot levels. All data observed on the meteorological tower will be recorded on strip charts in the reactor control room.

2.3.4 Short Term (Accidental) Diffusion Estimates

Conservative assessments of atmospheric diffusion conditions following a design basis accident have been made by us from the applicant's meteorological data and appropriate diffusion models. In the evaluation of short-term (0-2 hours at the exclusion distance and 0-8 hours at the low population zone distance) accidental releases from the plant building and vents, a ground-level release considering a building wake factor, cA, of 1103 square meters was assumed. The relative concentrations for the various time periods following an accidental release were calculated using the diffusion model described in Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident For Pressurized Water Reactors."

The relative concentration which is exceeded no more than five percent of the time for the 0-2 hour time period following an accidental release is 7.7×10^{-4} seconds per cubic meter at the exclusion distance of 1027 meters (measured from the outside edge of the containment building). This relative concentration is equivalent to dispersion conditions produced by Pasquill Type F stability with a wind speed of 0.5 meters per second. The relative concentrations estimated at the outer boundary of the low population zone for the various time periods following an accidental release will be provided in a supplement to this report upon the resolution of the low population zone radius matter as discussed in Section 2.1 of this report.

2.3.5 Long-Term (Routine) Diffusion Estimates

The staff has made estimates of average atmospheric diffusion conditions from onsite wind speed, direction and atmospheric stability data measured at the 10-meter level during calendar year 1975. Data recovery was 98 percent. The "Straight-Line Trajectory Model" described in Regulatory Guide 1.111, "Methods of Estimating Atmospheric Transport and Dispersion of Gaseous Effluents from Light-Water-Cooled Reactors," which was used in the assessment assumes a mixture of elevated and groundlevel releases. Noncontinuous and intermittent gaseous releases were evaluated separately from continuous releases. The calculations also include an estimate of maximum increase in calculated relative concentration and deposition due to valley recirculation of airflow not considered in the straight-line trajectory model. As described in Regulatory Guide 1.111, a downvalley adjustment factor of five was applied to the east and southeast sectors and an upvalley adjustment factor of 1.5 was applied in the north northwest sector. A summary of the relative concentration and deposition values used in the dose estimates are presented (in Section 5.5 of the Arkansas Final Environmental Statement).

2.3.6 Conclusions

The onsite meteorological measurements program will continue in operation during the lifetime of the plant to provide sufficient meteorological data for estimating radiation doses to the public as a result of routine or accidental releases of radioactivity to the atmosphere, and for initiating protective measures to protect the health and safety of the public.

The applicant has provided sufficient information concerning those meteorological conditions which are of importance to the safe design and siting of the plant. The applicant's onsite meteorological program conforms to the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," and has produced data which adequately describe site atmospheric dispersion conditions and which was used by the staff to make both conservative and realistic estimates of atmospheric dispersion characteristics for accidental and routine gaseous releases, respectively, from the plant. This program will continue in operation during the lifetime of the plant to provide sufficient meteorological data for estimating radiation doses to the public as a result of routine or accidental releases of radioactivity to the atmosphere, and for initiating measures to protect the health and safety of the public.

2.4 Hydrology

2.4.1 Hydrologic Description

The ANO-2 site is on the northern floodplain of Dardanelle Reservoir, about six miles upstream of Dardanelle Dam on the Arkansas River. The Arkansas River is a major waterway with 150,000 square miles of drainage area controlled by more than 24 reservoirs. The site is about 259 miles upstream from the mouth of the Arkansas River; the furthest upstream reservoirs are more than 700 miles away.

The minimum navigation pool level of Dardanelle Reservoir is elevation 336 feet mean sea level and the reservoir normally varies between 336 and 338 feet mean sea level to provide two feet of storage for hydropower generation. Plant grade is elevation 353 feet mean sea level and plant ground floor levels are a foot higher.

The plant will take its cooling water from Dardanelle Reservoir, which is part of the Arkansas River Navigation System. Future upstream diversions are not expected to be large enough to affect plant operation since the water is committed to maintain minimum navigation depths. Natural diversions, such as landslides or flood-caused rerouting, are also considered unlikely. Even if such a diversion were to occur, the storage available in the emergency cooling pond is sufficient for safe shutdown. The ultimate heat sink for the plant includes the Dardanelle Reservoir and a 14-acre man-made emergency cooling water storage pond. The pond is filled by runoff from surrounding slopes. If necessary, makeup to the pond can be accomplished by a water line from the city of Russellville water supply system. A spillway limits the static water level in the pond to a maximum of 347 feet mean sea level; its bottom is at 341 feet mean sea level. The ultimate heat sink is discussed further in Section 9.0 of this report.

2.4.2 Flooding

The greatest flood of record occurred in May 1943 with an estimated maximum runoff rate of 683,000 cubic feet per second. Dardanelle Dam is designed to discharge up to 900,000 cubic feet per second without exceeding a maximum water level of 338 feet mean sea level, and can safely pass a substantially more severe probable maximum flood. The levees along the river channel in this area are designed for a maximum discharge of 830,000 cubic feet per second.

The U.S. Army Corps of Engineers estimate the probable maximum flood at Dardanelle Dam to have a maximum runoff rate of 1,500,000 cubic feet per second, and a corresponding reservoir elevation of 353 feet mean sea level. We concur with this estimate. To determine the corresponding water level at the site, the applicant conservatively assumed a straight line variation in levels between elevation 353 feet mean sea level at Dardanelle Dam and elevation 389.5 feet mean sea level on the downstream side of Ozark Dam, the next dam 51 miles upstream. The resulting estimated peak probable maximum flood level at the site is elevation 358.0 feet mean sea level.

The applicant estimated maximum waves due to wind activity coincident with a probable maximum flood water level of 358 feet mean sea level of 2.5 feet. Maximum wave runup from these waves on safety-related structures would be to elevation 368 feet mean sea level. Our independent analyses of potential wave action, using computational techniques discussed in Regulatory Guide 1.59, "Design Basis Floods For Nuclear Power Plants," indicate that the applicant's analyses are conservative.

The spillway and exit channel of the emergency cooling pond have been designed to pass safely a standard project flood for its local drainage area. A standard project flood is about half as great as a probable maximum flood and is considered to represent the most severe precipitation conditions reasonably characteristic of the region, based on historical hydrometeorology and excluding extremely rare occurrences. The spillway is composed of concrete slabs linked together with exposed galvanized reinforcing rods. To aid in preventing piping under the slabs, a metal waterstop was driven between the slabs parallel to the axis of the spillway. The embankment sections adjacent to the spillway were originally sized to provide one foot of freeboard above the maximum pond elevation that would be reached during a standard project flood. Based on standard practice by the Corps of Engineers and others, we required that freeboard allowance be increased to three feet; or that the dike be erosion protected. The applicant elected to place riprap on both faces (upstream and downstream) and the crest of the embankments. We performed an independent analysis of the pond runoff characteristics for a storm of probable maximum intensity using standard Weather Bureau (now National Oceanic and Atmospheric Administration) precipitation estimates and a synthetically developed runoff model for the pond drainage area. This analysis indicated that the spillway and exit channel could safely pass a probable maximum flood without loss of pond inventory if proper erosion protection for the embankment sections were provided. We concluded that the riprap design criteria proposed by the applicant was acceptable for this purpose. However, to assure the continued integrity of the riprap during the life of the project, we have added a requirement in Section 4.7.4.1 of the ANO-2 Technical Specifications for the surveillance of the riprap and the spillway of the ultimate heat sink pond.

The effects at the site of arbitrarily assumed upstream dam failures were investigated independently by both the applicant and the staff. In all cases it was determined that the maximum water level at the site would be less than that produced by a probable maximum flood, even though upstream dam failures could cause dams further downstream to fail.

Ice flooding can occur under extreme conditions, but we consider the controlling flood conditions to be those associated with a probable maximum flood. In this area, there are no known cases where flooding caused by ice has been more severe than precipitation-induced flooding.

All safety-related structures and equipment are located above elevation 369 feet mean sea level, the probable maximum flood level, or are protected from flooding and wave runup by structures which can be made watertight as discussed in Section 9.0 of this report. For a local probable maximum storm, there are several openings where resulting ponded water could conceivably enter safety-related structures. These doors and hatches are equipped with intrusion alarms monitored in the control room and are normally closed. We have concluded that the applicant's flood protection design is acceptable. The plant will be shutdown, with an appropriate emergency plan to protect safety-related facilities in the event of a severe flood on the Arkansas River.

2.4.3 Water Supply

During normal operation the circulating and service water systems of Unit 2 will utilize a cooling tower. Makeup to this system is expected to be a maximum of 50 cubic feet per second from Dardanelle Reservoir. This will be taken from the intake canal common to Units 1 and 2. Since Unit 1 has a once-through cooling system, the combined water intake will be 1700 plus 50 cubic feet per second. Blowdown from the Unit 2 cooling tower is expected to be a maximum of 20 cubic feet per second and will be returned with the Unit 1 discharge. The combined consumptive use of both units operating at full load will be about 50 cubic feet per second.

The emergency cooling pond will serve as a heat sink for normal plant shutdown of either unit, as a source of water for simultaneously shutting down both units in the event of a loss of Dardanelle Reservoir water inventory, or a plant accident. The applicant has stated that the pond size (84 acre-feet) is sufficient to dissipate the total heat transferred to the Unit 1 and 2 service water systems as a result of a design basis accident in one unit and a normal plant shutdown of the other unit, while limiting the cooling pond water temperature to a maximum of 129 degrees Fahrenheit. The worst condition for maximum heat rejection to the pond would be a design basis loss-of-coolant accident in Unit 1 concurrent with a normal shutdown of Unit 2. We performed independent analyses of the ability of the emergency cooling pond to accept the heat rejected under the above stated conditions. We conclude that the applicant's estimate of maximum cooling pond water temperature is conservative and that pond inventory is sufficient for 30 days without makeup.

2.4.4 Low Water Considerations

Daily streamflow records for the period of January 1923 to September 1957, collected at the Dardanelle gauging station just below the Dardanelle Dam, have been adjusted by the Corps of Engineers to reproduce flows as they would have been regulated by the complete system of upstream dams. The minimum daily average flow as computed in this study was 400 cubic feet per second during the driest critical month of record; Units 1 and 2 require 1,750 cubic feet per second of cooling water.

It is possible for the inflow to Dardanelle Reservoir to be zero under very exceptional circumstances involving emergency operation of upstream dams. These conditions would exist for only a few hours, however, during which time there would either be adequate water in storage in the reservoir, or the plant could be shutdown and safety maintained in shutdown by using the emergency cooling pond for a period of 30 days or more. Similarly, a decrease of level in Dardanelle Reservoir to below plant pump intake levels, which could result in the event of failure of the Dardanelle Dam, would still not lead to an unacceptable situation since the emergency cooling pond would be available for shutdown and cooldown.

Plant operators would be notified of a failure of Dardanelle Dam by an alarm in the control room that is automatically activated when the reservoir level has dropped one foot below the normal minimum operating level of 336.0 feet mean sea level. The applicant has stated that 30 minutes are required to operate all six sluice gates to transfer the service water system to the emergency cooling pond, and that a minimum time of approximately 85 minutes would be available before the reservoir level could drop below the minimum required submergence level of elevation 327.3 feet mean sea level. The emergency cooling pond, to be kept at a normal level of 347 feet mean sea level (84 acre-feet), will provide a shutdown-cooldown source of water. The pond will be replenished by natural runoff, or in the event natural runoff is not sufficient, from the Russellville water supply. Pond level will be monitored daily by the applicant.

2.4.5 Groundwater

Groundwater in the upper overburden at the site fluctuates with the level in Dardanelle Reservoir, but at the site is generally found about ten feet below the surface sloping toward the reservoir. The lower bedrock zones are low-yield artesian sources. Domestic wells located down gradient from the plant site extend into this bedrock; therefore, any contaminated water accidently spilled at the plant will migrate very slowly through the relatively impermeable clayey overburden toward the lake and should have no effect on water supplies taken from the artesian bedrock aquifer.

The only use of groundwater in the vicinity of the site is for local domestic purposes. Shallow domestic wells in the general vicinity are located up gradient from the plant site; therefore, contamination from the plant is not possible. The possibility of contamination of groundwater, and/or migration of such contaminants to the reservoir is very remote because of the affinity of radionuclides for surface clays, and extremely low permeabilities. These factors should negate any significant or long distance travel of contaminated water.

No potable water supply is drawn from Dardanelle Reservoir or from the Arkansas River downstream of Dardanelle Dam because of its salinity.

2.5 Geology, Seismology and Foundation Engineering

2.5.1 Geology and Seismology

We have reviewed the regional and site geological and seismological conditions as presented in the Final Safety Analysis Report. The Final Safety Analysis Report adequately appraises conditions pertinent to an evaluation of the site geology, and confirms the conditions as described in the Preliminary Safety Analysis Report which was reviewed at the construction permit stage by the staff and its advisors, the Seismological Investigations Group of the National Oceanic and Atmospheric Administration and the United States Geological Survey, and reported in the Safety Evaluation Report dated April 20, 1972.

During our present review, no new information, which would alter the National Oceanic and Atmospheric Administration's conclusion has come to our attention. Therefore, we have concluded that our earlier finding remains valid. That finding was that 0.20g and 0.10g are adequate values for the safe shutdown earthquake and operating basis earthquake, respectively.

The Atomic Energy Commission's Safety Evaluation Report, dated June 6, 1973, concluded that there were no geologic conditions present at the Arkansas Nuclear One, Unit 1 site which would preclude the safe operation of the nuclear reactor. An earlier report, the U.S. Geological Survey Report dated August 16, 1968, supports these conclusions. Further, the Atomic Energy Commission's Safety Evaluation Report, dated April 20, 1972, for the ANO-2 plant concluded that the additional site investigations for ANO-2 have confirmed that foundation conditions beneath the two ANO units are similar, and therefore concluded that the seismic acceleration values for ANO-1 are applicable for ANO-2. Since these documents were published, no additional geologic or seismologic evidence has been discovered which would change these earlier conclusions.

2.5.2 Foundation Engineering

The site foundation material is composed of a stiff clay soil overlying shale with interbedded sandstone and siltstone. The rock surface has low relief and slopes gently to the southwest.

The plant facilities are located on a broad, nearly flat bench adjacent to the floodplain of the Arkansas River. This bench is at an elevation of about 353 feet. Surface relief within the limits of the site is less than 10 feet.

Surficial soils at the site consist of eight to thirty feet of tan clay and silty clay overlying the uppermost bedrock unit which consists of 70 to 90 feet of hard black shale. The top four to eight feet of the uppermost bedrock unit is weathered to hard tan clay, grading with depth to a soft gray shale. The uppermost bedrock unit is underlain by approximately 60 feet of light to dark gray, hard shale with minor sandstone interbeds. Rock penetrated below the shale consisted of gray, dense, fine grained, well cemented, horizontally bedded sandstone. A few thin limestone strata were encountered at depth.

All seismic Category I structures except the emergency cooling pond inlet and outlet structures and electric utility manholes are founded on unweathered shales. Unconfined compressive strength of competent bedrock samples range from 2420 to 4690 pounds per square inch. The average compressive strength is 3460 pounds per square inch. A site seismic survey indicated compression wave velocities from 10,000 to 14,500 feet per second. Such velocities are associated with a dense, competent rock. There is no karstic or cavernous terrain in the area around the site. Due to the properties and nature of the shale bedrock foundation, both we and our consultant, the Corps of Engineers, conclude that the bearing capacity of the bedrock is very high and that settlement of structures founded thereon will be negligible. The cooling pond inlet and outlet structures and electric utility manholes are relatively small reinforced concrete structures founded in the overburden soils. These structures are lightly loaded; therefore, bearing capacity and settlement are not a problem. The overburden soil at the site is primarily cohesive and is not subject to liquefaction. Based on the above conditions, we conclude that the overburden soils will provide adequate support for these lightly loaded structures.

During excavation of the foundation for Unit 2 all rock surfaces were cleaned and coated with gunite within 24 hours after exposure to prevent slaking of the shale during construction. No swelling or heave of the foundation rock was noted prior to placement of concrete.

Compacted fill specifications were designated based on the maximum density as determined by American Society for Testing and Materials Designation D1557, Method D. Compaction requirements varied as discussed below depending on the use of the backfill material.

Fill placed beneath structural slabs was granular material placed in lifts not exceeding eight inches uncompacted thickness. Fill beneath structures was moisture conditioned and compacted to 95 percent of the maximum density. Granular backfill was placed against the exterior walls of structures and was compacted to 90 percent of the maximum dry density. Impervious material for embankments, pond liner and backfill consisted of silty or sandy clay. These materials were moisture conditioned and compacted to 95 percent of the maximum dry density. Random backfill material was placed in areas outside the limits of structural backfill and compacted to 90 percent of the maximum dry density. We and our consultant, the Corps of Engineers, agree that the compacted fill described in the Final Safety Analysis Report will provide acceptable support for fill supported seismic Category I facilities.

The applicant has provided an evaluation of lateral earth pressure for both static and dynamic conditions. Static earth pressure diagrams were constructed following the basic principles outlined in "Soil Mechanics in Engineering Practice" by Terzaghi and Peck, second edition. Dynamic lateral earth pressures were calculated based on the Mononobe-Okabe method. We concur with the applicant's use of these procedures.

A seismic Category I emergency cooling pond was constructed to serve as the ultimate heat sink. The pond is primarily a shallow excavation in the overburden soils. The cooling pond slopes were excavated to two and one half horizontal to one vertical. An impervious blanket was provided at the bottom and on the side slopes of the pond to limit any seepage loss.

Stability analyses were made in accordance with the modified swedish slip circle methods. The earthquake load used in the analysis was introduced as a static horizontal load equivalent to the safe shutdown earthquake of 0.2g times the appropriate mass of soil and water. All slopes were evaluated under conditions of normal water level, rapid drawdown, and normal water level combined with earthquake loading.

The minimum computed factor of safety was 2.6 for normal water level conditions and 1.1 for normal water level with earthquake loading. We and our consultants, the Corps of Engineers, conclude that the ultimate heat sink pond has been constructed in accordance with Preliminary Safety Analysis Report design criteria and is acceptable.

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 Conformance with General Design Criteria

Arkansas Nuclear One - Unit 2 was designed and is being constructed on the basis of the proposed AEC General Design Criteria which were published July 11, 1967. Design and construction were thus initiated and proceeded to a significant extent based upon the criteria proposed in 1967. Since July 15, 1971, when the Atomic Energy Commission published the General Design Criteria of Appendix A of 10 CFR Part 50 the applicant has attempted to comply with the newer criteria to the extent practical. Recognizing work already accomplished and design commitments made, the applicant discusses in Section 3.1 of the Final Safey Analysis Report the design of ANO-2 with respect to the criteria of July 15, 1971. As a result, our technical review assessed the plant against the General Design Criteria now in effect and we have concluded that the plant design conforms to the intent of these newer criteria.

3.2 Classification of Structures, Components and Systems

3.2.1 Seismic Classification

Criterion 2 of the General Design Criteria requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

Structures, systems and components important to safety that are required to be designed to withstand the effects of a safe shutdown earthquake and remain functional have, in general, been properly classified as seismic Category I items. All other structures, systems and components that may be required for operation of the facility have been designed to other than seismic Category I requirements including those portions of Category I systems which are not required to perform a safety function. Structures, systems, and components important to safety that have been designed to withstand the effects of a safe shutdown earthquake and remain functional are identified in an acceptable manner in Tables 3.2-1 and 3.2-2 of the Final Safety Analysis Report.

The basis for acceptance in our review has been conformance of the applicant's designs, design criteria and design bases for structures, systems and components important to safety with the Commission's regulations as set forth in General Design Criterion 2 and industry codes and standards.

We conclude that structures, systems and components important to safety that are designed to withstand the effects of a safe shutdown earthquake and remain functional have, in general, been properly classified as seismic Category I items in conformance with the Commission's regulations and industry codes and standards and are acceptable.

3.2.2 System Quality Group Classification

Criterion 1 of the General Design Criteria requires that nuclear power plant systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

Fluid system pressure-retaining components important to safety have, in general, been designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed. The applicant has identified those fluid-containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety where reliance is placed on these systems: (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintenance in the safe shutdown condition, and (3) to contain radioactive material. These fluid systems have, in general, been classified in an acceptable manner in Tables 3.2-3 and 3.2-5 of the Final Safety Analysis Report and on system piping and instrumentation diagrams in the Final Safety Analysis Report.

The applicant has applied Quality Groups A, B, C, and D in Regulatory Guide 1.26, "Quality Group Classifications and Standards" to the fluid system pressure-retaining components important to safety. These components that are classified Quality Group A, B, C, or D have been constructed to the codes and standards identified in Table 3.2-4 of the Final Safety Analysis Report.

The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping and valves in fluid systems important to safety with the Commission's regulations as set forth in General Design Criterion 1, the requirements of the Codes specified in Section 50.55a of 10 CFR Part 50 and industry codes and standards.

We conclude that fluid system pressure-retaining components important to safety that have been designed, fabricated, erected and tested to quality standards in conformance with the Commission's regulations and industry codes and standards are acceptable.

Components of the reactor coolant pressure boundary as defined by the rules of 10 CFR Part 50, Section 50.55a have been properly identified and classified as American Society of Mechanical Engineers Section III, Class A or Class 1 and American Society of Mechanical Engineers Code for Pumps and Valves, Class 1, components in Table 5.2-1 of the Final Safety Analysis Report. These components within the reactor coolant pressure boundary have been constructed in accordance with the requirements of the applicable codes and addenda as specified by the rules of 10 CFR Part 50, Section 50.55a, Codes and Standards.

We conclude that construction of the components of the reactor coolant pressure boundary in conformance with the Commission's regulations provides reasonable assurance that the resulting quality standards are commensurate with the importance of the safety function of the reactor coolant pressure boundary.

The American Society of Mechanical Engineers Code Cases specified in Section 5.2.1.4 of the Final Safety Analysis Report whose requirements have been applied in the construction of pressure-retaining American Society of Mechanical Engineers Section III, Code Class A or Code Class 1, components within the reactor coolant pressure boundary (Quality Group Classification A), are in accordance with those code cases in Regulatory Guides 1.84, "Code Case Acceptability - ASME Section III Design and Fabrication," and 1.85, "Code Case Acceptability - ASME Section III Materials," that are acceptable to the Commission. We conclude that compliance with the requirements of these code cases, in conformance with the Commission's regulations, is expected to result in a component quality level that is commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

3.3 Wind and Tornado Design Criteria

All seismic Category I structures exposed to wind forces are being designed to withstand the effects of the design wind. The design wind specified has a velocity of 80 miles per hour based on a recurrence interval of 100 years.

The procedures used to transform the wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gust factors are in accordance with American Society of Civil Engineers Paper No. 3269, "Wind Forces on Structures."

All seismic Category I structures exposed to tornado forces and needed for the safe shutdown of the plant are designed to resist a tornado with a 300 miles per hour tangential wind velocity and a 60 miles per hour translational wind velocity. The atmospheric pressure drop associated with the design tornado is three pounds per square inch in three seconds. An appropriate spectrum of tornado-generated missiles is also postulated as will be discussed in Section 3.5 of this report.

The procedures used to transform tornado wind velocity into pressure loadings are similar to those used for the design wind loadings. The pressure drop associated with the design tornado is treated as a static uniform load applied on vertical and horizontal projected areas of the structures.

Tornado missile effects were determined using procedures to be discussed in Section 3.5 of this report. The total effect of the design tornado on seismic Category I structures was determined by the appropriate combination of the individual effects of the tornado wind pressure, pressure drop and tornado associated missiles. Tornado-generated loads were then combined with other applicable loads as will be discussed in Section 3.8 of this report. Structures are arranged on the plant site and protected in such manner that collapse of structures not designed for tornadoes will not affect safety-related structures and systems.

The criteria used and the procedures utilized to account for loadings on seismic Category I structures and components induced by the design wind and tornado specified for the plant provide a conservative basis for engineering design and assure that such environmental forces are adequately represented.

The use of these procedures provides reasonable assurance that, in the event of wind or a tornado, the structural integrity of all seismic Category I structures will not be impaired and, consequently, seismic Category I systems and components located within these structures will be adequately protected. Conformance with these criteria is an acceptable basis for satisfying the requirements of General Design Criterion 2.

3.4 Water Level (Flood) Design

3.4.1 Flooding Effects on Seismic Category I Structures

The design flood level resulting from the most unfavorable condition or combination of conditions that produce the maximum water level of the site is discussed in Section 2.4 of this report. The hydrostatic effect of the flood has been considered in the design of all seismic Category I structures exposed to water head. The manner in which design basis flood loads have been combined with other applicable loads is discussed in Section 3.8 of this report.

The criteria used and the procedures utilized to account for loadings on seismic Category I structures induced by the design flood and/or highest ground water level specified for the plant provide a conservative basis for engineering design and assure that such environmental forces are adequately represented.

The use of these criteria provides reasonable assurance that, in the event of floods and/or high ground water, the structural integrity of the plant seismic Category I structures will not be impaired and, consequently, seismic Category I systems and components located within these structures are adequately protected and may be expected to perform their intended safety functions. Conformance with these design procedures is an acceptable basis for satisfying the requirements of General Design Criteria 2.

3.4.2 Protection of Essential Equipment

The Arkansas Nuclear One - Unit 2 design is based on a probable maximum flood elevation of 358 feet mean sea level. All safety-related systems and components are either located on floors above elevation 369 feet mean sea level, which is the probable maximum flood combined with wave runup or are protected by the following measures: wall thickness in seismic Category I structures below flood level have a minimum wall thickness of two feet, waterstops are provided in all construction joints below flood level, the number of openings in walls and slabs below flood level is kept to a minimum, watertight doors and equipment hatches are installed, watertight seals are provided for all penetrations below the flood level, and administrative procedures are established to assure that all watertight doors will be locked closed in the event of a flood warning. A requirement to close the doors protecting safety-related equipment has been included in Technical Specification Section 3/4.7.5.

As a result of our review, we conclude that the design for the protection of essential equipment from flooding and from the design basis flood meets the requirements of General Design Criterion 2, with respect to the protection of essential equipment from the effects of ground water flooding and from the design basis flood and is, therefore, acceptable.

3.5 Missile Protection Criteria

The plant seismic Category I structures, systems and components are shielded from, or are designed for, various postulated missiles. Missiles considered include tornado generated missiles and various postulated missiles generated within the plant.

Adequate information has been provided indicating the structures, shields and barriers that are designed to resist the effects of missiles. The missiles applicable to each of these structures, shields and barriers are also adequately identified and their characteristics defined.

The analysis of structures, shields and barriers to determine the effects of missile impact was accomplished in two steps. In the first step, the potential damage that could be done by the missile in the immediate vicinity of impact was investigated. This was accomplished by estimating the depth of penetration of the missile into the impacted structures. Furthermore, secondary missiles are prevented by fixing the target thickness well above that determined for penetration. In the second step of the analysis, the overall structural response of the target when impacted by a missile was determined using established methods of impactive analysis. The equivalent loads of missile impact, whether the missile is environmentally generated or accidentally generated within the plant, were combined with other applicable loads as is discussed in Section 3.8 of this report.

The design procedures used to determine the effects and loading on seismic Category I structures by design basis missiles selected for the plant provide a conservative basis for engineering design to assure adequate protection from the effects of missile impacts.

The use of these criteria for protection from postulated missiles provides reasonable assurance that resulting loads and effects will not impair the structural integrity of seismic Category I structures, or result in any loss of function of safety-related systems and components contained in such structures. We have concluded that conformance with these criteria is an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4.

3.5.1 Tornado Missile Protection

Our review for tornado missile protection was based on General Design Criteria 2 and 4. We found that all safety-related equipment necessary for a cold shutdown was adequately protected from tornado missiles with the exception of the underground diesel fuel lines leading from the emergency diesel fuel oil tank vault to the diesel generator room and the sluice gate within the emergency cooling pond intake structure.

The applicant subsequently provided additional protective covering over the diesel fuel lines in addition to the previous three-foot soil cover, and provided further protective features in the design of the emergency cooling pond intake structure.

We therefore conclude that the Arkansas Nuclear One - Unit 2 facility is adequately protected against tornado missiles and is acceptable.

3.5.2 Internally Generated Missiles

Pressurized components and rotating machines have the potential to become missile sources within the facility. Protection against such missiles is achieved by proper orientation of components and systems, by use of missile barriers and by physically separating redundant safety-related systems or components from each other and from nonsafety-related systems.

As a result of our review, we conclude that the design is in conformance with General Design Criterion 4 as it relates to structures housing essential systems and to the systems being capable to withstand the effects of internally generated missiles, Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," as it relates to protection of spent fuel pool systems and spent fuel assemblies from internal missiles, and Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power

Plants," as it relates to the design of the intake structure to withstand the effects of internal missiles and is, therefore, acceptable.

3.5.3 Turbine Missiles

The Arkansas Nuclear One - Unit 2 has a non-peninsular turbine orientation, so that the containment building, as well as other structures housing safety-related systems are within the low trajectory turbine missile strike zones. Although the risks associated with postulated turbine missiles were not reviewed at the construction permit stage, at the staff's request, the applicant presented a turbine missile analysis in the Final Safety Analysis Report. The applicant concluded that damage to safety-related equipment from potential turbine missiles is extremely unlikely, and that the unit is adequately protected. The applicant has indicated that measures such as frequent valve testing will be adopted to reduce the probability for destructive overspeed to an acceptable level.

Our review indicates that the probability of a turbine missile damaging the containment, or main steam line, could be greater than that estimated by the applicant, conservatively in the range of 10^{-5} to 10^{-6} per turbine year with respect to the proposed plant-turbine configuration. However, with satisfactory measures to protect against turbine overspeed, the probability of turbine missile damage is acceptably low. Accordingly, the ANO-2 technical specifications will include provisions for inservice inspection to minimize the likelihood of an overspeed event.

3.6 Protection Against Dynamic and Environmental Effects Associated with the Postulated Rupture of Piping

3.6.1 Piping Inside Containment

The applicant has referenced Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," as the basic document used to establish the design criteria for piping systems inside containment. In addition, the standard ANSI-N176 (draft 3, dated June 1974) is also referenced for postulating the types of breaks which occur at the break locations. The difference between Regulatory Guide 1.46 and ANSI-N176 is in the design break geometry assumptions.

The applicant also submitted a document, "Design Basis Pipe Breaks for the ANO-2 Reactor Coolant System," dated May 1976, based on the methodology of Combustion Engineering topical report CENPD-168, "Design Basis Pipe Breaks for CE Two Loop Plants," and specific ANO-2 analysis and design parameters. This document has been reviewed by the staff and found to be acceptable for the reactor coolant system piping analysis and calculation of limited break areas for determining reactor cavity pressurization.

We have reviewed the applicant's criteria for protection against dynamic effects associated with the postulated rupture of piping inside containment. Results of the review indicate that the analytical method for determining pipe motion subsequent to rupture and pipe restraint dynamic interaction is adequately described and referenced in the Final Safety Analysis Report. The design criteria used for identifying high energy fluid piping and postulating pipe break locations are consistent with the criteria of Regulatory Guide 1.46.

On the basis of information contained in the applicant's Final Safety Analysis Report we have concluded that the applicant's criteria constitutes an acceptable design basis for satisfying General Design Criteria 1, 2, 4, 14 and 15.

3.6.2 Piping Outside Containment

Plant design criteria applied to the design of the facility will accommodate the effects of postulated pipe breaks and cracks, including pipe whip, jet effects, and environmental effects. The provisions for protection against the dynamic effects associated with postulated pipe ruptures outside containment will mitigate the consequences of such ruptures so that the reactor can be shut down safely and maintained in a safe condition. The means used to protect safety-related systems and components include physical separation, enclosure within suitable designed structures, pipe whip restraints and equipment shields. Protection against pipe failure outside containment is in accordance with a letter from A. Giambusso dated December 12, 1972, "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment." The applicant has analyzed high energy piping systems for the effects of pipe whip, jet impingement and environmental effect on safety-related systems and structures.

High energy systems that were evaluated for the above effects include the main steam, main feedwater, emergency feedwater, chemical and volume control and steam generator blowdown systems. Areas evaluated to determine the effects of pipe whip and jet impingement as well as the environmental effects on the ability of safety-related systems and components to effect safe shutdown include the control room, safety-related electrical equipment, instrumentation and cabling, shutdown cooling systems, auxiliary feedwater system, and the essential portion of the service water system.

The applicant has presented an analysis on the effect of the above stated high energy line breaks outside containment on safety-related systems. The plant basis includes the ability to sustain a high energy pipe break accident coincident with a single active failure and retain the capability for safe cold shutdown. For postulated pipe failures, the resulting environmental effect will not preclude the habitability of the control room, and will not cause the loss of function of electric power supplies, controls and instrumentation needed to complete a safety action.

The applicant has also presented an analysis on the effect of the moderate energy line breaks outside containment on safety-related systems. The moderate energy systems are designed to meet the criteria set forth in Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures In Fluid Systems Outside Containment". The moderate energy systems that were analyzed include the service water system, shutdown cooling system, low pressure portion of the emergency feed-water system, containment spray system, emergency core cooling system and fire protection system. We have evaluated the analysis and conclude that a postulated pipe crack in a moderate energy line will not cause loss of function of any safety-related system.

Based on our review, we conclude that the applicant has adequately designed and protected areas and systems required for safe plant shutdown following postulated events, including the combination of pipe failure and single active failure. The plant design meets the criteria set forth in (1) A. Giambusso's letter cited above with respect to protection of safety-related systems and components from a postulated high energy line break, (2) the Branch Technical Position APCSB 3-1 with respect to protection of safety-related systems and components from a postulated moderate energy line failure and (3) Mechanical Engineering Branch Technical Position 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," with respect to the criteria used for postulating pipe rupture and leakage locations in high and moderate energy pipes outside containment, and is, therefore, acceptable.

3.7 Seismic Design

3.7.1 Seismic Input

The seismic design response spectra applied in the design of seismic Category I structures, systems, and components was developed using the procedures of Bechtel topical report BC-TOP-4, "Seismic Analysis of Structures and Equipment for Nuclear Power Plants," which has been reviewed and approved by the staff. The specific percentage of critical damping values used in the seismic analysis of seismic Category I structures, systems and components are in conformance with the recommendations of Regulatory Guide 1.61, "Damping Values for Seismic Analysis of Nuclear Power Plants." The synthetic time history used for seismic design of seismic Category I plant structures, systems and components is adjusted in amplitude and frequency content to obtain response spectra that envelop the design response spectra specified for the site.

The procedures, cited above, used to generate the design response spectra and conformance to the recommendations of Regulatory Guide 1.61 provide reasonable assurance that the seismic inputs to design analysis of seismic Category I structures, systems, and components are adequately defined so as to form a conservative basis for the design of such structures, systems and components to withstand seismic loadings.

3.7.2 Seismic System and Subsystem Analysis

Modal response spectrum multi-degree-of-freedom and normal mode-time history methods were used for the analysis of all seismic Category I structures, systems and components. The vibratory motions and the associated mathematical models account for the soil structure interaction and the coupling of all coupled seismic Category I structures and plant equipment. Governing response parameters have been combined by the square root of the sum of the squares to obtain the modal maximums when the modal response spectrum method is used. The absolute sum of responses was used for closely spaced frequencies. Horizontal and vertical floor spectra inputs used for design and test verification of structures, systems and components were generated by the normal mode-time history method. Torsional loads have been adequately accounted for in the seismic analysis of the seismic Category I structures. Vertical ground accelerations were assumed to be two-thirds of the horizontal ground acceleration and the horizontal and vertical effects were combined simultaneously. Constant vertical load factors were employed only where analysis showed sufficient vertical rigidity to preclude significant vertical amplifications in the seismic system being analyzed.

We have reviewed the Final Safety Analysis Report and applicable amendments and find the seismic system and subsystem dynamic analysis methods and procedures used by the applicant to be an acceptable basis for seismic design.

3.7.3 Seismic Instrumentation Program

The type, number, location and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude and phase relationship of the seismic response of the containment structure corresponds to the recommendations of Regulatory Guide 1.12, "Instrumentation For Earthquakes."

Supporting instrumentation will be installed on seismic Category I structures, systems, and components in order to provide data for the verification of the seismic responses determined analytically for such Category I items.

We conclude that the seismic instrumentation program proposed by the applicant complies with Regulatory Guides 1.12 and is acceptable.

3.8 Design of Category I Structures

3.8.1 Concrete Containment Structures

Reactor coolant systems are enclosed in prestressed concrete containment as described in Section 3.8.1 of the Final Safety Analysis Report. The containment structure has been designed in accordance with applicable subsections of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, and American Concrete Institute (ACI) Code 318 to resist various combinations of dead loads, live loads, environmental loads including those due to wind, tornadoes, operating basis earthquake, safe shutdown earthquake and loads generated by the design basis accident including pressure, temperature and associated pipe rupture effects.

Static analysis for the containment shell and base are based on methods previously applied. Likewise, the liner design for the containment employs methods similar to those previously accepted.

The choice of the materials, the arrangement of the anchors, the design criteria and design methods are similar to those evaluated for previously licensed plants. Materials, construction methods, quality assurance and quality control measures are covered in the Final Safety Analysis Report and, in general, are similar to those used for previously accepted facilities.

Prior to operation, the containment will be subjected to an acceptance test in accordance with the Regulatory Guide 1.18, "Structural Acceptance Test For Concrete Primary Reactor Containments," during which the internal pressure will be 1.15 times the containment design pressure.

The criteria used in the analysis, design and construction of the concrete containment structures to account for anticipated loadings and postulated conditions that may be imposed upon the structures during its service lifetime are in conformance with established criteria, codes, standards and specifications and are, therefore, acceptable.

The use of these criteria as defined by applicable codes, standard and specifications; the loads and loading combinations, the design and analysis procedures; the structural design criteria; the materials, quality control and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within and outside the containment, the structure may be expected to withstand the specified design conditions without impairment of its structure in the performance of its safety function. Conformance with these criteria codes, specifications, and standards constitutes an acceptable basis for satisfying the requirements of General Design Criteria 2, 4, 16, and 50.

3.8.2 Concrete and Steel Internal Structures

The containment interior structure consists of a shield wall around the reactor, secondary shield walls and other interior walls, compartments and floors. The major code used in the design of concrete internal structures is American Concrete Institute Code (ACI) 318-63, "Building Code Requirements for Reinforced Concrete." For steel internal structures, the American Institute of Steel Construction (AISC) specification, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," is used.

The containment concrete and steel internal structures are designed to resist various combinations of dead and live loads, accident induced loads, including pressure and jet loads, and seismic loads. The load combinations used cover those cases likely to occur and include all loads which may act simultaneously. The design and analysis procedures that are used for the internal structures are the same as those approved in previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-63 Code and in the AISC Specification for concrete and steel structures, respectively.

The containment internal structures are designed and proportioned to remain within limits established by the staff under the various load combinations. These limits are, in general, based on the ACI 318-63 Code and on the AISC Specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction, their fabrication, construction and installation, are in accordance with the ACI 318-63 Code and the AISC Specification for concrete and steel structures, respectively.

The criteria used in the analysis, design and construction of the internal structures of the containment, to account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetime, are in accordance with established criteria, codes, standards and specifications and are, therefore, acceptable.

The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; and the materials, quality control and special construction techniques; provides reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures may be expected to withstand the specified design conditions without impairment of their structural integrity in the performance of their safety function. Conformance with these criteria, codes, specifications and standards constitutes an acceptable basis for satisfying the requirements of General Design Criteria 2 and 4.

3.8.3 Other Seismic Category I Structures Other Than Containment

Seismic Category I structures other than containment and structures interior to it are built from structural steel and concrete members. The structural components consist of slabs, walls, beams and columns. The major code used in the design of concrete Category I structures was the ACI 318-63, "Building Code Requirements for Reinforced Concrete."

For steel seismic Category I structures, the AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," was used.

The concrete and steel seismic Category I structures are designed to resist various combinations of dead loads; live loads; environmental loads including winds, tornadoes, operating basis earthquakes and the safe shutdown earthquake and by loads generated by postulated ruptures of high energy pipes such as reaction and jet impingment forces, compartment pressures, and impact effects of whipping pipes.

The design and analysis procedures that are used for these seismic Category I structures are the same as those approved on previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-63 Code and in the AISC Specification for concrete and steel structures, respectively.

The various seismic Category I structures are designed and proportioned to remain within limits established by the Regulatory staff under the various load combinations. These limits are, in general, based on the ACI 318-63 Code and on the AISC Specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction, their fabrication, construction and installation, are in accordance with the ACI 318-63 Code and the AISC Specification for concrete and steel structures, respectively, except for the reinforcing steel in the walls of the spent fuel pool.

The end anchorage of this reinforcing terminated into the front face instead of the rear face of the adjoining walls. This error in detailing invalidated the wall edge conditions used in the original pool analysis. The applicant has completed a thorough re-analysis of the pool walls using the as-built configuration and found that with some additional bracing and reinforcement of an interior wall, the spent fuel pool will be safe for use and will adequately resist the imposed loadings. The applicant has tested the spent fuel pool by filling the pool with water and measuring the resultant deflections. The applicant has also inspected for cracks and other signs of distress. The measured deflections were found to be within their predicted values and no signs of distress were observed. The program also requires inspection of the pool structure after the spent fuel is loaded, and long-term monitoring of the pool structure. We have reviewed this matter and conclude that the inspection and monitoring program and the additional supports and reinforcing ensures a safe usable structure and is acceptable. We will require that the appropriate portions of the long term inspection and monitoring program be included in the ANO-2 technical specifications.

The applicant has used a mechanical connector (trade name DYWIDAG) for reinforcing steel joints that has not been previously accepted by the staff. This connector was used only in some areas of the auxiliary building where it was necessary to provide temporary construction openings in some walls and floors which necessitated joints in the reinforcing steel to permit closure of the openings. The locations of the connectors were determined in such a fashion that the loads transferred by the connectors were not the maximum loads in the reinforcing bars. In addition, the connectors used were stronger than required for the size of reinforcing bars that were spliced. The applicant has submitted sufficient laboratory test data to demonstrate the adequacy of the connectors to perform their intended function. In addition, these connectors have been used for many years in other commercial engineered structures. On the basis of our review, we find the use of these connectors acceptable for use in this particular application.

The criteria used in the analysis, design and construction of all the plant seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime, are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory staff, with the exceptions as noted above.

The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria and the materials quality control and special construction techniques provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within the structures, these structures may be expected to withstand the specified design conditions without impairment of their structural integrity in the performance of their safety function. Conformance with these criteria codes, specifications, and standards constitutes an acceptable basis for satisfying the requirements of General Design Criteria 2 and 4.

Foundations of seismic Category I structures are described in Section 3.8.5 of the Final Safety Analysis Report. Primarily, these foundations are reinforced concrete of the mat type. The major code used in the design of these concrete mat foundations is ACI 318-63. These concrete foundations have been designed to resist various combinations of dead loads; live loads; environmental loads including winds, tornadoes, operating basis earthquakes; and the safe shutdown earthquake; and loads generated by postulated ruptures of high energy pipes.

The design and analysis procedures that were used for these seismic Category I foundations are the same as those approved on previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-63 Code. The various seismic Category I foundations have been designed and proportioned to remain within limits established by the staff under the various load combinations. These limits are, in general, based on the ACI 318-63 Code modified as appropriate for load combinations that are considered extreme. The materials of construction, their fabrication, construction and installation, are in accordance with the ACI 318-63 Code.

The criteria used in the analysis, design and construction of seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory staff.

The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; and the materials, quality control and special construction techniques; provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents, the foundations may be expected to withstand the specified design conditions without impairment of their structural integrity in the performance of their safety function. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the requirements of General Design Criteria 2 and 4.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

Piping Vibration Operational Test Program

The applicant has agreed to perform a piping preoperational vibration dynamic effects test program to check the vibration performance of piping important to safety. The preoperational vibration dynamic effects test program that will be conducted on safety-related American Society of Mechanical Engineers (ASME) Class 1, 2 and 3 piping systems and their restraints during startup and the initial operating conditions constitutes an acceptable program.

This program will provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and operating modes associated with the design operational transients. The tests, as planned, will develop loads similar to those experienced during reactor operation. A commitment to proceed with such a program constitutes an acceptable design basis for satisfying the applicable requirements of General Design Criterion 15.

Seismic Qualification of Mechanical Equipment

The applicant has submitted procedures which use acceptable dynamic testing and analysis techniques to confirm the adequacy of seismic Category I mechanical equipment, including their supports, to function during and after an earthquake of magnitude up to and including the Safe Shutdown Earthquake. Subjecting the equipment and supports to these dynamic testing and analysis procedures provides reasonable assurance that in the event of an earthquake at the site, the seismic Category I mechanical equipment will continue to function during and after a seismic event.

Implementation of these dynamic testing and analysis procedures constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 14.

3.9.2 ASME Code Class 2 and 3 Components

All seismic Category I pressure retaining systems, components and equipment outside of the reactor coolant pressure boundary, including active pumps and valves, are designed to sustain normal loads, anticipated transients, the operating basis earthquake, and the safe shutdown earthquake within stress limits which are consistent with those outlined in Regulatory Guide 1.48, "Design Limits and Loading Combinations."

The specified design basis combinations of loading as applied to the design of the safety-related American Society of Mechanical Engineers (ASME) Code Class 2 and 3 pressure-retaining components in systems classified as seismic Category I provide reasonable assurance that in the event an earthquake should occur at the site, or an upset, emergency or faulted plant transient should occur during normal plant operation, the resulting combined stresses imposed on the system components may be expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without gross loss of structural integrity. The design load combinations and associated stress and deformation limits specified for all ASME Code Class 2 and 3 components, including the active pumps and valves, constitute an acceptable basis for design and satisfying General Design Criteria 1, 2 and 4.

The applicant's operability assurance program for active ASME Class 2 and 3 seismic Category I pumps and valves includes component testing, or a combination of tests and predictive analysis supplemented by seismic qualification testing of motors, operators, and component appendages. This program provides assurance that such components can withstand postulated seismic loads in combination with other significant loads without loss of structural integrity, and can perform the "active" function (i.e., valve closure or opening or pump operation) when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated. The applicant's component operability assurance program constitutes an acceptable basis for implementing the requirements of General Design Criterion 1 as related to operability of ASME Code Class 2 and 3 active pumps and valves.

The criteria used in developing the design and mounting of the safety and relief valves of ASME Code Class 2 systems will provide adequate assurance that, under discharging conditions, the resulting stresses are expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of their function.

The criteria used for the design and installation of overpressure protection devices in ASME Code Class 2 systems are consistent with Regulatory Guide 1.67, "Installation of Overpressure Protection Devices" and constitute an acceptable design basis in meeting the applicable requirements of General Design Criteria 1, 2 and 4.

3.9.3 Reactor Vessel Supports and Reactor Internals

We were informed on May 7, 1975 by a licensee of a pressurized water reactor, Virginia Electric and Power Company, that an asymmetric loading resulting from a postulated pipe rupture at a particular location in the reactor coolant loop had not been taken into account in the original design analysis of the reactor coolant system and reactor internals (which in this case included the fuel elements) for North Anna, Units 1 and 2 (Docket Nos. 50-338 and 50-339). This loading results from the forces induced on the internals within the reactor vessel by transient differential pressure conditions within the vessel during the postulated pipe rupture. In addition, the asymmetric loading from the transient differential pressures that would exist around the exterior of the reactor vessel from the same postulated pipe rupture were not included in the original design analysis. However, the symmetric loadings from such a pipe rupture were included in the original analysis of the reactor coolant system and reactor internals.

The maximum load calculated on the reactor vessel supports occurs for a postulated break at or very near the cold leg of the reactor pressure vessel nozzle. The maximum loading on the internals results from postulated ruptures in the cold leg between the reactor vessel nozzle and the primary coolant pump. These postulated ruptures may result in pipe whip, thrust, jet impingement, reactor cavity asymmetric pressure and internals response loads.

The term "reactor internals response loads" refers more specifically to the dynamic response of the reactor vessel internals to a very short time pressure differential that would develop across the core barrel if the postulated break occurred. This pressure differential would travel across and down the core barrel, and then up through the reactor internals resulting in the asymmetric loads on the core barrel. These loads are transferred through the core barrel flange to the vessel supports. In addition, the transient pressure distribution in the reactor vessel cavity results in asymmetric loads on the reactor pressure vessel which in turn are transferred to the supports.

It is our opinion that the question related to the adequacy of the reactor coolant system and internals is generic in nature and therefore, applies to all pressurized water reactor facilities.

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 will require 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 currently indicates that the response to our information requests will be completed in the latter part of 1977.

The applicant has informed us that the hydraulic blowdown model to be used to determine the loads on the reactor coolant system components will be in accordance with the model described in the Combustion Engineering, Inc., topical report CENPD-42, "Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Conditions."

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 force to be included by Combustion Engineering, Inc. in a description of the CEFLASH-4B code. The submittal of the CEFLASH-4B thermal hydraulic model is currently scheduled for December 1977. We expect to complete our review of the CEFLASH-4B methodology by the end of 1978. It is Combustion Engineering's position that the calculational methodology of CENPD-42 will be shown to be conservative by the use of the CEFLASH-4B methodology.

At the time of completion of our review of the new Combustion Engineering thermal hydraulic model represented by the CEFLASH-4B code we will require the applicant to show that the loads which have previously been calculated for the ANO-2 reactor coolant system components using the CENPD-42 thermal hydraulic methodology are conservative when compared to loads calculated by the staff approved CEFLASH-4B methodology. We will condition the operating license to require that in the event the conservatism of the existing ANO-2 loss-of-coolant accident analyses is not sufficiently demonstrated by comparison with the newly approved CEFLASH-4B methodology, the applicant will (1) utilize the approved CEFLASH-4B model in determining the loads on the ANO-2 reactor coolant system and internals, (2) determine the effects of the loads by computing the resultant stresses and strains and the actual margins to failure in the reactor coolant system components and in the reactor vessel internals and (3) demonstrate that the integrity of the fuel assembly components is maintained throughout the loading caused by the loss-of-coolant accident. In the event the conservatism of the existing ANO-2 analyses is not adequately demonstrated, as discussed above, we will also require the applicant to provide: (1) justification that the fuel can be adequately cooled if there is significant calculated grid deformation, or (2) modification of the fuel spacer grids to include the structural capability to withstand the combined loading of a safe shutdown earthquake and a loss-of-coolant accident.

We intend to condition the operating license to require that the determination of the conservatism of the ANO-2 analyses and the evaluation utilizing the CEFLASH-4B methodology, if found to be necessary, be completed by the applicant within an eighteen month period of the date of the staff's approval of the CEFLASH-4B methodology. Upon receipt and review of the information at that time we will determine what modifications to ANO-2, if any, will be required to assure that acceptable margins of safety are maintained. If modifications are necessary we will require the applicant to implement them.

Until the completion of our generic review, we have concluded that it is acceptable to continue the licensing of facilities for operation because of the low probability of the rupture of the reactor coolant system cold leg between the reactor vessel nozzle and primary coolant pump, as would be required to develop loads of significant magnitude.

3.10 Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment

The applicant has stated that the safety-related electrical equipment has been seismically qualified by prototype tests or by analysis and that the test program meets the requirements of IEEE Std 344-1971, "Seismic Qualification for Class I Electric Equipment for Nuclear Power Generating Stations." The applicant has described the seismic qualification program, summarized the test and/or analysis results, and identified the test documentation. A portion of this information was included in the sections of the Combustion Engineering Topical Report CENPD-182, "Seismic Qualification of CE Instrumentation and Electrical Equipment," which are applicable to ANO-2.

The applicant's seismic qualification program for seismic Category I instrumentation and electrical equipment is not entirely consistent with the current staff position that adequate additional assurance must be provided to assure that equipment qualified under the criteria of IEEE Standard 344-1971 will function when subjected to seismic excitation. The applicant was advised in a meeting on July 30, 1975 in Bethesda, Maryland with the staff that a certain number of selected electrical components may have to be retested in order to meet current staff requirements for seismic qualification of electrical equipment. An NRC task group visited the ANO-2 facility on December 3, 1975 and April 29, 1976 to determine specifically which of the electrical equipment within the balance-of-plant scope and the nuclear steam supply system scope, respectively, would have to be requalified. Much of this equipment was tested to criteria which are comparable to the IEEE Standard 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class IE Equipment For Nuclear Power Generating Stations." Therefore our inspection focused on the equipment which was not tested using multifrequency and multi-axis testing methods and on the equipment which was qualified by analysis.

Subsequent to the meetings noted above, the staff has participated in additional meetings with the applicant and has made additional requests for information. The remaining outstanding items of our review of the seismic qualifications of the ANO-2 seismic Category I equipment are identified in detail in our letter to the applicant dated September 7, 1977.

Based on our review of the applicant's test procedures and test results, we have concluded that the seismic qualifications of the electrical safety-related equipment is acceptable, subject to the resolution of those items identified in our letter to the applicant dated September 7, 1977. We will report our evaluation of these outstanding items in a supplement to this report.

3.11 Environmental Design of Mechanical and Electrical Equipment

The applicant has identified and classified all safety-related equipment according to the environmental design categories depending upon their location and functional requirements. The design criteria for safety-related equipment installed inside the containment which must operate during and subsequent to an accident, are that this equipment shall be capable of functioning under the accident and post-accident temperature, pressure, humidity, radiation and chemical environment conditions for the time periods required. The applicant has documented that reactor protection system and engineered safety feature equipment are qualified for use under the required environmental service conditions in accordance with IEEE Standard 323-1971, "IEEE Trial Use Standard, General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations."

During our review, the applicant was requested to submit test procedures and test results for equipment qualification of the reactor protection system, engineered safety feature actuation system and selected balance-of-plant safety-related Class IE equipment. In response, the applicant submitted test procedures and test results for the equipment qualification of the reactor protection system and engineered safety features actuation system, and submitted additional information to support their qualification program for the balance-of-plant safety-related equipment.

We have reviewed the above test procedure and the supplemental information presented and conclude that the qualification program is acceptable in part. To enable the staff to complete the evaluation, the applicant was requested to submit the qualification test procedures and test results for 1) the Fisher Porter transmitters, 2) the Agastat relays, and 3) the Potter-Brumfield relays. Recently the applicant informed the staff that the qualification testing for some of the above-mentioned items is in the process of being completed and the information requested will be submitted shortly.

Also our review revealed that for a main steam line break accident, a temperature transient inside containment may exceed the design temperature for which the Class

IE instrumentation and controls have been qualified. Although the staff recognizes that this transient exists for a short period of time, it is not apparent that the Class IE safety equipment inside containment will retain their functional integrity when subjected to this transient which involves temperatures about 410 degrees Fahrenheit for about 100 seconds. Therefore the applicant was requested to identify all the Class IE equipment that would be subjected to this transient and submit analysis or test results that demonstrates that this equipment will retain its functional integrity during this transient. In response to our concerns the applicant submitted an analysis summary, based on slab modeling techniques, to show that the effects of this transient of short duration superheated steam on all safety-related Class 1E equipment is negligible with respect to the functional operability of the equipment. This analysis is currently being reviewed by the staff. Further evaluation and discussion of the main steam line break environmental qualification envelope is provided in Section 6.2.1 of this report.

Based on our review, we conclude that the qualification of the safety-related equipment satisfies the Commission's requirements as stated in Section 7.1 of this report and is acceptable conditioned only on the satisfactory resolution of the items identified above. We will report the results of the evaluation of the outstanding items identified above in a supplement to this report.

4.0 REACTOR

4.1 Summary Description

The reactor core design presented in ANO-2 Final Safety Analysis Report is similar to that reviewed and approved for the Calvert Cliffs 1 and 2 plants, except that the ANO-2 plant will use fuel assemblies with a 16x16 fuel rod array, while the Calvert Cliffs cores were made up from 14x14 fuel rod assemblies. The initial power of the ANO-2 core is 2815 megawatts thermal, compared to the initial power level of 2560 megawatts thermal for Calvert Cliffs 1 and 2 plants.

The ANO-2 core is cooled and moderated by light water at a pressure of 2250 pounds per square inch absolute. The moderator coolant contains boron as a neutron absorber. The concentration of boron in the coolant is varied as required to control reactivity changes including the effects of fuel burnup. The reactor coolant is recirculated through the core by reactor coolant pumps. Heat transfer from the primary to the secondary system is accomplished in two steam generators. The steam generators are vertical shell U-tube type evaporators with integral moisture separating equipment.

Power generation in the reactor core is controlled by the control element assemblies each consisting of a group of five individual absorber rods connected at the top end to a spider structure which couples to a control element drive mechanism extension shaft. These assemblies are of two types, those with rods containing full length absorber material to control the reactivity of the core under operating conditions and those with rods containing part length absorber section to control axial power distribution.

4.2 Mechanical Design

4.2.1 Fuel

The Arkansas Nuclear One, Unit 2 core is composed of 177 fuel assemblies and 81 control element assemblies. The fuel assemblies are arranged to approximate a right circular cylinder with an equivalent diameter of 123 inches and an active length of 150 inches. The fuel assembly, which provides for 236 fuel rod positions, consists of five guide tubes welded to spacer grids and is secured at the top and bottom by end fittings. The guide tubes each displace four fuel rod positions and provide channels which guide the control element assemblies over their entire length of travel. In selected fuel assemblies, the central guide tube houses in-core instrumentation.

Each control element assembly consists of five neutron absorber rods assembled in a square array with one rod in the center. The rods are connected to a spider structure, which is coupled to the control element drive mechanism shafting. There are a total of 81 control element assemblies; 73 are full length and eight contain part length neutron absorber columns.

The fuel is low enrichment uranium dioxide in the form of pellets. These pellets are placed in Zircaloy tubing, which is pressurized with helium gas. The high pressure helium aids the heat transfer between the fuel and the cladding and also minimizes cladding creepdown. The fuel rods are positioned by eleven Zircaloy-4 spacer grids of the leg-spring type and a bottom grid of Inconel 625.

The Arkansas Nuclear One, Unit 2 reactor will be the first to use Combustion Engineering 16x16 fuel. This fuel will be longer than the previous Combustion Engineering 14x14 design. Because of this and the fact that there will be more fuel rods per fuel assembly, the fuel rods will operate at lower linear heat generation rates. The cladding also has a larger thickness-to-diameter ratio than the 14x14 design.

Evaluation of the Combustion Engineering 16x16 fuel design has been based upon engineering analyses, mechanical tests, and in-reactor operating experience. Additionally, the performance of the design will be subject to continuing surveillance in Arkansas Nuclear One, Unit 2 and in other operating reactors manufactured by Combustion Engineering. These programs will continually provide confirmatory and current design performance information.

Combustion Engineering and Arkansas Power and Light Company will conduct a surveillance program that involves six precharacterized assemblies in the first core. Two precharacterized fuel assemblies will be placed in each fuel region. Since ANO-2 will be the first plant to reload the 16x16 fuel assemblies, we have required that Arkansas Power Light Company perform a supplemental surveillance program that includes a 100 percent visual inspection of all fuel assemblies as they are withdrawn from the core after use. A description of this program is provided in Section 4.2.1.1.10 of the Final Safety Analysis Report. This supplemental surveillance program will be a proof test to give final assurance that no long-term detrimental behavior has occurred.

One of the major thermal analysis considerations reviewed by the staff is related to fuel densification. The initial density of the fuel pellets and the size, shape and distribution of pores within the fuel pellet influence the densification phenomenon. The effects of densification on the fuel rod will increase centerline temperature and the stored energy, increase the linear thermal output, increase the probability for local power spikes (augmentation), and increase the potential for cladding collapse.

Combustion Engineering has conducted an extensive study of fuel densification and has developed a conservative time-dependent description of the densification process as described in the Combustion Engineering topical report CENPD-118, "Densification of Combustion Engineering Fuel," dated June 1974. These densification kinetics along with data on fuel swelling, thermal expansion, fission gas release, fuel relocation, thermal conductivities, cladding creep, and other properties, have been combined in a detailed fuel performance evaluation model, which is presented in the Combustion Engineering topical report CENPD-139, "Combustion Engineering Fuel Evaluation Model," dated July 1, 1974. This model is used to calculate fuel temperature and stored energy, changes in linear thermal output and augmentation (power spike) factors. We have reviewed CENPD-139 and concluded that the fuel performance evaluation model is a generically acceptable method of describing the fuel behavior, as discussed in our acceptance letter to Combustion Engineering, dated December 4, 1974, and this model is applicable to the Arkansas Nuclear One, Unit 2 fuel. There are several reasons for applicability of the generic model: (1) the specific fuel fabrication process is tied to the densification model through resintering tests, which are used to determine the amount of incore fuel densification, and (2) the thermal performance computer code is compared with a body of experimental data whose design parameters include those of the Combustion Engineering fuel.

The Combustion Engineering System 80 fuel of the ANO-2 type has been demonstrated to densify very little and, therefore, should not be prone to form axial gaps between the fuel pellets during densification. Combustion Engineering has submitted for review a computer code that will calculate time-to-collapse of Zircaloy cladding in a pressurized water reactor environment. This code is described in the report CENPD-187, "CEPAN Method of Analyzing Creep Collapse of Oval Cladding." We have reviewed this code and found it acceptable. The applicant has performed time-to-cladding collapse calculations using the CEPAN code and the worst case combination of material properties and component dimensions including the allowable manufacturing tolerances. We conclude that based on the results of these calculations which indicate that cladding collapse will not occur during the design life of the fuel and the conservative usage of the input data that there is adequate assurance that cladding collapse will not occur for the ANO-2 fuel.

An important aspect of the behavior of the reactor core during a loss-of-coolant accident is the calculation of the combined loads on the fuel due to blowdown forces and design-basis earthquake. The applicant has submitted a topical report, CENPD-178, "Structural Analysis of Fuel Assemblies for Combined Seismic and Loss-of-Coolant Accident Loading" addressing this matter. The applicant has also submitted the results of structural tests on the fuel assembly to determine the dynamic load deflection characteristics and damage limits. This matter is discussed further in Section 3.9.3 of this report.

Mechanical tests to demonstrate the effects of flow-induced vibration and consequent fretting and corrosion have been performed on test assemblies and on full size (14x14) fuel assemblies to demonstrate that flow induced vibration fretting or wear is acceptably low. Similar full scale hot flow testing of 16x16 assemblies has been performed to substantiate these results for the new 16x16 design. The staff has reviewed the applicant's summary report No. PED-76-003P prepared by Combustion Engineering Incorporated on the results of the flow test of a 16x16 fuel assembly. The submittal was adequate with the exception that insufficient information was provided on determination of loss coefficients for fuel assembly entrance, exit and spacer grids. In response to our request for information, the applicant provided additional information and test data which acceptably confirms that the local loss coefficients for the spacer grids and the fuel assembly entrance and exit are consistent with the design values used in the thermal hydraulic analyses.

We conclude that Arkansas Power and Light Company has acceptably established the basis for its design on previous experience of Combustion Engineering fuel and various generic studies performed by Combustion Engineering.

4.2.2 Reactor Vessel Internals

The materials for construction of components of the reactor internals have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code.

The materials for reactor internals exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion on all materials is expected to be negligible.

The controls imposed on reactor coolant chemistry provide reasonable assurance that the reactor internals will be adequately protected during operation from conditions which could lead to stress corrosion of the materials and loss of component structural integrity.

The requirements and controls on welding processes provide reasonable assurance that no deleterious hot cracking will be present during the assembly of austenitic stainless steel components. All weld filler metal will be of selected composition to produce welds with at least five percent delta ferrite. Tests and examinations in accordance with Section III of the ASME Code, Summer 1972 addenda, assure that adequate delta ferrite levels are met.

The controls imposed to avoid sensitization during fabrication and processing of austenitic stainless steels satisfy the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

We have concluded that the material selection, fabrication practices, examination procedures, and protection procedures described by the applicant provide reasonable assurance that the austenitic stainless steel used for reactor internals will be in a metallurgical condition which precludes susceptibility to stress corrosion cracking during service. The use of materials proven to be satisfactory by actual service experience and conformance with the recommendations of the Regulatory Guide constitutes an acceptable basis for meeting the applicable requirements of General Design Criteria 1 and 14.

With regard to flow-induced vibration testing of reactor internals, the applicant has proposed Fort Calhoun and Maine Yankee reactor internals test programs as the established prototype for ANO-2. The applicant has further proposed to conduct an augmented reactor internals inspection program in addition to full compliance with the guidelines established in Regulatory Guide 1.20, "Vibration Measurements on Reactor Internals," for other than prototype plants. To provide continuous assurance of reactor internals structural integrity and performance, the applicant will install and operate a neutron noise monitoring system at the ANO-2 facility.

The applicant's augmented inspection program, in addition to vibration testing and subsequent visual inspection as part of the ANO-2 preoperational tests, provides added confirmation of the capability of the structural elements of the reactor internals to sustain flow-induced vibrations. The proposed program exceeds the requirements of Regulatory Guide 1.20, "Vibration Measurements on Reactor Internals" for nonprototype plants.

We have reviewed the preoperational vibration test program proposed by the applicant for verifying the design adequacy of the reactor internals under loading conditions that will be comparable to those experienced during operation. The comparison of tests, from the Fort Calhoun and Maine Yankee plants, the predictive analysis provided in the ANO-2 Final Safety Analysis Report, augmented reactor internals inspection and continuous noise monitoring will provide adequate assurance that the reactor internals can be expected to withstand flow-induced vibrations without loss of structural integrity during their service lifetime. We have concluded that the proposed preoperational vibration test program, together with augmented reactor internals inspection and continuous noise monitoring, constitutes an acceptable basis for demonstrating the design adequacy of General Design Criteria 1 and 4.

4.2.3 Control Rod System Structural Materials

The mechanical properties of structural materials selected for the control rod system components exposed to the reactor coolant satisfy Appendix I of Section III of the ASME Code, or Part A of Section II of the Code, and also the staff position that the yield strength of cold worked austenitic stainless steel should not exceed 90,000 pounds per square inch.

The requirements and controls on welding processes provide reasonable assurance that no deleterious hot cracking will be present during the assembly of austenitic stainless steel components. All weld filler metal will be of selected composition to produce welds with at least five percent delta ferrite; and tests and examinations in accordance with Section III of the ASME Code, Summer 1972 addenda, will assure that adequate delta ferrite levels are met.

The controls imposed in the application and processing of austenitic stainless steels to avoid sensitization satisfy the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices performed in accordance with these recommendations provide added assurance that stress corrosion cracking will not occur during the design life of the components.

The compatibility of all materials used in the control rod system in contact with the reactor coolant satisfies the criteria for Articles NB-2160 and NB-3120 of Section III of the Code. Both martensitic and precipitation-hardening stainless steels have been given tempering or aging treatments in accordance with staff positions.

Conformance with the codes and Regulatory Guide indicated above, and with the staff positions on the allowable maximum yield strength of cold worked austenitic stainless steel and minimum tempering or aging temperatures of martensitic and precipitation-hardened stainless steels, constitutes an acceptable basis for meeting the requirements of General Design Criterion 26.

4.2.4 Reactivity Control System

Reactor power can be controlled by movement of control rods and a soluble chemical neutron absorber (boric acid). The reactor control system directs the control element drive mechanism to insert, hold, withdraw, or trip the control element assemblies. The chemical and volume control system provides the second means, by varying the concentration of boric acid in the coolant to effect relatively slow reactivity changes.

The rod control system consists of 73 clusters of full length rods and eight clusters of part length rods to shape the reactor power distribution, and to compensate for changes in reactivity resulting from fuel burnup. Each cluster has five absorber rods. A rod cluster control assembly comprises a group of individual neutron absorber rods fastened at the top end to a common spider assembly. The absorber material used in the control rods is boron carbide, which is black to thermal neutrons.

The full length rod cluster control assemblies are divided into two groups: control and shutdown. The control group compensates for reactivity changes due to variations in operating conditions of the reactor, i.e., power and temperature variations. The control and shutdown groups provide adequate shutdown margin, in case of reactor trip. Shutdown margin is defined as the amount by which the core would be subcritical at hot shutdown conditions if all rod cluster control assemblies are tripped assuming that the highest worth assembly remains fully withdrawn and assuming no changes in xenon or boron concentration or part length rod cluster control position.

The function of the part length rods is to control axial neutron flux shape and axial xenon oscillations should they occur. The part length rods are on manual control.

The soluble neutron absorber (boric acid) is varied to control long-term reactivity changes. These long-term reactivity changes consist of (1) fuel depletion and fission product buildup, (2) cold to hot, zero power reactivity change, (3) reactivity change produced by intermediate term fission product buildup such as xenon and samarium, and (4) due to burnable poison depletion.

Based on our review of the design criteria applied and the tests performed on the reactivity control system of ANO-2 we conclude that reasonable assurance is provided that this system may be expected to withstand the imposed loads associated with normal reactor operation, anticipated operational transients, postulated accidents, and seismic events without gross loss of structural integrity or impairment of function. Compliance with these design criteria forms an acceptable basis for satisfying the mechanical reliability stipulations of General Design Criterion 27.

4.3 Nuclear Design

The nuclear design of the ANO-2 reactor is in many respects similar to the Combustion Engineering System 80 design previously reviewed and approved by the staff. The principal difference is that the ANO-2 core consists of 177 fuel assemblies whereas the System 80 design utilizes 241 fuel assemblies. ANO-2 will be the first operating plant to employ Combustion Engineering's 16x16 fuel rod assembly design. The core average linear heat generation rate at 100 percent of rated power is 5.53 kilowatts per foot.

4.3.1 Design Bases

We have reviewed the design bases used by the applicant to establish the core design and the designs of the reactivity and power distribution control systems. We have established that these design bases are consistent with all applicable General Design Criteria of 10 CFR 50, Appendix A. Those design bases that are important to the safety of the plant are discussed below.

4.3.2 Power Distribution Control

The applicant's basis for power distribution control is that the power distributions produced during all phases of normal operation are no worse than those assumed as initial conditions in the safety analyses. Specifically, the peak linear heat generation rate must be maintained below the value used as the initial condition in the loss-of-coolant accident analysis. This value, 13.0 kilowatts per foot, coincides with a limit on the total peaking factor of 2.30. Also, the power distribution must be controlled to maintain the departure from nucleate boiling ratio initial condition in the loss of flow analysis and certain rod drop analyses. This departure from nucleate boiling ratio limit is dependent on the location of the axial peak as measured by the axial shape index. It varies from a minimum of 1.50 to a maximum of 2.10. Further discussion of the derivation of this limit is presented in Section 15.0 of this report.

The applicant has performed extensive power distribution calculations to demonstrate that the design limits described above can be met during normal operation. These calculations simulated the reactor behavior during both steady-state operation and during typical load-following maneuvers. The results of these calculations show that the maximum steady-state peaking factor, excluding uncertainties, is 1.76. This value occurs at beginning-of-life.

For comparison with the limit of 2.30 given above, the calculated value must be augmented by appropriate uncertainties and allowances. These consist of the engineering factor, an augmentation factor to account for fuel densification effects, and an uncertainty allowance to account for calculational errors. The applicant has supplied an estimate of the calculational uncertainty which we are reviewing as a part of our overall evaluation of the core protection calculator system. Pending completion of our review, we have established that a value of 1.10 is acceptably conservative.

$$F_Q^T = F_Q^N \times 1.03 \times 1.03 \times 1.10$$

Where:

F_Q^N is the maximum steady state peaking factor excluding uncertainties,
F_Q^T is the total peaking factor

Engineering Factor = 1.03

Augmentation Factor = 1.03

Calculational Uncertainty Factor = 1.10

Using the equation, the maximum total peaking factor under steady-state conditions is 2.05. The difference between 2.05 and 2.30 represents margin to accommodate load following operations.

Similarly, we have compared the expected (calculational) axial and radial power distributions for both steady state and load following with the design power distributions used to evaluate the overpower margin (percentage increase in power relative to rated power required to produce a minimum departure from nucleate boiling ratio assumed in the safety analyses. Our review indicates that the expected distributions, including uncertainties, produce greater margin than do the design distributions.

The applicant has recently submitted revised loss-of-coolant accident analyses which assume an initial condition value of 14.5 kilowatts per foot as compared to the value of 13.0 kilowatts per foot discussed above. As we state in further detail in Section 6.3 of this report, we will report our evaluation of the revised loss-of-coolant accident analyses in a supplement to this report.

Core Operating Limit Supervisory System

The applicant plans to employ a new reactor monitoring system, designated the Core Operating Limit Supervisory System (COLSS), to continuously monitor important reactor characteristics and establish margins to operating limits. This system, which consists of software executed on the plant computer, will utilize the output of the incore detector system to synthesize the core average axial power distribution. Rod positions taken from the control rod position indication system, together with pre-calculated radial peaking factors, will be used to construct axially dependent, radial power distributions. Using this information, together with measured primary coolant flow, pressure and temperature, COLSS will establish the margin to the operating limits on maximum linear heat generation rate and minimum departure from nucleate boiling ratio. The system will also monitor azimuthal flux tilt and total power level and generates an alarm if any of these limits are exceeded. The margins to all of these limits except azimuthal tilt are continuously displayed to the operators; the tilt can be displayed at the request of the operator. The operator will monitor these margins and take corrective action if the limits are approached. These actions include improving power distribution by moving full length or part length rods, reducing power or by changing thermal hydraulic conditions, i.e., coolant inlet temperature and primary system pressure.

A description of the COLSS algorithms and an uncertainty analysis of the calculations performed by COLSS is presented in Combustion Engineering Topical Report CENPD-169-P "COLSS-Assessment of the Accuracy of PWR Operating Limits as Determined By the Core Operating Limit Supervisory Systems" dated July 1975. This report is still under review by the staff. However, we have reviewed the algorithm in enough detail to allow us to conclude that the methods employed in COLSS to determine power distributions are acceptable. The axial power distribution synthesis methods are the same as those used at existing Combustion Engineering plants for periodic processing of incore detector data. Similarly, the use of precalculated information to determine radial peaking factors is consistent with the approach now used to establish monitoring limits on existing reactors.

Most of our current COLSS review effort is directed toward establishing the uncertainty associated with COLSS processing. This evaluation includes determining the combined effect on the COLSS output of uncertainties in measured precalculated in put and errors associated with the use of the calculational techniques employed in COLSS. We will report on the results of this review in a supplement to this safety evaluation.

Reactivity Coefficients

The reactivity coefficients are expressions of the effect on neutron multiplication of changes in such core conditions as power, temperature, pressure, and void content. These coefficients vary with fuel burnup. The applicant has presented calculated values of these coefficients and has also evaluated the accuracy of these calculations.

We have reviewed the calculated values of the reactivity coefficients and have concluded that they adequately represent the full range of expected values. We have also concluded that the reactivity coefficients used in the safety analysis conservatively bound the expected values including uncertainties.

The predicted total power coefficient is strongly negative for all reactor conditions through core life thus satisfying the requirements of Criterion 11 of the General Design Criteria. The applicant will measure the moderator temperature coefficient and the power coefficient during startup tests to check the calculated values and to further ensure that conservative coefficient values were used in the accident analysis.

Control

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup and fission product buildup, a significant amount of excess reactivity will be built into the core. The applicant has provided sufficient information relating to core reactivity balance for the first core and has shown that means are incorporated into the design to control excess reactivity at all times.

Control of both excess reactivity and power level will be achieved with movable control element assemblies and through the variation of boron concentration in the reactor coolant. Calculations made by the applicant show that sufficient additional control element assembly worth has been provided to accommodate the reactivity effects of the steam line break accident at any time during the core life allowing for the most reactive control element assembly stuck in the fully withdrawn position and also allowing for calculational uncertainties. These worths will be verified during startup tests. In addition, the chemical and volume control system will be capable of shutting down the reactor by adding soluble boron poison, maintaining the reactor at least five percent subcritical when refueling. This combination of control systems satisfies the requirement of Criterion 26 of the General Design Criteria.

The plant will be operated at steady-state full power with only one bank of the full-length control element assemblies slightly inserted. Limited insertion of the full-length control rods will permit compensating for fast reactivity changes (e.g., that required for power level changes and for the effects of minor variations in moderator temperature and boron concentrations) without impairing shutdown capability.

Rod insertion will be controlled by the power-dependent insertion limits that will be given in the Technical Specifications. These limits will (1) ensure that there is sufficient negative reactivity available to permit the rapid shutdown of the reactor with ample margin, and (2) ensure that the worth of a control rod that might be ejected in the unlikely event of an ejected rod accident will be no worse than that assumed in the accident analysis.

Soluble boron poison will be used to compensate for slow reactivity change including those associated with fuel burnup, changes in xenon and samarium concentration, buildup of long-life fission products, burnable poison rod depletion, and the large moderator temperature change from cold shutdown to hot standby. The soluble boron poison system will provide the capability to take the reactor at least ten percent subcritical in the cold shutdown condition.

We have reviewed the calculated rod worths and the uncertainties in these worths, based upon appropriate comparison of calculations with experiments. On the basis of our review, we have concluded that the applicant's assessment of reactivity control is suitably conservative, and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability assuming the most reactive control element assembly is stuck in the fully withdrawn position. We have concluded that the control element assembly and soluble boron worths are acceptable for use in the accident analysis.

Stability

The stability of the reactor to xenon-induced power distribution oscillations and the control of such transients have been discussed by the applicant. Due to the negative power coefficient, the reactor is inherently stable to oscillations in total reactor power.

The core will be stable to axial xenon oscillations at beginning-of-life but will become unstable as burnup progresses. The applicant has provided sufficient information to show that axial oscillations will be detected and controlled before any safety limits are reached, thus preventing any fuel damage. The core will be stable to both radial and azimuthal xenon oscillations throughout core life.

Analytical Methods

The applicant has described the computer programs and calculational techniques used to calculate the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of these methods to predict experimental results. We conclude that the information presented adequately demonstrates the ability of these analytical methods to calculate the reactor physics characteristics of the ANO-2 core.

4.4 Thermal and Hydraulic Design

The principal criterion for the thermal-hydraulic design of a reactor is avoidance of thermally induced fuel damage during normal steady-state operation and during anticipated operational occurrences. The applicant used the following design limits to satisfy this criterion:

(1) The margin to departure from nucleate boiling will be chosen to provide a 95 percent probability with 95 percent confidence that departure from nucleate boiling will not occur on a fuel rod having the minimum departure from nucleate boiling ratio during normal operation and any anticipated operational occurrence. The preliminary design used a minimum allowable limit of 1.30 on the departure from nucleate boiling ratio.

(2) Operating conditions are selected to ensure hydraulic stability within the core, thereby preventing premature departure from nucleate boiling ratio.

(3) The peak temperature of the fuel will be less than the melting point (2805 degrees Centigrade unirradiated and reduced by 32 degrees Centigrade per 10,000 megawatt days per metric ton uranium) during normal operation and anticipated operational occurrences.

The thermal and hydraulic design parameters for the reactor are listed in Table 4.1 of this report. A comparison of these parameters with those of Calvert Cliffs 1 & 2 is given in the table. The ANO-2 design shows an increase in total power and a decrease in flow rate and core flow area. Within the reactor vessel the internals design is basically the same except smaller for those components affected by the smaller core.

The applicant has submitted supplementary information on reactor vessel flow model testing in Amendment 35 to the Final Safety Analysis Report. The applicant has established representative values for the flow distribution parameters for ANO-2 by interpolation of test data obtained from reactor vessel flow model test on earlier Combustion Engineering reactor geometric configurations. The staff finds the values used to be acceptable for the ANO-2 design.

TABLE 4.1

REACTOR DESIGN COMPARISON

THERMAL AND HYDRAULIC DESIGN PARAMETERS (NOMINAL)	ANO-2	Calvert Cliffs 1 & 2
Performance Characteristics		
Reactor Core Heat Output, thermal megawatts	2815	2560
Reactor Core Heat Output, millions of British thermal units per hour	9608	8737
System Pressure, pounds per square inch absolute	2250	2250
Minimum DNBR at Nominal Conditions (full power)	2.26	2.18
Coolant Flow		
Total Flow Rate, millions of pounds per hour	120.4	128.8
Average Velocity Along Fuel Rods, feet per second	16.4	14.2
Average Mass Velocity, millions of pounds per hour per square foot	2.60	2.33
Coolant Temperature, degrees Fahrenheit		
Nominal Inlet	553.5	543.4
Vessel Outlet	612.0	595.0
Average in Vessel	582.75	569.2
Nominal Outlet of Hot Channel (full power)	652	642.9
Heat Transfet at 100 percent Power		
Active Heat Transfer, Surface Area, square feet	51,000	48,400
Average Heat Flux, British thermal units per hour per square foot	182,200	178,000
Maximum Heat Flux, British thermal units per hour per square foot	425,800	527,900
Average linear heat rate of fuel rod only, kilowatts per foot	5.34	5.94
Maximum Clad Temperature, degrees Fahrenheit		
Clad Surface at Nominal Pressure	657	657
Fuel Temperature, degrees Fahrenheit		
Maximum at 100 percent Power	3420	4170
CORE MECHANICAL DESIGN PARAMETERS		
Fuel Rod Array	16x16	14x14
Number of Fuel Assemblies	177	217
Fuel Rods per Assembly	224-236	164-176
Fuel Assemblies Overall Dimensions, inches	7.980 x 7.980	7.980/7.980
Number of Spacer Grids per Assembly	12	8
Fuel Rods		
Number	40,716	36,896
Outside Diameter, inches	0.382	0.440
Clad Thickness, inches	0.025	0.026
Clad Material	Zircaloy 4	Zircaloy 4
Fuel Pellets		
Material	Sintered Pellets	Sintered Pellets
Length, inches	0.390	0.650

TABLE 4.1 (Continued)

CORE MECHANICAL DESIGN PARAMETERS (continued)	ANO-2	Calvert Cliff 1/2
Fuel Enrichment, weight percent uranium two thirty five		
Region A	1.93	2.01
Region B	2.27	2.51
Region C	2.94	2.91
Control Element Assemblies		
Number of Full/Part Length	73/8	77/8
NUCLEAR DESIGN PARAMETERS		
Hot Channel Factors		
Heat Flux		
Total Heat Flux Factor	2.62	3.0
Enthalpy Rise		
Nuclear, Enthalpy Rise Factor	1.55	1.65

Compared to the 14x14 fuel rod array, the 16x16 fuel assemblies in ANO-2 contain a great number of fuel rods of smaller diameter and pitch. The 16x16 fuel assembly has more heat transfer surface area and fuel rod linear footage, thereby reducing the peak heat flux and linear heat rate. This results in an increased thermal margin relative to a 14x14 fuel assembly at a given power density.

The margin to departure from nucleate boiling at any point in the core is expressed in terms of the departure from nucleate boiling ratio. For the ANO-2 departure from nucleate boiling ratio analysis the applicant used the W-3 correlation. The thermal-hydraulic design calculations for ANO-2 were performed using the Combustion Engineering COSMO/INTHERMIC codes. These codes are described in the topical report CENPD-161, "TORC CODE - A Computer Code For Determining the Thermal Margin of a Reactor Core," which includes a description of the TORC code. CENPD-161 has been reviewed by the staff and found to be acceptable for the ANO-2 reactor core thermal hydraulic analyses. The COSMO/INTHERMIC code pair has been found to give conservative results with respect to TORC calculations.

The applicant was requested to use applicable 16x16 fuel assembly departure from nucleate boiling data to support the thermal-hydraulic design basis used for steady-state and limiting transient analyses. The Combustion Engineering departure from nucleate boiling test program was previously conducted with an axially uniform heat flux distribution applied to electrically heated rod bundles representative of 14x14 and 16x16 fuel assemblies. The assemblies utilized Combustion Engineering standard spacer grids. Data from these test were used to develop a critical heat flux correlation, CE-1, which is used in conjunction with the TORC subchannel analysis code. In May 1976, the staff reviewed and approved the CE-1 correlation with the restriction that it was acceptable for uniform axial heat flux data with respect to predicting critical heat flux.

Combustion Engineering has since extended the critical heat flux test program to include axially nonuniform heat flux distributions. Preliminary results of the additional experimental studies were reported in Amendment 36 to the Final Safety Analysis Report. The data were evaluated using the TORC analysis code and the CE-1 correlation, with the addition of the Tong F-factor to account for the nonuniform heat flux.

The applicant has shown that the W-3 correlation (including a five percent departure from nucleate boiling ratio penalty to account for the unreviewed nonuniform axial heat flux data) compared with the CE-1 correlation (derived from data on 14 x 14 and 16 x 16 type bundles) is conservative for use on 16 x 16 fuel for pressures above 1800 pounds per square inch absolute. Therefore, use of the W-3 correlation (including penalty) for ANO-2 is acceptable.

With regard to rod bowing, we are in the process of developing criteria for evaluating the effect of rod bow on the departure from nucleate boiling ratio for application to the Combustion Engineering 16 x 16 fuel assemblies. We will establish an appropriately conservative interim penalty prior to startup. Based on our evaluation of 14 x 14 fuel assemblies, we anticipate no penalty for ANO-2 during the first cycle of operation.

The thermal-hydraulic design of the core for the ANO-2 plant was reviewed. The scope of the review included design criteria and the steady-state analysis of the core thermal-hydraulic performance.

We conclude that the thermal-hydraulic design of the core conforms to the Commission's regulations and to applicable regulatory guides and staff technical positions and is acceptable subject to the establishment of the value of the penalty to account for rod bowing effects prior to issuance of the operating license.

5.0 REACTOR COOLANT SYSTEM

5.1 Summary Description

The ANO-2 reactor is a pressurized water reactor with two coolant loops. The reactor coolant system circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to the steam generators. Each coolant loop in the reactor coolant system consists of one hot leg pipe between the reactor vessel outlet and the steam generator inlet and two cold leg pipes from the steam generator outlets to the reactor vessel inlet. Each cold leg contains a reactor coolant pump. The reactor coolant system contains a pressurizer connected to one of the hot leg pipes. A reactor quench tank (together with the interconnecting piping and instrumentation necessary for operational control) is provided to receive, condense and cool steam discharge from the pressurizer safety valves. All the above components are located in the containment building.

The system is similar to plants such as San Onofre 2 and 3, which have received a construction permit and Calvert Cliff's Units 1 and 2, which were reviewed and for which operating licenses were issued.

5.2 Integrity of the Reactor Coolant Pressure Boundary

The design loading combinations specified for ASME Code Class 1 reactor coolant pressure boundary components have been appropriately categorized with respect to the plant condition identified as normal, upset, emergency or faulted. The design limits used by the applicant for these plant conditions are consistent with the criteria recommended in Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components." Use of these criteria for the design of the reactor coolant pressure boundary components will provide reasonable assurance that (1) in the event an earthquake should occur at the site, or (2) other system upset, emergency or faulted conditions should develop, the resulting combined stresses imposed on the system components will not exceed the allowable design stresses and strain limits for the materials of construction. Limiting the stresses and strains under such loading combinations provides a basis for the design of the system components for the most adverse loadings postulated to occur during the service lifetime without loss of the system's structural integrity. The design load combinations and associated stress and deformation limits specified for ASME Code Class 1 components constitute an acceptable basis for design in satisfying the related requirements of General Design Criteria 1, 2 and 4.

The applicant has identified the active components within the reactor coolant pressure boundary for which operation is required to safely shut down the plant and maintain it in a safe condition in the event of a safe shutdown earthquake or design basis accident. The applicant has agreed to utilize an operability assurance program, in addition to stress and deformation limits, to qualify active valves. This program includes valve testing, or a combination of tests and predictive analysis, supplemented by seismic qualification testing of valve operator systems to provide assurance that active components (1) will withstand the imposed loads associated with normal, upset, emergency and faulted plant conditions without loss of structural integrity, and (2) will perform the "active" function under conditions comparable to those expected when safe plant operation or shutdown is to be effected, or the consequences of a seismic transient or of an accident are to be mitigated.

The applicant's component operability assurance program constitutes an acceptable basis for implementing the requirements of General Design Criterion 1 as related to the operability of American Society of Mechanical Engineers Code Class 1 active valves.

The criteria used in the design and mounting of the safety and relief valves of American Society of Mechanical Engineers Code Class 1 systems will provide adequate assurance that, under discharging conditions, the resulting stresses are expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function. The criteria used for the design and installation of overpressure relief devices in American Society of Mechanical Engineers Code Class 1 systems are consistent Regulatory Guide 1.67, "Installation of Overpressure Protection Devices" and Section III of the Code and constitute an acceptable design basis in meeting the applicable requirements of General Design Criteria Nos. 1, 2, 4, 14 and 15.

5.2.1 Fracture Toughness

Compliance with Code Requirements

We have reviewed the material selection, toughness requirements, and extent of materials testing performed by the applicant to provide assurance that the ferritic materials used for pressure retaining components of the reactor coolant boundary will have adequate toughness under test, normal operation, and transient conditions. The reactor vessel is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition, including Addenda through Summer 1970.

The ferritic material of the reactor pressure vessel beltline was qualified by impact testing to meet the acceptance standards of both Appendix G of 10 CFR Part 50 and Appendix G of Section III of the ASME Code. The remaining ferritic material of the reactor pressure vessel was tested to meet the design requirements of Section III of the ASME Code, 1968 Edition, including Addenda through Summer 1970. The reference nil ductility transition temperature of the reactor vessel beltline material, determined by Charpy V-notch and drop-weight testing, is zero degrees Fahrenheit and the reference nil ductility transition temperature of the remaining material in the reactor coolant system was estimated to be fifty degrees Fahrenheit.

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 nonductile behavior or rapidly propagating fracture can be established for the pressure retaining components of the reactor coolant boundary.

Operating Limitations

The reactor will be operated in a manner that will minimize the possibility of rapidly propagating failure. The pressure-temperature limit curves, for all phases of plant operation, were established using the available impact test data and conservative reference nil ductility transition temperature estimates to perform a fracture toughness calculation by the methods of the ASME Code, Appendix G, Summer 1972 Addenda and using the additional requirements of 10 CFR Part 50, Appendix G.

We have reviewed the operating limit curves in the Final Safety Analysis Report and conclude that they are acceptable. We will require that acceptable pressure-temperature limit curves are included in the ANO-2 technical specifications.

We conclude that the use of Appendix G of the ASME Code as a guide in establishing safe operating limitations, and the use of results from the available fracture toughness tests and conservative estimates performed in accordance with the code and the Commission's regulations will ensure adequate safety margins during operation, testing, maintenance, and postulated accident conditions.

5.2.2 Reactor Vessel Materials Surveillance Program

The toughness properties of the reactor vessel beltline material will be monitored throughout service life with a material surveillance program that will meet the requirements of American Society for Testing Materials Standard E 185-73, except that the beltline specimen material was chosen at random from the six beltline plates rather than in accordance with ASTM E 185-73. This program also complies with Appendix H of 10 CFR Part 50 except the specimen holders will be attached to the vessel cladding. Combustion Engineering has submitted a Topical Report, CENPD-155, "CE Procedures for Design, Fabrication, Installation and Inspection of Surveillance Specimen Holder Assemblies," dated September 1974. We have evaluated this report and concluded, based on our evaluation as presented in our letter to Combustion Engineering, dated May 15, 1975, that the procedures for design, fabrication, installation and inspection of surveillance specimen holder assemblies described in this report are acceptable. On the basis of the information provided in CENPD-155, we conclude that the method of attaching capsule holders to the vessel clad is acceptable and results in no degradation of the vessel base material. 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.

Changes in the fracture toughness of the reactor vessel beltline material caused by exposure to neutron radiation will be assessed properly and adequate safety margins against the possibility of vessel failure can be provided if the material surveillance requirements of Appendix H, 10 CFR Part 50 are met. Compliance with these specifications and regulations ensures that the surveillance program constitutes an acceptable basis for monitoring radiation induced changes in the fracture toughness of the reactor vessel material, and satisfies the requirements of General Design Criterion 31.

5.2.3 Fracture Toughness of Class 2 and Class 3 Components

We have reviewed the requirements for fracture toughness testing and properties and conclude that they provide assurance that the pressure retaining ferritic materials of all Code Class 2 and Class 3 components (outside as well as within the reactor coolant system) will have adequate toughness under test, normal operation, and transient conditions. The ferritic materials are specified to meet the toughness requirements of the ASME Code, and those of the staff.

The fracture toughness tests and properties required by the ASME Code and the staff provide reasonable assurance that safety margins against the possibility of nonductile behavior or rapidly progagating fracture have been established for the pressure-retaining ferritic materials of all Code Class 2 and Class 3 components, both within and outside the reactor coolant system.

5.2.4 Materials Compatibility with Reactor Coolant

We have reviewed the materials of construction for the reactor coolant pressure boundary to ensure that the possibility of serious corrosion or stress corrosion is minimized. All materials used are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion of all materials except carbon and low alloy steel will be negligible. For these materials, conservative corrosion allowances have been provided for all exposure surfaces of carbon and low alloy steel in accordance with the requirements of the ASME Code, Section III, and the external nonmetallic insulation to be used on austenitic stainless steel components conforms with the requirements of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels."

Further protection against corrosion problems will be provided by control of the chemical environment. The composition of the reactor coolant will be controlled; and the proposed maximum contaminant levels, as well as the proposed pH, hydrogen overpressure, and boric acid concentrations, have been shown by tests and service experience to be adequate to protect against corrosion and stress corrosion problems.

The controls imposed on reactor coolant chemistry are in conformance with the recommendations of Regulatory Guide 1.44, "Control of Sensitized Stainless Steel," and provide reasonable assurance that the reactor coolant pressure boundary components will be adequately protected during operation from conditions that could critically lead to stress corrosion of the materials and loss of structural integrity of a component.

The instrumentation and sampling provisions for monitoring reactor coolant water chemistry provides adequate capability to detect changes on a timely basis and to effect corrective actions within limits which preclude stress corrosion attacks to unacceptable levels. The use of materials of proven performance and the conformance with the recommendations of the cited Regulatory Guides constitute an acceptable basis for satisfying the requirements of General Design Criteria 14 and 31.

5.2.5 Fabrication and Processing of Ferritic Material

Materials selection, toughness requirements, and extent of materials testing proposed by the applicant provide assurance that the ferritic materials used for pressure retaining components of the reactor coolant boundary, including the reactor vessel and its appurtenances, will have adequate toughness under test, normal operation, and transient conditions.

The ferritic materials are specified to meet the toughness requirements of the ASME Code, Section III, including Summer 1972 Addenda. In addition, materials for the reactor vessel are specified to meet the additional test requirements and acceptance criteria of Appendix G, 10 CFR 50.

The fracture toughness tests and procedures required by Section III of the ASME Code, as augmented by Appendix G, 10 CFR 50, for the reactor vessel, provide reasonable assurances that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure retaining components of the reactor coolant boundary.

The results of the fracture toughness tests to be performed in accordance with the ASME Code and NRC regulations provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and NRC regulations constitutes an acceptable basis for satisfying the requirements of General Design Criterion 31.

The controls imposed on welding preheat temperatures and weld cladding are in conformance with the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel." These recommendations provide reasonable assurance that cracking of components made from low alloy steels will not occur during fabrication, and will minimize the possibility of subsequent cracking due to residual stress being retained in the weldment.

Conformance with Regulatory Guide and Codes mentioned above constitutes an acceptable basis for meeting the requirements of General Design Criteria 1 and 14.

5.2.6 Fabrication and Processing Austenitic Stainless Steels

Within the reactor coolant pressure boundary, no components of austenitic stainless steel exceed a yield strength of 90,000 pounds per square inch.

The controls imposed upon components constructed of austenitic stainless steel used in the reactor coolant pressure boundary and for the reactor vessel and its appurtenances satisfy the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel" and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants."

The requirements of controls on welding processes provide reasonable assurance that no deleterious hot cracking will be present during the assembly of austenitic stainless steel components. All weld filler metal is of selected composition to produce welds with at least five percent delta ferrite. Tests and examinations in accordance with Section III of the ASME Code, Summer 1972 Addenda, were performed to assure that adequate delta ferrite levels were met.

Material selection, fabrication practices, examination procedures, and protection procedures performed in accordance with the procedures and recommendations mentioned above provide reasonable assurance that the austenitic stainless steel in the reactor coolant pressure boundary will be free from hot cracking (microfissures) and in a metallurgical condition which precludes susceptibility to stress corrosion cracking during service. Conformance with the Regulatory Guides and staff positions mentioned constitutes an acceptable basis for meeting the requirements of General Design Criteria 1 and 14.

5.2.7 Steam and Feedwater System Materials

The mechanical properties of materials selected for Class 2 and 3 components of the steam and feedwater systems will satisfy Appendix I of Section III of the ASME Boiler and Vessel Pressure Code, or Parts B and C and Section II of the Code. The fracture toughness properties of the ferritic materials will satisfy the requirements of Articles NC-2300 and ND-2300 of Section III, Summer 1972 Addenda, of the ASME Code and are acceptable.

The requirements and controls on welding processes provide reasonable assurance that no deleterious hot cracking will be present during the assembly of austenitic stainless steel components. All weld filler metal is of selected composition to produce welds with at least five percent delta ferrite. Tests and examination in accordance with Section III of the ASME Code, Summer 1972 Addenda, were performed to assure that adequate delta ferrite levels were met.

Controls imposed in the application and processing of austenitic stainless steels satisfy the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

Fabrication and heat treatment practices performed in accordance with these requirements provide reasonable assurance that stress corrosion cracking will not occur during the design life of the plant. The control placed upon concentrations of leachable impurities is nonmetallic thermal insulation used on austenitic stainless steel components of the steam and feedwater systems in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation of Austenitic Stainless Steel."

The onsite cleaning and cleanliness controls during fabrication satisfy the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." The precautions taken in controlling and monitoring the preheat and interpass temperatures during weld of carbon and alloy steel components conform to the recommendations given in Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low-Alloy Steel."

Conformance with the codes, standards, Regulatory Guides, and NRC positions mentioned constitutes an acceptable basis for assuring the integrity of steam feedwater systems, and for meeting the requirements of General Design Criterion 1.

5.3 Reactor Vessel Integrity

We have reviewed all factors contributing to the structural integrity of the reactor vessel and conclude there are no special considerations that make it necessary to consider potential reactor vessel failure for this plant. The bases for our conclusion are that the design, materials, fabrication, inspection, and quality assurance requirements of the plant conform to applicable Commission regulations and Regulatory Guides, and to the rules of the ASME Boiler and Pressure Vessel Code, Section III 1968 Edition including Addenda and applicable Code cases through Summer 1970.

Operating limitations on temperature and pressure will be established for this plant in accordance with Appendix G, "Protection Against Nonductile Failure," of ASME Code Section III, and Appendix G of 10 CFR Part 50.

The integrity of the reactor vessel is assured because the vessel:

(1) Was designed, analyzed, and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and pertinent Code Cases;

(2) Was made from materials of controlled and demonstrated high quality;

(3) Will be inspected and tested to provide substantial assurance that the vessel will not fail because of material or fabrication deficiencies;

(4) Will be operated under conditions and procedures and with protective devices that provide assurance that (a) the reactor vessel design conditions will not be exceeded during normal reactor operation, and (b) the vessel will not fail under conditions of any of the postulated accidents;

(5) Will be subjected to periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under service conditions; and

(6) May be annealed to restore the material toughness properties if this becomes necessary.

5.4 Inservice Inspection Program

To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be inspected prior to startup and periodically throughout the life of the plant.

The design of the reactor coolant system incorporates provisions for access for inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1970 Edition. This edition of the code does not address the inspection of ASME Code Class 2 or 3 components. The applicant has indicated that equipment will be provided to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel.

We require that the applicant's inservice inspection program comply with the provisions of the ASME Code, Section XI as required by 10 CFR Part 50, Section 50.55a(g). This will include examination of Code Class 2 and 3 components as well as Class 1 components. The program will consist of a preservice inspection and an inservice inspection plan. We will require the applicant to submit an updated inservice inspection plan based on Section XI of the ASME Code, 1974 Edition, including all addenda through Summer 1975.

The conduct of periodic inspection and hydrostatic testing of ASME Code Class 1, 2 and 3 components in accordance with the requirements of ASME Code Section XI will provide reasonable assurance that evidence of structural degradation or loss of leaktight integrity occurring during service will be detected in time to permit corrective action before the safety function of the component is compromised. Compliance with the inservice inspection required by this code constitutes an acceptable basis for satisfying the requirements of General Design Criteria 32, 36, 39, 42 and 45.

5.5 Leakage Detection System

Coolant leakage within the containment may be an indication of a small through-wall flaw in the reactor coolant pressure boundary.

The leakage detection system to be used to detect leakage to the containment includes diverse leak detection methods, has sufficient sensitivity to measure small leaks, will identify the leakage source to the extent practical, and provides suitable control room alarms and readouts. The major components of the system are the containment atmosphere particulate and radiogas monitors, and level indicators on the containment sumps. Indirect indications of leakage are obtainable from the containment pressure, humidity, and temperature indicators.

The leakage detection systems provided to detect leakage from components of the reactor coolant pressure boundary satisfy the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," and provide reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions.

Compliance with the recommendations of Regulatory Guide 1.45 constitutes and acceptable basis for satisfying the requirements of General Design Criterion 30.

5.6 Components and Subsystem Design

5.6.1 Reactor Coolant Pump Flywheel Integrity

The probability of a loss of pump flywheel integrity can be minimized by the use of suitable material, adequate design, preservice spin testing and inservice inspection.

The applicant's selection of materials, fracture toughness tests, design procedures, preservice overspeed spin testing program, and inservice inspection program for reactor coolant pump flywheels have been reviewed and found acceptable on the basis of conformance with Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," and established industry codes and standards.

The probability of loss of reactor coolant pump flywheel integrity is minimized because it has been designed and spin tested at 125 percent of normal operating speed. In addition each finished flywheel was given a 100 percent volumetric examination after spin testing using ASME Section III procedures and acceptance criterion. Inservice inspection of the flywheel will be performed in accordance with the recommendations of Regulatory Guide 1.14.

The potential for the reactor coolant pump flywheel to become a missile in the event of a rupture in the pump suction or discharge sections of reactor coolant system piping is under generic study by the staff. The Electric Power Research Institute has contracted with Combustion Engineering, Incorporated, to perform a 1/5 scale reactor coolant pump research program. The objective of the program is in part, to obtain empirical data to substantiate or modify current mathematical models used in predicting pump performance during a postulated loss-of-coolant accident.

We have determined that additional protective measures, such as prevention of excessive pump overspeed or limitation of potential consequences to safety-related equipment, are technically feasible. If the results of the generic investigations of this matter indicate that additional protective measures are necessary to assure that an acceptable level of safety is maintained, we will require that such measures be implemented.

We conclude that the provisions for material selection and flywheel design, and inservice inspections in accordance with Regulatory Guide 1.14 ensure adequate flywheel integrity and constitutes an acceptable basis for satisfying the requirements of General Criterion 4.

5.6.2 Steam Generator Tube Integrity

The steam generators are vertical shell, U-tube heat exchangers with integral moisture separating equipment. The primary purpose of the steam generators is to transfer heat from the primary to secondary side for the generation of steam. The primary reactor coolant flows through the inverted U-tubes giving up heat to generate steam on the shell side of the secondary loop. The tube and tube sheet barriers prevent the transfer of activity generated within the core to the secondary system. Since the steam generators provide a heat sink for the primary reactor coolant system they are at a higher elevation than the reactor core. The elevation difference creates natural circulation capability sufficient to remove core decay heat following coastdown of all reactor coolant pumps.

The bases for determination of steam generator U-tube minimum wall thickness required to sustain combined loss-of-coolant accident plus safe shutdown earthquake loads has been addressed in sufficient detail in the ANO-2 Final Safety Analysis Report. The applicant submitted "Arkansas Steam Generator Structural Analysis of Tubes for Pipe Rupture Accidents," CENC-1262, an analyses of sufficient depth and detail to justify minimum tube wall thickness to sustain the faulted condition loads.

The materials used in Class 1 and Class 2 components of the steam generators were selected and fabricated according to codes, standards, and specifications acceptable to the staff. Procedures for monitoring the secondary coolant chemistry are in agreement with established staff technical positions.

The staff has evaluated the measures that will be taken to assure that the steam generator tubes in the Arkansas Nuclear One, Unit 2 facility will not be subjected to conditions that will cause degradation of integrity. We have also evaluated the provisions made by the applicant to detect such degradation, should it occur, before it has progressed far enough to affect the safety of the plant.

The facilities, steam generators, and operating procedures described in the Final Safety Analysis Report are of a more recent vintage than those that have been associated with steam generator tube degradation in some operating plants.

Regarding the newer plants, including Arkansas Nuclear One, Unit 2, nuclear steam supply vendors of pressurized water reactors that have experienced significant steam generator tube corrosion have redesigned the steam generators and made significant changes in the secondary system water chemistry. The affected nuclear steam supply system vendors are obtaining experimental data on tube material compatibility in simulated secondary coolant conditions so that the new pressurized water reactor plants should not have extensive localized corrosion.

For the Arkansas Nuclear One, Unit 2 steam generators, current regulatory requirements are considered sufficient to ensure plant safety and we have concluded that these measures are adequate. There is no reason to believe that plant safety will be compromised by steam generator tube degradation. Our conclusions are based on the following considerations:

(1) The steam generators will be of advanced design with improved secondary water flow characteristics, providing more tolerance for occasional lack of water chemistry control.

(2) The applicant will use an all-volatile type of water chemistry that has been shown by service experience to minimize the probability of tube degradation.

(3) Provisions for monitoring the secondary water chemistry will be included. These will be used to detect the presence of deleterious impurities before significant tube degradation can occur.

(4) Provisions for monitoring reactor coolant leakage to the secondary side are included in the design, and the limits on such leakage that will be imposed will ensure that tube degradation, should it occur, will be detected before it develops into serious deterioration of integrity.

(5) We will require in the technical specifications that periodic inservice inspections of the steam generator tubes be performed in accordance with the

recommendations of Regulatory Guide 1.83, "Inservice Inspection of Pressurizer Water Reactor Steam Generator Tubes."

(6) The design of the steam generators permits inservice inspection of the tubes by methods that will detect incipient tube degradation. Tubes that could further degrade to marginal conditions can be taken out of service by plugging.

We conclude that the steam generators have been designed to permit inservice inspection of all Code Class 1 and 2 components including individual tubes as recommended in Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," and Section XI of the Code. Conformance with Regulatory Guide 1.83 and Section XI of the Code constitutes an acceptable basis for meeting the applicable portions of Criteria 1 and 32 of the General Design Criteria.

Conformance with applicable codes, standards, staff positions, and Regulatory Guides constitutes an acceptable basis for meeting the applicable requirements of General Design Criteria 14, 15 and 31.

5.6.3 Residual Heat Removal

The residual heat removal is accomplished by use of the shutdown cooling system. The shutdown cooling system is used in conjunction with the main steam and feedwater system and the auxiliary feedwater system to reduce the temperature of the reactor coolant system in post shutdown periods from normal operating temperature to the refueling temperature. The initial phase of the cooldown is accomplished by heat rejection from the steam generators to the condenser or atmosphere. After the reactor coolant temperature and pressure have been reduced to approximately 300 degrees Fahrenheit and 300 pounds per square inch absolute, the shutdown cooling system is put into operation to reduce the reactor coolant temperature to the refueling temperature and to maintain this temperature during refueling.

The shutdown cooling system includes two shutdown cooling heat exchangers and uses the two low-pressure safety injection pumps and the two containment spray pumps. Two normally closed motor operated valves in the single shutdown cooling suction line in addition to the manual valves provide isolation of the shutdown cooling system from the reactor coolant system.

During shutdown cooling, a portion of the reactor coolant flows through the shutdown cooling nozzles located on the hot leg pipe and splits into two lines before reaching the shutdown cooling heat exchangers. This flow is circulated through the shutdown cooling heat exchangers by the low-pressure safety injection pumps and the containment spray pumps and returned to the reactor coolant system through the four low-pressure safety injection lines.

The cooldown rate is controlled by adjusting flow through the heat exchangers with throttle valves on the discharge of each heat exchanger. The flow controller maintain a constant total shutdown cooling flow to the core by adjusting the heat exchanger bypass flow to compensate for changes in flow through the heat exchangers.

The containment spray pumps and shutdown cooling heat exchangers are also used to remove heat from the recirculating containment sump water following a loss-of-coolant accident. The containment spray system is actuated following a trip of two out of the four high containment pressure signals. The containment spray actuation signal starts the spray pumps and opens the spray isolation valves. The spray water is discharged into the containment upper region through spray nozzles arranged on headers. The containment spray pumps initially take suction from the refueling water tank. When a low level is reached in the refueling water tank, a low level signal generates a recirculation actuation signal which automatically transfers the pump suction to the containment sump. During the recirculation mode spray water is cooled by the shutdown cooling heat exchangers prior to discharge into the containment.

We have reviewed the ANO-2 shutdown cooling system design and found that an appropriate number of isolation valves are provided to prevent overpressurization of the system due to single active failure of fluid components or any single active or passive failure of the electrical components. Of the components required to be operated in initiating shutdown cooling, only the two shutdown cooling isolation valves, 2CV-5084-1 and 2CV-5086-2 are located inside the containment building. We requested additional information concerning the possible consequences if one of the motor operated valves (2CV-5038, 2CV-5084 or 2CV-5086) at the suction of the low pressure safety injection pumps should be closed either when one of the pumps is started or while the pumps are in operation during shutdown cooling. The applicant responsed by making a modification to provide for the initiation of an alarm in the control room in the event of closure of one of the above valves. The operator can then open the valve to correct the situation by use of handwheels which are provided on these valves to allow local operation in the event an electrical failure in the valve operator or the associated cabling precludes remote operation. The applicant has stated that similar pumps at another facility (Calvert Cliffs) have been known to operate without suction for about one hour with only noncatastrophic damage to the seals. Based on this previous experience, we conclude that the consequences of the postulated failure are acceptable. The system can be brought to a cold shutdown (212 degrees Fahrenheit) condition within 36 hours with only one of the two cooling trains operable. The shutdown cooling system design is also discussed in Section 7.6.2 of this report.

We conclude that the shutdown cooling system design is acceptable for the ANO-2 plant.

5.7 Overpressure Protection

The reactor coolant pressure boundary is protected against overpressurization by two spring-loaded safety valves located on top of the pressurizer. The steam release from these valves discharges to the pressurizer relief tank through a common header. The pressurizer safety valves limit the reactor coolant system to less than 110 percent of the design pressure (2500 pounds per square inch absolute) following a loss of load incident from 100 percent power.

The applicant has assumed that the loss of load does not trip the reactor immediately, but that a delayed reactor trip does occur due to a high pressurizer pressure signal. No credit is taken for the action of the pressurizer spray, letdown, heat transfer to pressurizer walls, or turbine bypass system, but credit was taken for action of the secondary safety valves and high pressurizer pressure reactor trip. The calculated primary safety valve flow rate is less than the total rated capacities of the safety valves (331,200 pounds per hour versus 790,000 pounds per hour). The large difference in the flow rate is due to the applicant's conservative sizing procedure. We find the 7.9×10^5 pounds per hour capacity of the safety valves acceptable.

Overpressure protection for the steam side of the steam generators and the main steam line piping up to the main steam isolation valve is provided by a total of ten spring-loaded, open bonnet safety valves which discharge to the atmosphere. The safety valves will be flange-mounted on each of the two steam lines upstream of the steam line isolation valves outside the containment. ASME code requirements for the overpressure protection of the low pressure side (secondary side) are satisfied if the pressure stays below 110 percent of the design pressure (1200 pounds per square inch absolute) during the worst possible overpressure transient (100 percent loss of turbine generator load without simultaneous reactor trip). The valves are conservatively sized to pass a steady-state steam flow equivalent to the maximum expected power level at the design pressure of the main steam system. The maximum calculated steam flow rate through the secondary safety valves for loss of load with delayed reactor trip is 8.9×10^6 pounds per hour. This flow rate is 57 percent of the rated flow rate of the ten safety valves (15.55×10^6 pounds per hour at 1100 pounds per square inch absolute).

We have reviewed the design of the overpressure protection devices for the ANO-2 plant and conclude that they demonstrate an adequate design approach to relief capacity for the primary and secondary systems for the events considered above as set forth in the Final Safety Analysis Report.

There have been several reported incidents of reactor vessel overpressurization in pressurized water reactors during startup and shutdown in which the limitation of 10 CFR Part 50, Appendix G have been exceeded.

By letter dated December 27, 1976, we requested that the applicant provide us with (1) an analysis of the reactor coolant system response to pressure transients that could potentially occur during startup and shutdown, (2) a description of any modifications determined to be necessary in order to preclude exceeding the limits of Appendix G to 10 CFR Part 50, and (3) a schedule for the implementation of the modifications. In addition, we requested that the applicant identify short-term measures which will be taken to reduce the likelihood that overpressurization events will occur in the interim period until the permanent design changes can be made. Prior to a decision on the issuance of an operating license authorizing power operation, we will require that the applicant agree to operate with acceptable short-term measures and commit to the implementation of an acceptable long-term solution. We will report on the results of our review of this matter in a supplement to this report.

5.8 Loose Parts Monitoring

Occasionally, miscellaneous items such as nuts, bolts, and other small items have become loose parts within reactor coolant systems. In addition to causing operational inconvenience, such loose parts can damage other components within the system or be an indication of undue wear or vibration. For such reasons, we have encouraged applicants over the past several years to participate in programs designed to develop an effective on-line loose parts monitoring system. Recently, prototype loose parts monitoring systems have been developed and are presently in operation or being installed at several plants.

ANO-2 will utilize a loose parts monitoring system which will monitor acoustically any loose parts which would accumulate in the bottom of the reactor vessel or in each steam generator. The loose parts monitoring system will provide the operator an immediate audible and visual alarm of any loose parts in these areas. The loose parts monitoring system will include facilities whereby prerecordings of sounds normal to the reactor can be played for comparison. In addition, the recording facilities will be capable of recording sounds from any of the monitored locations for analysis and record. We are continuing our review of the ANO-2 loose parts monitoring system and will report the results of our review in a supplement to this report.

6.0 ENGINEERED SAFETY FEATURES

6.1 General

Engineered safety features are those structures, systems, and components necessary to mitigate design basis accidents, including the postulated loss-of-coolant accident, and high energy pipe break breaks. They are designed to seismic Category I requirements, and to function with complete loss of offsite power. Systems and components are provided with sufficient redundancy so that failure of a single component will not result in loss of the safety functions.

The sections following cover containment systems, the emergency core cooling system, and habitability systems. Additional engineered safety features are evaluated in Sections 7.0, 8.0, 9.0, and 10.0 of this report. The residual heat removal system is discussed in Section 5.6.3 of this report.

6.2 Containment Systems

The containment systems for Arkansas Nuclear One - Unit 2, include the containment structure, containment heat removal systems, containment isolation system, containment combustible gas control systems and provisions for containment leakage rate testing.

6.2.1 Containment Functional Design

The containment consists of a steel-lined, prestressed, reinforced concrete structure with a net-free volume of 1,780,000 cubic feet. The containment structure houses the nuclear steam supply system, including the reactor, steam generators, reactor coolant pumps, and pressurizer, as well as certain components of the plant engineered safety features systems. The containment is designed for an internal pressure of 54 pounds per square inch gauge and a temperature of 300 degrees Fahrenheit.

The applicant has analyzed the containment pressure response for postulated loss-of-coolant accidents in the following manner. Mass and energy release rates to the containment for postulated reactor coolant system pipe breaks were calculated by Combustion Engineering utilizing the CEFLASH-4 and FLOOD-MOD 2 computer codes. These data were then used as input to the Bechtel COPATTA computer program, which performs transient thermodynamic calculations with appropriate considerations of containment heat removal systems and structural heat sinks to calculate the containment pressure response.

The applicant has analyzed a number of reactor coolant system pipe break accidents including a spectrum of break locations and sizes. The postulated double-ended pipe rupture at the pump suction of the reactor coolant system resulted in the highest calculated containment pressure which was about 53 pounds per square inch gauge. The loss of one of the two containment spray trains and full emergency core cooling system operation were conservatively assumed for the evaluation.

We have also analyzed the containment pressure response to a postulated double-ended cold leg pump suction break using the CONTEMPT-LT computer code. Our analysis was based on the mass and energy release data, the containment structural heat sink data, and the containment heat removal systems performance data provided by the applicant. Conservative condensing heat transfer coefficients to the structures inside containment were also used. The peak calculated pressure was essentially the same as that calculated by the applicant for the worst postulated loss-of-coolant accident. Based on the results of our analysis, we conclude that the applicant's analysis is acceptable provided that the following matter related to the single failure analyses is resolved to the staff's satisfaction.

The applicant has analyzed a spectrum of main steam line break accidents to determine the containment pressure and temperature response. The total mass and energy released to the containment consists of that in the steam generator volume at the beginning of the accident plus the contribution from blowdown of the feedwater system. Calculation of the release rate of the mass and energy initially contained in the steam generator is based on previously accepted calculational methods, as set forth in the Combustion Engineering Standard Safety Analysis Report (CESSAR), utilizing the SGN III computer code. We have required that calculation of the mass and energy contributed by the feedwater system include the application of the most conservative single failure to the feedwater system. We require additional information regarding the single failure analysis performed by the applicant including a justification that the mass and energy added from the feedwater system have been conservatively accounted for. We will report our evaluation of this matter in a supplement to this report.

The applicant has analyzed the transient pressure response within various containment interior compartments. The compartments investigated are the reactor cavity, reactor cavity wall pipe penetration, and steam generator compartment.

The mass and energy release rates were calculated utilizing the CEFLASH-4 computer program and this data was used as input to the Bechtel developed subcompartment analysis program the COPATTA code, to determine the pressure response within the subcompartments. For the subcooled portion of the blowdown calculation, the applicant has used the extended Moody critical flow model.

In our confimatory analysis we modified the applicant's mass and energy release rate data to conservatively account for the difference that would be expected if the HENRY/FAUSKE subcooled flow model, which we believe is more appropriate, were used.

Based on the results of our confimatory analysis using the RELAP-3 code, we conclude that the design pressures for the containment interior compartment are acceptable.

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 safety-related equipment. As a result there is a generic concern regarding the capability of safety-related equipment to remain operable in the accident environment which would 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 assumption of dry steam blowdown and using conservative assumptions for minimizing containment heat transfer coefficients with a conservative treatment of the thermodynamics of condensate behavior.

In the ANO-2 Final Safety Analysis Report, an analysis for a spectrum of main steam line breaks postulated to occur inside containment was provided by the applicant which resulted in a peak calculated atmosphere temperature of 415 degrees Fahrenheit exceeding 289 degrees for about 70 seconds. Component heat transfer calculations were used to justify the adequacy of an environmental qualification temperature of 289 degrees Fahrenheit. The results of these analyses indicate that the component thermal response will be less than the peak temperature to which the equipment has been environmentally qualified. We have requested additional information from the applicant on these analyses and will report our conclusions in a supplement to this report.

The applicant has analyzed the consequences of inadvertent actuation of the containment spray system on the containment vessel. A pressure drop of 3.36 pounds per square inch inside the containment was calculated by the applicant. The containment external design pressure is 3.5 pounds per square inch gauge. In our confirmatory analysis, we calculated a pressure drop inside the containment essentially the same as the applicant. Therefore, we conclude that the external design pressure of the containment is acceptable.

We have evaluated the containment system functional design in accordance with the General Design Criteria and, in particular, Criteria 16 and 50. We are unable to conclude on the acceptability of the containment internal design pressure and temperature until we complete our review of the applicant's analysis of the containment pressure response due to a postulated main steam line break accident. We will report our findings on these matters in a supplement to the Safety Evaluation Report.

6.2.2 Containment Heat Removal Systems

The containment spray system and the containment cooling system are provided to remove heat from the containment following a loss-of-coolant accident. Any of the following combinations of equipment will be capable of providing the cooling capacity required to maintain the peak pressure at less than design pressure for the spectrum of assumed break sizes:

(1) Two trains of the containment spray system,

(2) One train of the containment spray system and two containment cooling units, or

(3) Four containment cooling units.

The containment spray system serves only as an engineered safety feature and performs no normal operation function. It is a seismic Category I system consisting of redundant piping, valves, pumps and spray headers. All active components of the containment spray system are located outside the containment building. Missile protection is provided by direct shielding or physical separation of equipment. The containment sump is covered by a protective screen assembly designed to prevent debris from entering the sump which could damage the containment spray or safety injection systems.

A high containment pressure will cause the engineered safety features actuation system to automatically operate the containment spray systems. The spray pumps and valves can also be operated manually from the control room. The spray pumps will initially take suction from the refueling water storage tank. When the water in the refueling water storage tank reaches a low level, which occurs about a half hour or more after a loss-of-coolant accident, the spray pump suction is manually transferred to the containment sump. The applicant's analysis of the net positive suction head available to the spray pumps indicates that sufficient water will have been delivered to the containment by the start of the recirculation phase to provide the net positive suction head required for continued operation of the spray pumps. The analysis was performed consistent with the recommendations of Regulatory Guide 1.1, "Net Positive Suction Head For Emergency Core Cooling and Containment Heat Removal System Pumps," and is acceptable.

The containment cooling system is used during both normal and accident conditions. Four equal capacity containment cooling units are provided, with each unit containing two sets of cooling coils and bypass dampers. Cooling water is supplied to one set of the cooling coils by the chilled water system during normal plant operation, and to the other set of coils by the service water system in the event of a loss-of-coolant accident.

During normal plant operation, three of the four containment cooling units are required to provide sufficient cooling. Upon receipt of a containment isolation actuation signal, the idle containment cooling unit is automatically started, service water flow is initiated and bypass dampers are opened for all units.

The containment cooling system is a seismic Category I system. The cooling units are located outside the secondary concrete shield for missile protection, and are accessible for periodic testing and inspection during normal plant operation.

We have reviewed the containment heat removal systems for conformance with General Design Criteria 38, 39 and 40, and have found them to be acceptable.

6.2.3 Containment Air Purification and Cleanup Systems

The containment spray system is used for iodine removal from the containment atmosphere following a postulated loss-of-coolant accident. Sodium hydroxide is added to the containment spray solution to enhance the iodine scrubbing function of the system. Redundant spray additive pumps are used to raise the pH of the spray solution to a value between 8.4 and 8.7. Sodium hydroxide addition is continued during the recirculation phase until the additive pump is stopped by a low-level signal from the sodium hydroxide storage tank. Analyses have shown that the recirculation water pH will not exceed 10.7 at any time.

We have evaluated this system and conclude it is effective for removal of elemental iodine, and iodine absorbed on airborne particulate matter. We calculated first order removal coefficients for elemental and particulate iodine of 10 and 0.36 inverse hours, respectively, in an estimated effective containment volume of 1.52×10^6 cubic feet which represents 85 percent of the total free volume of the containment. The elemental iodine removal effectiveness is assumed to diminish after a decontamination factor of 100 has been achieved in the containment atmosphere. The long term sump pH of 8.8 is considered adequate to maintain the decontamination factor of 100 for the elemental iodine.

6.2.4 Containment Isolation Systems

The containment isolation system is designed to isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of closed systems and isolation valves, is provided so that no single valve or piping failure can result in loss of containment integrity. Containment building penetration piping up to and including the external isolation valve is designed as seismic Category I equipment, and is protected against missiles which could be generated under accident conditions. Containment isolation will occur automatically upon receipt of a containment isolation actuation signal of high containment pressure (five pounds per square inch gauge). All fluid penetrations not required for operation of the engineered safety features equipment will be isolated. Remotely operated isolation valves have been provided for those safety-related systems which will not be automatically isolated.

The containment purge system consists of one inlet and one outlet line each fitted with two redundant fifty-four inch butterfly valves in series designed to close in less than five seconds. In our review of the containment isolation provisions, we noted that the applicant has proposed to purge the containment approximately four times per year. We have required that the containment purge system's operation be limited to one percent of the time per year (about 85 hours) or that ANO-2 must meet the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operation." The Technical Specification therefore limits the purge system operation to 85 hours or less as stated in Section 3/4.6 of the technical specifications.

We have reviewed the containment isolation system for conformance to General Design Criteria 54, 55, 56 and 57. We conclude that the system meets the General Design Criteria and is, therefore, acceptable.

6.2.5 Combustible Gas Control in Containment

Following a loss-of-coolant accident, hydrogen may accumulate inside the reactor building. The major sources of hydrogen generation include:

(1) a chemical reaction between the fuel rod cladding and the steam resulting from vaporization of the emergency core cooling water;

(2) corrosion of aluminum by the alkaline spray solution; and

(3) radiolytic decomposition of the cooling water in the reactor core and the building sump.

The applicant's analysis of post-loss-of-coolant accident hydrogen generation is consistent with the guidelines of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." The concentration limit specified by the applicant for actuating the hydrogen control equipment is three percent by volume which is calculated to occur about three days after a loss-of-coolant accident. This concentration is well below the lower flammability limit of four percent by volume. We have performed a similar analysis of hydrogen generation in the containment following a loss-of-coolant accident and our results indicate that the applicant's analysis is conservative.

Two full capacity electric recombiners, located inside containment, are provided. The recombiner system incorporates several design features that are intended to assure the capability of the system to remain operable in the event of an accident. Among these are:

(1) Seismic Category I design;

(2) Institute of Electrical and Electronic Engineers (IEEE-Standard 279-71) requirements for the wiring and electrical equipment;

(3) Protection from missile and jet impingement from broken pipes; and

(4) Redundant to the extent that no single component failure disables both recombiners.

We have reviewed and accepted the design and functional capability of the proposed recombiner.

The applicant has also provided a controlled containment purge system as a backup to the recombiner system in accordance with the guidelines of Regulatory Guide 1.7.

Based on our review of the systems provided for combustible gas control following a loss-of-coolant accident, we conclude that the systems meet the recommendations of Regulatory Guide 1.7 and are, therefore, acceptable.

6.2.6 Containment Leakage Testing Program

The containment design includes provisions and features to test penetrations, isolation valves and the overall containment leak rate. However, we will require additional information regarding the applicant's leakage testing program to determine compliance with the requirements of Appendix J to 10 CFR 50. We will conclude on the acceptability of the applicant's leakage testing program in a supplement to the Safety Evaluation Report.

6.3 Emergency Core Cooling System

6.3.1 Design Basis

The basic design of the emergency core cooling system for ANO-2 is functionally similar to that developed for other Combustion Engineering plants, such as Calvert Cliffs 1/2 and San Onofre 2/3. In ANO-2, one low pressure safety injection header feeds four cold legs, whereas in Calvert Cliffs 1/2 and San Onofre 2/3, two low pressure headers feed all four cold legs. The safety injection tanks for Calvert Cliffs 1/2 has an operating pressure of 200 pounds per square inch gauge whereas those for San Onofre 2/3 and ANO-2 have a minimum operating pressure of 600 pounds per square inch gauge.

The system is designed to provide emergency core cooling for postulated accidents where it is assumed that a failure in the reactor coolant system piping results in loss-of-coolant from the system greater than the makeup capacity of the charging pumps. The subsystems provided are of such number, diversity, reliability, and redundancy that no single failure of emergency core cooling equipment occurring during a loss-of-coolant accident will result in inadequate cooling of the reactor core. The subsystems are designed to function over a range of reactor coolant system pipe break sizes, up to and including the flow area associated with a postulated double-ended break in the largest reactor coolant pipe. The emergency core cooling system is also designed to protect against steamline break consequences.

6.3.2 System Design

The ANO-2 system consists of safety injection tanks, a high pressure injection subsystem, a low pressure injection subsystem, and provision for recirculating flow from the containment sump. Various combinations of these subsystems assure core cooling for the complete range of postulated break sizes.

In the event of a loss-of-coolant accident, the system will operate initially in the injection and subsequently in the recirculation mode. In the injection mode, high pressure safety injection will be provided by two high pressure safety injection pumps--each is sized to deliver water at a rate sufficient to maintain level in the reactor vessel, matching boiloff assuming 25 percent spill at the time the safety injection system switches into recirculation mode (not less than 30 minutes after the loss-of-coolant accident). If offsite power is not available, the high pressure pumps and associated injection valves are supplied with emergency power, one pump from each of the two diesel generators. This permits the automatic operation of one full capacity system in the event of simultaneous loss of outside power and the failure of any active component, including an emergency diesel generator.

Each of the injection lines is provided with a check valve and a motor-operated stop valve to isolate this subsystem from the reactor coolant system. Opening of these stop valves will be actuated by the safety injection actuation signal. The pumps take their suction initially from the borated water in the refueling water tank and, after that tank is nearly exhausted, borated water is recirculated from the containment sump. A design requirement of the refueling water tank is that it must have sufficient capacity for at least 30 minutes of delivery at the full capacity of all safety injection and containment spray pumps.

Four safety injection tanks, 1850 cubic feet each with a minimum of 1413 cubic feet of borated liquid, are provided to reflood the core during the initial stages of a loss-of-coolant accident involving large pipe breaks. Adequate fluid is contained in the tanks to accomplish this function with one tank discharging through the break. Each tank is connected to one of the cold legs of the reactor coolant system by a line with two check valves and a normally open, remotely operated valve in series. The safety injection tank will, therefore, inject water automatically whenever the pressure in the reactor coolant system falls below the safety injection tank pressure of 610 pounds per square inch gauge.

During normal operation, the motor-operated valve is maintained in the open position and the check valves prevent high pressure reactor coolant from flowing into the lower pressure safety injection tanks. During shutdown operation, when reactor coolant system pressure drops to 650 pounds per square inch gauge, the safety injection tanks are depressurized to 400 pounds per square inch gauge manually. When reactor coolant system pressure drops to 415 pounds per square inch gauge, the safety injection tank isolation valves are closed. These isolation valves are interlocked with the pressurizer pressure measurement channels to prevent closing of the valves above 700 pounds per square inch gauge and also to open these valves automatically as reactor coolant pressure is increased to 700 pounds per square inch gauge during startup. After the valves are opened, they will be locked open in the control room.

The low pressure injection system consists of two pumps, each rated at 3250 gallons per minute design capacity and each supplied with emergency power from separate diesel generators. For the injection mode of operation, these pumps will also supply borated water from the refueling water tank. Sizing of the low pressure safety injection pump is governed by the shutdown cooling function.

When essentially all of the water in the refueling water tank has been injected, suction for the high pressure pumps is automatically transferred to the containment sump for the recirculation mode of operation. The low pressure pumps are tripped. In the recirculation mode of operation, the emergency core cooling systems will provide long-term core cooling by recirculating the spilled reactor coolant, the injected water, and the containment spray drainage, collected in the containment sump, back to the reactor.

All of the emergency core cooling system subsystems are designed to accomplish their functions when operating on either offsite power or emergency (onsite) power. In the event of a loss-of-offsite power concurrent with a single failure in the emergency power supply system, the safety injection tanks (which require no electrical power), plus one high head and one low head injection pump would provide the minimum required emergency core cooling systems flow. We have reviewed the information presented by the applicant concerning the available net positive suction head for the emergency core cooling system pumps. The high and low pressure pumps are located in safeguards rooms in the lowest level of the auxiliary building. This location maximizes the available net positive suction head for safety injection pumps.

The method used by the applicant to calculate net positive suction head is consistent with Regulatory Guide 1.1. No credit is taken for containment pressure increase due to heating of the atmosphere. In determining the available net positive suction head, it was assumed that containment pressure was in equilibrium with the maximum calculated containment sump water temperature. The applicant states that the calculated available net positive suction head for the high pressure safety injection pumps at the system runout flow of 825 gallons per minute is approximately 20 feet. The net positive suction head required as determined by the pump test curves furnished by the manufacturer is 18 feet at 825 gallons per minute.

The Advisory Committee on Reactor Safeguard's (ACRS) letter report, dated February 10, 1972, on the construction permit application indicated that the ACRS wished to review the final design of the safety injection tank system prior to fabrication and installation of the system in ANO-2. Accordingly, the applicant submitted a report on safety injection system improvements for ANO-2. This report, submitted on February 10, 1972, and supplementary information dated January 26, 1973, was reviewed by the staff and by the Advisory Committee on Reactor Safeguards at its 155th meeting, March 8-10, 1973.

The applicant in this report made a parametric study of the performance of the emergency core cooling system based on the previous Interim Acceptance Criteria. Safety injection tank parameters in the study were pressure, total volume, water/gas volume ratios, line sizes, break size and gas pressure in the fuel rod pellet-to-clad gap. The tank pressures evaluated in this study ranged between 200 pounds per square inch gauge and 600 pounds per square inch gauge. The water/gas volume ratio was increased with increasing gas pressure to achieve the performance desired at each tank size. As a result of the study, the safety injection design for ANO-2 was modified as follows:

Parameter	New	Old
Operating pressure (pounds per square inch gauge)	600	200
Total volume per tank (cubic feet)	1850	1600
Water fraction (percent)	79.8	55
Gas mass per tank (pounds)	1049.2	695.5

The staff concluded that adoption of the 600 pounds per square inch gauge design value in conjunction with other safety injection tank system parameters defined by the applicant in his letter of January 26, 1973, was acceptable for the design, fabrication, and installation of the safety injection tank system. The Advisory Committee on Reactor Safeguards, by letter dated March 13, 1973, also reviewed the proposed design and found it acceptable for ANO-2 subject to other information developed during plant construction.

The staff has reviewed the final design of the safety injection system and has concluded that it is consistent with the modified design previously accepted and that no further information has been developed which would reasonably be expected to alter previous conclusions. We, therefore, find the safety injection system design to be acceptable for ANO-2 provided that Appendix K emergency core cooling system analysis performance results are acceptable.

6.3.3 Performance Evaluation

We are continuing our review of the ANO-2 emergency core cooling system performance analysis. The results of our ANO-2 emergency core cooling system evaluation will be provided as a supplement to this report and will 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 valves within containment.

Containment Back Pressure

Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation includes the effect of operation of all installed containment pressure reducing systems and processes. The corresponding reflood rate in the core will then be reduced because of the resistance to steam flow in the reactor coolant loops.

Following a loss-of-coolant accident, the pressure in the containment building will be increased by the addition of steam and water from the primary reactor system to the containment atmosphere. After initial blowdown, heat transfer from the core, primary metal structures, and steam generators to the emergency core cooling systems, water will produce additional steam. This steam, together with any emergency core cooling systems water spilled from the primary system, will flow through the postulated break into the containment. This energy will be released to the containment during both the blowdown and later emergency core cooling systems operational phases; i.e., the reflood and post-reflood phase.

Energy removal occurs within the containment by several means. Steam condensation on the containment walls and internal structures serves as a passive energy heat sink that becomes effective early in the blowdown transient. Subsequently, the operation of the containment spray system will remove steam from the containment atmosphere. When the steam removal rate exceeds the rate of steam addition from the primary system, the containment pressure will decrease from its maximum value.

The emergency core cooling systems containment pressure calculations for Arkansas Nuclear One, Unit 2, were performed using the Combustion Engineering emergency core cooling systems evaluation model. The staff reviewed the model and published a Status Report on October 15, 1974, which was amended November 13, 1974. We concluded that the Combustion Engineering containment pressure model was acceptable for emergency core cooling systems evaluation. We required, however, that justification of the plant-dependent input parameters used in the analysis be submitted for our review of each plant.

This information was submitted for Arkansas Nuclear One, Unit 2, by Amendment 32 dated October 31, 1975. The applicant has reevaluated the containment net-free volume, the passive heat sinks, and operation of the containment heat removal systems with regard to the conservatism for the emergency core cooling systems analysis. The evaluation was based on measurements within the containment and from as-built drawings to which a margin was added. The containment heat removal systems were assumed to operate at their maximum capacities, and minimum operational values for the spray water and service water temperatures were assumed.

We have concluded that the plant-dependent information used for the emergency core cooling systems containment pressure analysis for Arkansas Nuclear One, Unit 2, is conservative and, therefore, the calculated containment pressures are in accordance with Appendix K to 10 CFR Part 50 of the Commission's regulations.

Performance Evaluation Status

We are continuing our review of the ANO-2 emergency core cooling system performance analysis. Our performance evaluation conclusions will be provided in a supplement to this report.

6.3.4 Tests and Inspections

The applicant will demonstrate the operability of the emergency core cooling system by subjecting components to system tests and component tests.

The objective of the system tests will be to verify that the safety injection system will respond as required to perform its intended function. The tests will be performed during each scheduled refueling shutdown. A test safety injection actuation signal will be applied to initiate operation of the system. The safety injection pump motors may be deenergized for this test. Circuit tests and pump starting and operation may be demonstrated at any time. The system test will be considered satisfactory by the applicant, if control board indication and visual observation indicate that all components have received the test safety injection actuation signal in proper sequence of timing. This will verify that the appropriate pump breakers have opened and closed, and that all valves have completed their travel. During reactor operation, instrumentation channels used to initiate safety injection are checked during each shift, while safety injection actuation signal logic circuitry will be tested on a monthly basis.

In addition, the applicant will conduct tests to verify the proper operation of the safety injection system components. These tests supplement the system level tests by verifying acceptable performance of each active component in the safety injection system. The tests will include cycling of all check valves to ensure proper operation and checking of refueling water tank level and safety injection tank pressure and level instrumentation channels.

The safety injection pumps will be started every month. Acceptable performance will be demonstrated by conformance to the requirements of 10 CFR Part 50.55a relating to the testing of pumps. The flow is recirculated to the refueling water tank.

The applicant states that the emergency core cooling system will be generally tested as suggested by Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems For Pressurized Water Reactors," with the exception that the capability of the system to recirculate water from the containment sump into the reactor coolant system will not be verified by in-plant testing. We will require test verification of the capability of the high pressure safety injection pumps to operate in the recirculation mode in coincidence with other required pumps taking suction from the containment sump. The purpose of this testing is to demonstrate that conditions that could adversely affect the emergency core cooling system performance, such as inadequate net positive suction head, air binding or vortex formation at the sump intakes, do not occur. We will report our evaluation of this matter in a supplement to this report.

6.3.5 Conclusions

Acceptability of the emergency core cooling system for the full spectrum of postulated break sizes and locations is dependent on resolution of the concerns noted in Sections 6.3.3 and 6.3.4 above. We will report on these matters in a supplement to this report.

6.4 Habitability Systems

The emergency protective provisions of the control room related to the accidental release of radioactivity and toxic gases are evaluated in this section. Relevant portions of the control room ventilation system are described here but are further described and evaluated in Section 9.4 of this report.

6.4.1 Radiation Protection Provisions

The applicant will meet General Design Criterion 19, "Control Room," of Appendix A to 10 CFR Part 50, by use of concrete shielding and by use of a redundant 2000 cubic feet per minute recirculating filter serving the combined control room zone of Units 1 and 2. In addition, a redundant 333 cubic feet per minute charcoal filter is installed to supply pressurization air to the control room, minimizing infiltration of unfiltered air. The control room has vestibule entrances (two doors in series) to further minimize infiltration.

The applicant upgraded the original habitability design in response to staff concerns involving adequate operator radiation protection. The modified system is described in Amendment 39 to the Final Safety Analysis Report. The vestibules and pressurization filters were added to the design as a result of these concerns.

In the event of a high radiation signal at the fresh air intakes, the control room zone of Unit Nos. 1 and 2 is isolated from the outside and from the balance of the normal control room air conditioning equipment. The same signal actuates the emergency filters. The system is configured such that the makeup air used to pressurize the control room is filtered through four inches of charcoal. This results in a filter credit of 99 percent iodine removal. The recirculated air is filtered through at least two inches of charcoal resulting in a filter credit of 95 percent iodine removal.

We have performed operator dose calculations assuming a design basis loss-of-coolant accident. The resultant doses are within the guidelines of General Design Criterion 19. We therefore conclude that the control room radiation protection is acceptable.

6.4.2 Toxic Gas Protection Provisions

Control room habitability following a postulated toxic gas release is required to ensure that operators can continue to monitor and, if required, control plant operations. Chlorine has been identified as the only material that, if released, would pose a potentially serious operator hazard. Chlorine is stored in one ton cylinders about 500 feet from the control room air intakes. Provisions such as quick-acting chlorine detectors and self-contained breathing apparatus will be provided to protect the operator against a chlorine release. We have reviewed these provisions against the guidelines of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," and have found them to be adequate. We conclude that the plant's toxic gas protection is acceptable.

6.5 Engineered Safety Feature Systems Materials

The mechanical properties of materials selected for the engineered safety features satisfy Appendix I of Section III of the American Society of Mechanical Engineers (ASME) Code, or Parts A, B and C of Section II of the Code, and the NRC position that the yield strength of cold worked stainless steels shall be less than 90,000 pounds per square inch.

The controls on the pH of the reactor containment sprays and the emergency core cooling water following a postulated loss-of-coolant accident are adequate to ensure freedom from stress corrosion cracking of the austenitic stainless steel components and welds of the containment spray and emergency core cooling systems throughout the duration of the postulated accident to completion of cleanup.

The requirements and controls on welding processes provide reasonable assurance that no deleterious hot cracking will be present during the assembly of austenitic stainless steel components. All weld filler metal was of selected composition to produce welds with at least five percent delta ferrite; and tests and examinations in accordance with Section III of the ASME Code, Summer 1972 addenda, were required to assure that adequate delta ferrite levels are met.

Controls imposed in the application and processing of austenitic stainless steels, for components of the engineered safety features to avoid sensitization, satisfy the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

Fabrication and heat treatment practices performed in accordance with these requirements provide added assurance that stress corrosion cracking will not occur during the postulated accident time interval. The control of the pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, are in accordance with Regulatory Guide 1.7, "Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident," and provide assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution by corrosion of containment metal or cause serious deterioration of the containment. The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on austenitic stainless steel components of the engineered safety features are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

Conformance with the codes and Regulatory Guides mentioned above, the staff positions on the allowable maximum yield strength of cold worked austenitic stainless steel, and the minimum level of pH of containment sprays and emergency core cooling water constitute an acceptable basis for meeting applicable requirements of General Design Criteria 35, 38, and 41.

7.0 INSTRUMENTATION AND CONTROL

7.1 General

The Commission's General Design Criteria, Institute for Electrical and Electronics Engineers (IEEE) Standards including IEEE Criteria for Protection Systems for Nuclear Power Generating Stations (IEEE Standard 279-1971, "Criteria For Protection Systems For Nuclear Power Generating Stations") and applicable Regulatory Guides for Power Reactors have been utilized as the bases for evaluating the adequacy of the protection and control systems. Specific documents employed in the review are listed in Appendix B to this report.

The review of the protection and control systems was accomplished by comparing the designs with the applicable portions of those of Calvert Cliffs Unit 1 and 2 plants.

Our review concentrated on those areas which are unique to Arkansas Nuclear One - Unit 2, for which new information has been received, or which have remained as continuing areas of concern during this and prior reviews of similar designed plants.

We have required the applicant to submit a final design drawing package for the reactor protection system and the engineered safety feature systems which is in sufficient detail and scope to enable the staff to conduct an independent review. We have reviewed selected final design drawings and conclude that the drawings presently docketed are adequate for this operating license review.

A site review for the purpose of identifying the physical arrangements and installation of electrical equipment to assure that the system designs are implemented in accordance with the design criteria and bases was conducted on July 6, 7 and 8, 1977. We have identified the outstanding concerns resulting from our site visit in a letter to the applicant. The evaluation of the applicant's response to the majority of those items is being conducted by our Office of Inspection and Enforcement. We are continuing our review of the applicant's response to the remaining three outstanding items summarized below and will report our evaluation of these items in a supplement to this report.

(1) Documentation in the Final Safety Analysis Report of the short circuit tests conducted on electrical penetrations.

(2) Documentation of the separation criteria for redundant raceways where separation is less than one inch of air space.

(3) Documentation and justification describing reinstatement of nonsafety electrical loads on safety busses.

Therefore, conditioned on the resolution of the items identified above, we conclude that the site visit, completes this portion of our review and that the design satisfies the Commissions regulations, industry standards and regulatory guides cited above and in Section 8.1 of this report and is acceptable.

7.2 Reactor Trip System

7.2.1 General

The reactor trip system presented in the Final Safety Analysis Report has been significantly modified from the originally proposed system reviewed during the construction permit stage. The modified design utilizes a computer based system for deriving the low departure from nucleate boiling ratio and the high local power density trip functions, in addition to the analog hard wired system previously proposed.

The computer based system consists of (1) the sensor channels that provide input to the core protection calculators, (2) the logic channels that calculate and actuate bistables whenever the departure from nucleate boiling ratio or the local power density value exceeds a predetermined set point, and (3) the bistables that provide input to the reactor trip system.

The analog hard-wired system consists of (1) sensor channels that monitor various parameters and trip bistables whenever predetermined set points are exceeded, and (2) logic matrix channels that monitor the bistable trip and actuate a reactor trip on a two-out-of-four coincidence basis.

The computer based portion of the reactor trip system has not been previously used in safety-related systems and as such represented a new design approach for which additional review effort was required.

The review of the reactor trip system was conducted in two parts:

(1) The design of the computer based portion of the system which includes its testability, its operating bypass capabilities and its interaction with the analog hard-wired portion of the system.

(2) The design of the analog hard wired portion of the system, including portions of the hard-wired system that interface with the computer based portion such as the reactor pressure and reactor power channels, have been evaluated in this scope only to the extent of verifying their channel independence and the equipment qualification. However, their quality and their design features as required by computer based portion of the reactor trip system is included in the review of the computer based system.

The detailed description and our evaluation and conclusions for the computer based portion of the reactor trip system are discussed in Section 7.2.3 of this report. The analog hard-wired portion of the reactor trip system is evaluated and discussed in Section 7.2.2 of this report.

7.2.2 Reactor Trip System-Hardwired Analog Portion

The reactor trip system is comprised of four redundant and independent protection channels. Each channel monitors the following parameters and initiates a bistable trip which in turn actuates a set of three trip relays associated with each trip function.

(1) High linear power level
(2) High logarithmic power level
(3) High local power density
(4) Low departure from nuclear boiling ratio (DNBR)
(5) High pressurizer pressure
(6) Low pressurizer pressure
(7) Low steam generator no. 1 water level
(8) Low steam generator no. 2 water level
(9) High steam generator no. 1 water level
(10) High steam generator no. 2 water level
(11) Low steam generator no. 1 pressure
(12) Low steam generator no. 2 pressure
(13) High containment pressure

Each set of trip relay outputs are combined into three of six independent logic matrices representing all possible two-out-of-four trip combinations for the four protection channels. Each logic matrix contains four output relays. The output of the six logic matrices provide four redundant and independent trip paths to the undervoltage coils of the control rod power supply breakers. Thus, each logic matrix can interrupt the four trip paths, causing insertion of all rods. Each channel, logic matrix and trip path is completely testable during reactor operation.

The following sections address the problem areas revealed during our review and the resolutions concerning them.

Equipment Protection Trips

The applicant identified a loss of load and a high steam generator water level reactor trip functions as being required for equipment protection and not for plant safety. Although the high steam generator water level trip function is designed to satisfy the requirements of IEEE Standard 279-1971, the loss of load trip function and its associated bypass circuitry were not designed in full conformance with the above requirements. The applicant was advised that the introduction of other than safety grade trips into the reactor trip system could result in the degradation of the system and therefore the loss of load trip and its associated bypass circuitry should be designed to satisfy fully the requirements of IEEE Standard 279-1971. The applicant elected to modify the design and delete this trip function from the reactor trip system scope. We conclude that this modified design is acceptable.

Independence of Redundant Power Supplies

The system as presently designed provides two independent vital alternating current power supplies to energize each logic matrix circuit via a voltage comparator circuit. Four of the logic matrix power supply sources originate from one or the other of two redundant vital buses. However, two logic matrices receive power from both redundant vital buses. The design did not provide sufficient information to assure that adequate isolation between redundant buses is maintained. The applicant was requested to provide documentation to demonstrate that a single failure in these circuits would not compromise the independence of the vital buses, or modify the design to ensure vital bus independence.

The applicant has, in lieu of modifying the design, submitted test procedures and test results to demonstrate that a single failure in these circuits would not compromise independence of the vital buses. We have reviewed these test procedures and test results and conclude that the tests as documented are incomplete and as such are unacceptable at this time. The applicant was requested to submit additional information, based on tests, to show that (1) all failure modes have been addressed, (2) the maximum credible voltages that can be applied to these circuits, both alternating current and direct current voltages, have been addressed, and (3) surge voltage tests have been conducted as recommended in IEEE Standard 472-1974, "IEEE Guide for Surge Withstand Capability Tests."

The applicant has recently submitted additional information related to this matter. We will report resolution of this item in a supplement to this report.

7.2.3 Reactor Trip System - Digital Computer Portion

Introduction

The operating license application for ANO-2 is the first in the United States to propose to include stored program digital computers in a portion of the reactor protection system. The core protection calculator system (CPCS) is the digital computer based portion of the reactor protection system. It was designed by Combustion Engineering, the nuclear steam supplier for ANO-2. The remainder of the reactor protection system is conventional analog hard-wired equipment. The CPCS, in conjunction with the overall reactor protection system is designed to provide at least the same level of protection to the core as a conventional, hard-wired system.

The CPCS is designed to provide reactor protection for two conditions: (1) low local departure from nucleate boiling ratio (DNBR), and (2) high local linear power density. The remaining twelve of fourteen protective functions of the reactor protection system are accomplished 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 Section 7.2.2 of this report. Section 7.2.3 of this report summarizes our review of and conclusions relating to the CPCS. Appendix D to this Safety Evaluation Report addresses our review and conclusions relating to the CPCS in further detail.

At the construction permit stage of review for ANO-2, the proposed design of the reactor protection system was a conventional hard-wired analog system. In our Safety Evaluation Report for the ANO-2 construction permit application, we concluded that "the applicant was not able to provide analysis to convince us that no safety problems can arise from either axial or azimuthal Xenon oscillations." We also noted that the applicant indicated that automatic protection features would be provided to protect against such oscillations unless it could be shown that such protection was not required. Prior to the Final Safety Analysis Report submittal, the applicant indicated that the CPCS would be presented as the response to this matter and included the CPCS design and qualification in the Final Safety Analysis Report. Our evaluation for the CPCS is presented herein.

Digital computers offer potential advantages over analog circuitry for the proposed application. Safety limits are functions of several interacting process variables, which for practical purposes, cannot be completely defined by analog circuitry. Predictive parametric power distribution analyses and thermal margin analyses are combined with interacting process variables to define process limits which will assure that the fuel design limits are not exceeded.

Administrative procedures for the operation of the plant may also be simplified through the use of digital computers. Control rod position is not directly input to the circuitry of a conventional analog protection system. The rod pattern configuration must be controlled administratively in accordance with pre-calculated patterns to assure proper protective action by the analog circuits. This function is incorporated within the CPCS design. However, with the CPCS, the operating DNBR is limited as a function of axial shape index by administrative procedures.

The proposed digital protection system provides on-line routines for synthesis of the power distribution and evalution of the DNBR using measured inputs from the ex-core nuclear flux monitors, control element assembly (CEA) position indicators, and other sensor data such as core inlet temperatrure, primary system pressure and coolant pump speed. The on-line algorithms are a simplified version of the off-line design procedures; the simplification provides the reduced running time required for on-line processing but results in some loss of accuracy. The calculated thermal margin results must be compsenated for this loss of accuracy through the use of uncertainty factors. The magnitude of the uncertainty factors are sufficient to result in conservative thermal margin calculations in the CPCS compared to values obtained from more rigorous calculations in the plant computer. The CPCS permits power operation under plant process conditions which could lead to a trip in analog circuits since values of one process parameter which could lead to an adverse thermal margin may be compensated by favorable values of another; this can be permitted by the CPCS but not by current designs of analog circuits. At the same time, the CPCS provides more automated protection against violations of administrative procedures and control system malfunctions.

The primary disadvantages of the CPCS for the proposed application are the reliability uncertainty and the performance uncertainty posed by the lack of experience with digital computers in plant protective systems.

Many staff concerns relate to qualification and quality control procedures to assure the initial reliability and performance of the integrated system and to assure continued software reliability especially after implementation of future design changes or changes to program constants. However, the proposed limited applications for initiation of DNBR and high local power density reactor trips will permit the demonstration and evaluation of digital computers in the safety system without undue risk to the health and safety of the public. These trips are designed to prevent fuel damage during anticipated operational occurrences and normally perform no role in the prevention of postulated accidents, although they do provide the initial response to mitigate the consequences of the reactor coolant pump shaft seizure and the steam generator tube rupture accidents.

Our review of the CPCS included evaluation of conformance of the applicant's design, design criteria, design bases and qualification program to the Commission's regulations as set forth in the General Design Criteria and to applicable regulatory guides, branch technical positions, and industry standards. These are listed in Table 7-1, "Acceptance Criteria for Control," NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," LWR Edition dated September 1975. While these criteria have been adequate for the review of analog hard-wired type of protection systems, they were also useful for the review of the CPCS. Computer hardware design and qualification were evaluated through the existing criteria. Specific regulatory rules and guidance for the evaluation of stored computer programs do not exist to conduct the review. For future evaluation of computer based protection systems, the staff intends to formalize and document the specific criteria developed and utilized for the review of the CPCS.

To establish a basis for the review, the staff conducted a technical survey on applications of digital computers to nuclear power plants. The survey encompassed European, Canadian and United States applications of computers. The specific results of the survey concerned with protection systems defined the following key concerns that are to be addressed in a design and qualification program:

- Complexity/simplicity of software structure
- Testability of software
- Top-down design of software
- Bottom-up test structure
- Decentralization of the protection system with one to two of the most complex trips per computer

The main emphasis on previous computer based application effort has been on software, not hardware. It is noted that the specific emphasis was on software structure, design verification and testability. These points were also stressed by a recognized leader in software engineering, Dr. Barry Boehm in "Software and Its Impact: A Quantative Assessment," Datamation magazine, May 1973, pages 48-59. In formulating a review plan for the CPCS, the staff integrated the findings of the technical survey along with our normal review procedures for safety grade electrical hardware.

To conduct the review, a task force of staff personnel was selected. To cover the disciplines required for the review, the task force was composed of members from the Core Performance, Analysis, and the Electrical Instrumentation and Control Systems Branches. Furthermore, to aid the task force in the review of the software, consultants from Oak Ridge National Laboratory were also obtained.

As a result of technical surveys, literature surveys, and previous software experience of the staff and consultants, specific guidelines for evaluating the software were selected and used in the evaluation. These are discussed below:

(1) <u>Lack of Critical Design Faults</u>

Critical design faults are those type of faults that result in insufficient plant protection upon a challenge to the reactor protection system. Designer error is the main source of critical design faults; the review concentrated on evaluating the adequacy of the design bases, the computer program development, implementation, test plans, and test results to assess the potential existence of critical design faults.

(2) <u>High Software Reliability</u>

An evaluation of software reliability measures the adequacy of and adherence to a quality assurance program for the design, development, and qualification of software. The objective of high reliability is to minimize the number of errors in the final product. Components of our qualitative evaluation consisted of configuration control procedures, independent reviews, and documentation. As an additional component of reliability, the use of automatic audit and surveillance type programs to monitor hardware performance and execution of the stored programs will also be evaluated.

The scope of our safety review of the CPCS is from the sensor channels that provide input to the calculators to the channel bistable trip outputs for the DNBR and local power density trip. This safety review evaluates the proposed design, implementation, qualification test program, test results, bypasses, interlocks, periodic test program, the physical and functional interfaces between the protective system, the plant operator, and the remainder of the plant, for the core protection calculator system.

The review also considered the effects of CPCS unreliability on the unreliability of the plant protective system. Of specific concern were the increases in interchannel connections and channel interdependence from previous protection system designs; the complexity of the CPCS design; and potential common design errors in the CPCS hardware and computer software. The plant protective system is comprised of the reactor protection system and the reactor trip system. The reactor protection system is configured as a hybrid, combining analog function modules and the CPCS in its design. The analog modules and the CPCS perform comparable tasks. That is, they determine the need for protective action at the channel level and initiate the protective action, when required (i.e., two or more channel trips), at the system level. Both the analog modules and the CPCS provide interfaces from the reactor protection system to the reactor trip system. The reactor trip system acts to trip the reactor whenever two or more protective action signals are received from the reactor protection for the same trip function. Thus, the CPCS functions in the plant protective system are similar to those of an analog function modules.

The CPCS hardware and software are designed to meet the same design criteria as these analog modules. The requirements of 10 CFR 50 Appendix B for quality assurance during the design, development, qualification and design verification, including an independent design review, have been imposed on the CPCS hardware and software. Thus, it is our judgment that the effects of CPCS unreliability on the unreliability of the plant protection system does not significantly impact the assumptions used in the anticipated transients without scram analyses currently under review by the staff.

<u>Summary</u>

The CPCS review is incomplete at this time. A staff decision on the acceptability of the system for the licensing of ANO-2 has not been made. Major qualification test programs for the CPCS remain to be executed and evaluated. The acceptability of the results of these tests cannot be predicted at this time. However, based on satisfactory resolution of all of the staff's safety issues we see no reason at this time to conclude that the design is unacceptable.

The majority of the CPCS design information received prior to February 1977 has been reviewed, evaluated, and is reported herein. Additional design information, in the form of functional descriptions of the protection algorithms were resubmitted to the staff in February 1977. At first appraisal, it appears that the design has been significantly altered. We will make an assessment of this matter and report our evaluation in supplement to this report. Our evaluation of the applicant's qualification test programs and test results which have been docketed by February 1977 are also presented.

In response to the safety positions issued by the staff, the applicant has proposed several design modifications and retest programs. The items remaining to be submitted by the licensee for staff review include details on the design modifications, revised software specifications, revised computer program listings, qualification test plans, pre-operational and start-up test plans, and test reports. The applicant has also proposed several design modifications based on development experience to date. A review and evaluation of these design modifications, test plans, test results and technical specifications test results will be reported in a supplement to this report.

A detailed status summary of the twenty-seven safety positions developed by the staff is presented in Table 7.1 beginning on page 7-27 of this report.

System Description

The CPCS consists of six digital computers configured and implemented to provide protection in the form of the low DNBR and high linear power density trips. The system is composed of four redundant digital computers, referred to as the core protection calculators (CPCs) and two redundant computer based control element assembly calculators (CEACs). The CEACs provide each CPC with processed control element assembly position data. The CPCs acquire data from plant process sensors, the control element assembly position sensors, directly as well as via the CEACs, and perform the required calculations. Each CPC provides trip inputs to one of the four redundant and independent reactor trip system channels when the trip setpoints are exceeded. The functional configuration for the CPCS is presented in Figure 7.1 of this report.

Each control element assembly has two separate reed switch position transmitters. Redundant position indication is automatically and continuously monitored by the CEACs. In previous protection system designs, the operator was responsible for monitoring control element assembly positions and maintaining relative alignments. This was to ensure that the operational state of the plant remain within the design basis of the protection system. For the CPCS the margin to trip of the protection system is automatically adjusted as a function of control element assembly misalignment. For the CPCS, the operator is required to monitor DNBR as a function of axial shape index and maintain DNBR within prescribed limits. Failure of the core operating limit supervisory system (COLSS), a nonsafety system that monitors DNBR, would impair this function and would require more conservative DNBR operating limits.

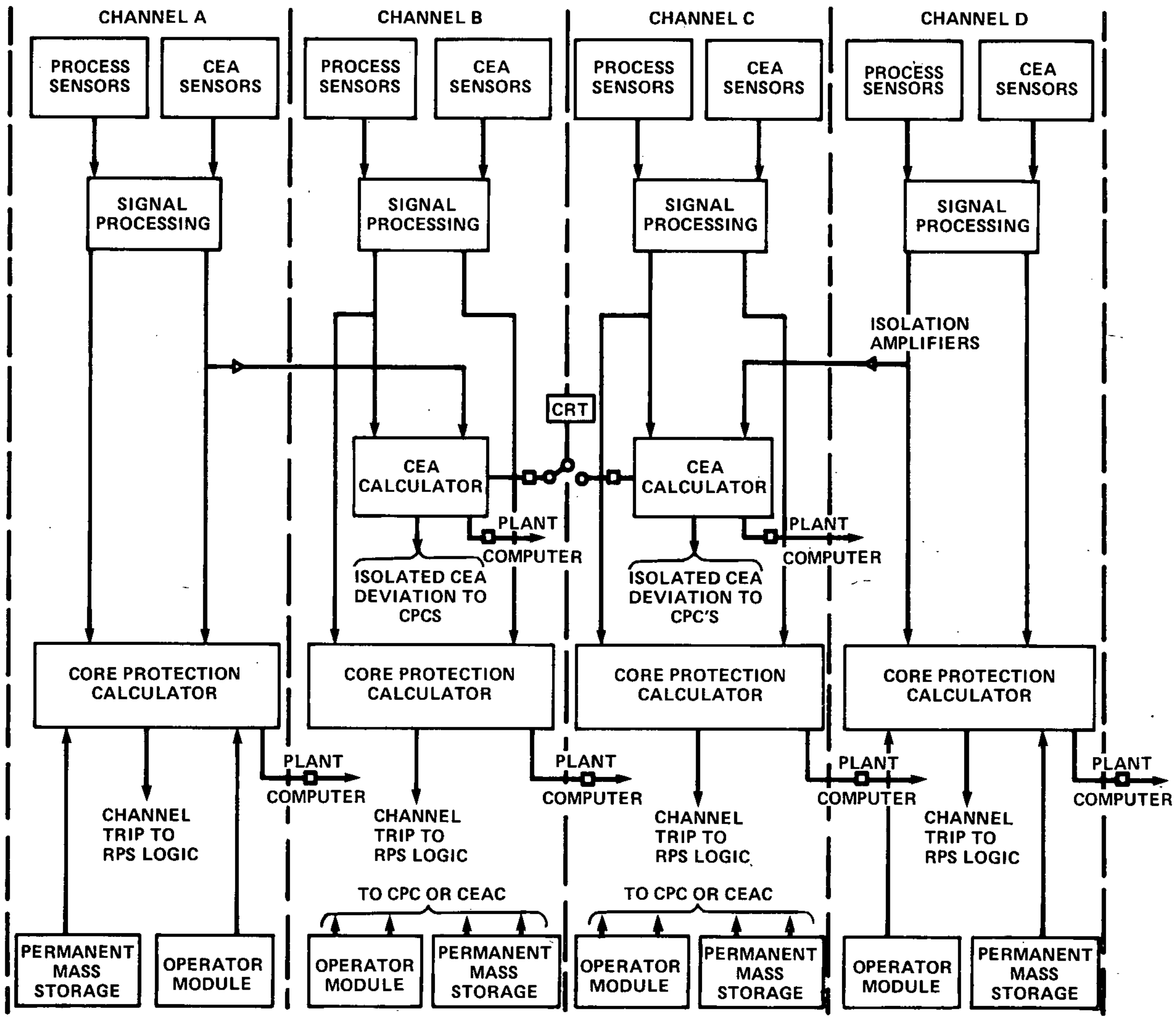

Figure 7.1 Core Protection Calculator (CPC) System Functional Configuration

All of the position data from reed switch position transmitters are routed to a CEAC. The CEAC's process these data to determine a deviation penalty factor for single rod deviation. The deviation penalty factor is a measure of the misalignment of the control element assembly and is used as a modifier to the radial peaking factor in the protection algorithms to account for control element assembly misalignment. To maintain channel independence between the individual CPCs isolation amplifiers are used in routing portions of the position signals to the CEACs as shown in Figure 7.1 of this report.

The shaft speed of each reactor coolant pump is measured by a sensor. The sensor output signal is proportional to pump speed, which in turn is used by the CPCs to calculate coolant flow.

Additional process sensor signals are also used by the CPCS. Functionally, these consist of hot and cold leg temperatures, pressurizer pressure, and ex-core flux signals. These sensor signals are used in the evaluation of the DNBR and linear power density trip variables.

The computer input signals consists of analog and digital signals. The analog signals consisting of sensor signals are converted to digital signals by means of an analog-to-digital converter. Digital input to the computer is received from the operator's module. The operator's module is an input/output device to the computer that allows the reactor operator to enter data and to interrogate the computer. Pulse inputs from the pump speed sensor are converted to digital signals which are proportional to pump speed. This is performed in the data input/output subsystem. Also, each CPC periodically reads the deviation penalty factor communicated from the CEAC's.

The CPC is a byte addressable, 65K byte computer wherein the trip algorithms are implemented and executed. The CEAC is the same type computer, and is used to process control element assembly position information.

The software for the core protection calculator is functionally structured in terms of modules. These consist of the system executive module, protection algorithm module, initialization module, system test module and the operator's module monitor. The system executive module provides for interrupt servicing, both internal and external, system startup and task scheduling. Fixed frequency clock interrupts, and external interrupt, cause execution of the schedule functions, which begins or continues execution of algorithms based on a predefined priority structure.

The initialization module verifies that time-dependent transients have died out of the data base and initiates execution of the algorithms stored in memory. The operator's module monitor detects keyboard input, and when in the display mode, updates values of displayed points. Each protection algorithm in the system is priority structured for execution, and is executed at a predetermined frequency.

Finally, the system test module performs automatic on-line testing and provides automatic interface capability for all off-line testing.

Each CPC provides outputs for three continuous displays of calculated results. The displays consist of DNBR margin, local power density margin, and calibrated power based on measured neutron flux. These displays provide the operator with information on the status of each channel.

Each CPC also provides binary outputs and relay driver outputs. The binary outputs drive the operator's module digital display meter. The relay outputs drive status lights, annunciators, and the matrix relays in the reactor trip system. Upon loss of power the system fails in the tripped state.

An operator's module is provided for each protection channel. It is designed to permit the operator to monitor system status, performance, and to enter selected data to the system. The data entered are constants for use in the protection algorithms. These data are called addressable constants and consists of thermal calibration constants, an azimuthal tilt factor, and other data. Each of these constants may vary with time and reactor conditions and operator input provides a means of updating.

A permanent mass storage unit upon which the protection algorithms, test programs, and test data are stored is provided for each channel. This is the unit from which computer memory is initially loaded or reloaded in the event that it is necessary. The unit is also used during periodic testing of the calculators.

Staff Review Methodology

The review and safety evaluation of computer programs has presented some problems. Common logic, developed by the same programmers and implemented in the same way in each protection channel is a potential source of common mode failure due to design errors.

However, this is not new as the same problem exists in analog hard-wire technology. For software review, the issue is sharpened in that the number of operations and the complexities of the logic in a computer program exceed those found in hard-wire protection systems. Thus, the safety review of computer-based protection systems requires a greater effort than for analog hard-wire protection systems.

The review methodology employed for the safety evaluation was the audit principle. The licensee and the vendor have objected to the depth of staff audits that were conducted, however, the staff considers that such depth is justified in view of the system complexity and the continued design and development of the system by the applicant during the safety review. In addition to the standard round one and round two questions, the staff also held numerous meetings and conducted field audits in its evaluation of the system. Minutes of these meetings and audits are listed in the bibliography to this report.

Details on the safety evaluation of the CPCS are presented in Appendix D to this report.

CPCS Review Status Summary

The disposition of the 27 safety positions reported herein is as follows:

(1) The applicant has responded to and fully implemented to the staff's satisfaction five of the safety positions generated by the staff. These issues, designated as items 2, 3, 6, 7, and 17 in Table 7.1 of this report are categorized as closed issues.

(2) The applicant has responded to and partially implemented the requirements of 15 of the safety positions. The applicant's proposed response, as we understand it, is acceptable. However, we have to review new docketed material and also monitor and audit the implementation of the position before a final evaluation is made. These positions, designated as items 1, 4, 5, 8, 10, 11, 12, 13, 14, 15, 16, 21, 22, 23 and 24 in Table 7.1 of this report are categorized as outstanding issues.

(3) The applicant's response to four of the safety positions are unacceptable to the staff. These issues, designated as items 9, 18, 19, and 20 in Table 7.1 of this report are also categorized as outstanding issues.

(4) We have not completed our review of the applicant's response to positions 25, 26 and 27.

7.2.4 Conclusion

We have reviewed the ANO-2 reactor trip system as described in the Final Safety Analysis Report and conclude that the reactor trip system meets the staff's requirements as stated in Section 7.1 of this report and is acceptable, conditioned on the resolution of items discussed in Sections 7.2, 7.2.3, 3.10. and 3.11.

7.3 Engineered Safety Features Systems

7.3.1 General

The engineered safety feature actuation systems are comprised of sensor monitoring channels, basic logic channels, and two independent and redundant component actuations trains. Each channel, logic and actuated equipment train is fully testable during reactor operation.

Our review of the ANO-2 application included review of the selected schematic drawings of the circuitry pertaining to the actuation systems included in the Final Safety Analysis Report, including the circuitry pertaining to individual components such as pumps and valves of the systems. The significant areas of this review are discussed below.

7.3.2 Engineered Safety Features Actuation and Basic Logic

Each actuation system is identical except for the input parameters and includes four redundant and independent protection channels per trip input. Each actuation system logic is configured in the same manner as the reactor trip system with the four trip path outputs arranged into two independent, selective, two-out-of-four concidence logics. Each coincident logic actuates one of the two redundant groups of engineered safety feature equipment.

System actuation subsystems and associated trip input parameters identified in the ANO-2 design are the following:

(1) Containment isolation actuation system and penetration room ventilation system; high containment pressure or low pressurizer pressure.

(2) Safety injection actuation system and containment cooling actuation systems, low pressurizer or high containment pressure.

(3) Containment spray actuation system; high-high containment pressure and safety injection actuation signal.

(4) Recirculation spray actuation system; low refueling water tank level.

(5) Main steam line isolation system; low pressure in either of two steam generators.

(6) Emergency feedwater actuation system; low pressure in either of two steam generators and low steam generator level.

We have reviewed the engineered safety feature actuation descriptive information, which included the logic diagrams and selected schematic drawings included in the Final Safety Analysis Report. We conclude that the design of the engineered safety feature actuation system meets the staff's requirements as stated in Section 7.1 of this report and is acceptable, conditioned only on the satisfactory resolution of the items discussed in Sections 7.1, 7.3.3, 7.3.6, 3.10 and 3.11.

7.3.3 Changeover from Injection Mode to Recirculation Mode

The applicant has provided, as part of the engineered safety feature actuation, the recirculation actuation system subsystem that automatically initiates the changeover from the injection mode to the recirculation mode of operation before the refueling water storage tank is emptied following a loss-of-coolant accident. The design consists of four independent level transmitters actuating independent and redundant logic trains on a two-out-of-four basis when a predetermined minimum setpoint is reached. The logic initiates opening of reactor building sump valves and terminates the flow from the refueling water storage tank.

Our review of the schematics, included in the Final Safety Analysis Report, indicated that the design (de-energize to trip) could initiate the changeover from injection mode to recirculation mode prematurely when subjected to a single failure. This design feature would degrade the emergency core cooling system below an acceptable level and would not be acceptable. The applicant was, therefore, requested to modify the design to preclude inadvertant actuation or demonstrate that a single failure, such as a short circuit, an open circuit, or a single failure of a power source, would not cause premature actuation of the recirculation actuation system.

In response, the applicant submitted a modified design. The modified design utilizes an auctioneered power supply arrangement from redundant vital buses to supply power to two of the four level transmitters that actuate the recirculation actuation system logic. This design modification is similar to the power supply configuration provided in the reactor trip system and the engineered safety feature system basic logic matrix. We have reviewed the logic diagrams and selected schematic drawings included in the Final Safety Analysis Report and conclude that the modified design of the recirculation actuation system is acceptable conditioned only on the satisfactory resolution of the concerns identified in Section 7.2.2 of this report.

7.3.4 Main Steam Isolation System

The system consists of two fully testable, independent and redundant sensor and actuating logic trains. Each actuation logic is configured in the same manner as the reactor trip system with the four trip path outputs arranged into two independent, selective two-out-of-four coincident logics. Each coincident logic actuates one of two redundant groups of the system's equipment. Actuation of the system automatically closes both main steam isolation valves, the feedwater isolation valves, and trips the feedwater and condensate pumps. During our review we have identified the following concerns.

(1) The system design presently incorporates one normally closed air-operated steam atmospheric dump valves in each loop between the steam generator and main steam isolation valves which are controlled by non-Class IE equipment. We initially concluded that the non-Class IE equipment did not provide adequate assurance that these valves would remain closed during a steam line break accident. The applicant was requested to verify that the consequences of inadvertent opening of the atmospheric dump valve during a steam line break accident are acceptable or to modify the design to ensure that the atmospheric

dump valves will remain closed when required. In a recent amendment to the Final Safety Analysis Report, the applicant submitted a modified design that provides a redundant main steam isolation signal to each of two solenoid valves associated with each atmospheric dump valve. The applicant has submitted the final schematics and has verified that the solenoid valves and the associated circuitry is qualified Class IE equipment. We have reviewed the selected final design drawings and conclude that the modified design, as implemented, satisfies the single failure criterion and meets the staff's requirements as stated in Section 7.1 of this report and is therefore acceptable.

(2) The present design also uses a turbine trip signal via non-Class IE equipment to trip the turbine stop and control valves during a steam line break accident. This signal is derived from a set of non-Class IE undervoltage relays located in the control element drive mechanism control system. Thus, in the event of a main steam line break accident the reactor trip system trips the Class IE breakers, supplying power to the control rod drives, which in turn trip the undervoltage relays mentioned above and generates the signal to trip the turbine. This trip signal is used as back-up protection to isolate the intact steam generator in the event one of the main steam isolation valves fail to close. The staff has completed the review of these systems on a generic basis. Based on our review and the staff's conclusions as published in NUREG-0138 Issue No. 1, "Treatment of Non-Safety Grade Equipment in Evaluation of Postulated Steam Line Break Accidents," we conclude that the applicant's design satisfies the staff's requirement as stated in the above referenced document and is therefore acceptable.

Our review of the main steam isolation system included the review of the logic diagrams and selected schematic diagrams included in the Final Safety Analysis Report. We have concluded that the design meets the staff's requirements as stated in Section 7.1 of this report and is therefore acceptable.

7.3.5 Emergency Feedwater System

The auxiliary feedwater system is comprised of two fully testable redundant and independent subsystems. Automatic initiation of the system is accomplished by logic selection and comparison of the steam generator level, and steam generator pressure. In the event of a main steam line break accident, the control system insures that the emergency feedwater flow is automatically aligned with the unaffected steam generator.

During our review of the emergency feedwater system the applicant was requested to modify the design and provide sufficient power diversity so that there is not complete reliance on any one source of energy for emergency feedwater system operation. The applicant's response, provided in the question and response section of the Final Safety Analysis Report in response to questions numbered 020.35, 020.54, 222.22 and 222.90, indicated that the design would be modified prior to startup following the first regularly scheduled refueling outage to provide direct current power to all electrically operated components associated with the turbine driven pump train. Additional evaluation and discussion regarding this matter is provided in Section 10.5 of this report.

In addition, the applicant was requested to provide documentation that demonstrates that the steam generator level transmitters used in the plant protection system (i.e., the reactor trip system and emergency feedwater system) will retain the necessary accuracy to initiate and monitor protective action during steam generator blowdown condition, as required by IEEE Std. 279-1971. In response to our concerns, the applicant submitted an analysis that demonstrates that for the worst case conditions during a main steam line break blowdown condition, the dynamic effects affecting the steam generator level transmitters during blowdown would contribute to less than a two percent error in the measured variable value. Justification for the small errors is due to small diameter of primary piping, long sensing lines and the use of condensation pots to assure adequate reference leg volume. Based on the analysis submitted and the small errors postulated in the level sensing elements during a steam line break accident, we conclude that this design satisfies the staff's requirements stated in Section 7.1 of this report and therefore is acceptable.

We have reviewed the design of the emergency feedwater system which included the logic diagrams and selected schematics provided in the Final Safety Analysis Report. We conclude that the design meets the staff's requirements as stated in Section 7.1 of this report and is acceptable. The operating license will be conditioned to require the implementation of the auxiliary feedwater pump power diversity matter as discussed in Section 10.5 of this report.

7.3.6 Selective Actuation System Logic Power Supplies

There are two independent, selective, two-out-of-four coincident actuation system logics per system actuated. Each selective actuation logic serves one redundant group of engineered safety feature equipment and is powered from four vital power supplies. Our review of the power supplies for the selective actuation logics revealed that the redundant actuation logics were both powered from the same power sources. It is our concern that a single failure can compromise the independence of the four vital power supplies and can result in loss of all engineered safety feature actuation and protection functions.

The applicant has submitted a modified design for the two selective two-out-of-four coincidence actuation system logic per engineered safety feature system train. Each selective actuation logic serves only one of the redundant groups of engineered safety feature equipment and is powered from two instead of the previously proposed four vital power supplies. This design is now similar to the design used in the basic reactor trip system and engineered safety feature logic actuation system, in that only one vital power supply from each of the two redundant trains is being utilized to actuate the required equipment in each train. Based on our review of the functional logic diagrams and selected final design schematics included in the Final Safety Analysis Report, we conclude that this design satisfies the staff's requirements stated in Section 7.1 of this report and, therefore, is acceptable conditioned only on the satisfactory resolution of the item identified in Section 7.2.2 of this report. We will report the final resolution of this matter in a supplement to this report.

7.4 Systems Required for Safe Shutdown

Our evaluation of the ANO-2 application included review of the selected schematic drawings of the circuitry pertaining to systems required for safe shutdown, including the circuitry used to initiate operation of individual components, e.g., pumps and valves. In addition, we have reviewed the instrumentation and controls provided for effecting hot shutdown with potential capability for subsequent cold shutdown from outside the control room.

We conclude that the design meets the staff's requirements, as stated in Section 7.1 of this report, including the requirements of the General Design Criterion 19 and is therefore acceptable.

7.5 Safety-Related Display Instrumentation

We have reviewed the designs for the instrumentation systems that provide information to the operator to enable him to perform the required manual safety functions and for post-accident and incident surveillance. The significant areas of review are discussed below. We conclude that the design provides an equivalent or improved degree of surveillance as compared with previously accepted designs and is therefore acceptable.

7.5.1 Accident and Post-Accident Monitoring

Our requirements with regard to accident and post-accident monitoring instrumentation are that the instrumentation should be:

(1) Qualified for accident environment (post-accident instruments only).

(2) Redundant with at least one channel recorded.

(3) Energized from onsite power supplies and designed in accordance with the requirements of IEEE Standard 279-1971.

We have reviewed the portions of the Final Safety Analysis Report which addresses the applicant's conformance to the above requirements. The applicant identified in

Table 7.5-1 of the Final Safety Analysis Report the parameters that are used for accident and post-accident monitoring.

We conclude that the design for accident and post-accident monitoring satisfies our requirements as stated in Section 7.1 of this report and provides an equivalent degree of surveillance as compared to the designs previously licensed and is therefore acceptable. We are currently reviewing the adequacy of the applicant's list of parameters deemed essential for accident and post-accident monitoring and will report our evaluation of this matter in a supplement to this report.

7.5.2 Inoperable Bypass Status Indication

The inoperable bypass status indication was reviewed. The present design provides automatic bypass status indication at systems level. The applicant's design includes a system level manually initiated inoperable status pushbutton for each channel in addition to the automatic inoperable status indication. These pushbuttons are located in the control room and are used to supplement administrative procedures that identify safety-related system inoperability. We have reviewed this design and conclude that the design conforms to the staff's recommendations as stated in Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," and is acceptable.

7.6 Other Systems Required for Safety

We reviewed the selected schematic drawings of the circuitry related to other systems required for safety. Significant areas of review are discussed below. We have concluded that the design of these systems meets the staff's requirements as stated in Section 7.1 of this report and is acceptable, subject only to the resolution of the items discussed in Sections 7.6.3 and 7.6.4.

7.6.1 Safety Injection Tank Isolation Valves

Each of the four safety injection tanks are provided with a motor operated isolation valve. The control circuits for these valves are designed to automatically open and prevent closing of these valves whenever the reactor coolant pressure is above 700 pounds per square inch gauge. In addition, whenever a safety injection actuation signal is present the valves will automatically open when above 700 pounds per square inch gauge. The interlock functions are derived from two independent and diverse instrument channels. Each channel operates two of the four isolation valves. Valve position indication is provided in the control room to monitor the status of these valves at all times. In addition, the motor circuit breakers for each valve will be maintained open under administrative control to assure that the valves will remain open during reactor operation.

Our review of the design of the safety injection tank isolation valves included review of the logic diagrams, and selected schematic diagrams included in the Final Safety Analysis Report. We have concluded that the design meets the staff's requirements stated in Branch Technical Position EICSB Number 4 of Appendix A of the Standard Review Plan and is, therefore, acceptable.

7.6.2 Shutdown Cooling System Low Pressure to High Pressure Isolation

Two motor-operated valves are provided in series in the inlet line from the reactor coolant system to the shutdown cooling system. These valves are normally closed and are only open for shutdown cooling after the system pressure is reduced below 300 pounds per square inch gauge. The valves will also close automatically whenever the system pressure is increased above 300 pounds per square inch gauge. The interlock functions are derived from two independent pressure instrument channels of diverse principles, with each valve controlled by only one of the channels.

During our review the applicant was requested to (1) verify that the consequences of an inadvertent valve closure of these valves during the shutdown cooling mode would not degrade the core cooling system below an acceptable level, or (2) modify the design to make the consequences of such a failure acceptable. Section 5.6.3 contains additional information regarding the evaluation of this design.

Our review of the shutdown cooling system low pressure to high pressure isolation included the review of the logic diagrams and selected schematics included in the Final Safety Analysis Report. We conclude that the design meets the staff's requirements as stated in Section 7.1 of this report and the staff's requirements stated in Branch Technical Position EICSB Number 3 of Appendix A of the Standard Review Plan and is, therefore, acceptable.

7.6.3 Safety-Related Fluid Systems

The applicant was requested to address inadvertent actuation of all electrically-operated passive and active components in the safety-related fluid systems and evaluate the effects relative to the single failure criterion, and identify the degree of conformance to the staff's position as stated in Branch Technical Position EICSB Number 18 of Appendix A of the Standard Review Plan.

Our review of the proposed design concludes that the design does not conform fully with the requirements stated in the above position and is unacceptable. We have identified a recirculation valve to the refueling water storage tank (valve number 2CV-5628-2) which, if inadvertently actuated to close during the initial stages of safety injection, could compromise the functional integrity of the emergency core cooling system. The applicant submitted a design to preclude inadvertent closure of this valve due to a single electrical failure. Our review of the design concludes that our requirements as stated in EICSB Position 18 have been implemented only in part and as such, the design is unacceptable.

The applicant was requested to modify the above designs to provide redundant position indication in the control room which would meet the single failure criterion. In addition, the applicant will be required to submit the detailed schematics for this valve to demonstrate how the design criteria and our requirements have been implemented prior to a decision or issuance of the operating license. We will report our final evaluation of this item in a supplement to this report.

7.6.4 Reactor Coolant Pump Coast Down Capabilities

Our concern regarding reactor coolant pump coastdown capabilities in the event that the pump breakers failed to isolate the power supplies during an underfrequency condition was reviewed with the applicant. In view of the fact that a generic resolution of this problem is being pursued by the applicant (and the staff), we deferred application of the Branch Technical Position EICSB Number 15 (stated in Appendix A of the Standard Review Plan) until completion of this generic study.

In the interim it has been established that until it can be demonstrated by analysis that an underfrequency condition will not prevent the pumps from performing their coastdown function, the tripping of the reactor coolant pump breakers is considered to be a required safety function. The applicant was required to either:

(1) Modify the design to include the capability to automatically trip the reactor coolant pump breakers for underfrequency conditions, and provide underfrequency trip sensors designed in accordance with IEEE Standard 279-1971 and qualified as Class IE devices or,

(2) Provide an analysis demonstrating that an underfrequency condition will not prevent the pumps from performing their coastdown function. This analysis should include:

    (a) The limiting underfrequency condition.

    (b) A histogram of underfrequency transients showing these degraded frequency occurrences versus frequency decay rate.

    (c) Information showing the duration of underfrequency transients and maximum anticipated frequency decay rate.

In response in Amendment 43 to the Final Safety Analysis Report, the applicant submitted an analysis to demonstrate that in the event the pump breakers failed to isolate the power supplies during an underfrequency condition, the reactor protection system would trip the reactor in sufficient time to preclude the reactor from going below the minimum departure from nucleate boiling ratio limits. The analysis was based assuming the worst case condition of 6.47 Hertz per seconds frequency decay rate occurring at 30 percent rated load and a power factor of 0.98. Although the submitted worst case analysis indicates that the design provides sufficient margin between the expected and the minimum departure from nucleate boiling ratio values, the applicant was requested to submit additional analyses to demonstrate that the consequences of underfrequency degradation at intermediate and high operating loads would also be acceptable. We conclude that the applicant's design is acceptable, subject to the satisfactory documentation, review and acceptability of the additional analyses.

7.7 Control Systems Not Required for Safety

Systems not required for safety identified in the Final Safety Analysis Report are:

(1) Reactor control system
(2) Reactor coolant pressure control system
(3) Pressurizer level control system
(4) Feedwater control system
(5) Steam bypass control system
(6) Boron control system
(7) Load dispatch system
(8) Incore instrumentation system
(9) Core operating limit supervisory system (COLSS)
(10) Plant computer system

Our review of these systems compared the ANO-2 design with those designs provided in recently licensed plants. The following systems incorporate major design changes that have not been previously reviewed:

(1) A computer-based core operating limit supervisory system (COLSS) is used in plant control to ensure that the operator maintains the reactor system within the conditions assumed in the safety analysis,

(2) The plant computer system provides group sequencing for the control element assemblies, and

(3) The reactor control system positions the control element assemblies.

The designs of COLSS and the plant computer system and their interaction with the plant protection system are currently being reviewed and evaluated by the staff as safety systems. Our evaluation and conclusions are discussed in Section 7.2.3 of this report.

Regarding the reactor control system, Amendment 32 to the Final Safety Analysis Report identified a redesign of the part length control element assemblies. The major design change was the conversion of the control circuitry of the drive mechanisms for these rods from a non-tripping to a tripping system. To provide for the completion of our review with regard to spurious withdrawal or insertion of the control element assemblies, the applicant was requested to provide a failure mode and effects analysis that addressed inadvertent tripping of the two part length control element assembly subgroups. In addition, the applicant was requested to submit the final electrical schematics that describe this system and define their alternating current power distribution arrangement.

In Amendment 36 the applicant provided a general description of this system and submitted a general logic arrangement drawing. Based on the information presented we could not support the applicant's claim at that time that the design precluded tripping both part length control element assembly subgroups when subjected to a single failure.

Based on our review of information subsequently provided by the applicant we determined that a single failure in the "zero crossing detector" module could cause both part length control element assembly subgroups to drop. The applicant was requested to and agreed to modify the design by providing a barrier between the two part length control element assembly circuit board cards to preclude simultaneous degradation of both circuit boards.

Based on our review of the final design of the part length control element assemblies, we conclude that the design satisfactorily precludes tripping both part length subgroups when subjected to a single failure and is, therefore, acceptable.

The dropping of part length control element assembly subgroups is also discussed in Section 15.3 of this report.

With respect to the nonsafety control system, with the exception of the core operating limit supervisory system, we find that the ANO-2 design is similar to those designs of recently licensed plants. We have concluded that the differences are minor and do not affect our previous conclusions and that the designs of these control systems are also acceptable.

We will report the final evaluation of the core operating limit supervisory system in a supplement to this report.

7.8 Electrical Penetrations

The electrical penetrations are designed and tested to meet the requirements of IEEE Standard 317-1972, "Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations."

Our review of the circuits associated with the electrical penetrations verified that sufficient backup protection is provided to assure that the functional integrity of the penetrations is maintained. The applicant has provided backup breakers in series with the primary breakers for the high voltage circuits (i.e., 6900 volts alternating current, and 480 volts alternating current. For the low voltage circuits, the applicant provided fuses as backup protection to the primary breakers and justified this design by analysis.

We, therefore, conclude that this design satisfies the requirement of General Design Criterion 50 and is acceptable.

7.9 Cable Separation Criteria

We have reviewed the applicant's criteria and procedures for maintaining the integrity, physical independence and for identification of safety-related equipment and circuits. During our review we identified areas where adequate separation was not provided or the information provided was too general to complete our evaluation. These specific areas are identified below.

On the basis of our review we conclude that, conditioned on the satisfactory resolution of the items identified in Sections 7.1 and 7.9.4, the separation criteria for the ANO-2 design provides an equivalent or improved degree of separation as compared to designs of recently licensed plants and is acceptable.

7.9.1 Separation Between Class 1E and Non-Class 1E Raceways in the Cable Spreading Area and Main Control Room

During our initial review the separation criteria between nonsafety trays and Class IE trays was found to be incomplete. The initial design did not address conditions where trays carrying non-Class IE cables are interposed between trays carrying redundant Class IE cables. In response to our concern the applicant modified their design and amended the Final Safety Analysis Report to include criteria for areas where non-Class IE raceways are interposed between redundant Class IE raceways. We have reviewed the modified criteria and the design requirement which provide barriers or flush fitting solid top and bottom covers on the non-Class IE raceway when they are interposed between redundant Class IE raceways. We conclude that the modified design satisfies the staff's requirements stated in Section 7.1 of this report and therefore is acceptable.

In addition, Amendment 39 identified a change in the criteria for cable splices. This change permits cable splices in non-Class IE power cables with long runs. In our review of this modification the applicant has assured us that these non-Class IE power cables will not be run with or associated with any Class IE circuit raceway. We conclude that this design change satisfies the staff's requirements stated in Section 7.1 of this report and therefore is acceptable.

7.9.2 Separation Between Wiring in Instrument Cabinets

The applicant identified that in selected instrument cabinets non-Class IE wiring is bundled together with Class IE wiring. Although we recognize that the nonsafety wiring is associated with low energy circuits, the design did not originally appear to provide adequate assurance that a single failure would not degrade the independence of the redundant safety channels. The applicant was requested to either modify the design by providing adequate separation between Class IE wiring and non-Class IE wiring inside the cabinets or demonstrate that a single event in these nonsafety circuits (e.g., "electrical noise") would not degrade the protection system operability below an acceptable level.

In response, the applicant submitted a detailed analysis describing their design, identified the maximum credible potential faults that can exist on these circuits, and identified the types of isolation devices and noise rejection capabilities of such devices. Based on our review of the analysis and verification of the design during our site visit, we find that the applicant provided sufficient justification to assure that faults imposed on the non-Class IE circuits routed with Class IE circuits inside the safety-related cabinets would not degrade the safety systems below an acceptable level. Therefore, we conclude that this design is acceptable.

7.9.3 Criteria for Physical Identification of Safety-Related Equipment

In Amendment 36 the applicant identified a change to the criteria for physical identification of safety-related equipment by deleting information in Section 8.3.1.5 of the Final Safety Analysis Report, which was previously reviewed and accepted by the staff. This change significantly modified the degree of conformance of the design to the requirements stated in Section 4.22 of IEEE Standard 279-1971 and was, therefore, unacceptable. Subsequently, the applicant amended the Final Safety Analysis Report to incorporate the criteria for physical identification of safety-related equipment as previously accepted. We have reviewed the amended criteria and have verified the implemented design during our site visit. We conclude that the design satisfies our requirements as stated in Section 7.1 of this report and is, therefore, acceptable.

We will report the resolution of this item in a supplement to this report.

7.9.4 Separation Criteria Between Redundant Class IE Circuits in Metal Conduits

The applicant utilizes metal conduits for various Class IE essential circuit routings. During our review we identified areas where redundant channel wiring routed in separate and independent metal conduits, was routed in close proximity (i.e., one inch apart) to each other without provisions for barriers other than the conduit itself. Although we recognize that the metal conduits may be a valid barrier for certain types of events, we do not consider that conduits alone are adequate barriers for all types of events. The applicant was requested to review the 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.

Incidents such as a fire in an open tray crossing under redundant conduits was cited as an example that may effect the cables inside the conduit and degrade the system circuits below an acceptable level. The applicant committed to evaluate their design and will advise us regarding their findings on this concern. We will review the applicant's response when submitted and report our evaluation in a supplement to this report.

7.10 Protection System Response Time Testing

The ANO-2 facility technical specifications include requirements for periodic response time testing of the protection system and the applicant is currently developing test procedures that would verify response time requirements. We conclude that these measures are acceptable. The applicant was requested to submit sample test procedures of typical safety channel response time testing to verify the adequacy of the design. Our position regarding response time testing is that until experience with the ANO-2 design or other identical designs demonstrate that the protection system response times, including sensor response time, do not change over long intervals of operating experience, the response time testing should be repeated periodically. Therefore, we will require that the system response time test be repeated not less frequently than every 18 months. Accordingly, the technical technical specifications will include a requirement specifying the test program. With the above technical specification as is presently included in Section 3/4.3 of the technical specifications, we consider this matter resolved.

TABLE 7.1

CORE PROTECTION CALCULATOR SYSTEM POSITIONS

A listing of the staff positions that developed during our evaluation of the CPCS is presented below. Each position's number and title is followed by the section number of Appendix D to this report in which the position is discussed in further detail and the position's current status. We will report our further evaluation of the outstanding issues in a supplement to this report.

(1) Uncertainty Associated with the Algorithms, Section 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 during startup to demonstrate the adequacy of the axial power synthesis by comparing to in-core measurements and analysis for various power conditions.

(2) Conservatism of the CPCS Response to Dropped Control Element Assemblies, Section 3.5, Closed

We require three-dimensional transient power distribution studies be performed to assure that effects of dropped off-center CEA's are conservatively predicted by each of the four CPC channels. Our concerns are the adequacy of delta temperature power basis for rapid transients when ex-core sensors are not available.

(3) I/I Converter Isolation Device, Section 4.1.2, Closed

It is the staff's position that the current-to-current (I/I) converter isolation devices be qualified in accordance with specified criteria, and that the results of the qualification tests be submitted for our review including the test plan, test set-up, test duration and acceptability requirements.

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

We will require that the applicant identify their design basis events for the control element assembly calculator (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 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.

(6) Position Isolation Amplifiers, Section 4.1.4, Closed

It is the staff's position that the isolation amplifiers be qualified in accordance with the specified criteria and that the results of the qualification tests be submitted for our review, including the test plan, test set-up, test procedures and acceptability requirements.

(7) Protected Memory, Section 4.2.1, Closed

The ANO-2 memory protection hardware causes instruction attempting to write into protected memory to be converted into read instructions. No safety credit is allowed for this feature unless failures in the system that result in attempts to write in protected memory are annunciated to the operator. Furthermore, if safety credit is desired, we shall require that a status lamp seal indicate the state of operation.

(8) Time Interval of Periodic Testing, Section 4.2.1, Outstanding

(a) The applicant is to develop an acceptable analysis of the CPCS reliability in accordance with the requirements of Section 4.4 of IEEE Standard 279-1971 and Sections 4.2 and 4.3(1) and (7) of IEEE Standard 338-1971. This analysis will provide the basis for evaluating the performance data obtained in parts 8B and 8C and for establishing and modifying the periodic test interval after the initial operation period.

(b) Completion of the supplemental qualification testing identified in the staff's July 7, 1976 letter and documentation of the system reliability during this test interval is required.

(c) During the first six months of operation, the periodic test interval should be significantly more frequent than the proposed 30 days. The interval could in part be based on the results of (a) and (b) above. All failures during this period should be carefully recorded, classified and analyzed. At the end of the six-month period, the performance of the CPCS should be analyzed using the model developed in (a) above and the operational reliability assessed. Based on these results, the test interval could then be modified.

(9) System Functional Testing, Section 4.2.1, Outstanding

The applicant has not provided definitive and adequate procedures for periodically checking and verifying the functional operation of the CPCS in accordance with the requirements of General Design Criterion 21, "Protection System Reliability and Testability" and the guidelines of Section 4.3 of IEEE Standard 338-1971, "Criteria for the Periodic Testing of Nuclear Power Generating Station

Protection Systems." To verify the functional performance of the CPCS and to insure adequate overlap, the periodic test program should be modified to include procedures for testing each trip function in each channel from sensor input to the CPCS to trip output to the reactor trip system. The procedures should be sufficient to verify that the protective action for each function will ensue for the expected extremes for each sensor.

(10) <u>Periodic Testing, Addressable Constants, Section 4.2.1, Outstanding</u>

For any changes to addressable constants, the test program will identify calculation errors which may or may not be actual errors. We will require that the applicant develop practical techniques and procedures for verifying calculated results after changes to addressable constants.

(11) <u>Environmental Performance Qualification, Section 4.2.4, Outstanding</u>

In accordance with the requirements of IEEE Standard 279-1971, Sections 4.1, 4.3, and 4.4, and IEEE Standard 323-1971, Section 4.3, prior to initial operation, a satisfactory environmental test of the integrated system (exclusive of sensors) should be performed or an acceptable analysis clearly establishing the adequacy of component testing is required for staff evaluation. The staff's July 7, 1976 letter includes the requirements for this qualification testing.

(12) <u>Electrical Noise and Isolation Qualification, Sections 4.1.4 and 4.2.4, Outstanding</u>

Tests for electrical isolation separation and noise susceptibility will be required. The applicant shall develop and submit for approval test plans 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.

(13) <u>Sensor Qualification, Section 4.2.4, Outstanding</u>

For those unique sensors (reactor coolant pump speed and control element assembly position) the applicant is required to submit documentation to verify their environmental performance qualification.

(14) <u>Seismic Qualifications, Section 7.4.2.5, Outstanding</u>

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 3.11, Outstanding

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.

(16) Quality Assurance Plan, Section 4.3, Outstanding

The results from our recent audits of the hardware and the software have served to focus our concerns upon the quality assurance program used for system development. Upon evaluation of these results, we have concluded that the applicant is not complying to the quality assurance plan with regard to the following 10 CFR Part 50 criteria:

(1) Criterion 1, Quality Standards and Records, Appendix A - General Design Criteria for Nuclear Reactor Plants, and

(2) Appendix B - Quality Assurance Criteria for Nuclear Power Plants and Reprocessing Plants.

The bases for this conclusion are deviations from stated positions, the lack of documented system software development guidelines, and design errors uncovered in our review to date.

As stated in Appendix B of 10 CFR Part 50, the Quality Assurance Program must be applied to the design, fabrication, construction and testing of the structures, systems, and components of the facility. It is our position that a Quality Assurance Plan is required for the core protection calculator system to embrace all activities from the current frozen design (as of November 24, 1975) to the final design of the installed system. An effective quality assurance program is required to minimize design errors and is an important component to the qualitative reliability of the system. The acceptability of the Quality Assurance Plan and the compliance with the plan must be assessed by the staff prior to the completion of the safety evaluation.

In addition to the criteria stated in Appendix B of 10 CFR Part 50, the staff desires to emphasize the positions entitled, "Performance Qualification of Software Change Procedures," with respect to the Quality Assurance Plan.

(17) Performance Qualification Testing Section 4.4, Closed (See Position 24)

For evaluation of dynamic test results, we will require submittal of FORTRAN Codes test results for selected cases to permit comparison with CPCS Performance Qualification test results. In addition, we will require that transient analyses be performed for selected dynamic test cases using the codes normally employed by Combustion Engineering for Section 15.0 of Final Safety Analysis Report transient analyses. This will enable the staff to determine if the time to trip output on the CPCS based on projected DNBR of 1.3 is reached. The trip signal input to the more sophisticated codes will be the time to trip for respective cases on the CPCS.

The staff will accept for review the Phase II test results previously obtained on the plant system. However, all software revisions since those tests must be implemented in accordance with qualified change procedures (see "Qualification of Software Change Procedure"), and all Phase II test cases must be repeated on the FORTRAN version of the final program. If final test results are not essentially identical to previous results, a repeat of Phase II test on the plant system configuration will be required.

(18) Burn-In Test, Section 4.1.4, Outstanding

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) Qualification of Software Change Procedures, Section 4.4, Outstanding

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 applicants proposed procedures 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 systems.

(e) Step 4 need not be repeated for future software revisions. All software design changes and revisions to constants in memory (except addressable constants) are subject to documentation, review and approval by the Regulatory staff.

(20) Data Link to Plant Computer, Section 4.2.3, Outstanding

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 system 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.

(21) Checksum, Section 4.2.1, Outstanding

Our review of the paper tape memory dump representing the frozen design revealed that the checksum values are not the same in all redundant channels. For consistency and inspection purposes, we require that a procedure be implemented that will result in checksum agreement between corresponding blocks of all redundant computer channels in the system. Furthermore, the checksums in each channel must be available for inspection purposes through the operator's module.

(22) Timeout Error Detection for Penalty Factor Transmission, Section 4.3, Outstanding

We have noted that the write instruction designed to transmit the penalty factor from each CEAC to each of the CPC's does not have an error response routine for Input/Output (I/O) timeout. Since all other I/O operations in the system have this feature, we shall require that the CEAC-penalty factor write commands be likewise provided with error test and response routines.

(23) Watchdog Timer, Section 4.3, Outstanding

(a) Core Protection Calculator (CPC)

We shall require an automatic (hard-wired) trip of the associated protection channel upon timeout of the watchdog timer. From the safety review of the design information submitted to date, we have concluded that a significantly larger number of the CPC's safety functions would be monitored if the watchdog timer reset command were moved from the clock interrupt handler to the trip sequence program. In the interest of safety we require that the watchdog timer be reset from this trip sequence program.

(b) Control Element Assembly Calculator (CEAC)

Upon timeout of the watchdog timer, we require that the "fail bit" be set in the CEAC output. From the safety review of the design information submitted to date, we have concluded that a significantly larger number of the CEAC's safety functions would be monitored if the watchdog timer reset command were moved from the clock interrupt handler to the penalty factor algorithm module. In the interest of safety we require that the watchdog timer be reset from the penalty factor algorithm module.

(24) <u>Phase II Test and Test Report, Section 4.4, Outstanding</u>

Upon review of the Combustion Engineering topical report CENPD-222 "Core Protection Calculator System (CPCS) Phase II Design Qualification Test Report," we have concluded that the computer program has not been tested to quality standards commensurate with the importance of the safety functions to be performed. On this basis, we find the Phase II Test Report unacceptable, including the test procedures and acceptance criteria utilized for the tests. Furthermore, the test report is incomplete in the analysis of test cases. This has raised concerns about the functional adequacy of the system. Because of these deficiencies, we do not consider the Phase II Test Report as an acceptable verification of the CPCS computer program.

Our major areas of concern are as follows: (Sequence is of no significance - all concerns are of equal importance.)

(a) Of the 36 static test cases, 18 failed the stated acceptance criteria. Coding error was the prime cause of not satisfying criteria for 14 cases. For verification purposes, the coding error was deliberately inserted into the FORTRAN simulation program to generate erroneous simulation results to compare with the results produced by the Core Protection Calculator System (CPCS). These procedures are unacceptable. Also, these actions are in direct violation of the stated test procedures described in Section 7A.4.7.6, Design and Performance Qualification Testing of the Final Safety Analysis Report.

From the test report, it appears that the coding errors of a fixed point multiplication overflow and a floating point multiplication underflow were detected in the execution of the static test cases and of the dynamic test cases. The execution of test cases with known coding errors in the computer program violates the test procedures stated in Section 7A.4.7.6 of the Final Safety Analysis Report.

Thus, because of the procedures used in the execution of the Phase II test cases (both static and dynamic test cases), we find the test results unacceptable. Also because of the large error tolerances used for evaluating acceptability of test results, we conclude that the verification of the correct implementation of the CPCS protection algorithms is not shown in the test report.

We shall require that the verification of the correct implementation of the protection algorithms be conducted with procedures which as a minimum are described in Section 7A.4.7.6 of the Final Safety Analysis Report. In addition, acceptance criteria must be specified and justified.

(b) As a result of the analysis of the test cases, several computer program changes have been proposed and are identified in the test report. In general, the test report does not provide the basis of change, such as test case results with explanations of why the change is required. In order to conduct an independent review of the proposed changes, the staff requires the basis for all proposed changes to the program. This must include all changes identified in the test report along with supporting test cases and explanations such as the results for and explanation of dynamic test cases 11 and 21.

(c) In the discussion on dynamic test acceptance criteria, it is assumed that the initial steady state deviations may be applied as a uniform bias throughout the transient. An analysis to support this assumption will be required.

(d) In the discussion of dynamic test case 15, the eight-second delay in trip time is attributed to improper initial conditions of the test case. For this dynamic test case, it is not clear that the delay in trip is uniquely attributable to initial offset in parameters. We shall require that the applicant provide the detailed information to support this conclusion.

(e) The analysis of the selected dynamic test cases presented in the report are incomplete. Trip time data for each channel and the FORTRAN simulation trip time are presented, but the response comparisons are only conducted for the trip point. No quantitative evaluation of error and error time history between the state variables presented for the FORTRAN simulation and the corresponding CPCS state variables is made. The lack of this comparison does not allow for an assessment of the implementation adequacy of the dynamic algorithms. We shall require that a comparison of the response of principle state variables from the FORTRAN simulation be made with the corresponding state variables of the CPCS. The resulting error history should be sufficiently small (and acceptable to the staff) to demonstrate adequate implementation of the dynamic algorithms. Furthermore, a summary table of trip times for all of the dynamic test cases is required for review purposes.

(f) The excore detector readings presented in the description do not appear to include sufficient variation in relative magnitude to test all of the various correction options inherent in the local power density trip functional program. We require documentation to clearly demonstrate that the shape correction routines were all correctly implemented and tested.

(g) In evaluating the static test cases, the staff had difficulty in assessing test procedures, the input parameters used in the test cases and the analysis of the limited number of intermediate and output parameters presented for the testing. In evaluating the input data that were used in the static test cases, we found that the input had been modified for greater than 50 percent of the cases.

To evaluate the above problems, the applicant's test plan must be provided on which the Phase II test report is based. The test plan should include acceptance criteria, the procedures used in the testing, a description of and objectives of each test case, the input data to be used in each case, and especially the parameters and variables to be recorded and analyzed for each test case.

(h) The Phase II test report does not address a test observed by the staff during which a channel failed to trip. (Trip Report - Demonstration of CPC Testing -November 24-26, 1975.) We shall require an analysis of this case as part of the test program.

(i) Static Test Acceptance Criteria: The error bands specified for static test acceptance criteria must be clarified and justified. The clarification should include identification and qualification of error components comprising the overall uncertainty band, and description of how they are combined to obtain the overall uncertainty tolerance. All CPC error components inherent in the Phase II test configuration must be included in the analysis; i.e., analog to digital conversion error, simulator errors, noise effects, and processing error in the digital computation must be quantified and justified in supporting documentation.

Support data must be provided to justify the increase in acceptance criteria ($\pm$ five percent error) due to the Power Utility Plant Simulator (PUPS) output hardware, cabling, and noise effects.

(j) Dynamic Test Acceptance Criteria: The previous position of the staff on "Performance Qualification Testing" requires that transient analysis be performed for selected dynamic test cases using the codes normally employed for Section 15.0 of Final Safety Analysis Report transient analyses.

The required time of trip to prevent DNBR from going below 1.3 as determined by these analyses should be specified for applicable cases as one of the acceptance criteria.

(k) Scaling Errors: The staff will require evidence that steps have been taken to preclude additional errors in the scaled range of program variables such as occurred for Static Test Case 14.

(l) Round Off Errors: The staff will require further analysis of the Static Test Case II error. It is not clear why the results should be sensitive to an exact equality of two different instrument signals, since the inherent measurement error makes such a comparison meaningless. Discuss provisions which are being taken to assure that other errors of similar logic origin do not exist. The logic should be justified and the deviation in results due to this logic should be quantified.

(m) Auto Restarts: The effect of recurring auto restarts that occurred during the use of test procedures C and D should be discussed. The staff will require details of the program changes designed to resolve this problem. The staff will also require details of the testing planned to conclusively demonstrate that the problem is resolved.

(n) Dynamic Test Cases: A repetition of the dynamic tests will be required. The dynamic test cases are the primary basis for evaluation of the dynamic algorithms. The staff regards that it is necessary to demonstrate the qualification of the corrected design as identified by Position 18 (Burn-In Test). All design changes as identified by the applicant and the staff cannot be adequately evaluated without this testing.

(25) Maintainability of the Core Protection Calculator System, Section 7.4.2.1, Outstanding

IEEE Standard 279-1971, Section 4.21, "System Repair," identifies maintainability as one of the requirements for the reactor protection system. The discussions in Sections 7A.4.8.2.1 and 7A.4.7.2.3 of the ANO-2 Final Safety Analysis Report do not adequately address the maintainability of the Core Protection Calculator System (CPCS). Industrial experience with process computer systems has identified several concerns regarding maintainability of digital computer systems over the operating life of the plant. These concerns are summarized as follows:

(a) Lack of standardization in hardware and software design has led to difficulties in identifying second sources of parts supply.

(b) The short commercial life cycle of electronic parts compared to plant operating life has resulted in obsolescence of equipment and unavailability of spare parts.

(c) Suppliers' and users' lack of experience, trained technicians to maintain equipment.

(d) Incomplete maintenance and trouble shooting procedures and system documentation has made maintenance difficult.

As a result of these concerns, and since the ANO-2 represents the first system of its type for use in a reactor protection system, we require that the CPCS maintainability plan for the life of the plant be documented and docketed for the regulatory staff's review and evaluation. In addition to the information presented in the Final Safety Analysis Report, the plan should address the following:

(a) The maintenance actions (i.e., preparation, failure verification and fault location, replacement part procurement, repair and verification tests) required.

(b) The maintenance diagnostic and repair features (e.g., displays and controls, external accessibility, test points, cables and connectors, internal accessibility, manuals and test equipment).

(c) Hardware and software maintenance support to be provided by vendors (and/or others) and personnel qualification and training to support this maintenance service.

(d) Hardware and software maintenance to be provided to the applicant and personnel qualifications and training to support this maintenance.

(26) Optical Isolator, Section 4.1.4, Outstanding

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.)

(27) Periodic Testing of Isolation Devices, Section 4.2.1, Outstanding

The unique design of the CPCS relies on many isolation devices (i.e., optical isolators for control element assembly calculator to core protection calculator data transfer and control element assembly position signals) to maintain electrical independence among the protection channels. The ability of these devices to maintain the isolation among channels is one of the bases for accepting the design of the CPCS. It is our concern that failures of the isolation characteristics of these devices would seriously compromise the ability of the CPCS to function. The current periodic test procedures do not include provision for verifying that the isolation characteristics of these devices has not failed. Therefore, it is our position that periodic tests to verify the isolation characteristics of those isolation devices used to ensure channel independence should be performed. We will require that the applicant submit, for our review and approval, a test procedure for periodically checking the isolation characteristics.

8.0 ELECTRIC POWER

8.1 General

General Design Criteria 17 and 18 and the following Regulatory Guides and standards were utilized as the primary bases for evaluating the adequacy of the electric power systems of the Arkansas Nuclear One - Unit 2 plant.

(1) Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between their Distribution Systems."

(2) Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies."

(3) Regulatory Guide 1.32, "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants."

(4) Regulatory Guide 1.41, "Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments."

(5) IEEE Standard 308-1971, "Criteria For Class 1E Electric Systems for Nuclear Power Generating Stations."

8.2 Offsite Power Systems

The switchyard at the site consists of a 500 kilovolt yard and a 161 kilovolt yard interconnected by a 600 million volt ampere auto transformer. This switchyard supplies power to both units at the site. The 500 kilovolt system is a two bus breaker-and-a-half arrangement which provides terminal facilities for three 500 kilovolt transmission lines, a unit auxiliary transformer and the bus tie auto transformer. The 161 kilovolt system is a four element ring bus arrangement which provides terminal facilities for two 161 kilovolt transmission lines, the bus tie auto transformer and one of the three startup transformers at the site. This startup transformer is shared between both Unit 1 and Unit 2. The bus tie auto transformer interconnects the 500 kilovolt and the 161 kilovolt yards. This auto transformer provides terminal facilities for two additional startup transformers with one dedicated for each unit at the site.

The 500 kilovolt transmission lines leave the site on divergent rights-of-way. Although cross-over of transmission lines and common right-of-way occurs in discreet sections along the transmission network, the routing precludes loss of all offsite power due to a single structural failure in any of the transmission towers.

Primary and back-up protective relaying systems have been provided for each 500 kilovolt and 151 kilovolt circuit in addition to circuit breaker failure protection. Two independent and separate sources of direct current control power are supplied to the switchyard from station batteries complete with battery charger and distribution system. Loss of either direct current source will not inhibit the ability to supply offsite power to the station.

The offsite power to ANO-2 is supplied from the switchyard to two station startup transformers providing two immediate access sources. A third source of offsite power may be made available by manually removing the generator disconnect links, thus permitting backfeed through the main transformer to the unit auxiliary transformer. However, credit for this third source of power has not been assumed in our evaluation.

Each of the station transformers are provided with double secondary windings. One of the secondary windings feed the nonessential 6.9 kilovolt split bus distribution system. The other secondary winding feeds the nonessential 4.16 kilovolt split bus distribution system. The power supply to the nonessential 4.16 kilovolt bus in turn feeds one of two essential 4.16 kilovolt buses through a 1200 ampere breaker.

Normally, power to the split bus distribution system is derived through the unit auxiliary transformer. On a unit trip, the system is automatically transferred to the startup transformer. By preselection accomplished in the control room, one of the startup transformers is designated as the preferred reserve power source and the other startup transformer as its backup. Thus, if either startup transformer is out of service, the remaining transformer is available to automatically supply each portion of the split bus distribution system should the auxiliary transformer fail. Interlocks are provided to prevent paralleling the power sources to a common bus.

During our review the applicant verified that the startup transformer, which is shared between Arkansas Nuclear One, Units 1 and 2, has sufficient capacity to supply the required emergency loads of one unit while simultaneously supplying the necessary auxiliary loads in the other unit to achieve an orderly shutdown. In addition, the applicant has conducted power systems stability studies showing that a loss of the largest generating unit, or the most critical transmission line, will not adversely affect the stability of the remainder of the transmission or the ability to provide offsite power to the Arkansas Nuclear One - Unit 2 plant.

We have identified a concern regarding grid stability as a result of the July 5, 1976 grid voltage degradation at Millstone Nuclear Station. The applicant was requested to evaluate the Arkansas Nuclear One - Unit 2 design for 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. We are presently awaiting the applicant's response to our request for information regarding this matter.

We have reviewed the design of the offsite power system, which included the review of selected schematics presently included in the Final Safety Analysis Report. We have concluded that the design satisfies the applicable criteria outlined in Section 8.1 of this report and is therefore acceptable, subject only to the satisfactory resolution of the item identified above.

8.3 Onsite Power Systems

8.3.1 Alternating Current Power Systems

The alternating current emergency onsite power system is comprised of two redundant and independent distribution systems, each powered by one of two redundant diesel generators.

Each distribution system includes 4160 volt, 480 volt, and 120 volt load centers which provide power to the various safety loads. Each of the redundant load groups consist of a complement of safety equipment needed to achieve safe shutdown and/or to mitigate the consequences of a design basis accident. An additional third group of loads consisting of a third high pressure safety injection pump, a third service water pump and a third charging pump is provided as backup to the main load groups. The third group can be powered from either distribution system. The tie breakers for the third group are interlocked electrically and mechanically to prevent these loads from being simultaneously connected to both redundant onsite distribution systems. The interlocks and physical arrangement of these distribution systems satisfy the requirements of Regulatory Guide 1.6 and are acceptable.

The two diesel generators are each rated at 2850 kilowatts continuous, 3100 kilowatts for 2000 hours, and 3500 kilowatts for 30 minutes. The maximum emergency load that they will be required to carry is 2812 kilowatts. This is within the limit recommended by Regulatory Guide 1.9.

Each diesel generator is equipped with mechanical and electrical trip interlocks to ensure personnel protection and to prevent or limit equipment damage. An accident signal generated in the plant protection system will cause a bypass of the emergency diesel generator trips except the engine overspeed trip, the generator differential trip, and a two-out-of-three low lube oil pressure trip.

Each diesel generator is automatically and independently started upon loss of normal offsite power or on receipt of a safety injection actuation signal. The diesels are designed to attain rated speed and voltage within 15 seconds of receiving the start signal and automatically accept the engineered safety features loads in a predetermined sequence. The sequencing of loads is accomplished by time delay relays in the circuitry of each individual component associated within each load group.

Our review revealed that the design provisions to periodically test these time delay relays were inadequate to assure that the relays will retain their functional operability within the requirements established by the safety analysis. As a resolution to this concern the Technical Specifications in Section 3/4.8 will require that the applicant periodically verify that the automatic load sequencer timers are operable and are within the sequence time established by the safety analysis.

Each diesel generator and its auxiliary support system including the fuel supply system is seismically qualified and housed in a separate seismic Category I installation. The total onsite fuel storage capacity assures at least nine days of a diesel generator operation at full load. This subject is discussed further in Section 9.5 of this report.

The station vital 120 volt alternating current system consists of four redundant channels. The four redundant 120 volt alternating current vital distribution buses supply power to the plant protection system instrumentation and related circuits. Each alternating current instrument bus is supplied separately from an inverter. Each pair of inverters is normally supplied from one 480 volt emergency bus and is backed-up from a vital 125 volt direct current source.

Our review of the onsite alternating current power distribution system included the review of selected electrical schematics, and the descriptive information presented in the Final Safety Analysis Report. We conclude that the design satisfies the applicable criteria as outlined in Section 8.1 of this report and is acceptable.

8.3.2 Direct Current Power System

The vital direct current power system for ANO-2 is comprised of two batteries each with an assigned static battery charger and distribution board. The direct current system is compatible with the two-division split-bus configuration of the alternating current system. The 125 volt direct current batteries are each rated at 1300 ampere hours and are adequately sized for an eight hour emergency period to supply power to all safety loads without assistance from the battery charger. The static battery chargers provided are each rated at a 480 volts alternating current, three phase, 60 hertz input with a nominal output of 125 volts direct current. Each charger is capable of supplying all steady state direct current loads required under any condition of operation while recharging the battery to a fully charged state from a discharged condition within eight hours. In addition, a third battery charger is provided as backup to either of the redundant direct current distribution systems. The tie breakers to this charger are interlocked electrically and mechanically to prevent this charger from being simultaneously connected to both redundant direct current distribution systems.

Each of the two direct current distribution systems are seismically qualified and are housed in separate seismic Category I installations. Each battery room is provided with its own ventilation system.

On the basis of our review of the descriptive information provided and the review of selected schematics included in the Final Safety Analysis Report, we conclude that the design satisfies the applicable criteria as outlined in Section 8.1 and is acceptable.

9.0 AUXILIARY SYSTEMS

9.1 General

We have reviewed the design bases of the auxiliary systems, including their safety-related objectives, and the manner in which these objectives are achieved.

The auxiliary systems necessary for safe reactor operation or shutdown include: portions of the service water system; ultimate heat sink; portions of the chemical and volume control system; the heating, ventilation and air conditioning systems for the control room and portions of the auxiliary building including emergency diesel generator rooms, battery rooms, switchgear rooms, diesel generator fuel oil storage and transfer systems; and the diesel generator auxiliary systems.

The systems necessary to assure safe handling of fuel and adequate cooling of the spent fuel include: new and spent fuel storage facilities, the spent fuel pool cooling and cleanup system, the fuel handling system and portions of the fuel handling area ventilation system. The fire protection system was reviewed to verify that failures in the system would not affect safe plant shutdown.

We have reviewed those auxiliary systems or portions of the system whose failure would not prevent safe shutdown but could, either directly or indirectly, be a potential source of a radiological release to the environment. These systems include the equipment and floor drain system and portions of the chemical and volume control system.

Other systems that are nonsafety-related include the nonessential portions of service water systems, the component cooling water system, demineralized water system, condensate storage and transfer system, compressed air system and the nonessential heating and ventilation systems. The acceptability of these systems was based on determining that: (1) where the system interfaces or connects to seismic Category I systems or components, seismic Category I isolation valves will be provided to physically separate the nonessential portions from the essential system or components, and (2) the failure of nonseismic systems or portions of the systems will not preclude the operation of safety-related systems or components located in close proximity. We find the above listed systems meet our criteria and, therefore, find them acceptable.

ANO-2 shares some systems or portions of systems with ANO-1. These include: yard water supply loop of the fire protection system, portions of the ANO-2 liquid radwaste system, the solid waste handling and storage facilities, the auxiliary building fuel handling crane, one startup transformer and the electrical switchyard, station security system, the emergency cooling pond, and the control room (Units 1 and 2 located adjacent to each other).

We have determined that the sharing of these systems between ANO-2 and ANO-1 does not impair their ability to perform their safety functions. Based on our review of those systems and components to be shared we conclude that their design meets the requirements of General Design Criterion 5.

9.2 Fuel Storage and Handling

9.2.1 New Fuel Storage

The new fuel storage area is designed to provide dry storage for 63 new fuel assemblies (approximately one-third of a core). The racks have a spacing which is sufficient to maintain a multiplication factor, K-effective, of 0.95 or less even in the event that the storage area is flooded with unborated water. The new fuel storage pit and racks are designed to seismic Category I requirements and protected from tornado missiles traveling in the horizontal direction. Tornado missile protection is further discussed in Sections 3.5 and 9.1.2 of this report.

Based on our review, we conclude that the design of the new fuel storage facility is in conformance with General Design Criterion 62 as regards prevention of criticality and is, therefore, acceptable.

9.2.2 Spent Fuel Storage

The spent fuel will be stored underwater in the spent fuel storage pool. The spent fuel storage pool is a reinforced concrete seismic Category I structure with a stainless steel liner. The seismic Category I spent fuel storage racks are designed to accommodate 486 fuel assemblies (approximately two and three-fourths cores). The racks have a center-to-center spacing which is sufficient to maintain a K-effective of 0.95 or less even if the pool were inadvertently filled with unborated water. The spent fuel storage racks are designed to prevent fuel assemblies being placed in other than their prescribed locations.

The design of the spent fuel pool walls will prevent tornado missiles traveling in a horizontal direction from penetrating the pool. Although the building roof is not designed for tornado missile protection, we have accepted the design of the fuel building and spent fuel pool as regards to tornado missile protection. The facility is designed to prevent the cask handling crane from traveling over the spent fuel storage pool, thereby precluding damage to the stored fuel from a dropped cask.

Based on our review, we conclude that the design of the spent fuel storage facility meets General Design Criterion 61 and the recommendations of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Bases" and 1.29, "Seismic Design Classification," including seismic design and missile protection guidelines and is, therefore, acceptable.

9.2.3 Fuel Pool Cooling System

The spent fuel pool cooling and cleanup system is designed to remove the decay heat generated by the stored spent fuel assemblies, to maintain the purity and clarity of water in the spent fuel pool, refueling cavity and refueling water tank, and to maintain water level in the spent fuel storage pool.

The spent fuel pool cooling system consists of two 50 percent capacity spent fuel pool cooling pumps and one 100 percent capacity heat exchanger. One pump with the heat exchanger in operation can maintain the pool temperature at 120 degrees Fahrenheit or less with a total spent fuel inventory of six (one-third core) annual refueling batches. Two pumps with the heat exchanger in operation can maintain the pool temperature of 150 degrees Fahrenheit or less with a total spent fuel inventory of five (one-third core) annual refueling batches plus a full core emergency unloading in the pool.

The spent fuel pool cooling system is not designed to seismic Category I requirements. However, at our request, the applicant has analyzed the fuel pool cooling piping and the service water lines to and from the fuel pool heat exchanger for the forces associated with the safe shutdown earthquake. The applicant has agreed to provide seismic Category I pipe and equipment supports in the pool cooling and service water system where necessary.

Assured makeup water can be supplied from the seismic Category I service water system. Two redundant paths are provided, one from each of the service water headers. The fuel pool piping is so arranged that the pool cannot be inadvertently drained to uncover the fuel.

Based on our review, we conclude that the system design meets the intent of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," including provision of decay heat removal capability, and the recommendations set forth in Regulatory Guide 1.13, "Fuel Storage Facility Design Bases," and 1.29, "Seismic Design Classification," including seismic design, provisions to prevent uncovering the fuel, and provisions for assured makeup and is therefore, acceptable.

9.2.4 Fuel Handling System

The fuel handling system provides the means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves after it has been removed from the reactor. The fuel handling system also provides for the safe disassembly, handling and reassembly of the reactor vessel head and internals during refueling operations.

The system consists of the refueling machine, the control element assembly change machine, the control element assembly change mechanism, the fuel transfer equipment, the spent fuel handling machine, the new fuel elevator, and the auxiliary building crane.

The design of the fuel handling facility is such that there are no cranes other than the spent fuel handling machine, provided over the spent fuel storage area. The spent fuel handling machine is designed to seismic Category I requirements. Travel of the spent fuel cask handling crane is limited by two independent means of crane stops to preclude approaching the spent fuel pool; thus dropping or tipping of a spent fuel cask into the spent fuel pool is not possible.

The 100-ton capacity bridge crane is provided to handle the spent fuel shipping cask in the Arkansas Nuclear One, Units 1 and 2 auxiliary buildings. The crane transports the spent fuel cask to and from the rail car bay and to the ANO-1 and ANO-2 cask loading pits. The only safety-related equipment over which the cask must be carried is the relay room. The applicant has performed an analysis to demonstrate that the three-foot six-inch thick reinforced concrete relay room ceiling slab can protect the safety-related equipment located in the relay room from damage from a postulated cask drop. We conclude the analysis is acceptable. A control interlock is provided to limit the height to which the cask can be lifted above the relay room ceiling slab. As a backup to this control interlock, safety slings are provided to support the cask directly from the crane trolley structure to prevent a cask drop.

Based on our review, we have concluded that the fuel handling system design is in conformance with the recommendations of Regulatory Guide 1.13, including the recommendation regarding protection of the spent fuel storage facility from the impact of unacceptable heavy loads carried by overhead cranes and is, therefore, acceptable.

9.3 Water Systems

9.3.1 Service Water System

The service water system is designed to provide cooling water to the safety-related plant systems as well as the nonsafety-related auxiliary systems. The essential portions of the service water system supply cooling water to the emergency diesel generator cooling heat exchangers, the control room emergency heating ventilation and air conditioning condensing units, the shutdown cooling heat exchangers, the emergency core cooling system pump coolers, the containment cooling units and the safety-related area unit coolers. The service water system also serves as the seismic Category I backup water supply to the emergency feedwater system.

The service water system consists of two full capacity trains with three 100 percent capacity service water pumps. The third pump is for standby. All service water pumps are located inside the seismic Category I intake structures and protected from tornado missiles as well as internal missiles. The pumps are powered by redundant emergency buses. Each service water train supplying water to safety-related equipment is designed to seismic Category I requirements, and is isolated from the other train and from nonsafety-related portions of the service water system by seismic Category I automatic isolation valves or normally closed manually operated valves. The service water system is designed to use the Dardanelle Reservoir as the water supply during normal plant operation. An alternate water supply for the system is available from the seismic Category I emergency cooling pond, which will be used during normal plant shutdown and accident conditions. The underground piping connecting the emergency cooling pond and the service water intake structure is designed to seismic Category I requirements and is adequately protected from tornado missiles by three feet of earth above the piping. Further discussion of protection from tornado missiles may be found in Section 3.5 of this report.

The sluice gate at the emergency cooling pond is designed to seismic Category I requirements and will normally be locked open. The sluice gate is protected from tornado missiles.

The service water supply to the spent fuel pool cooling heat exchanger is the nonessential portion of the service water system. The applicant has analyzed this part of the service water piping for safe shutdown earthquake loading and has provided the necessary seismic Category I supports. Section 9.1.3 of this report has further details.

Based on our review, we conclude that the service water system design is in conformance with the requirements of General Design Criterion 44 regarding the capability of the system to transfer heat from systems and components important to safety to an ultimate heat sink under normal and accident conditions and to meet the single failure criterion. We further conclude that the system design meets the requirements of General Design Criteria 45 and 46 as regards to system design that allows performance of periodic inspections and testing.

9.3.2 Ultimate Heat Sink

The ultimate heat sink is designed to dissipate heat from the service water system for safe shutdown of the plant. As discussed in Section 9.2.1 of this report, the Dardanelle Reservoir provides primary heat sink for Arkansas Nuclear One, Units 1 and 2 during normal plant operation, while the emergency cooling pond provides the seismic Category I backup source for plant safe shutdown under normal or accident conditions. The emergency cooling pond, the sluice gate and piping between the pond and the service water intake structure are designed to seismic Category I requirements.

The applicant has submitted the results of an analysis which demonstrates the capability of the emergency cooling pond to serve both Arkansas Nuclear One, Units 1 and 2, assuming a loss-of-coolant accident in one unit, while the other unit is shutdown and is being cooled down. The analysis includes values for heat rate and total integrated heat for fission product and heavy element decay, rejected heat from station auxiliary systems, sensible heat, and the summation of the above, for a period of 30 days. Conservative meteorology was assumed. We find this analysis is applicable and acceptable.

Based on our review, we conclude that the ultimate heat sink design is in conformance with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink For Nuclear Power Plants," and, therefore, is acceptable.

9.4 Process Auxiliaries

9.4.1 Equipment and Floor Drainage Systems

The equipment and floor drainage system is designed to collect potentially radioactive fluids and discharge them to the waste management system. The system also collects certain nonradioactive fluids and discharges them to the sealed interceptor; the clarified effluent from the interceptor is conveyed to the intake canal.

The auxiliary building drains are arranged so that leakage in one engineered safety feature train does not flow into the rooms of the other train through the drainage system. Level indication and alarms in the control room are provided for those sumps in the auxiliary building which serve safety-related pump rooms. Therefore, the flooding of both trains of the engineered safety feature components is prevented.

Based on our review, we conclude that the equipment and floor drainage system is designed to protect safety-related areas and components from flooding and to prevent the inadvertent release of radioactive liquids to the environment due to piping or tank failure and is, therefore, acceptable.

9.4.2 Chemical and Volume Control System

The chemical and volume control system is designed to control and maintain reactor coolant inventory and also to control the boron concentration in the reactor coolant through the process of makeup and letdown. The system is also designed to collect reactor coolant pump seal controlled bleedoff, provide auxiliary pressurizer spray, maintain the primary water chemistry and purity and process the effluent reactor coolant to recover the boron and makeup water. The system charging pumps and associated valves and piping would be utilized for high pressure injection of borated water into the reactor cooling system upon receipt of a safety injection actuation signal.

The safety-related portion of the chemical and volume control system is designed to seismic Category I requirements. The system is capable of borating the reactor through either one of two flow paths and from either one of two boric acid sources. The portion of the system that would be utilized for emergency high head injection has sufficient redundancy to meet the single failure criterion. There are three charging pumps, each one having the full capacity required for high pressure injection. The pumps are powered by two separate emergency power buses. All portions of the system that contain boric acid solution are provided with heat tracing to maintain the fluid temperature above the boron solubility limit.

We reviewed the adequacy of the applicant's design for performing necessary functions of the chemical and volume control system during normal, abnormal, and accident conditions. We conclude that the design conforms to applicable regulations, guides, staff technical positions, and industry standards and is acceptable.

9.4.3 Failed Fuel Detection System

A single detector is installed in the reactor coolant chemical and volume control system letdown stream. The detector continuously monitors the letdown stream by measuring the gross gamma radiation and the concentration of iodine-135. The measurement is recorded in the control room with an alarm in the control room. Based on our review of this system, we conclude that the system will function as an acceptable failed fuel monitor.

9.5 Heating, Ventilation and Air Conditioning Systems

9.5.1 Control Room System

The control room heating ventilation and air conditioning system is designed to maintain the control room within the thermal and air quality limits required for operation of plant controls and uninterrupted safe occupancy during normal operation, shutdown and post-accident conditions.

The control room system consists of two 100 percent normal air conditioning systems, of which one is required to provide the necessary cooling and ventilation for the control room, computer room, and the visitors viewing gallery. Adequate filtered fresh air makeup is provided to the system during normal plant operation. The normal control room air conditioning system is continuously monitored with alarms for high radiation and high chlorine levels. In the event of high radiation or high chlorine levels, the normal air conditioning system is automatically de-energized and the control room is isolated by closing of the isolation dampers in the air supply and return ducts. The single supply and single return isolation dampers are designed to seismic Category I requirements and each equipped with two solenoid valves which are powered by engineered safety feature direct current power sources.

The control room emergency air conditioning systems and air filtering systems are provided for operation in the event of a design basis accident, high radiation or high chlorine, when the normal air conditioning system is de-energized and the control room is isolated. It consists of two unit coolers, two water cooled compressor condensing units and two redundant filter trains. The control room emergency air conditioning and the emergency air filtering systems are designed to recirculate air inside the control room without outside air makeup during emergency conditions. The systems are designed to seismic Category I requirements, meet the single failure criterion, and are protected from tornado missiles. The redundant equipment is powered by separate emergency buses.

The control room supply and return air ductwork is equipped with ionization type fire detectors to alarm in the control room in the presence of products of combustion in the ventilation system.

Based on our review, we conclude that the control room heating, ventilating and air conditioning system design meets the requirements of General Design Criterion 19, as regards to the capability to operate the plant from the control room during normal and accident conditions. We conclude that the system design is acceptable.

9.5.2 Auxiliary Building System

The auxiliary building is served by separate ventilation systems, each designed to meet the specific requirements of the area.

Each of the emergency diesel generator rooms is provided with two exhaust fans. Each exhaust fan is capable of limiting the diesel generator room temperature to a maximum of 115 degrees Fahrenheit. The air intakes, shutoff dampers, exhaust fans, ductwork supports and the temperature controls of the diesel generator room ventilation systems are designed to meet seismic Category I requirements and are protected from tornado missiles. The exhaust fans discharge air to the atmosphere, away from the air intakes, at a sufficiently high velocity to preclude recirculation of exhaust air into the room. Each ventilation system is powered by its associated diesel generator.

Each battery room and its corresponding equipment room is served by an independent exhaust system, designed to remove air from the high point of each battery room at the minimum rate of twenty air changes an hour. Each exhaust system is equipped with an air flow switch to alarm in the control room when the discharge air flow is not established. An indicating light in the control room shows each exhaust fan status.

Each of the two switchgear rooms is provided with two 100 percent capacity seismic Category I air cooling units. Each cooling unit consists of a filter, a cooling coil and a centrifugal fan. The unit takes air from the switchgear room and returns cooled air to the same room. The unit coolers are connected to redundant trains of essential service water system for cooling water supplies. Each switchgear room is also equipped with an exhaust fan designed to seismic Category I requirements and protected from tornado missiles.

Based on our review, we conclude that the auxiliary building heating, ventilation and air conditioning system will perform its intended function and, therefore, is acceptable.

9.5.3 Auxiliary Building Radwaste Area Engineered Safety Features Unit Coolers

Seismic Category I unit coolers are provided in the auxiliary building for the following engineered safety feature rooms: the shutdown heat exchanger rooms, the high pressure safety injection pump rooms, the charging pump rooms, the emergency feedwater pumps area, the electrical equipment rooms and the boric acid makeup pump room. Redundant unit coolers are provided for each engineered safety feature area.

Based on our review, we conclude that the engineered safety feature room air cooling system design is capable of performing its intended function and, therefore, is acceptable.

9.5.4 Fuel Handling Area System

The fuel handling and storage area is served by an independent ventilation system. The ventilation system maintains a slight negative pressure in this area. The system supplies air from one supply air handling unit to the fuel handling floor and storage pool area. The ventilation air from these spaces is then exhausted to a containment flue through a multifilter unit consisting of a roughing filter, a high efficiency particulate filter, a charcoal adsorber, and two exhaust fans, one of which serves as a standby. In the event of high radiation in the area, the ventilation system will remove the contamination from the space to avoid the spread of airborne radioactive particles to other areas.

The fuel handling area ventilation system does not provide seismic Category I isolation dampers in the air supply and exhaust ducts for isolation of the fuel handling area after a postulated fuel accident. We have evaluated and accepted the system design based on the fact that the applicant has performed a fuel handling accident analysis without taking credit for the isolation of the fuel handling floor.

Based on our review, we conclude that the fuel handling floor heating, ventilating and air conditioning system design is acceptable.

9.6 Diesel Generator Auxiliary Systems

9.6.1 Diesel Generator Fuel Oil Storage and Transfer System

The diesel generator fuel oil storage and transfer system is designed to provide an independent fuel oil supply train for each diesel generator.

The system contains two underground storage tanks, each of which is sized to provide sufficient fuel oil supply to operate one diesel generator for four and one-half days (22,500 gallons per tank). A cross connection with two normally closed valves on the suction side of the fuel oil transfer pumps permits transfer of fuel oil to a diesel engine from either diesel fuel oil storage tank. Thus the total fuel oil storage in both tanks can supply sufficient fuel oil to one operating diesel generator for nine days while supplying post loss-of-coolant maximum electrical load demands. Each diesel generator is supplied by a train consisting of one day tank with a two and one-half hour fuel oil storage capacity and one fuel oil transfer pump. The system is designed with sufficient flexibility to meet the single failure criterion. The fuel oil transfer pumps are powered by the emergency bus associated with the particular diesel generator train. The underground storage tanks, day tanks, pumps and connecting piping are designed to seismic Category I requirements and are protected from tornado missiles. The underground fuel oil storage tanks are designed to resist the loadings imposed by the probable maximum flood. Fuel oil is supplied to the underground storage tanks from one above ground 185,000 gallon, nonseismic fuel oil storage tank. The underground fuel oil piping from the fuel oil storage tanks is protected from tornado missiles by six feet of earth above the piping.

Based on our review of the diesel generator fuel oil storage and transfer system design, we conclude that the design provides the redundancy and independence for systems essential to safety and meets the recommendations of Regulatory Guide 1.29, "Seismic Design Classification," with regard to seismic design, has adequate capacity and can perform its designated safety functions and is, therefore, acceptable.

9.6.2 Other Diesel Generator Auxiliary Systems

The other diesel generator auxiliary systems include the diesel generator cooling water system, the diesel generator starting system, the diesel generator lubrication system, and the diesel generator combustion air intake and exhaust system.

The diesel generator cooling water system is designed to maintain the temperature of the diesel engine within a safe operating range. The system is a closed cooling system and the heat is rejected to the essential portion of the service water system. The system is designed to seismic Category I requirements. The makeup water for the system expansion tank is from the condensate storage tank which is not designed to seismic Category I requirements. The applicant has provided the results of an analysis which concluded that the diesels can operate at full load continuously for seven days without makeup to the diesel generator cooling water system expansion tank. We agree with this evaluation and conclusion.

Each diesel generator is provided with two independent compressed air starting trains. Each train consists of an air compressor and air storage tank. Each tank is capable of providing five starts without recharging from the compressors. The starting air system is designed to seismic Category I requirements.

Each diesel generator is provided with a lubrication system designed to assure adquate lubrication of bearings and other wearing parts. Lube oil cooling is provided by the diesel generator cooling water system. The system is designed to seismic Category I requirements.

Each diesel generator takes suction from the diesel generator room for its combustion air. The air intake into the room is designed to seismic Category I requirements, and protected from tornado missiles. The diesel engine exhausts are discharged at a sufficient distance away from the air intakes to avoid recirculation of the combustion products back into the diesel generator rooms. The diesel exhaust system is designed to seismic Category I requirements. However, it is not protected from tornado missiles. In response to our concerns regarding potential damage of the diesel exhaust system due to tornado missiles, the applicant has submitted the results of an analysis which concludes that the design of the diesel exhaust system can withstand a postulated tornado missile impact and the calculated reduction in cross-sectional area of the exhaust stack will not cause loss of the diesel generator operating at rated load. We have performed an independent evaluation and agree with the applicant's analysis of the tornado missile effects on the diesel exhaust system.

We have reviewed and evaluated the design, and conclude that the system will perform its intended function and therefore is acceptable.

9.7 Fire Protection

We require that the objectives of the fire protection program in a nuclear power plant include (1) the minimization of the occurrence of fires in safety-related areas, (2) the prompt detection and extinguishment of such fires when they occur, (3) the assurance that the capability to safely shut down the plant is maintained, and (4) the provision of reasonable assurance that a fire does not cause the release of a significant amount of radioactive material.

The Commission's criteria for fire protection are set forth in General Design Criterion 3. For the implementation of General Design Criterion 3, guidance is provided in Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants," and Appendix A to Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," as set forth in the LWR Edition of the Standard Review Plan, NUREG-75/087.

The ANO-2 fire protection system is designed to comply with General Design Criterion 3 and with applicable standards and guides of the National Fire Protection Association, and the State of Arkansas Codes and Regulations. Structures and systems important to safety are designed and located, consistent with other safety requirements, to minimize fire hazard. Noncombustible and heat resistant construction materials are used throughout the plant, wherever practicable. Areas that are essential for plant operation or contain safety-related equipment are protected by fire barriers. Fire detection and suppression systems are provided to minimize the effects of fires on safety-related systems and structures. The fire protection system includes water sprinkler systems, standpipe and hose systems, portable carbon dioxide extinguishers and fire detection and alarm systems.

The fire protection water supply system is common to both reactor units. Fire water for ANO-2 is supplied by two fire pumps located in the Arkansas Nuclear One-Unit 1 (ANO-1) intake structure. The two fire pumps take suction from separate service water bays which are normally supplied from the Arkansas River (which flows into the Dardenelle Reservoir) through the intake screens. The service water bays can also be supplied from the emergency cooling water pond.

Two vertical shaft centrifugal fire pumps are provided, each with a design capacity of 2500 gallons per minute at a discharge head of 125 pounds per square inch gauge. One is an electric motor driven pump. The other is a diesel engine driven pump having an eight hour fuel supply tank located in the same room as the pump; additional fuel is available from onsite fuel storage tanks. An automatic electric jockey pump maintains a pressure of 125 pounds per square inch gauge on the fire water piping system when the system is not in use. Both fire pumps are arranged to start automatically when a large amount of flow drops the pressure on the system - the electric pump starts at 110 pounds per square inch gauge and the diesel pump starts at 90 pounds per square inch gauge. Both fire pumps can be started manually from the control room. Both fire pumps will continue to run until they are shut off manually.

Each of the two fire pumps has a separate discharge into a twelve inch underground fire loop which encircles both ANO-1 and ANO-2. Valving is arranged so that a single break in the discharge piping will not remove both fire pumps from service. All yard fire hydrants, automatic water suppression systems, and interior fire hose lines are supplied by the fire loop. Sectionalizing valves of the post indicator type are provided on the fire loop to allow isolation of various sections for maintenance or repairs.

Interior fire hose stations with fifty feet of one and one-half inch hose have been provided throughout the plant. The nozzles on hose lines are of the adjustable spray type; in areas of potential electrical fires, they are of a type rated for electrical fires. Automatic water spray systems actuated by rate compensation heat detectors protect certain high hazard equipment such as the outside oil filled transformers, hydrogen seal oil unit, feedwater pump lube oil reservoir, diesel fuel storage vaults and the cable spreading room.

Portions of the fire water system that are within engineered safety feature equipment rooms are supported in accordance with seismic Category I requirements.

Portable dry chemical and carbon dioxide fire extinguishers will be distributed throughout the plant in accordance with the National Fire Protection Association requirements.

The plant has a prototype signaling system which transmits fire alarm and supervisory signals to the control room. Smoke detectors of both the ionization and photo-electric types have been provided in selected areas of the plant and in certain heating, ventilating and air conditioning systems which include the cable spreading room, the control room, computer room, emergency diesel generator rooms and associated fuel oil day tank enclosures, engineered safety feature switchgear rooms, containment cable penetration areas, air ducts of the control room ventilation system and other ventilation exhaust ductwork and areas where a potential for fire exists. Annunciation and alarms are provided in the control room upon activation of fire and smoke detection systems or actuation of any automatic fire protection system.

Fire barriers of a three-hour fire rating have been provided to isolate various safety-related areas, or to isolate a hazard from safety-related areas. Fire barriers enclose the turbine lube-oil storage room, control room, battery rooms, cable spreading room, emergency diesel generator rooms, fuel oil storage vaults, and the engineered safety features switchgear areas. Fire barriers also separate the turbine building from the auxiliary building.

Following the implementation of the modifications of the fire protection systems and administrative controls resulting from our review, the ANO-2 technical specifications will be modified to include limiting conditions for operation and surveillance requirements for the existing fire protection systems and administrative controls.

In accordance with the recommendations set forth in NUREG-0050, "Recommendations Related to Browns Ferry Fire," we are conducting additional reviews and evaluations of nuclear power plant fire protection programs. We have not yet completed our review of this subject for the ANO-2 plant. The applicant indicates that submittal of the remainder of the Fire Hazards Analysis and of the responses to requested additional information will be completed by about November 30, 1977. As a result of our review additional requirements could be imposed on the ANO-2 plant to further improve the capability of the fire protection program.

Based on our evaluation of the ANO-2 facility completed to date, we conclude that the facility fire protection program presently meets the applicable guidelines in effect prior to the issuance of Branch Technical Position 9.5-1 and, for the interim, is acceptable. We have determined that sufficient flexibility exists in the plant's design to allow the implementation of design changes that may be necessary to assure conformance with the provisions of Appendix A to Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." Our final evaluation and conclusions regarding our review of the applicant's Fire Hazards Analysis Report, the applicant's evaluation of the fire protection program and any required modifications to the facility fire protection program will be reported in a future report.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion system is of conventional design, similar to those of previously approved plants. The system is designed to remove thermal energy from the reactor coolant by two steam generators and convert it to electrical energy by the turbine driven generator. The condenser transfers unusable heat in the cycle to the circulating water. The entire system is designed for the maximum expected thermal output from the nuclear steam supply system.

In the event of a turbine trip or a large load reduction, the heat transferred from the reactor coolant to the steam generators is dissipated through the turbine bypass system to the condenser, or, if the condenser is not available, through the power operated atmospheric relief valves and safety valves to the atmosphere.

10.2 Turbine Generator

The turbine generator is a tandem-compound type consisting of one double-flow high pressure turbine and two double-flow low pressure turbines. The rotational speed is 1800 revolutions per minute. The turbine electrohydraulic control system controls the steam flow through the turbine by modulating the turbine governor valves.

The turbine control system is designed to trip the turbine under the following conditions: turbine overspeed, condenser low vacuum, excessive thrust bearing wear, reactor trip, generator trip, low bearing oil pressure, loss of electrohydraulic control power, excessive vibration, high exhaust hood temperature, moisture separator drain system high level, loss of stator coolant, low hydraulic fluid pressure, loss of both speed signals and manual trip from the control room or at the turbine. Overspeed protection is accomplished by two independent systems, a mechanical overspeed trip device and an electrical overspeed trip device. The mechanical overspeed sensor will trip the turbine stop and control valves and the combined intermediate stop and intercept valves if 110 percent of rated speed is reached. The electrical backup overspeed sensor will trip these turbine inlet valves at 112 percent of rated speed. Because of the redundancy in the turbine overspeed protection system, the turbine is, therefore, protected from excess overspeed.

Based on our review of the turbine generator overspeed protection system design, we conclude that the system can perform its designated safety functions and is, therefore, acceptable.

10.3 Main Steam Supply System

The steam generated in each of the two steam generators is routed to the turbines by two main steam lines. Each main steam line contains five American Society of Mechanical Engineers Code safety valves, one air operated atmospheric dump valve and one main steam isolation valve. The main steam supply system, from the steam generators up to and including the main steam isolation valves, are designed to seismic Category I requirements.

The main steam isolation valves are air operated, fast closing stop-check type and designed to provide positive isolation against steam flow from either direction in the event of a postulated steam break accident. They will prevent blowdown of both steam generators by closing in less than four seconds after receipt of a signal requiring isolation. Redundant solenoid air supply and vent valves are provided to each valve to insure that no single electrical failure will prevent main steam isolation valve closure.

Portions of the seismic Category I main steam piping (upstream of the main steam isolation valve) are located outdoors without protection from tornado missiles. In response to our concerns regarding potential damage of the safety-related main steam piping due to tornado missiles, the applicant has submitted the results of an analysis which conclude that the design of the main steam piping can withstand a postulated tornado missile impact without failure of the piping. We have performed an independent evaluation and agree with the applicant's analysis of the tornado missile effects on the outdoor seismic Category I main steam piping and conclude that the piping can withstand the tornado missile impact.

Based on our review, we conclude that the main steam supply system design is in conformance with the single failure criterion, the seismic recommendations of Regulatory Guide 1.29, "Seismic Design Classification," and valve closure time requirements. We conclude that the design of the main steam supply system is acceptable.

10.4 Circulating Water System

The circulating water system is a closed loop system which removes the heat rejected from the main condensers to a natural draft cooling tower. In the event a failure occurs to a circulating water system component inside the turbine building, there is a potential of discharging circulating water into the turbine basement at a rate of approximately 486,000 gallons per minute. At this flow rate, the applicant stated that the entire circulating water inventory would be pumped into the turbine building in approximately fifteen minutes. The resulting water level in the turbine building would be 358 feet-three inches which is two feet-nine inches below the design flood elevation at the plant site. There are no paths by which this water could enter any safety-related structure. There is no safety-related equipment in the turbine building that can be effected due to flooding.

We conclude that the circulating water system design is acceptable.

10.5 Emergency Feedwater System

The emergency feedwater system is designed to supply water to the steam generators for reactor coolant system sensible and decay heat removal when the normal feedwater system is not available. The emergency feedwater system will be utilized during certain periods of normal startup and shutdown, in the event of malfunctions such as loss of offsite power, and also in the event of accidents. The emergency feedwater system is designed to seismic Category I requirements and is protected from tornado missiles.

The system contains one motor driven pump and one turbine driven pump. Each pump has a capacity of 575 gallons per minute. Steam supply to the turbine driven pump is taken from either one of the two main steam lines at a point upstream of the main steam isolation valves. The motor driven pump is connected to the emergency power bus.

The emergency feedwater pumps normally take suction from the condensate storage tank. In Amendment 38 to the Final Safety Analysis Report the applicant modified the system design to connect the emergency feedwater pump's suction to the startup and blowdown demineralizer system and the condensate storage tank. The condensate storage tank has a capacity of 200,000 gallons with a minimum reserve capacity of 120,000 gallons of condensate for the emergency feedwater system water supply. This minimum reserve supply of water is sufficient for cooldown of the reactor coolant system to the temperature and pressure at which shutdown cooling system can be placed in operation. Redundant trains of the seismic Category I essential service water system are connected to the emergency feedwater system pump suctions for available water supplies.

In the event an emergency feedwater actuation signal is received by the emergency feedwater system, both pumps will start. Simultaneously, all valves in the discharge lines will open. However, if isolation of a steam generator is required, as in the case of a postulated main steam line break, the emergency feedwater actuation system will open only the valves leading to the intact steam generator.

Some of the motor operated valves associated with the turbine driven auxiliary feedwater pump are powered by alternating current emergency buses. This arrangement does not meet the power diversity requirements set forth in Branch Technical Position APCSB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants". We have expressed our concern over this matter to the applicant. In response, the applicant has committed to modify the system to meet the power diversity requirement during the first refueling period.

This will require the valves in the turbine driven pump discharge and suction lines and the steam supply lines to be powered by direct current emergency power supplies We find this commitment acceptable. We will appropriately condition the operating license to ensure the implementation of this commitment.

Based on our review, and the applicant's commitment to modify the emergency feedwater system to meet the power diversity guidelines set forth in Branch Technical Position APCSB 10-1 during first refueling period, we conclude the emergency feedwater system design is acceptable.

10.6 Water Hammer

Events such as damage to the feedwater system piping at Indian Point 2 on November 13, 1973, and at other plants, could originate as a consequence of recovering of the feedwater sparger in the steam generator or uncovering of the steam generator feedwater inlet nozzles. Subsequent events in turn lead to the generation of a pressure wave that is propagated through the pipes and could result in unacceptable damage.

We are currently evaluating this problem on a generic basis for all pressurized water reactors. It is our position that the applicant must demonstrate during plant operation that unacceptable damage such as experienced at Indian Point 2 and Calvert Cliffs would not result at the Arkansas Nuclear One - Unit 2 facility. We will 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 wish to review the test procedures prior to execution of the tests and will require that the tests be performed before the plant reaches full power operating conditions.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

The radioactive waste management systems are designed to provide for the controlled handling and treatment of liquid, gaseous and solid wastes. Since the construction permit was issued, the applicant has modified the radwaste system to reduce radioactive releases. These modifications include the addition of a regenerative waste processing system in the liquid radwaste system, and a startup and blowdown demineralizer system. The gaseous and solid radwaste systems have not changed from that described in Section 3.1.7 of the construction permit Safety Evaluation Report issued April 1972.

Based on our evaluation as described below, we find the gaseous, liquid and solid radwaste and associated process and effluent radiological monitoring systems to be acceptable. Our evaluation regarding the capability of the liquid and gaseous radwaste treatment systems to meet the dose design objectives of Appendix I to 10 CFR Part 50 may be found in Section 3.2.4 of the Commission's Final Environmental Report for ANO-2 dated June 1977. The calculated doses are compared with the design objectives of Appendix I in Section 5.5.1.7 of the Final Environmental Statement.

11.2 Liquid Waste Summary

11.2.1 Description and Evaluation

The modified liquid radioactive waste systems are described in the Evironmental Statement for the operating license stage. Subsequent to the publication of the Safety Evaluation Report for the construction permit, the liquid radioactive waste systems were modified to include a startup and blowdown demineralizer system and a regenerative waste processing subsystem.

The startup and blowdown demineralizer system will process blowdown water from the steam generator blowdown system to maintain secondary system water purity. The system consists of heat exchangers, demineralizers, and equipment for regeneration of the demineralizer. Processed blowdown will be returned to the condenser for water conservation.

Liquid wastes from regeneration of the Unit 2 startup and blowdown demineralizers, and from regeneration of the Unit 1 condensate demineralizers will be collected and processed in the regenerative waste processing subsystem prior to recycling or discharge. This system consists of waste tanks and two evaporators. The design capacity of each evaporator is 14,000 gallons per day and the design capacity of the startup and blowdown demineralizers is 260,000 gallons per day. The average expected waste flows to the regenerative waste processing subsystem and the startup and blowdown demineralizer system are 6,400 gallons per day and 21,600 gallons per day, respectively. The difference between the expected flows and design capacity flows and the equipment redundancy will provide adequate reserve capacity for processing surge flows. We consider the system capacity and system design to be adequate for meeting the demands of the station during anticipated operational occurrences.

We have determined that during normal operations including equipment downtime and anticipated operational occurrences, the radioactivity released from the liquid radwaste system of Arkansas Nuclear One - Unit 2 will be approximately 0.23 curies per year excluding tritium and dissolved gases. Release of tritium from the plant will be approximately 340 curies per year.

11.3 Gaseous Waste Summary

11.3.1 Description and Evaluation

The gaseous radioactive waste system and building ventilation systems are described in the operating license Environmental Statement. The gaseous radioactive waste systems described in Section 3.1.7 of the construction permit Safety Evaluation Report have not been modified in the applicant's Final Safety Analysis Report.

The gaseous waste system will collect and process gases stripped from the primary coolant along with miscellaneous tank cover gases.

The gaseous waste system consists of two compressors, one surge tank, and three waste gas decay tanks. None of the system components are designed to withstand a hydrogen explosion. The system will be designed to operate at positive pressure and will be purged with nitrogen gas to prevent air (oxygen) buildup as a result of infiltration. Hydrogen and oxygen concentrations in the gases entering the system and stored in the decay tanks will be monitored by an automatically sequenced gas analyzer. The gas analyzer will indicate when and where potential explosive mixtures are occurring. We find that the gaseous waste system provides for redundant instrumentation to annunciate and prevent the buildup of potentially explosive mixtures. We find the design provisions incorporated to reduce the potential of a hydrogen explosion to be acceptable.

We have determined that during normal operations, including anticipated operational occurrences, the radioactivity released annually from the radioactive gaseous waste systems will be 7600 curies of noble gases, 0.03 curies of Iodine-131, 810 curies of tritium, 25 curies of argon-41, and 0.004 curies of particulates.

11.4 Solid Waste Summary

11.4.1 Description and Evaluation

The solid radioactive waste system is described in the operating license Final Environmental Statement dated June 1977. The solid radioactive waste system described in Section 3.1.7 of the construction permit Safety Evaluation Report has not been modified in the applicant's Final Safety Analysis Report.

The solid radwaste treatment system will be designed to collect and process wastes based on their physical form and need for solidification prior to packaging. "Wet" solid wastes, consisting of spent ion exchanger resins, concentrated evaporator bottoms, and spent filter cartridges, will be combined with a solidification agent and catalyst mixture to form a solid matrix and sealed in 50 cubic feet disposable steel liners. Liners will be filled by pumps which bring together radwaste and liquid solidification agents. Dry solid wastes, consisting of contaminated clothing and paper, and miscellaneous items such as tools and glassware, will be compacted into 55-gallon drums.

We have determined that the expected solid waste volumes and activities from ANO-2 to be shipped offsite annually will be approximately 13,000 cubic feet of "wet" solid waste containing approximately 1700 curies total and approximately 4,100 cubic feet of "dry" sold waste containing less than five curies total. Storage facilities to accommodate approximately 24 liners will be provided. Based on our estimate of expected solid waste volumes, we find the storage capacity adequate for meeting the demands of the plant.

Wastes will be packaged and handled in accordance with the requirements of 10 CFR Part 20, 10 CFR Part 71 and shipped to a licensed burial ste in accordance with the Commission's and the applicable Department of Transportation regulations.

11.5 Process and Effluent Radiological Monitoring

The process and effluent radiological monitoring system will be designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance and monitor and control radioactivity levels in plant discharges to the environs.

Liquid and gaseous streams will be monitored. Table 11.3-1 of this report indicates the proposed locations of continuous monitors. Monitors on effluent streams will automatically terminate discharges should radiation levels exceed a predetermined value; these monitors are identified in Table 11.3-1.

Systems which are not amenable to continuous monitoring or for which detailed radioisotopic analyses are required will be periodically sampled and the samples analyzed. The sampling system will provide representative primary and secondary

TABLE 11.3-1

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEM

	No.	Type of Detector	Medium	Sensitivity
CVCS Process	1	Gamma-Scintillation	Water	1×10^{-4}(Iodine-135)
Component Cooling Water	2	Gamma-Scintillation	Water	5×10^{-6}(Cesium-137)
Service Water, Containment Cooling Coils	2	Gamma-Scintillation	Water	5×10^{-6}(Cesium-137)
Service Water, Shutdown Cooling Heat Exchangers	2	Gamma-Scintillation	Water	5×10^{-6}(Cesium-137)
Service Water, Fuel Pool Heat Exchanger	1	Gamma-Scintillation	Water	5×10^{-6}(Cesium-137)
Steam Generator Sample Coolers	2	Gamma-Scintillation	Water	5×10^{-6}(Cesium-137)
Waste Management System[a]	1	Gamma-Scintillation	Water	5×10^{-6}(Cesium-137)
Regenerative Waste Processing System Common Header	1	Gamma-Scintillation	Water	3 x 10-7(Cesium-137)
Regenerative Waste Processing System Transfer Pumpsa	2	Gamma-Scintillation	Water	5 x 10-6(Cs-137)
Main Condenser Air Discharge	2	Beta-Gamma	Vapor	1×10^{-5}(Xenon-133)
Waste Gas[a]	1	Beta-Gamma	Gas,Air	1×10^{-5}(Xenon-133)
Penetration Rooms	2	Beta-Gamma	Air	1×10^{-5}(Xenon-133)
Hydrogen Purge/Containment Atmosphere Particulate	2	Gamma-Scintillation	Air	1.5 x 10-10(Cesium-137)
Hydrogen Purge/Containment Atmosphere Gas	2	Beta-Gamma	Air	1×10^{-5}(Xenon-133)
Fuel Handling Area Ventilation	1	Beta-Gamma	Air	1×10^{-5}(Xenon-133)
Radwaste Area Ventilation	1	Beta-Gamma	Air	1×10^{-5}(Xenon-133)
Containment Purge[a]	1	Beta-Gamma	Air	1×10^{-5}(Xenon-133)

[a]These monitors terminate the release when the radiation level exceeds a predetermined level.

liquid and gaseous samples as required to effectively monitor the operation of both units and to control radioactivity levels in plant discharges to the environs.

This system will provide samples from approximately fifty different primary and secondary system points, which can be sampled locally or at sample sinks located in the auxiliary building. Radioactive samples will be either delayed to allow decay of short-lived radioisotopes and/or shielded from the sample sink to protect personnel handling samples. If a critical sampling line becomes inoperable, there is at least one alternate path which can be used to obtain a similar sample.

We have reviewed the locations and types of effluent and process monitoring provided. Based on the plant design and on the continuous monitoring locations and intermittent sampling locations, we have concluded that all normal and potential release pathways will be monitored. We have also determined that the sampling and monitoring provisions will be adequate for detecting radioactive material leakage to normally uncontaminated systems and for monitoring plant processes which affect radioactivity released during normal operation, including anticipated operational occurrences.

On the basis of our review we conclude that the monitoring and sampling provisions meet the requirements of General Design Criteria 60, 63 and 64 and the recommendations of Regulatory Guide 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Material in Liquid and Gaseous Effluents from Light Water Cooled Nuclear Power Plants," and are acceptable.

11.6 Evaluation Findings

Our review of the radwaste systems included system capabilities to process the types and volumes of wastes expected during normal operations and anticipated operational occurrences in accordance with General Design Criterion 60, the design provisions incorporated in accordance with General Design Criterion 60 to control releases of radioactive material due to leakage overflows, the codes, standards, and seismic design classification applied to the system design. We have reviewed the applicant's system descriptions, process flow diagrams, piping of the radwaste treatment systems and for those auxiliary supporting systems that are essential to the operation of the radwaste treatment systems. We have performed an independent calculation of the releases of radioactive materials in liquid and gaseous effluents based on the calculational methods of NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)" dated April 1976.

We have determined as shown in Section 3.2.4 of the Commission's Final Environmental Statement dated June 1972 for ANO-2 that the level of routine radioactive releases is "as low as is reasonably achievable" in accordace with the requirements of 10 CFR Part 50.34a and conforms to the requirements of Appendix I to 10 CFR Part 50.

Our review of the process and effluent radiological monitoring systems included the provisions proposed for sampling and monitoring all station effluents in accordance with General Design Criterion 64, for providing automatic termination of effluent releases and assuring control over discharges in accordance with General Design Criterion 60 and Regulatory Guide 1.21, for sampling and monitoring plant waste process streams for process control in accordance with General Design Criterion 13, for conducting sampling and analytical programs in accordance with the guidelines in Regulatory Guide 1.21, and for monitoring process and effluent streams during postulated accidents. The review included piping and instrument diagrams and process flow diagrams for the liquid, gaseous, and solid radwaste systems and ventilation systems; and the location of monitoring points relative to effluent release points on the site plot diagram.

Based on the foregoing evaluation, we conclude that the above aspects of the proposed liquid and solid radwaste treatment and monitoring systems are acceptable. The basis for acceptance has been conformance of the applicant's designs, design criteria, and design bases for the radioactive waste treatment and monitoring system to the applicable regulations and guides referenced above, as well as the staff technical positions and industry standards.

12.0 RADIATION PROTECTION

12.1 General

We have reviewed the applicant's radiation protection program as described in Chapter 12 of the ANO-2, Final Safety Analysis Report. With respect to planning, designing and operating the station, the applicant has provided a radiation protection program which will meet the requirements of 10 CFR Part 20. The radiation protection program contains design and program features which are consistent with the guidelines of Regulatory Guide 8.8, "Information Relevant to Monitoring Occupational Radiation Exposures As Low As Is Reasonably Achievable," and Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposure As Low As Is Reasonably Achievable." In meeting these objectives, the applicant has used operating experience with Arkansas Nuclear One - Unit 1 and the aggregate experience of Combustion Engineering, Incorporated and the Bechtel Corporation.

Our review covered management policy and organization relating to radiation protection shielding and layout designs; area and airborne monitoring systems; ventilation, and the health physics program.

12.2 Radiation Protection Design Features

The shielding design was reviewed to determine whether it would minimize exposure to operating personnel (consistent with 10 CFR 20) during normal operations, anticipated operational occurrences, and during maintenance associated with operations and shutdown. Plant areas have been classified into radiation zones based on maximum design dose rates and expected frequency and duration of occupancy. Shielding issues resolved during the review included clarification of: (1) anticipated exposure to contract maintenance workers, and (2) radiation zones for low background counting and monitoring areas. The amendments to the Final Safety Analysis Report provided the basis for concluding that the shielding design is consistent with Regulatory Guide 8.8, or provides acceptable alternatives.

The applicant has provided the location, size, and shape of significant sources of radiation in the auxiliary and fuel building and containment structures. The applicant's source term calculations are based on (1) reactor operation at 2815 thermal megawatts, (2) a maximum failed fuel rate of one percent, and (3) an acceptable set of estimated leakage rates and partition factors. The primary shield calculations were performed with the GRACE, ANISN, FAIM and 2DBS codes. We find the assumptions used in the applicant's shielding calculations are conservative and acceptable.

Pipes, demineralizer tanks, evaporators, pumps, and sampling points containing radioactivity have been located in shielded areas or compartments and the applicant proposes to use labyrinths, shielded valve galleries and penetrations, reach rods, remote valve actuation and portable shielding to reduce unnecessary exposure during operation. Based on the applicant's design criteria, shield models and operating philosophy, we conclude that the shielding and layout permits compliance with 10 CFR Part 20 and should maintain exposures as low as is reasonably achievable.

12.3 Area Monitoring

The applicant's area radiation monitoring system is designed to (1) monitor dose rate in selected areas, (2) alarm when dose rates exceed a preset level, and (3) provide a continuous record of dose rates in selected plant areas. The applicant has or will provide twenty-three area monitors in locations where employees may be expected to encounter significant normal or abnormal dose rate conditions.

The applicant has or will provide six fixed and four portable airborne radioactivity monitors consisting of particulate, iodine, and gas measuring detectors. The applicant has indicated that the sensitivity of the instruments and/or analytical techniques used will be sufficient to detect fractions of maximum permissible concentrations.

On the basis of locations chosen, sensitivities, and alarm settings, we conclude that this is reasonable assurance that radiation levels within the plant will be adequately monitored, and that the area radiation monitoring system is acceptable.

12.4 Ventilation

The applicant's ventilation system is designed to ensure that personnel are not exposed to normal or abnormal airborne concentrations exceeding those in 10 CFR Part 20 by (1) maintaining air flow from areas of low radioactivity potential to areas of high radioactivity potential, (2) preventing recirculating air in the auxiliary and fuel buildings, (3) maintaining a negative pressure in the auxiliary and fuel buildings with respect to the atmosphere, and (4) periodically purging the containment structure with outside air through high efficiency particulate and charcoal filters. Various other areas of the plant will contain high efficiency particulate and charcoal filters to minimize the build-up of airborne radioactivity.

The applicant plans to also keep onsite inhalation exposures as low as is reasonably achievable by: (1) the use of respiratory protection devices in areas of high airborne radioactivity, (2) periodic body burden counting, (3) elimination of high airbone radioactivity following detection, (4) usage of temporary local exhaust and containment enclosures, and (5) appropriate decontamination of equipment and work areas prior to conducting maintenance activities.

Table 12.2-4 of the Final Safety Analysis Report provides estimates of the in-plant thyroid exposures derived from the applicant's airbone radioactivity source terms. These estimates are based on a calculational approach that is acceptable to us.

We conclude that the ventilation systems described in the Final Safety Analysis Report meet Regulatory Guide 8.8 design objectives and we conclude that the applicant has provided reasonable assurance that they can maintain airborne radioactivity in normally occupied areas below 10 CFR Part 20 limits.

12.5 Health Physics

The health physics program was reviewed to determine that the program will make an appropriate contribution to keeping radiation exposures as low as is reasonably achievable. Our review covered management policies, organization, facilities, monitoring equipment and procedures for controlling contamination and occupational radiation exposures. The applicant's stated policy for radiation protection is based on compliance with applicable NRC regulations, Regulatory Guides, and technical specifications. The health physics programs and radiation safety procedures include: personnel dosimetry by thermoluminescent dosimetry and/or film badges; protective clothing and respiratory protection including a respiratory training program; radiation exposure controls by barriers: locked doors; signs to discourage unauthorized entry into radiation areas; testing and calibrating monitoring instruments; training; and the maintenance of radiological records and reports.

Health physics facilities for normal operations include: shielded laboratory apparatus for counting air and wipe samples; a calibration room for monitoring instruments; a locker room for changing into protective clothing and respirators; a personnel and equipment decontamination room, and a shadow shielded partial body counter. The counting room will be shared with Unit 1 along with beta gamma and neutron sources for onsite calibration of dose rate instruments and dosimeters.

Health physics instruments for radiation surveys will consist of alpha, beta, gamma and neutron survey meters, as well as count rate meters at access control points. Other equipment to be used for radiation protection purposes includes protective clothing, respiratory devices, air samplers, self reading dosimeters, neutron film badges, lead shot, lead sheets, remote handling tongs, and laundry machines.

A radiation work permit system will be used by the applicant for control of entry into a radiation area. Exceptions are permitted only for entries immediately essential to provide for personnel or reactor safety and only by qualified persons carrying radiation monitoring equipment.

Persons will receive film badges or thermoluminescent dosimeters and pocket or extremity dosimeters as required by radiation work permit. Body burden counts will be performed periodically to confirm the adequacy of station contamination control practices.

On the basis of our review we conclude that the health physics program described in the application is of sufficient scope to maintain occupational exposures as low as is reasonably achievable as required by 10 CFR Part 20 and is acceptable.

12.6 Dose Assessment

The applicant's doses estimate of 196 man-rem for in-plant exposure did not include an estimate for man-rem associated with corrective maintenance. However, in response to a question the applicant revealed that another 200 to 400 man-rem per year could be allocated for those purposes. Data from NUREG-0109, "Occupational Radiation Exposure at Light Water Cooled Power Reactors, 1969-1975" indicates that routine and special maintenance accounts for about seventy percent of in-plant dose. We expect that exposure to personnel at ANO-2 will be in the order of 400-600 man-rem per year. This dose assessment for normal operations and anticipated maintenance is acceptable.

13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

The Arkansas Power and Light Company is responsible for the operation of Arkansas Nuclear One - Unit 2 (ANO-2). This responsibility is carried out by their Power Production Department. The Plant Superintendent, who is responsible for the safe operation of the Arkansas Nuclear One plant which includes both Unit 1 (ANO-1) and Unit 2 (ANO-2), reports to the Assistant Director of Power Production, who in turn reports to the Director of the Power Production Department.

The plant staff, under the direction of the Plant Superintendent and the Assistant Plant Superintendent, is responsible for the operation of both units located at the site. The plant staff for both units consists of approximately 185 full-time employees functioning in seven main groups: an operations group (about 55 people) responsible for operating the plant; a maintenance group (about 43 people) responsible for the mechanical and electrical maintenance at the plant; a technical support group (about 30 people) responsible for radiation protection, radiochemistry support, water quality control and plant performance evaluation; an instrument and controls group (about 38 people) responsible for the maintenance of instrument and controls systems; a quality control group (about three people) responsible for review and inspection of quality related activities; a nuclear engineering group (about four people) responsible for monitoring and evaluating core physics and core performance; and an administrative group responsible for administrative services.

The plant operations group is under the supervision of the Supervisor of Plant Operations. Reporting to him will be two Assistant Supervisors of Plant Operations, each having primary responsiblity for one unit. Reporting to them are the plant operating shifts. The minimum shift composition for the operation of ANO-2 will consist of at least five persons, one of whom will hold a senior operators license, and two of whom will hold an operators license. A separate shift crew with no responsibility for ANO-2 will operate ANO-1.

The applicant has stated that the education and experience of plant operation, technical and maintenance support personnel will meet the requirements set forth in American National Standards Institute Standard N18.1-1971, "Standard for the Selection and Training of Personnel for Nuclear Power Plants." This meets Regulatory Guide 1.8, "Personnel Selection and Training." We have reviewed the qualifications of key supervisory personnel assigned to the ANO-2 station. We find them acceptable since the qualifications of key supervisory personnel, with regard to educational background, experience, and technical specialties, are in accord with those defined in ANSI N18.1-1971.

Primary offsite technical support for plant operation will be provided by the Power Production Department. Within this department the main support will be provided by several groups reporting to the Assistant Director of Power Production. These groups are the quality assurance group, the nuclear projects group, the nuclear fuel group and the licensing group. The fuel management group utilized the Nuclear Activities Department of Middle South Services for assistance in fuel cycle and core calculational matters.

We conclude that the applicant's organizational structure and qualifications of the plant personnel meet Regulatory Guide 1.8, "Personnel Selection and Training" and are satisfactory to provide an acceptable operating staff. We further conclude that the applicant has the necessary resources to provide offsite technical support for the operation of the facility.

13.2 Training Program

The Plant Superintendent has the overall responsibility for the conduct and administration of the plant training program. At the station level, the day-to-day administration of the training program is carried out by the Training Coordinator. The program for formal education and training of the facility staff has been designed to meet the individual needs of the participants, depending upon their backgrounds, previous training and expected job assignment. The program conforms to the requirements set forth in American National Standards Institute standard N18.1-1971 and 10 CFR Part 55.

The nuclear training program provides a flexible, effective means of preparing personnel for station operations and license examinations. Arkansas Power and Light Company will conduct or contract for the teaching of each segment of the training program. Certain segments are conducted by Combustion Engineering, Inc.

The training provided for personnel who will be licensed consists of the following discrete segments: basic mathematics, nuclear preparatory, nuclear fundamentals, radiation training, observation training, systems and procedure training, reactor simulator training, onsite training, and cold license audit exam and review.

A comprehensive training program also is conducted for the training of professional technical personnel and for technicians and repairmen involved in all station instrumentation, control and station monitoring systems. This specialists' training is provided by Combustion Engineering, the equipment manufacturers or at other nuclear and nonnuclear facilities in the Arkansas Power and Light Company system. All station personnel receive general employee training, as applicable to their normal duties, consisting of appropriate plans and procedures, radiological health and safety, industrial equipment, and the station emergency plan.

Plans for requalification training and replacement training conform to the requirements of 10 CFR Part 50, 10 CFR Part 55, Appendix A and follows the guidance given in American National Standards Institute Standard N18.1.

Complete records of all training administered will be maintained.

On the basis of our review, we conclude that the training programs and schedules for all staff members are acceptable for the preoperational test program, for operator licensing examinations and for fuel loading.

13.3 Emergency Planning

The applicant has formulated and submitted an Emergency Plan which describes the program for coping with emergencies within and beyond the site boundary. This Emergency Plan which has been approved for the operation of ANO-1 is also applicable to ANO-2. The plan describes the organization, including the responsibilities and duties assigned to station personnel, for coping with radiological as well as other emergencies. During an emergency, all onsite activities will be under the direction of an Emergency Coordinator and, if necessary, an Emergency Control Officer will coordinate the supportive services required from the appropriate outside agencies. The means for notifying plant employers and outside agencies whose services may be required in emergencies is provided by various telephone and radio communication systems.

The emergency plan identifies various employees of the applicant with special qualifications for coping with emergency conditions. Included are personnel having expertise in the various technical disciplines. The plan also identifies other persons not employed by the licensee, such as area physicians, whose special qualifications could be utilized in providing assistance during radiological emergencies.

The plan provides a classification system for a broad spectrum of emergency situations. For radiation emergencies, in-plant monitors, together with environmental dose determinations, will be used for evaluating the release of radioactive materials. Assessment techniques will employ the use of meteorological isopleth overlays for estimating projected doses to the environs. Criteria have been established for the notification and participation of Federal, state and local agencies. The applicant has also established specific criteria for implementing protective measures, both within and outside the site boundary, for the protection of the public health and safety.

The emergency plan provides procedures for notifying, and written agreements reached with the following agencies: Arkansas State Department of Health, Arkansas Executive Office of Civil Defense and Disaster Relief, Arkansas Department of Public Safety, Russellville Fire Department, Pope County Sheriff's Department, Pope County Civil Defense Organization, Pope County Civil Defense Communications Division, Yell County Sheriff's Department, Yell County Civil Defense Organization, Johnson County

Sheriff's Department, Johnson County Civil Defense Organization, Logan County Sheriff's Department and the Logan County Civil Defense Organization. The plan identifies the point of contact and/or the principal official for each of the agencies.

The emergency plan and procedures will be reviewed on at least an annual basis. Notification lists will be maintained current. Meetings will be held with plant employees as well as appropriate offsite support personnel to discuss any changes in the Emergency Plan and procedures that may affect them.

The plant emergency facilities provide first aid and decontamination capability for the medical treatment of contaminated personnel. The plan specifies the monitoring equipment, decontamination supplies, and medical supplies available at strategic locations throughout the plant and at the emergency control center. The Arkansas Nuclear One - Unit 2 plant organization includes personnel trained in first aid. A written agreement has been established with the Millard-Henry Clinic to provide physicians for professional medical assistance. The applicant maintains an onsite ambulance for the transportation of contaminated and/or injured individuals to offsite treatment facilities. In addition, written agreement has been made with the Pope County Ambulance Service for backup assistance in the event of multiple injuries. Written agreements have been made with Saint Mary's Hospital and the University of Arkansas Medical Center for the offsite treatment of injured personnel.

All plant personnel will receive indoctrination on the provisions of the emergency plan and procedures, and are expected to be familiar with the content and relation to their job. Each employee will receive training in the basic principles of radiological safety, including the use of protective clothing and radiation monitoring equipment. More extensive training is provided for the members of the various emergency teams. Outside agencies whose assistance may be required in emergencies will be briefed on the emergency plan and their related duties. Specialized training will be provided for the medical support personnel. Training is also accomplished through participation in drills which are conducted at least annually. Where applicable, the drills may involve the physical response of offsite support personnel.

The plan provides criteria for re-entry into evacuated areas following an accident. Guidelines have been established for emergency exposures during re-entry including those associated with life saving actions.

We have reviewed the applicant's emergency plan and conclude that it meets the requirements of 10 CFR Part 50, Appendix E, is responsive to the specific requirements of the staff, and provides an adequate basis for an acceptable state of emergency preparedness. Details and procedures to implement the emergency plan require inspection and evaluation by our Office of Inspection and Enforcement prior to the issuance of an operating license.

13.4 Review and Audit

The means for providing the review and audit of plant operations is described in Section 6.0, of the applicant's proposed standard technical specifications. The onsite group, the Plant Safety Committee, and the offsite group, the Safety Review Committee, are currently functioning in the manner described for Arkansas Nuclear One - Unit 1. We have approved these specifications and found they meet the provisions for review and audit described in Section 4.0 of American National Standards Institute standard N18.7-1972, "Administrative Controls for Nuclear Power Plants."

13.5 Plant Procedures And Records

All safety-related operating, maintenance and testing activities are to be conducted in accordance with approved, written procedures meeting the recommendations of Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)" and American National Standards Institute standard N18.7-1972. Areas covered include general station operating procedures, system operating precedures, emergency operating procedures, annunciator response procedures, procedures performed by non-licensed personnel including maintenance and testing activities and administrative control procedures. The applicant's provisions meet the requirements of 10 CFR Parts 50.54 (i), (j), (k), (l), and (m). Procedures addressing activities associated with safety-related structures, systems and components are and in the future will be forwarded to the Station Review Committee for review and comment. Upon approval by the Station Manager, a procedure becomes available for use.

We conclude that the provisions for preparation, review, approval and use of written procedures are acceptable.

The applicant has described his record keeping programs and has committed to keeping records according to American National Standards Institute Standard Nos. N18.7 and N45.2.9-1974. Specific records and their retention periods will be shown in the facility technical specifications.

Based on our review, we conclude that the applicant's provisions for maintaining records meet the position described in American National Standards Institute Standard N18.7-1972, "Administrative Controls for Nuclear Power Plants," and are acceptable.

13.6 Industrial Security

The applicant submitted an initial security plan for the Arkansas Nuclear One plant dated September 11, 1972. We reviewed the plan and ten subsequent revisions to the plan submitted between February 5, 1973 and October 31, 1976 and conclude that the security plan as amended is in conformance with existing criteria including Regulatory Guide 1.17 "Protection of Nuclear Power Plant 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, certair 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. The staff has reviewed these commitments and has determined that when properly implemented they will be acceptable.

The applicant has committed to implementing the nonconstruction portions of the security plan prior to the date of fuel loading. Certain procurement and construction activities will be implemented by the applicant prior to August 24, 1978, to upgrade the physical security measures for the plant site. This on-going upgrading of physical security is consistent with the graded implementation permitted by Part 73.55 and is acceptable.

We are performing a continuing review of the progress of the upgrading measures to be implemented by the applicant to assure conformance to the performance requirements of 10 CFR Part 73.55 on or before August 24, 1978.

14.0 INITIAL TESTS AND OPERATION

We have reviewed the information provided in the final safety analysis report, through Amendment 43, on the applicant's initial test program. This review included an evaluation of: (1) the applicant's organization and staffing for the development, conduct, and evaluation of the test program; (2) the qualifications and experience of the principal participants managing and supervising the test program; (3) the administrative controls that will govern the development, conduct, and evaluation of the test program; (4) the degree of participation of the plant operating and technical staff in the test program; (5) the applicant's requirements pertaining to the trial use of plant operating and emergency procedures during the test program; (6) the schedule to be followed; and (7) the methods for conducting individual tests and the acceptance criteria to be used in evaluating the test results for plant structures, systems, and components. The review also included an evaluation of the applicant's method of review of reactor plant operating experiences, conducted to determine where improvement or emphasis may be warranted in the initial test program.

The applicant proposed to conduct control rod testing that is not in full conformance with Regulatory Guide 1.68 "Preoperational and Initial Startup Test Programs for Water Cooled Power Reactors", in that there are no planned rod drop time measurements for each control element assembly at the cold no flow condition or at the hot no flow condition.

We indicated to the applicant that an acceptable response to this concern would be either a modification of the control element drive mechanism performance test summary in Section 14.0 of the Final Safety Analysis Report to include the scram time measurement of each control element assembly at the zero flow conditions or measurement of the rod drop times at other conditions which would form the boundary low flow conditions existing during operation of the plant when scram insertion of the control element assemblies could be required.

The applicant subsequently contended that zero flow testing is not necessary because tests with forced flow should yield more conservative results. The applicant also indicated that it was not planned to operate with fewer than two reactor coolant pumps in operation. We have, therefore, informed the applicant that omitting the zero flow rod droptime measurements would be acceptable only if the ANO-2 technical specifications prohibit rod withdrawal with fewer than two reactor coolant pumps in operation.

The applicant has not made a final determination regarding which of the two alternates indicated above will be implemented for the ANO-2 plant. Therefore, we will report our final evaluation of this concern in a supplement to this report.

Acceptance criteria for several transient and control system tests were not reviewed because the applicant stated that they are not available from the the nuclear steam system supplier or architect engineer at this time. These tests as listed by their numbers in Table 14.1-4 of the Final Safety Analysis Report are (3) load transient test, (4) control system checkout test, (7) turbine trip test, (8) load rejection test, (10) loss of offsite power test, and (18) steam dump valve bypass systems tests. The applicant has committed to establish criteria for those tests based on actual test conditions and beginning-of-life parameters or conditions and parameters that produce more conservative acceptance criteria. We find this to be an acceptable commitment. In addition, these acceptance criteria will be included in the test procedures which will be available for the Office of Inspection and Enforcement's review a nominal ninety days prior to the test.

The applicant has not proposed to conduct in-plant testing to demonstrate recirculation from the emergency core cooling system containment sump in conformance with Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors." The applicant has proposed to conduct out-of-plant scale model tests to demonstrate that adequate net positive suction head is available and vortexing will not occur under various flow conditions. The applicant will be required to submit a description of the sump model tests and the inplant testing that will be used to validate the calculations of head loss for the low pressure safety injection pump suction lines. This information will be reviewed by the staff prior to a decision on issuance of an operating license. The resolution will be reported in a future supplement to this report. This matter is also discussed in Section 6.3.4 of this report.

We have concluded, with the exception noted above, that the information provided in the application describes an acceptable initial test program that will demonstrate the functional adequacy of plant structures, systems, and components.

15.0 ACCIDENT ANALYSES

15.1 General

The applicant has performed safety analyses to evaluate the capability of the ANO-2 plant to withstand normal and abnormal operational transients and a broad sprectrum of postulated accidents without undue risk to the health and safety of the public. The events considered include all relevant types discussed in the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," LWR Edition, dated September 1975. The classification of postulated events with respect to the evaluation criteria applied by Combustion Engineering Inc., in the Combustion Engineering Standard Safety Analysis Report (CESSAR, Docket No. 50-470) is also applicable to the list of events analyzed in Section 15.0 of the ANO-2 Final Safety Analysis Report as listed below:

(1) Class I

- (a) Does not induce fuel failures.
- (b) Does not lead to a breach of containment barriers and fission product release.
- (c) Does not require operation of any engineered safety features.
- (d) Does not lead to significant offsite radiation exposures.

(2) Class II

- (a) May induce fuel failures.
- (b) May lead to a breach of barriers and fission product release.
- (c) May require operation of engineered safety features.
- (d) May result in offsite radiation exposures in excess of normal operational limits.

(3) Class III

- (a) Very low occurrence probability.
- (b) Provide information relevant to site acceptability and certain design and performance aspects of the plant.
- (c) May require operation of engineered safety features.
- (d) May result in significant offsite radiation doses within the limits of 10 CFR Part 100.

The classification of the events listed in Section 15.0 of the ANO-2 Final Safety Analysis Report is itemized in Table 15.1 of this report.

15.2 Input Parameters for Safety Analyses

Assumptions and parameters employed in the analyses were reviewed. The trip set-points used for the safety analysis are usually more conservative than the nominal plant operating values and are listed for each transient in the Final Safety Analysis Report. The rod drop time used was three seconds for the rods to reach the 90 percent insertion position. This rod drop time is included in the technical specification requirements. The local power density trip provides the necessary overpower protection for anticipated transients. The high linear power level trip is used only in the control element assembly ejection accident analysis.

The initial conditions (i.e., reactor power, pressurizer pressure, reactor coolant temperature at the core inlet, core power distribution, etc.) were selected for the analysis to be consistent with the core operating limits allowed by the core operating limit supervisory system (COLSS). The core operating limits are defined as a set of initial conditions for which the specified acceptable fuel design limits are not violated as a result of the most rapid decrease in thermal margin caused by an anticipated operational occurrence.

In the analyses where the initial full power conditions are used, a core power level of at least 2900 thermal megawatts was assumed. Assumption of 2900 thermal megawatts is three percent higher than the proposed license power level which accounts for uncertainties.

Core physics parameters used in the accident analyses have been reviewed and found to be suitably conservative to represent the most adverse conditions of the core design throughout the first burnup cycle with respect to reactivity coefficients, control rod worths, and local power peaking factors provided that operating configurations are restricted to considered patterns. The present core parameters used for ANO-2 analysis are for first core only, hence for reload core the results must be reanalyzed and reevaluated for each reload core.

TABLE 15.1

CATEGORIES OF TYPICAL TRANSIENTS AND FAULTS

Class I

Uncontrolled control element assembly withdrawal from a subcritical or low power condition, including control element assembly or temporary control device removal error during refueling

Control element assembly misalignment

Uncontrolled boron dilution

Loss of forced reactor coolant flow

Startup of an inactive reactor coolant loop

Loss of external electrical load and/or turbine trip

Loss of normal feedwater flow

Loss of all alternating current power to the station auxiliaries (station blackout)

Excessive heat removal due to feedwater system malfunctions

Class II

Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, which actuate emergency core cooling

Minor secondary system pipe break outside containment

Inadvertent loading of a fuel assembly into an improper position

Class III

Major secondary system pipe failure

Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant system (loss-of-coolant accident)

Waste gas decay tank rupture

Steam generator tube rupture

Control element assembly ejection accident

Fuel handling accident

Single reactor coolant pump shaft seizure

15.3 Anticipated Transients

A number of plant transients can be expected to occur with moderate frequency as a result of equipment malfunctions or operator error in the course of refueling and power operation during the plant lifetime. Such transients meet the criteria of Class I in the evaluation and classification discussed in Section 15.1 of this report. Various chemical and volume control system malfunctions which could lead to an unplanned boron dilution incident have been reviewed. The ones that allow the operator the shortest time for corrective action have been analyzed starting from plant conditions of startup, power operation (automatic and manual), hot standby, cold shutdown, and refueling. The results of the analyses of these events showed that the operator has thirty-two minutes to take corrective action if a boron dilution incident occurs during refueling. For power operation in the manual control mode, the fuel is maintained within thermal limits by the high pressurizer pressure trip; in the automatic control mode the operator has more than sixty-three minutes after receipt of the first control element assembly insertion alarm to take corrective action via a control element assembly trip.

We have reviewed the analyses submitted for anticipated transients to ascertain that the transients do not violate the specific criteria which follow:

(1) Pressure in the reactor coolant and main steam system should not exceed 110 percent of design pressure (Section III of ASME Boiler and Pressure Vessel Code).

(2) Clad integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio throughout the transient will satisfy the 95/95 criterion and that the maximum centerline temperature remains below the fuel melting point. The 95/95 criterion provides a 95 percent probability, at a 90 percent confidence level, that no fuel rod in the core experiences a departure from nucleate boiling.

(3) Other plant conditions of a more serious nature are not induced by the transient if other independent faults of a more serious nature have not occurred.

It was found that the most limiting transients in regard to core thermal margins were the loss-of-forced reactor coolant flow, part length control element assembly drop and control element assembly withdrawal transients. For these transients, the minimum value of the departure from nucleate boiling ratio was approximately 1.3, which is also the limiting value accepted by the staff as evidence that clad integrity has not been jeopardized.

The most limiting transients with respect to pressure within the reactor coolant system were the loss of external electrical load transient and control element assembly withdrawal from one percent power. The peak reactor coolant system pressure of 2553 and 2561 pounds per square inch absolute respectively, did not result in violation of the 110 percent overpressure limits.

The control element assembly incidents result in the most rapid transients and are, therefore, discussed in further detail in Section 15.3.1 of this report.

The boron dilution incident evaluation presented in Section 15.1.4 of the ANO-2 Final Safety Analysis Report is consistent with the proposed technical specification limits for refueling boron concentration, and is acceptable.

The control element assembly incidents result in the most rapid transients and are discussed in further detail in Section 15.3.1 of this report.

We conclude that the plant design is acceptable with respect to transient response to events that might occur during the plant lifetime and that anticipated transients would not lead to more serious plant conditions in the absence of other faults.

15.3.1 Control Element Assembly

The control element assembly misoperation events analyzed by the applicant include mispositioning of individual full- or part-length control element assembly drops, and dropping of part-length control element assembly subgroups. A subgroup is defined as any one set of four symmetrical control element assembly, which is controlled by the same control element drive mechanism control system.

The effect of any individually mispositioned control element assembly on core power distributions will be evaluated by the control element assembly calculators and appropriate radial peaking factor penalties will be transmitted to the core protection calculators (CPCs). The CPCs will, themselves, assess other changes in core conditions (e.g., changes in coolant temperature, axial power distribution, power level) and initiate a low departure from nucleate boiling ratio or high local power density trip if required. However, there are trip delay times associated with the CPC generated departure from nucleate boiling ratio and high local power density trips (0.75 and 1.15 seconds, respectively) and time is required to insert control element assemblies following scram. To ensure that the CPCs can accommodate all misoperation events, it must be demonstrated that the elapsed time between initiation of the event and the time the core approaches either the departure from nucleate boiling ratio or local power density limit is sufficient to allow for CPC scram initiation and control element assembly insertion. Therefore, the misoperating events of most interest are those that result in a rapid decrease in margin to safety limit.

The drop of a full-length control element assembly results in an increase in the radial peaking factor and an initial decrease in reactor power. Subsequent return to power (with the degraded power distribution) is possible if the reactor control system is in the automatic mode. For this condition, the initial power reduction causes a mismatch between reactor and turbine power thus initiating automatic withdrawal of control element assembly.

The response of the core for a part-length rod drop is more complicated. The part-length rod system to be employed on ANO-2 is unique in several respects.

The part-length control element assembly drive mechanism is of the same design as that used on the full-length control element assemblies. Thus the part-length control element assemblies are trippable and both the part-length and full-length control element assemblies will fall into the core upon initiation of a scram. Non-trippable drive mechanisms were employed on earlier Combustion Engineering plants. Also, the design of the part-length control element assembly itself is different. Each rod has three distinct axial sections: the lower fifty percent of the part-length control element assembly group is Inconel; the next forty percent of the part-length control element assembly consists of a follower section filled with water; the upper ten percent of the part-length control element assembly group consists of boron carbide (similar to the construction of a full-length control element assembly).

As a result of the three section design, the dropping of a single part-length control element assembly or control element assembly subgroup results in complex reactivity changes versus time. The following contributions to the reactivity transient were considered in the analyses: (1) reactivity associated with the movement of the Inconel section from one axial region of the core to another; (2) insertion of the follower (water) section of the part-length control element assembly into the core; and (3) insertion of the boron carbide section of the part-length control element assembly into the core. Both the time dependence and value of the peak reactivity are dependent on the relative motion of the part-length control element assembly and the scram control element assemblies.

The analyses of the nuclear steam supply system response (total power, coolant temperature, system pressure) was performed using the CESEC code. The detailed response of the core (hot channel power, heat flux, fuel and cladding temperatures, etc.) were calculated using the STRIKIN code. Since the consequences of a single control element assembly or bank drop are strongly dependent upon the axial power distribution that exists at the start of the transient, the analyses were performed using several different axial power distributions as initial conditions with each distribution characterized by an axial shape index.*

The results of these analyses show that all of the drop events are limited by the departure from nucleate boiling ratio. Some of the events result in close approach

*Axial shape index (ASI) =

$$\frac{\text{Power in the bottom half of the core - power in the top half of the core}}{\text{total core power}}$$

to the peak linear heat rate limits. For each case studied, the departure from nucleate boiling ratio assumed as an initial condition was varied until the minimum departure from nucleate boiling ratio reached during the transient is equal to 1.3. The initial departure from nucleate boiling ratios so determined, together with the values of initial departure from nucleate boiling ratios established in a similar manner in the loss of the flow analyses are then plotted versus axial shape index. A bounding curve is constructed that envelopes all points on the plot. This curve of minimum departure from nucleate boiling ratio versus axial shape index is an operating limit that must be observed by the operator during all phases of normal operation. This curve has been included as a limiting condition for operation in the technical specifications.

We have reviewed the analyses of the misoperation events and conclude that the general approach used to establish that these transients can be accommodated is acceptable. The applicant has not evaluated the effects of dropping both part length control element assembly subgroups on the basis that no single failure can cause this to happen. The staff conducted a detailed review of drawings of the electrical system and wiring routing diagrams to evaluate this basis. Our review determined that a single failure in the "Zero Crossing Detector" module could cause both part length control element assembly subgroups to drop. The applicant has agreed to put a barrier between the two affected circuit board cards in the module to preclude the degradation of the circuit boards for both subgroups. Therefore, we concur with his conclusion that no single failure can cause both subgroups of part length control element assembly to drop. This condition therefore need not be considered as an anticipated transient, for which we allow no fuel damage.

In addition to evaluating the design to single failures one can postulate events of extremely low probability which could result in the dropping of both part length control element assembly subgroups. These involve selective failures to only part length control element assembly subgroup components, and not to full length control rods. Dropping of any full length control rods along with part length control element assemblies provides sufficient negative reactivity to negate the positive reactivity insertion gained from dropping either or both of the part length control element assembly subgroups. The type of failure which can be postulated includes a fire burning only the part length control element assembly subgroup components in the control cabinet or in the raceways, or an earthquake shaking loose only part length control element assembly subgroup relays, or a missile impinging only on part length control element assembly subgroup controls or wiring. We consider such selective incidents to be of sufficiently low probability that they would be accidents, not anticipated transients. We have not required analysis of this accident because it is clearly less limiting than other accidents analyzed and evaluated in this report.

15.4 Postulated Accidents

The plant has been analyzed to evaluate the effects and potential consequences of postulated accidents due to single faults which have small to extremely remote probability of occurrences. Such accidents meet the criteria of Class II and III events in the evaluation and classification discussed in Section 15.1 of this report.

We have reviewed the accident analyses submitted by the applicant to assure completeness and conservatism in the analysis, and to evaluate the acceptability of results.

We selected, for detailed analysis, six highly unlikely accidents that are representative of the spectrum of types and physical locations of postulated causes in the ANO-2 design and that involve the various engineered safety feature systems. The analyses of these accidents are discussed in the following sections. The calculated effects on the core and the potential consequences of these accidents exceed or are expected to exceed those of all other postulated accidents that directly affect the ANO-2 design and are the same as those analyzed for previously licensed pressurized water reactor plants. The accidents analyzed were (1) control element assembly ejection, (2) reactor coolant pump rotor seizure, (3) feedwater system piping breaks, (4) steam piping breaks inside and outside of containment, (5) reactor coolant system piping breaks, and (6) fuel handling accident. We have calculated doses for the control element assembly ejection, the fuel handling accident and the loss-of-coolant accident and have included them in Table 15.6 of this report.

On the basis of our experience with the evaluation of the steam line break and the steam generator tube rupture accidents for pressurized water reactor plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary coolant and secondary coolant radioactivity concentrations. We will include appropriate limits on primary and secondary coolant activity concentrations in the technical specifications to be issued with the operating license. Table 15.2 of this report lists the assumptions used to calculate the doses from a postulated steam generator tube rupture and steam line break accident. Table 15.3 of this report lists the calculated doses.

The radioactive waste gas decay tanks are designed to seismic Category I requirements. Therefore, the total failure of these tanks is sufficiently improbable that 10 CFR Part 100 guideline doses are applicable. Our calculations indicate that doses for failure of these tanks would be well within 10 CFR Part 100 guidelines. Appropriate technical specifications will be placed on the maximum activity that can be stored in any one tank at any time such that single failure of active components, including the lifting or sticking of a safety or relief valve, will not result in radiological consequences that exceed small fractions of 10 CFR Part 100 guideline doses.

15.4.1 Control Element Assembly Ejection

The mechanical failure of a control rod mechanism pressure housing would result in the ejection of a control element assembly. The consequences of this would be a

TABLE 15.2

ASSUMPTIONS USED FOR TUBE FAILURE AND STEAM LINE FAILURE ACCIDENTS

(1) Power = 2900 thermal megawatts

(2) 2-hour $^{X}/Q = 6.6 \times 10^{-4}$ seconds per cubic meter at exclusion boundary

(3) Iodine decontamination factor of 10 between water and steam

(4) Primary and secondary coolant equilibrium concentrations as limited by technical specifications (1.0 microcurie per gram Iodine-131 equivalent and $100/\bar{E}$ microcuries per gram noble gases for primary coolant and 0.1 microcurie per gram Iodine-131 equivalent for secondary coolant)

(5) Primary to secondary leak rate as limited by technical specifications (one gallon per minute)

(6) For accidents assumed to occur in coincidence with an iodine spike, the primary coolant concentration is limited by the technical specifications for 48-hour periods (60 microcuries per gram Iodine-131 equivalent at 100 percent power)

(7) Primary-secondary coolant equilibrium reached at thirty minutes after the accident

(8) Loss of offsite power so that steam is released from secondary side relief valve

(9) Source spike factor of 500 after accidents

(10) 10 percent of iodine and noble gases fuel activity in gaps

(11) All releases through the secondary system

TABLE 15.3

CALCULATED OFFSITE DOSES DUE TO TUBE RUPTURE AND STEAM LINE FAILURE

Accident	Two-Hour Exclusion Boundary Thyroid (rem)	Two-Hour Exclusion Boundary Whole Body (rem)	Course of Accident Low Population Zone Thyroid (rem)	Course of Accident Low Population Zone Whole Body (rem)
Steam-line break (with coincident iodine spiking)	4	<1.0	<1.0	<1.0
Steam-generator tube rupture (with coincident iodine spiking)	81	<1.0	5	<1.0
Steam-line break (no coincident iodine spiking	2	<1	<1.0	<1.0
Steam-generator tube rupture (no coincident iodine spiking)	7	<1.0	<1.0	<1.0

rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

Although mechanical provisions have been made to make this accident extremely unlikely, the applicant has analyzed the consequences of such an event.

The methods used to perform the analysis have been reviewed by the staff and found to be consistent with Regulatory Guide 1.77 "Assumption Used for Evaluating a Control Rod Ejection for Pressurized Water Reactors." These include the use of the computer code PDQ to determine radial peaking factors and a point kinetics representation of the core (the CHIC-KIN code) utilizing Doppler weighting factors calculated with the TWIGL code.

The ejection analysis was performed for beginning-of-life and end-of-cycle for both full power and zero power. To ensure a conservative analysis, the maximum radial peaking factor and maximum ejected rod worth were assumed to occur for the same initial conditions, although calculations indicated that this is not necessarily true. Further conservatism was introduced by increasing both the maximum ejected rod worth and radial peak by ten percent to account for the uncertainties.

The results show that, in all cases analyzed, the enthalpy of the hottest pellet is below 280 calories per gram limit recommended in Regulatory Guide 1.77. The full power analyses produced peak enthalpies of 150 to 180 calories per gram. The zero power cases produced the highest values, 275 calories per gram for both beginning-of-life and end-of-life. However, the analyses that produced these zero power results were based on peaking factors that are significantly higher than expected values. The use of more realistic peaking factors would produce significantly lower values. Even when these conservative results are included, the analyses shows that prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten uranium dioxide can be assumed not to occur.

The applicant used 200 calories per gram as a clad damage threshold. We requested that clad damage be evaluated based on the number of fuel pins experiencing a departure from nucleate boiling ratio of less than 1.3; the clad damage criterion given in Regulatory Guide 1.77. These latter results show that full power cases produce clad damage in less than six percent of the fuel pins; the zero power results indicate that clad damage will occur in less than twelve percent of the fuel pins. This zero power analyses was based on the same conservative power distribution used in the calculation of peak enthalpy.

Based on the conformance of the analysis with the recommendations of Regulatory Guide 1.77 we conclude that the applicant's analysis of the control element ejection accident is acceptable.

In addition to the applicant's analysis of this event as discussed above, we have performed an evaluation of the consequences of the control element ejection accident using the following assumptions:

(1) Power is 2900 megawatts thermal.

(2) Eleven percent of the fuel rods suffer clad failure due to the rod ejection accident.

(3) Twelve percent of iodine and noble gas activity in the fuel is in fuel-to-clad gaps.

(4) Release of total gap activity in failed fuel to primary coolant.

(5) Primary to secondary coolant operational leakage is 1.0 gallon per minute.

(6) Loss of offsite power so that steam is released from secondary.

(7) Primary-secondary coolant equilibrium reached at 30 minutes after the accident.

(8) Standard steam line release meteorology.

The calculated doses are listed in Table 15.6 of this report. We find the calculated doses to be well within the guideline values of 10 CFR Part 100.

15.4.2 Reactor Coolant Pump Rotor Seizure

The analysis of an instantaneous seizure of a rotor of a reactor coolant pump during any allowed mode of operation has been reviewed. This event was evaluated by the applicant using computer codes CESEC and TORC. We have completed our review and have accepted the use of the TORC and COAST codes for the analysis of the reactor coolant pump rotor seizure accident.

Our review of the CESEC code has progressed to the point that there is reasonable assurance that the analysis results dependent on CESEC will not be appreciably altered by any methodology revision that may be required as a result of the staff's further review of the code. The parameters used as input to the applicant's analysis were reviewed and found to be suitably conservative. The results of the analysis showed that less than two percent of the fuel rods experienced departure from nucleate boiling. This assures that the fuel damage will be minimal and that there will not be consequential loss of core cooling capability. The analysis showed that the maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressures.

The staff concludes that the plant design is acceptable with regard to possible seizure of a rotor of a reactor coolant pump subject to the receipt of a commitment from the applicant to perform confirmatory tests in support of the utilization of the CESEC and COAST codes for the ANO-2 analyses. The staff will require that a description of the test program be submitted for review. Some of the verification tests are expected to be conducted in the preoperational test program while others will be performed at a specified level of power. The staff will require, as a part of the commitment noted above, that the needed data and test results, obtained with proper instrumentation, will be submitted to the staff and will also be used by the applicant to confirm the pretest predictions by the CESEC and COAST codes. The results of our completed program will be applicable to ANO-2.

15.4.3 Feedwater System Piping Breaks

We have reviewed the results of the feedwater line break analysis performed by the applicant. These analyses were performed to determine the effects of primary system overheating and overpressurization due to loss of heat sink. The heat sink in this case is the steam generator with a broken feedline.

The applicant has performed parametric studies to determine the limiting transient with respect to the feedwater line break size. To be conservative the applicant has varied the initial steam generator mass and break size to obtain the following limiting conditions at the time of reactor trip:

(1) downcomer level at the low steam generator water level trip setpoint of the intact steam generator;

(2) steam generator connected to the ruptured feedwater line empties;

(3) greatest energy content of the reactor coolant system without initiating a trip (primary system pressure at 2422 pounds per square inch absolute); and,

(4) greatest reactor coolant pressure change with respect to the time of the trip.

These assumptions will generate occurrence of the following three signals to trip the reactor:

(1) unaffected steam generator low water level trip signal;

(2) affected steam generator low pressure trip signal;

(3) high pressurizer pressure trip signal.

In addition, the applicant has taken into account reactivity feedback due to heatup of fuel and moderator and has assumed that the most reactive control element assembly is stuck following a reactor trip. Feedwater discharged out of the broken pipe was assumed to be saturated liquid.

The applicant's analyses included a spectrum of feedwater line breaks inside and outside containment, during various modes of operation, and with and without offsite power. The accident which resulted in the most severe transient was determined and evaluated using the CESEC code. The result of the worst feedwater line break showed that no fuel damage and no consequential loss of core cooling capability will result. The maximum pressure within the reactor coolant system did not exceed 110 percent of the design pressure.

On the basis of our review we conclude that the analyses and consequences of this accident have been acceptably analyzed.

15.4.4 Spectrum of Steam Piping Breaks Inside and Outside of Containment

We have reviewed the analyses and effects of steam line break accidents inside and outside containment during various modes of plant operation and with and without offsite power. Initially, the applicant performed the analysis taking credit for the moisture carryover in the steam. However, this moisture carryover model is presently under review by the staff and has not been approved. Hence, we requested the applicant to reanalyze the steam line break accident without taking credit for the moisture carryover model. The applicant has fulfilled our request with the submittal of Amendment 36 of the Final Safety Analysis Report.

The applicant in his first set of analyses using the moisture carryover model determined and evaluated the most severe steam line break accident. In the subsequent submittal, the applicant has analyzed four cases without using the moisture carryover model. These cases were as follows:

(1) Full load, two loop initial condition, nozzle break, without loss of alternating current power;

(2) Full load, two loop initial condition, nozzle break, with loss of alternating current power;

(3) No load, two loop initial condition, nozzle break, without loss of alternating current power; and,

(4) No load, two loop initial condition, nozzle break, with loss of alternating current power.

One loop initial condition cases were not analyzed without using moisture carryover because the analyses performed with the moisture carryover model indicated that two loop cases gave the worst results.

Without the use of the moisture carryover model, larger reactor shutdown control element assembly worth is required because more energy is removed from the primary system. Larger removal of energy cools the primary system more and results in larger positive reactivity feedback. To circumvent this problem, the applicant has modified the ANO-2 plant technical specifications to assure operation with sufficient reactor shutdown control element assembly worth to override this increased reactivity feedback. New shutdown control element assembly worth requirements are -8.6 percent in reactivity for full power and -5.0 percent in reactivity for the no load condition, compared to previous -5.8 percent reactivity for full power and -2.4 percent reactivity for no load cases.

The staff has requested the applicant to provide further details on the two-dimensional and one-dimensional power distributions which were used to synthesize the two-dimensional results for the steam line break were requested. The applicant has committed to provide this information. We will report our evaluation of this item in a supplement to this report.

The results of the analyses of the spectrum of steam line break accidents for ANO-2 showed that no fuel damage and no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio experienced by any fuel rod was shown to be greater than 1.3. The maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressure. These results are acceptable to the staff. However, the additional information indicated above is required to complete our review of the plant response to this accident.

15.4.5 Spectrum of Reactor Coolant System Piping Breaks

The applicant has submitted a loss-of-coolant accident analysis showing compliance with Section 50.46 and Appendix K to 10 CFR Part 50. The staff evaluation of this submittal and conclusions based on this submittal will be issued as a supplement to this report.

15.4.6 Loss-of-Coolant Accident

The containment model used to describe the dose mitigating effects of the engineered safety features for the ANO-2 plant include a low leakage single containment structure surrounding the reactor and a sodium hydroxide injection system operating in conjunction with the containment spray system. The purpose of the sodium hydroxide additive injection system is to increase the iodine removal capability of the spray following the hypothetical loss-of-coolant accident. The assumptions we used in evaluating the consequences of this accident are given in Table 15.4 of this report. The results indicate that the potential radiological consequences are within the guideline values of 10 CFR Part 100.

As part of the loss-of-coolant accident, we and the applicant have also evaluated the consequences of leakage of containment sump water which is circulated by 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 @ 3200 meters	1.2×10^{-4}
8-24 hours @ 3200 meters	7.6×10^{-5}
24-96 hours @ 3200 meters	3.0×10^{-5}
96-120 hours @ 3200 meters	8.2×10^{-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	1.17×10^{5}
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

emergency core cooling system outside the containment after a postulated loss-of-coolant accident. We have assumed the sump water 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." After the loss-of-coolanet accident, this water is circulated into the auxiliary building to be cooled. If a source of leakage should develop, a portion of the iodine could become gaseous and exit to the outside atmosphere. The applicant has estimated a low level leakage rate of about 2100 cubic centimeters per hour (0.54 gallons per hour) from components of the emergency core cooling system. Our calculation of the dose resulting from assumed leakage of that amount is small, and when added to the calculated loss-of-coolant accident dose at the low population zone, is still well within Part 100 guidelines. We have also considered the possibility of substantial amounts of leakage over a short term period, i.e., failure of a pump shaft seal. Calculated doses from iodine releases caused by such leakage, if not filtered before release, could result in doses which exceed Part 100 guidelines. Leakage from those areas in the auxiliary building in which emergency core cooling system components are housed is treated by a filtration system with an iodine removal efficiency of at least 90 percent for the elemental form and 70 percent for the organic forms, and we conclude that adequate provisions to limit doses from this pathway have been provided.

15.4.7 Fuel Handling Accident

A fuel handling accident may be postulated to occur within containment or within the spent fuel pool area of the auxiliary building. We have not completed our analysis of the accident within containment. For the spent fuel pool area, we have assumed that a fuel assembly was dropped in the spent fuel pool during refueling operations and that all of the fuel rods in the assembly were damaged, thereby releasing the volatile fission gases from the fuel rod gaps into the pool. The radioactive material that escaped from the fuel pool was assumed to be released to the environment over a two-hour time period with the iodine activity reduced by filtration through engineered safety feature system filters. The dose results are shown in Table 15.6 and the assumptions and parameters used in the analysis are shown in Table 15.5. The dose model and dose conversion factors employed in the analysis were in agreement with those given in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." Calculated doses for the fuel handling accident in the spent fuel pool are well within the guidelines of 10 CFR Part 100. We will report on our evaluation of the radiological consequences for the fuel handling accident within containment in a supplement to this report.

15.4.8 Postulated Radioactive Releases Due to Liquid Tank Failures

The consequences of component failures which could result in release of liquids containing radioactive materials to the environs were evaluated for components located outside the reactor containment. Considered in the evaluation were (1) the

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 @ 3200 meters	1.2×10^{-4}

TABLE 15.6

POTENTIAL OFFSITE DOSES DUE TO DESIGN BASIS ACCIDENTS

	Two-Hour Exclusion Boundary (1045 Meters)		Course of Accident Low Population Zone (3200 Meters)	
Accident	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
Loss-of-Coolant	236	5	269	2
Fuel Handling (In spent fuel pool area)	35	<1.0	5	<1.0
Rod Ejection	51	<1.0	4	<1.0

radionuclide inventory in each component assuming a one percent operating power fission product source term, (2) a component liquid inventory equal to eighty percent of its design capacity, (3) the mitigating effects of plant design including the location of storage tanks in curbed areas designed to retain spillage, and (4) the effects of site geology and hydrology.

The applicant has incorporated provisions in the design to retain releases from liquid overflows. The site is adjacent to Dardanelle Reservoir. In the event of a spill resulting in radionuclides entering the ground water, the ground water flow will move the spillage towards the reservoir.

Based on our evaluation, the potential tank failure resulting in the greatest quantity of activity released to the environment is failure of one of the boron management holdup tanks. The tank is assumed to contain radionuclides at twenty percent of primary coolant activity levels for the design basis fission product inventory stated above. In our evaluation, we have determined the liquid transit time for the leakage to the reservoir to be 2800 years. Considering the leakage transit time, the calculated radionuclide concentrations in Dardanelle Reservoir result in values that are small fractions of the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, for unrestricted areas. Based on the foregoing evaluation, we conclude that the provisions incorporated in the applicant's design to mitigate the effects of component failures involving contaminated liquids, are acceptable.

15.5 Anticipated Transients Without Scram

The applicant's response to the requirements of WASH-1270, "Anticipated Transients Without Scram for Water-Cooled Power Reactors," was submitted in a letter dated October 1, 1974. In its submittal, the applicant referenced two Combustion Engineering topical reports, CENPD-149, "Review of Reactor Shutdown System (RPS Design) for Common Mode Failure Susceptibility," and CENPD-158, "Anticipated Transients Without Reactor Trip." We have completed our review of these topical reports on December 9, 1975 and issued a "Status Report on Anticipated Transients Without Scram for Combustion Engineering Reactors," which reports the results of this review. The Advisory Committee on Reactor Safeguards reviewed this report and our status reports on other reactor vendors, at its 189th meeting on January 8-10, 1976. The Committee's interim report on this matter is contained in a letter dated January 14, 1976.

In our status report, we identified certain outstanding issues for which we require that applicants with Combustion Engineering plants provide additional information and modify their present design. We are currently developing a program of implementation of our requirements. When this program has been completed we will require that these modifications be implemented at ANO-2 as appropriate.

In addition we noted, in a letter dated March 29, 1977, that there were sufficient questions regarding the applicability of Combustion Engineering topical report

CENPD-158 - Rev. 1 to ANO-2 and therefore requested that the applicant provide anticipated transients without scram analyses applicable to ANO-2. The applicant's response, dated May 3, 1977, references their September 20, 1976 submittal in which they proposed to use the Combustion Engineering supplementary protection system. The applicant contends that incorporation of the supplementary protection system would eliminate the need for further analysis on the ground that the probability of an anticipated transient without scram event would have been reduced to an acceptably low value.

The staff is rereviewing the entire anticipated transients witout scram program and intends to publish a technical report. The staff report will include an evaluation of the supplementary protection system. It is noted, however, that the acceptance of the supplementary protection system alone to satisfy anticipated transients with scram requirements is unlikely. If the supplementary protection system is found to be unacceptable, the applicant would be required to provide further analyses and identify the design changes to meet anticipated transients without scram limits for ANO-2. The anticipated transients without scram program is being reviewed as a generic issue for ANO-2 as for other plants. We will require that any changes indicated to be needed as a result of the approved Combustion Engineering analyses shall be incorporated into the design in a timely manner.

With regard to the effect of this matter on the review and licensing process, at this time we see no reason to change the conclusion as stated in WASH-1270, that limitations on this account are not necessary or appropriate. This conclusion is based on our determination that the likelihood of an anticipated transient without scram event is very low considering the number of plants now in operational status, or expected to come into operation before our requirements can be fully implemented.

16.0 TECHNICAL SPECIFICATIONS

The technical specifications in a license define certain features, characteristics and conditions governing operation of a facility that cannot be changed without prior approval of the Commission. The finally approved technical specifications will be made a part of the operating license. Included will be sections covering safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

At the time of submittal of the Final Safety Analysis Report, the applicant had proposed technical specifications in Chapter 16. Shortly thereafter, we informed the applicant that we intended to use the Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors as the basis for development of the final technical specifications for ANO-2.

The Standard Technical Specifications for Combustion Engineering plants to be used as the basis for the plant technical specifications have been updated as a result of their application to technical specifications for other plants and also as a result of continued discussion with Combustion Engineering and applicants with Combustion Engineering reactors.

We are currently working with the applicant to finalize the technical specifications for ANO-2. On the basis of our review to date, we conclude that normal plant operation within the limits of the technical specifications will not result in potential offsite exposures in excess of the 10 CFR Part 20 limits. Furthermore, the limiting conditions for operation and surveillance requirements will assure that necessary engineered safety features will be available in the event of malfunctions within the plant.

17.0 QUALITY ASSURANCE

17.1 General

The description of the quality assurance program for the operational phase of the Arkansas Nuclear One, Unit 2 (ANO-2) is contained in Arkansas Power & Light Company topical report APL-TOP-1-A, "Quality Assurance Manual - Operations." Our evaluation of this quality assurance program is based on a detailed review of this information and discussions with representatives of Arkansas Power & Light Company (applicant) to assess if the applicant has the program and resources to comply with the requirements of Appendix B to 10 CFR Part 50 and supplemental guidance contained in the documents WASH 1284, "Guidance on Quality Assurance Requirements During the Operational Phase of Nuclear Power Plants"; WASH 1309, "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants"; and WASH 1283, "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants - Revision 1."

17.2 Organization

The organizational structure responsible for the operation of ANO-2 and for the establishment and execution of the operational phase quality assurance program is shown in Figure 17.1. The Senior Vice President of Production, Transmission and Engineering, who has the overall responsibility for the engineering, design, procurement, construction, operation and quality assurance of ANO-2, has delegated the responsibility for all activities related to the operation of ANO-2 to the Director of Power Production.

The Manager of Quality Assurance, who reports to the Assistant Director of Power Production, has been delegated the authority for implementation and control of the quality assurance program. The applicant quality assurance organization, which is independent of undue influences and responsibilities from schedules and costs, has the authority and freedom to: identify quality problems; initiate, recommend or provide solutions; and verify implementation of solutions. The quality assurance organization is given responsibility for: reviewing and approving quality related documents (e.g., instructions, procedures, drawings, and specifications); performing vendor prequalifications of quality assurance requirements; verifying by test or inspection that quality requirements are met for materials, components, processes and plant modifications; performing source inspections; documenting and reporting to responsible management any nonconformances discovered in the course of inspection, surveillance or audit; assuring corrective actions are effective and accomplished in

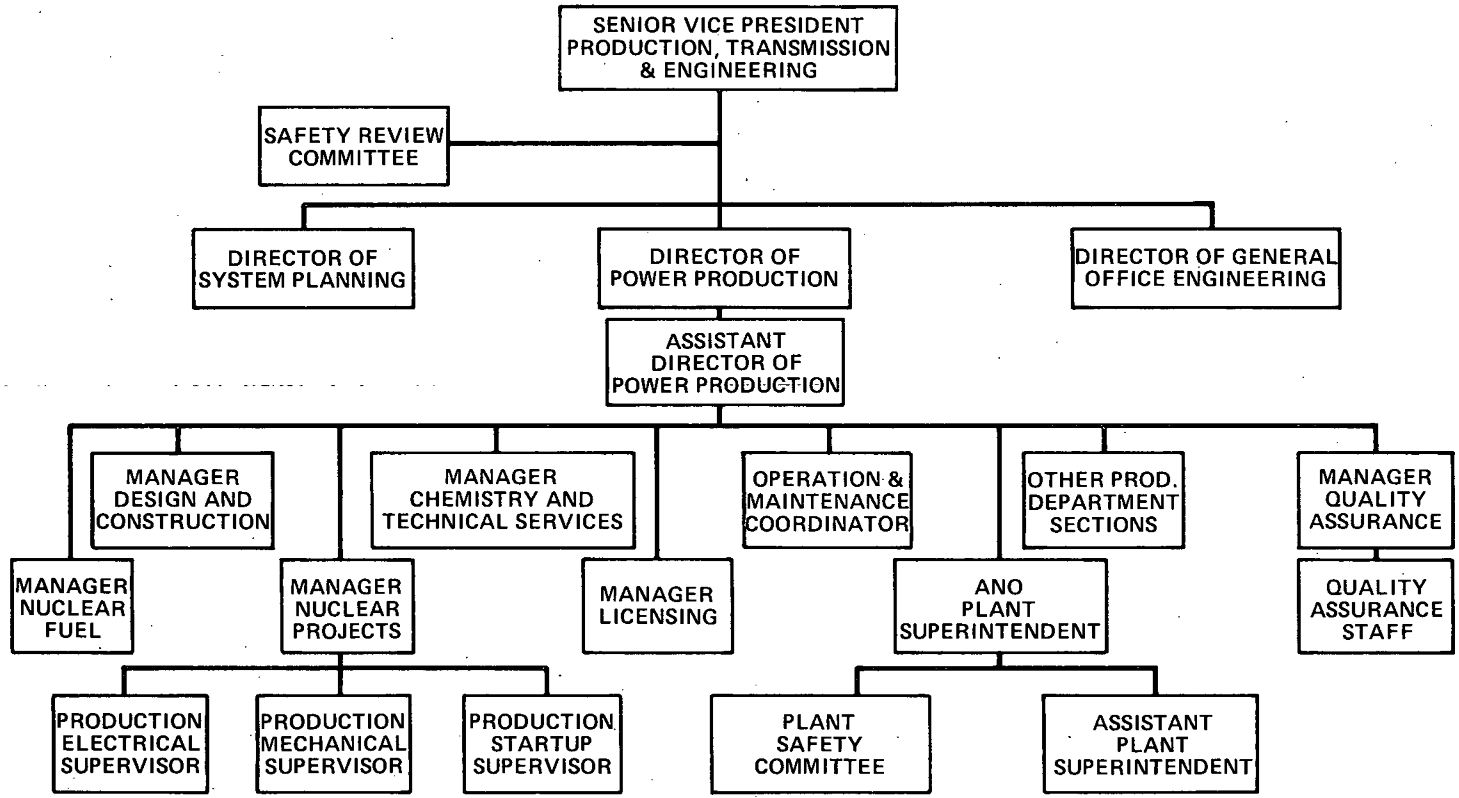

Figure 17.1 Arkansas Nuclear One - Unit 2, Arkansas Power and Light Company
Organization Chart

a timely manner; and surveillance and auditing of maintenance, repair and operation activities.

The Plant Superintendent is responsible for the safe and reliable operation of ANO-2. Disputes on quality assurance matters arising between individuals of the quality assurance organization and other applicant organizations are ultimately resolved by the Senior Vice President of Productions, Transmission and Engineering when the Manager of Quality Assurance fails to satisfactorily resolve these differences.

17.3 Quality Assurance

The quality assurance program for the operation of ANO-2 sets forth the quality assurance policies, which are established by the Senior Vice President of Production, Transmission, and Engineering and procedures which are contained in the quality assurance, quality assurance administrative, nuclear services, and master plant manuals. These manuals are the governing documents which control quality-affecting activities to comply with applicable requirements of Appendix B to 10 CFR Part 50. The applicant's quality assurance program, and revisions thereto, are reviewed and approved by the Manager of Quality Assurance and the Assistant Director of Power Production. The Senior Vice President of Production, Transmission and Engineering provides management assessment of the quality assurance program.

The quality assurance procedures, which are reviewed and approved by the Manager of Quality Assurance, encompass detailed controls for: translating codes, standards, regulatory requirements, technical specifications, engineering and process requirements into drawings, specifications, procedures, and instructions; developing, reviewing, and approving procurement documents, including changes; prescribing all quality-affecting activities by documented instructions, procedures or drawings; issuing and distributing approved documents; qualifying and certifying quality assurance and quality control personnel; purchasing items and services; identifying materials, parts, and components; performing special processes; inspecting and/or testing material, equipment, processes or services; calibrating and maintaining measuring and test equipment; handling, storing and shipping of items; identifying the inspection, test and operating status of safety-related items (plant Q list); identifying and dispositioning nonconforming items; correcting conditions adverse to quality; preparing and maintaining quality assurance records; and auditing of activities which affect quality.

An indoctrination and training program is established to assure that persons involved in quality-related activities are knowledgeable in quality assurance instructions and requirements and demonstrate a high level of competence and skill in the performance of their quality-related activities.

Quality is verified through surveillance, inspection testing, checking and audit of work activities. The quality assurance program requires that quality verification be performed by personnel other than those who performed the actual work activity. Inspections are performed using preplanned checklists by quality assurance and quality control personnel in accordance with written and approved inspection plans.

The qualifications of inspectors and their current status to conduct inspections, tests and examinations are based on applicable codes, standards, and AP&L training programs.

The quality assurance organization is responsible for the content and control of the audit program. Audits are performed in accordance with written procedures of checklists by appropriately trained personnel not having direct responsibility in the areas being audited. The audit activities, which are conducted yearly or on a more frequent basis as determined by the quality assurance organization, include an objective evaluation of quality assurance practices, procedures and instructions; work areas, activities, processes and items; the effectiveness of implementation of the quality assurance program; and compliance with policy directives.

The quality program requires both documentation of audit results and formal notification of the audit findings to the Manager of Quality Assurance and to management of the audited function. Audit findings, which indicate quality trends and the quality assurance program, are also reported to the Senior Vice President of Production, Transmission and Engineering. Management for the area audited implements the corrective action needed, if any. Follow-up audits are performed to determine that nonconformances are effectively corrected and that the corrective action precludes repetitive occurrences.

17.4 Implementation of the Quality Assurance Program

Adequacy of implementation of the applicants quality assurance program will be verified prior to licensing by the Commission's Office of Inspection and Enforcement who also will monitor the adequacy of quality assurance program implementation for the term of the operating license. If deviations or deficiencies in implementation of the quality assurance program are identified by Office of Inspection and Enforcement representatives, measures will be taken to assure that they are adequately corrected by the applicant and that steps are taken by the applicant to prevent their recurrence.

17.5 Conclusion

Based on our evaluation of the qualifications, duties, responsibilities, and authority for the various individual positions performing quality assurance functions, we conclude that the quality assurance organization for the Arkansas Power and Light Company has sufficient authority, and independence from undue influences of cost and schedule, to effectively conduct the operational quality assurance program for ANO-2. Our review and evaluation of the quality assurance program for the operational phase of ANO-2 has determined that the applicant's quality assurance program, as described in quality assurance topical report APL-TOP-1-A, provides a comprehensive system of planned and systematic controls such that quality-related activities will be conducted in accordance with the requirements of Appendix B to 10 CFR Part 50. We therefore conclude that the Arkansas Power & Light Company quality assurance program is acceptable for the operational phase of Arkansas Nuclear One - Unit 2.

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In its letter of February 10, 1972 the Advisory Committee on Reactor Safeguards indicated that certain matters would require further attention and resolution during the construction ANO-2.

Certain of these items are addressed further in this Safety Evaluation Report, as identified below. References are given to sections in this report for further discussion.

Margins for physics effects and additional thermal hydraulic analyses and testing	Section 4.0
Design parameter study for safety injection tank system	Section 6.3.2
Improvement of computer codes and research and development on other topics related to the emergency core cooling system analysis	Sections 4.3, 4.4 and 6.3
Inservice monitoring of vibration and loose parts	Sections 4.2.2 and 5.8
Containment hydrogen control system	Section 6.2.5
IEEE criteria for nuclear power plant protection system	Section 7.1
Anticipated transient without scram	Section 15.5
Other problems relating to large water reactors	Appendix C

The report of the Advisory Committee on Reactor Safeguards on the review of the application for an operating license for ANO-2 will be placed in the Commission's Public Document Room and will be included by the Commission staff in a supplement to this Safety Evaluation Report. The supplement will be published prior to the final determination regarding issuance of an operating license and will address the comments of the committee.

19.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are citizens of the United States. The applicant is not owned, dominated, or controlled by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any Restricted Data, but the applicant has agreed to safeguard any such data that might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes is involved. For these reasons, and in the absence of any information to the contrary, we find that the activities to be performed will not be inimical to the common defense and security.

20.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to financial data and information required to establish the financial qualifications of an applicant for a facility operating license are 10 CFR Part 50, Section 33(f) and 10 CFR Part 50, Appendix C. To assure that we have the latest information to make a determination of the financial qualifications of the applicant, it is our current practice to review this information during the later stages of our review of an application. We are reviewing the applicant's financial qualifications and will report the results of our evaluation in a subsequent report.

21.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

21.1 General

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licenses for facilities such as power reactors under 10 CFR Part 50.

21.2 Preoperational Storage of Nuclear Fuel

The Commission's regulations in 10 CFR Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

The applicant will furnish the Commission proof of financial protection in the amount of $1,000,000 in the form of a Nuclear Energy Liability Insurance Association Policy (Nuclear Energy Liability Policy, facility form No. NF-210). Further, the applicant will execute an Indemnity Agreement with the Commission effective as of the date of its preoperational fuel storage license. The applicant will pay the annual indemnity fee applicable to preoperational fuel storage.

21.3 Operating Licenses

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 protecton 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 $450 million.

Accordingly, a license authorizing operation of ANO-2 will not be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating lcense document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended so that the policy limits have been increased to meet the requirements of the Commission's regulations for reactor operation. Similarly, operating licenses will not be issued until an appropriate amendment to the present indemnity agreement has been executed. The applicant will be required to pay an annual fee for operating license indemnity as provided in our regulations, at the rate of $12 per thousand kilowatts of thermal capacity authorized in its operating licenses. On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the operating licenses, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licenses, including those as to proof of financial protection in the requisite amount and as to execution of an appropriate indemnity agreement with the Commission.

22.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, we have determined that, upon favorable resolution of the outstanding matters described herein, we will be able to 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 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.

Before an operating license will be issued to the applicant for operation of Arkansas Nuclear One - Unit 2, the facility must be completed in conformity with the construction permit, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power levels must be verified by the Commission's Office of Inspection and Enforcement prior to issuance of the license.

Further, before an operating license is issued, the applicant will be required to satisfy the applicable provisions of 10 CFR Part 140.

APPENDIX A

CHRONOLOGY OF RADIOLOGICAL SAFETY REVIEW

March 1, 1974	Application tendered by submittal of Amendment No. 20 to the application for license.
March 8, 1974	Staff letter advising applicant that an acceptance review of tendered application is being conducted.
March 22, 1974	Staff letter on operator requalification program.
March 25, 1974	Staff letter requesting reevaluation of security plan.
April 10, 1974	Bechtel Corporation letter on snubbers.
April 11, 1974	Staff letter accepting application for docketing and identifying deficient areas.
April 17, 1974	Application docketed.
April 17, 1974	Staff letter acknowledging receipt of application and transmitting Federal Register notices.
April 23, 1974	Notices of opportunity for hearing, receipt of application and consideration of issuance of facility operating license published in Federal Register.
May 15, 1974	Staff letter transmitting Amendment No. 1 to the construction permit.
May 16, 1974	Applicant letter requesting approval for use of certain codes and standards.
June 26, 1974	Applicant letter responding to staff's April 11, 1974 request for information.
July 3, 1974	Staff letter approving usage of ASME code cases.
July 11, 1974	Applicant letter responding further to staff's April 11, 1974 letter.

July 16, 1974	Amendment No. 21 docketed.
August 6, 1974	Staff letter establishing review schedule.
August 12, 1974	Amendment No. 22 docketed.
September 13, 1974	Staff letter rescheduling the reviews.
September 16, 1974	Amendment No. 23 docketed.
October 1, 1974	Applicant letter responding to the anticipated-transient-without-scram concern.
October 1, 1974	Applicant letter on date for submittal of emergency core cooling system analysis.
October 15, 1974	Amendment No. 24 docketed.
October 25, 1974	Staff letter on schedule for review of ECCS analysis.
November 19, 1974	Applicant letter on vibration measurements of reactor internals.
November 21, 1974	Staff letter requesting additional information on Quality Assurance Topical Report.
December 11, 1974	Staff letter requesting additional containment systems information.
December 17, 1974	Amendment No. 25 docketed.
December 27, 1974	Staff letter advising of status of reactor protection system review.
January 9, 1975	Applicant submittal of Revision 1 to Quality Assurance Manual for Operations.
January 13, 1975	Applicant letter responding to staff's letter of December 27, 1974 on core protection calculator system.
January 14, 1975	Applicant letter responding to staff's December 11, 1974 letter.
February 7, 1975	Staff letter requesting classification of quality assurance topical report.

February 13, 1975	Staff letter requesting information on operator training programs.
February 18, 1975	Staff letter requesting information on reactor internals topical report.
February 25, 1975	Amendment No. 26 docketed.
February 26, 1975	Staff letter requesting schedule for response to staff's letter of February 18, 1975.
March 4, 1975	Staff letter requesting information on reactor internal's vibration measurement topical report.
March 4 and 5, 1975	Staff meeting with applicant to discuss operating license review matters.
March 12, 1975	Staff letter requesting additional information on electrical, instrumentation and control systems.
March 18, 1975	Applicant letter responding to staff's February 13, 1975 letter.
March 20, 1975	Applicant letter responding to staff letters of February 26, 1975 and March 4, 1975.
March 27, 1975	Applicant letter transmitting detailed electrical drawings.
March 28, 1975	Applicant letter transmitting revision to quality assurance manual.
April 1, 1975	Staff letter requesting information on the industrial security plan.
April 2, 1975	Meeting on reactor internals and reactor vessel supports design.
April 3, 1975	Applicant letter responding to staff's March 12, 1975 letter.
April 9, 1975	Site visit by staff.
April 10, 1975	Department of the Army letter to staff on site foundation conditions.

April 14, 1975	Applicant letter on schedule for responding to the staff's April 1, 1975 letter.
April 17, 1975	Applicant letter reporting delay in completion of emergency core cooling system analysis.
April 18, 1975	Applicant letter on facility standard technical specifications.
April 18, 1975	Staff letter requesting additional information.
April 21, 1975	Amendment No. 27 docketed.
April 21, 1975	Staff letter on revisions to piping and instrumentation diagrams.
April 25, 1975	Staff letter requesting additional information.
May 1, 1975	Applicant letter addressing staff's April 28, 1975 letter.
May 5, 1975	Applicant letter providing response dates for questions of staff's April 18 and 25, 1975 letters.
May 6, 1975	Applicant letter transmitting detailed electrical drawings.
May 16, 1975	Applicant letter responding to staff's April 18, 1975 and April 25, 1975 letters.
May 23, 1975	Applicant letter regarding high density spent fuel storage racks.
May 23, 1975	Applicant letter transmitting industrial security plan information.
May 27, 1975	Amendment No. 28 docketed.
May 27, 1975	Applicant letter regarding vibration measurements of reactor internals.
June 2, 1975	Applicant letter transmitting detailed electrical drawings.
June 4, 1975	Staff letter requesting additional information.

June 6, 1975	Applicant letter requesting ASME code case approval.
June 12, 1975	Staff letter responding to applicant's letter of June 6, 1975 granting approval for use of ASME Code Case 1685.
June 16, 1975	Staff letter revising review schedule.
June 16, 1975	Applicant letter regarding portions of staff letters of April 18, 1975 and April 25, 1975.
June 16, 1975	Applicant letter stating disagreement with staff positions 110.8, 110.16 and 110.25.
June 17, 1975	Applicant letter providing dates for response to staff's June 4, 1975 letter.
June 19, 1975	Applicant letter transmitting core protection calculator system functional block diagrams.
June 19, 1975	Applicant response to core protection calculator system questions 241.7, 241.8, 242.27 and 222.25 through 222.48.
June 25, 1975	Staff letter requesting additional information.
June 26, 1975	Applicant letter transmitting detailed electrical drawings.
June 26, 1975	Staff letter describing an acceptable fuel surveillance program.
July 1, 1975	Staff letter on requirements for preoperational reactor internals vibration tests.
July 2, 1975	Amendment No. 29 docketed.
July 8, 1975	Applicant response to staff's June 25, 1975 letter.
July 15, 1975	Amendment No. 30 docketed.
July 15, 1975	Applicant's response to staff's June 26, 1975 letter.
July 16, 1975	Applicant's response to staff's July 1, 1975 letter.

July 17, 1975	Staff letter on steam generator minimum allowable tube wall thickness.
July 22, 1975	Applicant letter regarding review of post accident monitoring recorders.
July 25, 1975	Applicant letter regarding part length control element assembly design changes.
July 30, 1975	Meeting on appeals items.
August 5, 1975	Staff letter regarding summary of meeting on appeal items.
August 5, 1975	Meeting on steam generator minimum tube wall thickness.
August 6, 1975	Applicant transmittal of piping and instrumentation drawings.
August 8, 1975	Staff letter regarding summary of meeting on steam generator tube integrity.
August 21, 1975	Amendment No. 31 docketed.
August 22, 1975	Applicant letter transmitting detailed electrical drawings.
August 29, 1975	Applicant letter transmitting core protection calculator system hardware drawings.
September 3, 1975	Applicant letter submitting detailed electrical drawings.
September 4, 1975	Applicant letter regarding reactor coolant system piping analyses.
September 4, 1975	Staff letter requesting additional information.
September 10, 1975	Applicant letter appealing certain staff positions.
September 16, 1975	Staff letter requesting additional information on containment systems (Question 042.20).
September 18, 1975	Staff letter in response to applicant's letter of August 19, 1975.
September 18, 1975	Applicant letter regarding questions in staff's September 4, 1975 letter.

September 18, 1975	Staff letter regarding applicant's September 9, 1975 letter.
September 24, 1974	Staff letter requesting additional information (Questions 222.72 - 222.80).
September 24, 1975	Staff letter requesting additional information (Questions 222.81 - 222.100).
September 24, 1975	Applicant letter transmitting detailed electrical drawings.
September 26, 1975	Applicant letter regarding question number 042.20.
September 29, 1975	Staff letter regarding part length control element assembly revised design.
October 1, 1975	Meeting to discuss potential appeals items.
October 2, 1975	Meeting on blowdown forces on reactor coolant system.
October 2, 1975	Appeals meeting on nine items.
October 10, 1975	Staff letter requesting additional information (Questions 222.101 - 222.139).
October 13, 1975	Applicant letter responding to staff's September 29, 1975 letter on part length control element assemblies.
October 13, 1975	Applicant letter providing schedule for response to September 24, 1975 questions 222.72 through 222.80.
October 22, 1975	Staff letter stating position on spent fuel pool cooling system.
October 22, 1975	Applicant letter regarding the staff's letter of October 10, 1975.
October 28 and 29, 1975	Meeting on four items.
October 31, 1975	Staff letter requesting additional information (Questions 222.140 - 222.163).
November 3, 1975	Applicant letter responding to questions 241.6 and 241.32.

November 10, 1975	Amendment No. 32 docketed.
November 10, 1975	Applicant letter requesting approval of use of ASME Code Case 1625.
November 11, 1975	Staff letter requesting additional information.
November 12, 1975	Applicant letter responding to core protection calculator system questions 222.64, 222.68, 222.70, 222.72, 222.76 and 222.79.
November 14, 1975	Staff letter regarding reactor pressure vessel support systems.
November 26, 1975	Applicant letter responding to staff's October 10, 1975 letter.
December 3, 1975	Meeting of seismic qualification review team at site to review balance-of-plant scope.
December 4, 1975	Applicant letter responding to earlier request for information regarding CPC questions.
December 5, 1975	Applicant letter responding to earlier requests for information.
December 10, 1975	Applicant letter responding to question numbers 222.65 and 222.66.
December 11, 1975	Amendment No. 33 docketed.
December 15, 1975	Staff letter requesting additional information (Questions 310.20 - 310.24).
December 16, 1975	Applicant letter responding to question numbers 020.5 and 020.42.
December 19, 1975	Applicant letter regarding reactor vessel support analysis.
December 22, 1975	Applicant letter submitting responses to core protection calculator system questions.
December 23, 1975	Staff letter requesting additional information (Question 130.8).

January 6, 1976	Amendment No. 34 docketed.
January 6 and 7, 1976	Meeting on review of electrical drawings.
January 9, 1976	Applicant letter responding to core protection calculator system questions 222.145 and 222.146.
January 14, 1976	Staff letter on code approval for application to refueling water storage tank.
January 14, 1976	Applicant letter transmitting large size piping and instrumentation diagrams.
January 27, 1976	Applicant letter on submittal of next amendment.
January 28, 1976	Applicant letter submitting detailed electrical drawings.
February 2, 1976	Applicant letter responding to core protection calculator questions 222.109 and 222.152.
February 3, 1976	Applicant letter on Dywidag threadbar connectors in response to staff's December 23, 1975 letter.
February 4, 1976	Staff letter selecting Phase I test case results to be submitted.
February 9, 1976	Applicant letter on reactor protection system and engineered safety feature actuation system trip logic.
February 12, 1976	Meeting on draft safety evaluation report open items.
February 17, 1976	Applicant letter responding to question 222.83.
February 18, 1976	Amendment No. 35 docketed.
February 25, 1976	Staff letter requesting additional information (Questions 214.48 through 214.55 and 110.33).
March 2, 1976	Applicant letter responding to question 222.97.
March 4, 1976	Applicant letter responding to questions 214.39, 214.41 and 214.42.

March 8, 1976	Applicant letter regarding fire protection systems.
March 9, 1976	Applicant letter responding to question 222.129.
March 11, 1976	Applicant letter responding to requests of staff's letter of February 25, 1976.
March 22, 1976	Applicant letter transmitting detailed electrical drawings.
March 23, 1976	Applicant letter transmitting Draft Standard Tech Specs.
March 24, 1976	Staff letter regarding meeting to be held in Inspection and Enforcement Regional Office related to Appendix I evaluation.
March 26, 1976	Applicant letter submitting spent fuel pool wall deficiency report.
March 29, 1976	Received undated applicant letter transmitting Figure Numbers 8.3-55 and 8.3-50.
March 31, 1976	Amendment No. 36 docketed.
March 31, 1976	Applicant letter responding to requests for information numbered 222.78, 222.107, 222.121, 222.124, 222.130 and 222.133.
April 1, 1976	Staff letter requesting additional information on Dywidag connectors question 130.9 and CPC's.
April 7, 1976	Applicant letter providing schedule for responses to staff's April 1, 1976 letter.
April 29, 1976	Meeting at site to conduct seismic qualification review.
May 5, 1976	Applicant letter responding to staff's April 1, 1976 letter on Dywidag connectors (Question 130.9).
May 5, 1976	Amendment No. 37 docketed.
May 5, 1976	Meeting on spent fuel pool wall deficiency report.
May 7, 1976	Applicant letter transmitting large size drawings.

May 10, 1976	Staff letter providing draft model technical specifications.
May 11, 1976	Staff letter regarding distribution of safety analysis and environmental reports.
May 12, 1976	Staff letter enclosing drawing review outline for forthcoming computer protection system drawing review.
May 12, 1976	Applicant letter responding to CPC questions 222.74, 222.132, 222.150, 222.151, 222.152, 222.162, 222.164 and 222.74.
May 17, 1976	Applicant letter regarding the electrical, instrumentation and controls meeting discussions of January 6 and 7, 1976.
May 17, 1976	Applicant letter responding to staff's May 12, 1976 letter on core protection calculator system drawing review meeting.
May 17, 1976	Applicant letter regarding inadvertent recirculation actuation signal.
May 24, 1976	Staff letter informing of events and conclusions regarding anticipated transients without scram.
May 24, 1976	Applicant letter submitting fuel assembly mechanical design integrity reports.
May 25, 1976	Applicant letter submitting "Arkansas Steam Generator Tube Structural Analysis of Tubes for Pipe Rupture Accidents," in response to question 110.33.
June 9, 1976	Staff letter regarding the site exclusion area.
June 10, 1976	Meeting of seismic qualification review team with applicant in San Francisco.
June 14, 1976	Applicant letter transmitting criteria for design basis pipe break locations.
June 18, 1976	Staff letter requesting additional information.
June 30, 1976	Applicant letter documenting to CPC questions 222.124 and 222.167.

July 1, 1976	Amendment No. 38 docketed.
July 1, 1976	Applicant letter transmitting spent fuel pool wall deficiency report.
July 6, 1976	Applicant letter documenting delivery of core protection calculator hardware relay isolation test results to staff.
July 7, 1976	Staff letter addressing licensing schedule and stating twenty-three core protection calculator system positions.
July 8, 1976	Applicant letter responding to part four of question 222.82.
July 9, 1976	Staff letter requesting core protection calculator system information (Questions 222.166 - 222.173).
July 12, 1976	Applicant letter transmitting additions to technical specifications.
July 14, 1976	Applicant letter transmitting list of revised piping and instrumentation drawings.
July 16, 1976	Applicant letter indicating that responses to twenty-three core protection calculator system positions will be delivered by August 6, 1976.
July 19, 1976	Applicant letter transmitting dates for responses to the staff's July 9, 1976 letter.
July 20, 1976	Applicant letter regarding nonproprietary versions of previous submittals.
July 21, 1976	Applicant letter regarding the staff's July 7, 1976 letter.
July 29, 1976	Staff letter summarizing outstanding items and issues.
July 30, 1976	Staff letter requesting additional information on thermal expansion, vibration and transient response tests.
August 2, 1976	Staff letter requesting additional information.
August 3, 1976	Applicant letter confirming submittal of seismic analysis qualification information.

August 3, 1976	Applicant letter documenting submittal of core protection calculator system Phase I Test documentation.
August 4, 1976	Applicant letter regarding staff's letter of July 38, 1976.
August 5, 1976	Applicant letter responding to question 130.9.
August 5, 1976	Applicant letter transmitting seismic analysis for process protective cabinet 2 C15.
August 6, 1976	Applicant letter responding to twenty-three core protection calculator system positions issued in staff's July 7, 1976 letter.
August 6, 1976	Applicant letter responding to questions 222.166 through 222.173.
August 9, 1976	Applicant letter documenting submittal of information in support of their August 4, 1976 response to 222.166 and 222.167.
August 9, 1976	Applicant letter scheduling responses to question no. 413.24 of the staff's July 30, 1976 letter.
August 13, 1976	Applicant letter transmitting additional responses to core protection calculator (CPC) system positions 8A.
August 13, 1976	Applicant letter transmitting CPC seismic and environmental qualification test procedures (Positions 3, 16 and 13).
August 18, 1976	Applicant letter transmitting CPC Phase II Design Qualification Test Report dated June 1976.
August 19, 1976	Applicant letter requesting extension of dates for completion of construction.
August 20, 1976	Applicant letter transmitting CPC Position 9 procedures.
August 31, 1976	Applicant letter transmitting fuel assembly test reports.
September 1, 1976	Amendment No. 39 docketed.
September 2, 1976	Meeting on core protection calculator system.

September 3, 1976	Applicant letter submitting environmental (thermal) test procedure for process protective cabinet no. 2 C15 (Position 11).
September 9, 1976	Applicant letter correcting their September 3, 1976 letter on cabinet 2 C15 environmental test procedure.
September 14, 1976	Staff letter requesting antitrust information.
September 16, 1976	Staff letter issuing core protection calculator system position no. 24.
September 21, 22, and 23, 1976	Meeting for electrical drawing review on reactor protection system and engineered safety features actuation systems.
September 22, 1976	Applicant letter transmitting control element assembly calculator separation criteria (Position 4).
September 24, 1976	Applicant letter transmitting figure on core protection calculator system.
September 24, 1976	Applicant letter recording the minutes of the September 2, 1976 meeting.
September 28, 1976	Staff letter transmitting order extending construction completion data.
September 30, 1976	Applicant letter responding to the anticipated-transient-without-scram analysis questions of the staff's May 24, 1976 letter.
September 30, 1976	Staff letter on fire protection program evaluation.
September 30, 1976	Applicant letter documenting delivery of core protection calculator system burn-in test procedures.
October 6, 1976	Applicant letter transmitting Revision No. 3 to Quality Assurance Manual Operations.
October 8, 1976	Applicant letter indicating antitrust information to be submitted by January 1977.
October 8, 1976	Applicant letter transmitting core protection calculator software change procedures (Position 19).

October 12, 1976	Applicant letter transmitting information on CPC Phase II Tests (Position 24(k) and the use of equate statements).
October 14, 1976	Applicant letter transmitting fuel assembly pluck impact test report.
October 19, 1976	Staff letter regarding licensing schedule and CPC burn-in test procedures (Questions 222.170 and 222.171).
October 26, 1976	Staff letter requesting information (Questions 042.30 through 042.35).
October 26, 1976	Staff letter requesting information for review of technical specifications.
October 29, 1976	Applicant letter on fire protection responding to staff's September 30, 1976 letter.
November 5, 1976	Applicant submittal of revisions to technical specifications.
November 8, 1976	Staff letter regarding reactor vessel supports analysis.
November 11, 1976	Staff letter requesting information on the main steam line break analysis.
November 19, 1976	Applicant letter submitting seismic qualification information.
November 19, 1976	Staff letter accepting revision to the quality assurance topical report.
November 22, 1976	Applicant letter responding to staff's November 11, 1976 letter.
November 24, 1976	Amendment No. 40 docketed.
November 30, 1976	Applicant letter regarding technical specifications on inservice inspection of containment tendons.
December 2, 1976	Staff letter regarding core protection calculator system.
December 3, 1976	Staff letter regarding fire protection evaluation.

December 10, 1976	Applicant letter scheduling responses to staff's October 21, 1976 letter.
December 17, 1976	Staff letter transmitting fire protection sample technical specifications.
December 27, 1976	Staff letter on reactor vessel overpressurization.
January 4, 1977	Staff letter summarizing outstanding items and issues.
January 5, 1977	Applicant letter transmitting environmental qualification test reports in partial response to core protection calculator system positions 2, 3, 6 and 13.
January 7, 1977	Applicant letter on fire protection evaluation.
January 11, 1977	Applicant letter responding to staff's January 4, 1977 letter.
January 11, 1977	Applicant letter responding to staff's letter of October 19, 1976.
January 11, 1977	Applicant letter responding to staff's letter of October 26, 1976.
January 17 and 18, 1977	Meeting on draft Safety Evaluation Report open items.
January 24, 1977	Meeting on reactor systems open items.
January 25, 1977	Amendment No. 41 docketed.
February 1, 1977	Applicant letter transmitting revisions to technical specifications.
February 3, 1977	Staff letter requesting information in Enclosures A through F.
February 7, 1977	Applicant letter scheduling overpressurization response for June 30, 1977.
February 10, 1977	Applicant letter responding to outstanding items.
February 15, 1977	Applicant letter submitting CPC functional descriptions, a listing of CPC software modifications and a status report on each position.

February 17, 1977	Meeting on fuel surveillance, containment sump tests and low population zone.
February 18, 1977	Appeals meeting.
February 18, 1977	Staff letter on core protection calculator system Burn-In Test procedures.
February 22, 1977	Staff letter on fire protection evaluation.
February 25, 1977	Applicant letter documenting submittal of electromagnetic interference environmental test procedures.
February 25, 1977	Staff letter on new physical security plan rule.
February 25, 1977	Applicant letter submitting 16 x 16 fuel assembly flow test report.
February 28, 1977	Applicant letter transmitting main steam isolation system electrical drawing package.
March 8, 9, 10 and 11, 1977	Fire protection review team visit to site.
March 9, 1977	Applicant letter regarding the CPC protection algorithms.
March 10, 1977	Applicant letter stating concern about licensing schedule delays.
March 11, 1977	Staff letter on fuel handling inside containment.
March 14, 1977	Applicant letter responding to staff's February 18, 1977 letter regarding core protection calculator system periodic test interval.
March 14, 1977	Applicant letter transmitting seismic qualification information.
March 14, 1977	Applicant letter responding to core protection calculator (CPC) position no. 25.
March 14, 1977	Applicant letter transmitting CPC system Phase I Test Procedure.
March 14, 1977	Applicant letter transmitting CPC Interim Burn-In Test report.

March 14, 1977	Applicant letter transmitting reed switch position transmitter radiation qualification analysis.
March 15, 1977	Applicant letter transmitting emergency core cooling system small break analysis information.
March 22, 1977	Staff letter requesting information on selection of instrumentation trip setpoint values.
March 23, 1977	Applicant letter responding to staff's letter of February 3, 1977 on environmental qualification of electrical switchgear and motor control center equipment.
March 28, 1977	Applicant letter transmitting information on seismic response spectra in cabinet 2 C15 module mounting points.
March 29, 1977	Staff letter requesting information on core protection calculator system (Questions 222.175 and 222.176).
March 29, 1977	Staff letter requesting information on 16 x 16 fuel assembly flow test (fretting aspects) report.
March 30, 1977	Applicant letter on feedwater and main steam line break analyses.
March 31, 1977	Applicant letter responding to Enclosure F (ECCS Analyses) of item 1 (long-term emergency core cooling analyses) of staff's February 3, 1977 letter.
April 4, 1977	Applicant letter on service water pump motor seismic analysis.
April 5, 1977	Staff letter responding to applicant's letter of March 10, 1977 on licensing schedules.
April 6, 1977	Applicant letter on fuel handling accident inside containment.
April 7, 1977	Applicant letter on exclusion area mineral rights.
April 11, 1977	Applicant letter responding to question no. 222.172.
April 14, 1977	Staff letter requesting information and stating positions on fire protection program.

April 15, 1977	Applicant letter on licensing schedule.
April 22, 1977	Staff letter transmitting format for reporting hourly meteorological data.
April 25, 1977	Applicant letter responding to Enclosure 1 of staff's March 29, 1977 letter on core protection calculator system.
April 25, 1977	Applicant letter transmitting photographs of differential pressure transmitters in support of the seismic qualification review.
April 25, 1977	Applicant letter on preoperational test outstanding items.
April 25, 1977	Applicant letter responding to staff's March 29, 1977 letter on fuel assembly flow test report.
April 25, 1977	Applicant letter transmitting information on separation of Class IE and non-Class IE circuitry in instrument cabinets.
April 25, 1977	Combustion Engineering Power Systems letter regarding licensing schedule.
May 2, 1977	Staff letter to ACRS transmitting draft SER on CPCS.
May 3, 1977	Applicant letter on applicability of anticipated-transients-without-scram topical reports.
May 4, 1977	Staff letter transmitting Instrusion Detection Systems Handbook.
May 5, 1977	Staff letter requesting financial qualifications and reactor systems information (Questions 214.57 - 214.59).
May 17, 1977	Applicant letter transmitting fire protection information.
May 20, 1977	Advisory Committee on Reactor Safeguards Visit to Combustion Engineering Inc. offices.
May 20, 1977	Combustion Engineering letter transmitting syllabus of slides for the May 20 ACRS Subcommittee meeting.
May 23, 1977	Staff letter responding to Combustion Engineering Power Systems letter of April 25, 1977.

May 23, 1977	Staff letter requesting additional information on main steam line break analysis (Question No. 214.53).
May 25, 1977	Applicant letter transmitting CPC final Burn-In Test report.
May 25, 1977	Applicant letter transmitting EMI Test Report.
May 25, 1977	Applicant letter transmitting updated physical security plan.
May 25, 1977	Applicant letter transmitting emergency core cooling systems large break analyses.
May 31, 1977	Amendment No. 42 docketed.
May 31, 1977	Applicant letter on part length control element assembly.
May 31, 1977	Applicant letter on remote shutdown test.
June 2, 1977	Meeting on redundant valve position indication and part length control element assemblies.
June 3, 1977	Applicant letter transmitting antitrust review information.
June 13, 1977	Applicant letter on control rod drop tests.
June 13, 1977	Applicant letter transmitting responses to core protection calculator system positions nine, ten, fifteen, eighteen, twenty-six and twenty-seven.
June 17, 1977	Applicant letter responding to main steam line break analyses questions.
June 21, 1977	Staff letter requesting information on core protection calculator system.
June 24, 1977	Advisory Committee on Reactor Safeguards Subcommittee meeting in Russelville, Arkansas.
June 30, 1977	Advisory Committee on Reactor Safeguards Electrical Subcommittee meeting on core protection calculator systems in Washington, D.C.
June 30, 1977	Staff letter to Advisory Committee on Reactor Safeguards transmitting memorandum report on the core protection calculator system.

July 5, 1977	Applicant letter transmitting responses to staff's February 3, 1977 letter on environmental qualification of equipment.
July 6, 7 and 8, 1977	Staff meeting with applicant at the plant site to conduct the instrumentation and control systems audit inspection of the plant.
July 7, 1977	Applicant letter responding to staff's letter of July 7, 1976 on Position 20 of core protection calculator system.
July 18, 1977	Applicant letter responding to open items 7.6.3 and 7.7 from the meeting of June 2, 1977.
July 18, 1977	Applicant letter submitting corrections to antitrust information previously submitted on June 3, 1977.
July 18, 1977	Applicant letter responding to staff letter of March 29, 1977 on backup trip analysis for core protection calculator system.
July 19, 1977	Applicant letter discussing the core protection calculator Position No. 20.
July 19, 1977	Applicant letter transmitting supplements to the core protection calculator functional description documents and the data base documentation.
July 19, 1977	Applicant letter responding to staff's letter of May 5, 1977 on questions 214.58 and 214.59.
July 19, 1977	Applicant letter responding to staff's letter of May 5, 1977 on fuel assembly local loss coefficients.
July 22, 1977	Amendment 43 docketed.
July 22, 1977	Staff letter requesting information on fire protection program.
July 25, 1977	Staff letter to Advisory Committee on Reactor Safeguards transmitting a letter report on the core protection calculator system.
July 27, 1977	Staff meeting with applicant to discuss draft technical specifications.

July 27, 1977	Applicant letter transmitting test procedure on Fisher & Porter instrumentation.
July 28 and 29, 1977	Staff meeting with applicant to review the core protection calculator Phase I test program.
July 29, 1977	Applicant letter concerning applicant's appeal of core protection calculator system Position 20.
August 1, 1977	Applicant letter transmitting proposed Revision 4 to Quality Assurance Manual for Operations report.
August 4, 1977	Staff letter requesting information on environmental qualifications in regard to main steam line break analysis.
August 8 and 9, 1977	Staff meeting with applicant to audit the core protection calculator system Phase II Test and software burn-in test.
August 9, 1977	Applicant letter noting a change in the applicant's designated officer for signature of certain documents.
August 10, 1977	Staff meeting with applicant on draft technical specifications.
August 16, 1977	Applicant's appeal meeting with staff on core protection calculator Position 20.
August 16, 1977	Applicant letter scheduling response to staff's August 4, 1977 letter for September 30, 1977.
August 18, 1977	Applicant letter providing the short-term overpressurization protection program.
August 19, 1977	Staff letter requesting information on instrumentation and control systems (question 222.177) and items identified in staff's July 6, 7 and 8 site visit.
August 24, 1977	Staff letter stating position on mass and energy release calculation for the main steam line break analysis.
August 29, 1977	Staff letter transmitting fire protection program guidance, "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance."

August 30, 1977	Applicant letter transmitting initial Fire Hazards Analysis submittal.
August 30, 1977	Staff letter to Advisory Committee on Reactor Safeguards transmitting letter reports on core protection calculator.
August 31, 1977	Applicant's letter transmitting test report for input fault and surge testing of power supplies.
August 31, 1977	Applicant letter transmitting information on loose parts monitoring system.
September 2, 1977	Applicant letter transmitting information of fuel assembly local loss coefficients.
September 2, 1977	Applicant letter transmitting information on effects of boron precipitation on long-term emergency core cooling.
September 6, 1977	Applicant letter transmitting drawings in support of the fire protection program review.
September 6, 1977	Applicant letter transmitting information on low population zone radius.
September 6, 1977	Applicant letter transmitting preoperational test plan for core protection calculator system.
September 6, 1977	Applicant letter transmitting information on financial qualifications.
September 7, 1977	Staff letter requesting additional information and stating several positions.
September 8, 1977	Staff letter to applicant stating acceptance of Revision 4 to Quality Assurance Topical Report (APL-TPO-1A).
September 13, 1977	Applicant letter transmitting information on the core protection calculator system.
September 14, 1977	Applicant letter regarding Section 6 of technical specifications.
September 14, 1977	Staff letter to ACRS transmitting letter reports.

September 15, 1977	Staff meeting with applicant to discuss outstanding items.
September 16, 1977	Staff letter stating position on core protection calculator system Phase II Test and the software burn-in test.
September 19, 1977	Applicant letter requesting approval of Quality Assurance Topical Report (APL-TOP-1A) for implementation on ANO-2.
September 20, 1977	Staff letter to applicant in response to applicant's appeal of core protection calculator Position 20.
September 21, 1977	Applicant letter transmitting responses to questions on fire protection program.
September 21, 1977	Applicant letter requesting staff to review the reactor protection system and engineered safety feature actuation system as a two-out-of-three logic system.
September 27, 1977	Applicant letter tranmitting documents on the low population zone radius.
September 30, 1977	Applicant letter responding to items identified in the staff's visit to the site on July 6, 7 and 8, 1977.
September 30, 1977	Applicant letter regarding staff approval of the Quality Assurance Topical Report (APL-TOP-1A) for implementation at ANO-2.
September 30, 1977	Applicant letter transmitting emergency core cooling system small break analysis results.
September 30, 1977	Staff letter approving the implementation of Quality Assurance Topical Report (APL-TOP-1A) for ANO-2.
October 5, 1977	Staff letter requesting information on the system qualification final test report for the core protection calculator system.
October 5, 1977	Applicant letter transmitting several test procedures and additional information on the core protection calculator system.
October 7, 1977	Applicant letter transmitting a supplement to the core protection calculator integrated system burn-in test procedure.

October 11, 1977	Applicant letter transmitting supplemental response to question 222.88 on grid frequency decay rates.
October 11, 1977	Applicant letter transmitting the long-term overpressurization protection program.

APPENDIX B

BIBILOGRAPHY FOR
ARKANSAS NUCLEAR ONE - UNIT 2
SAFETY EVALUATION REPORT

NOTE: Documents referenced in or used to prepare this Safety Evaluation Report may be obtained at the source stated in the bibliography or, where no specific source is given, at most major public libraries. Correspondence between the Commission and the applicant (Final Safety Analysis Report, Environmental Report, and application) and Commission Rules and Regulations and Regulatory Guides may be inspected at the Commission's Public Document Room, 1717 H Street, N.W., Washington, DC. Correspondence between the Commission and the applicant may also be inspected at the Arkansas Polytechnic College Library, Russellville, Arkansas. Specific documents relied upon by the Commission's staff and referenced in this Safety Evaluation Report are listed as follows:

METEOROLOGY

1. Gross, E., "The National Air Pollution Potential Forecast Program." ESSA Technical Memorandum WBTM NMC47, National Meteorological Center, Washington, DC, 1970.

2. Holzworth, G. C., "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States," AP-101, Environmental Protection Agency, Office of Air Programs, Research Triangle Park, North Carolina, 1972.

3. Huschke, R. E., "Glossary of Meteorology," American Meteorological Society, Boston, Massachusetts, 1959.

4. Korshover, J., "Climatology of Stagnating Anticyclones East of the Rocky Mountains, 1936-1970." NOAA Technical Memorandum ERL ARL-34, Ari Resources Laboratories, Silver Spring, MD, 1971.

5. List, R. J., (ed), "Smithsonian Meteorological Tables," Smithsonian Institution, Washington, DC, 1971.

6. Marshall, J. L., "Lightning Protection," John Wiley and Sons, Inc., New York, page 190, 1973.

7. Riordan, P., "Extreme 24-hour Snowfalls in the United States: Accumulation, Distribution, and Frequency," Special Report ETL-SR-73-4. U.S. Army Engineer Topographic Laboratories, Fort Belvoir, Virginia, 1973.

8. Sagendorf, J. E., A Program for Evaluating Atmospheric Dispersion from a Nuclear Power Station. NOAA Technical Memorandum ERL ARL-42. Air Resources Laboratory, NOAA, Idaho Falls, Idaho, 1974.

9. SELS Unit Staff, "National Severe Storms Forecast Center, 1969: Severe Local Storm Occurrences, 1955-1967," ESSA Technical Memorandum WBTM FCST-12, Office of Meteorological Operations, Silver Spring, MD.

10. Tattelman, P., and Gringorten, I., "Estimated Glaze Ice and Wind Loads at the Earth's Surface for the Contiguous United States," Air Force Surveys in Geophysics, No. 277, AFCRL-TR-0664, Bedford, Massachusetts, 1973.

11. Thom, H.C.S., "New Distribution of Extreme Winds in the United States," Journal of the Structural Division, Proceedings of the American Society of Civil Engineers - July 1978, pages 1787-1801, 1968.

12. Thom, H.C.S., "Tornado Probabilities," Monthly Weather Review, October-December 1963, pages 730-737, 1963.

13. U.S. Department of Commerce, Environmental Data Services, "Local Climatological Data, Annual Summary with Comparative Data - Little Rock, Arkansas," Published annually.

14. U.S. Department of Commerce, Environmental Data Service, "Climatic Atlas of the United States," Environmental Science Service Administration, Washington, DC, 1968.

15. U.S. Department of Commerce, Environmental Data Service, "Local Climatological Data, Annual Summary with Comparative Data - Fort Smith, Arkansas," Published annually.

16. U.S. Department of Commerce, Environmental Data Service, "Storm Data," Published monthly, Asheville, NC.

17. U.S. Naval Weather Service, "Worldwide Airfield Summeries, Volume VIII, Part 5, United States of America (Mississippi Valley Area)," Federal Clearinghouse for Scientific and Technical Information, Springfield, Virginia, 1969.

DESIGN CRITERIA

18. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute, 1973.

19. ASME Boiler and Pressure Vessel Code, 1968 Edition, Section III, "Nuclear Vessels," American Society of Mechanical Engineers.

20. ASME Boiler and Pressure Vessel Code, 1971 Edition, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

21. ANSI B31.7, 1969 Edition, "Nuclear Power Piping," American National Standards Institute.

22. Draft ASME Code for Pumps and Valves for Nuclear Power, (dated November 1968), American Society of Mechanical Engineers.

23. ASME Boiler and Pressure Vessel Code, 1968 and 1971 Editions, Section VIII, Division 1, "Pressure Vessels," American Society of Mechanical Engineers.

24. ANSI B31.1.0-1967, "Power Piping," American National Standards Institute.

25. API Standard 620, Fourth Edition, February 1970, "Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks," American Petroleum Institute.

26. API Standard 650, Third Edition, July 1966, "Welded Steel Tanks for Oil Storage," American Petroleum Institute.

27. AWWA D100-67, "AWWA Standard for Steel Tanks - Standpipes, Reservoirs, and Elevated Tanks for Water Storage," American Water Works Association.

28. "Wind Forces on Structures," Final Report of the Task Committee on Wind Forces of the Committee on Load and Stresses of the Structural Division, Transactions of the American Society of Civil Engineers, 345 East 47th Street, New York, NY 10017, Paper No. 3269, Vol. 126, Part II, 1961, p. 1124-1198.

29. Gwaltney, R. C., *Missile Generation and Protection in Light Water-Cooled Power Reactor Plants*, ORNL-NSIC-22, September 1968.

30. Williamson, R. A., and Alvy, R. R., "Impact Effect of Fragments Striking Structural Elements," Holmes and Narver, Revised Edition, 1973.

31. American Institute of Steel Construction, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," 101 Park Avenue, New York, NY 10017, Sixth Edition, 1969.

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APPENDIX C

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS-GENERIC MATTERS

The Advisory Committee on Reactor Safeguards (Committee) periodically issues a report listing various generic items applicable to large light-water reactors. These are items which we and the Committee, while finding present plant designs acceptable, believe have the potential of adding to the overall safety margin of nuclear power plants, and as such should be considered for application to the extent reasonable and practicable as solutions are found, recognizing that such solutions may occur after completion of the plant. This is consistent with our continuing efforts toward reducing still further the already small risk to the public health and safety from nuclear power plants. The Committee report concerning these generic items on which this Appendix is based was issued to the Commission on February 24, 1977 in a letter from Committee Chairman M. Bender to Commission Chairman M. Rowden.

The status of staff efforts leading to resolution of all these generic matters is contained in our Status Report on Generic Items periodically transmitted to the Committee. The latest such Status Report is contained in a letter from B. Rusche to M. Bender dated January 31, 1977.

For many of the items identified in the Committee's generic items letter of February 24, 1977, we have provided in this Safety Evaluation Report specific discussions applicable to the ANO-2 plant. For those items applicable to ANO-2 which have not yet progressed to where specific action can be initiated relevant to individual plants, our Status Report on Generic Items referred to above provides the appropriate information.

These items are listed below with the appropriate section numbers of the Safety Evaluation Report where such discussions are to be found. The numbering corresponds to that in the February 24, 1977 report of the Committee.

Group II - Resolution Pending

(1) Turbine Missiles - This item is resolved for the ANO-2 plant by the periodic testing to be performed by Arkansas Power and Light Company (Section 3.5.3 of this report).

(2) Effective Operation of Containment Sprays in a LOCA - This item is resolved for the ANO-2 plant by the use of sodium hydroxide additive to the sprays (Section 6.2. of this report).

(3) Possible Failure of Pressure Vessel Post Loss-of-Coolant Accident by Thermal Shock - This item is under generic review as indicated in our status report to ACRS dated January 31, 1977.

(4) Instruments to Detect (Severe) Fuel Failures - This item is partly resolved, as reported in the February 24, 1977 letter from the Committee to the Commission. Instrumentation to detect fuel failures associated with normal operation and transients (limited fuel failures) has been shown to be adequate. The adequacy of instrumentation to detect failures associated with more rapid events during which substantial fuel failures could occur has not been demonstrated and this concern is considered unresolved. Further work is necessary to determine (a) the adequacy of current instrumentation for these rapid events, and (b) the need for additional instrumentation. Research administered by the Office of Reactor Safety Research and studies conducted under contracts administered by the Office of Nuclear Reactor Regulation should provide the information required to evaluate instrumentation limitations and needs. In the interim, we have not identified any credible event (transient or accident sequence) for which a rapid fuel failure detection system would prevent "substantial" fuel failure (including fuel melt) and loss of coolable geometry.

(5) Monitoring for Excessive Vibration or Loose Parts Inside the Pressure Vessel - This item is resolved for ANO-2 plant as discussed in the sections of this report (Sections 4.2.2 and 5.6 of this report).

(6) Non-Random Multiple Failures - This item is under generic review as indicated in our Status Report to ACRS dated January 31, 1977 (also Section 15.5 of this report)

(7) Behavior of Reactor Fuel Under Abnormal Conditions - This item is under generic review as indicated in our Status Report to ACRS dated January 31, 1977.

(8) Boiling Water Reactor Recirculation Pump Overspeed During Loss-of-Coolant Accident - not applicable to the ANO-2 plant design.

(9) The Advisability of Seismic Scram - A seismic scram is not proposed for the ANO-2 plant and we will not require such a scram - see letter dated May 19, 1977, from E. Case, Acting Director, Office of Nuclear Reactor Regulation, to Committee Chairman Bender; subject, "The Advisability of a Seismic Scram."

(10) Emergency Core Cooling System Capability for Future Plants - This item is under generic review as indicated in our status report to ACRS dated January 31, 1977.

Group II A - Resolution Pending - Items since December 18, 1972

(1) Control Rod Drop Accident (Boiling Water Reactors) - not applicable to the ANO-2 plant design.

(2) Ice Condenser Containments - not applicable to the ANO-2 plant design.

(3) Rupture of High-Pressure Lines Outside Containment - This item is resolved for the ANO-2 plant by Arkansas Power and Light Company by compliance with staff requirements (Section 3.6.2 of this report).

(4) Pressurized Water Reactor Pump Overspeed During a Loss-of-Coolant Accident - This item is resolved for the ANO-2 plant by the applicant by compliance with current staff requirements (Section 5.6.1 of this report).

(5) Isolation of Low Pressure from High Pressure Systems - This issue is resolved for the ANO-2 plant by Arkansas Power and Light Company by compliance with staff requirements (Sections 5.6.3 and 7.6.2 of this report).

(6) Steam Generator Tube Failures - This item is resolved for the ANO-2 plant by the measures Arkansas Power and Light Company will take to control secondary water chemistry (Section 5.6.2 of this report).

(7) ACRS/NRC Periodic 10-Year Review of all Power Reactors - This item is under generic review as indicated in our status report to ACRS dated January 31, 1977.

Group II B - Resolution Pending - Items Added Since February 13, 1974

(1) Computer Reactor Protection System - This item is under review for the ANO-2 plant as discussed in Section 7.2.3, Table 7.1 and Appendix D to this report.

(2) Qualification of New Fuel Geometries - This item is not totally resolved for the ANO-2 plant. Our evaluation of the Combustion Engineering 16x16 verification program is complete. However, determination of the rod bow penalty and the fuel surveillance program have not been completed (Section 4.0 of this report).

(3) Behavior of Boiling Water Reactor Mark III Containments - not applicable to the ANO-2 plant design.

(4) Stress Corrosion Cracking in Boiling Water Reactor Piping - not applicable to the ANO-2 plant design.

Group II C - Resolution Pending - Items Added Since March 12, 1975

(1) Locking Out of Emergency Core Cooling System Power-Operated Valves - This item is resolved for the ANO-2 plant by the Arkansas Power and Light Company commitment to lock out power to appropriate valves (Section 7.6.1 of this report).

(2) Design Features to Control Sabotage - On February 24, 1977 the Commission published new requirements for the physical protection of nuclear power plants against acts of sabotage (10 CFR 73.55). This item will be resolved for the ANO-2 plant by the compliance of Arkansas Power and Light Company with the new regulations (Section 13.6 of this report).

(3) Decontamination and Decommissioning of Reactors - This item is under generic review as indicated in our status report to ACRS dated January 31, 1977.

(4) Vessel Support Structures - This item is resolved for the ANO-2 operating license by Arkansas Power and Light Company commitment to provide analyses, including asymmetric forces, of the design of the vessel supports (Section 3.9.3 of this report).

(5) Water Hammer - This item is under generic review as indicated in our status report to ACRS dated January 31, 1977 and as indicated in Section 10.5 of this report.

(6) Maintenance and Inspection of Plants - This item is resolved for the ANO-2 plant by compliance with Commission staff requirements applicable to operating license applications (Section 11.0 of this report.)

(7) Behavior of Boiling Water Reactor Mark I Containments - not applicable to the ANO-2 plant design.

Group II D - Resolution Pending - Items Added Since April 16, 1976

(1) Safety-Related Interfaces Between Reactor Island and Balance-of-Plant - This item pertains to standard plants. Accordingly, it does not apply to ANO-2 which is not a standard plant.

(2) Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electric Equipment - This item is not addressed in this report except as a general requirement for environmental qualification of equipment (Sections 3.11 and 6.2 of this report).

APPENDIX D

to

ANO-2

Safety Evaluation Report

on

Core Protection Calculator System

Appendix D

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APPENDIX D

CORE PROTECTION CALCULATOR SYSTEM

D.1 General

Further details of the staff's evaluation of the core protection calculator system (CPCS) are presented in the following sections. In Section D.2, the design bases for the system are reviewed and evaluated. In Section D.3, an evaluation of the protection algorithms is presented. An evaluation of system hardware, stored computer program, and qualification testing is presented in Section D.4.

In Section D.5, a review of the uncertainties of the minimum departure from nucleate boiling ratio (DNBR) is presented. A staff calculation is presented in Section D.6 and references are provided in the bibliography.

D.2 Design Basis

D.2.1 Discussion

The design bases for CPCS cores require that sufficient thermal margin be maintained under conditions of normal operation to preclude the violation of specified fuel design limits in event of any anticipated operational occurrence, i.e., any operational transient event that is expected to occur one or more times during the life of the plant. When considering such events, initial process conditions are assumed to be within the limits designated in the plant technical specifications. Safety analyses must demonstrate that anticipated transients initiated within the process limits at any time during the core life will not violate the minimum DNBR limit or the high local power density limit which could result in damage to some fuel.

The reactor operator is responsible for maintaining the steady state DNBR at or above a predetermined axial shape dependent DNBR operating limit curve. This provides sufficient DNBR margin to prevent a DNBR degradation to a value of less than 1.3 during the course of any of the anticipated operational occurrences. Similarly, the peak local power density during normal operation will be maintained at a limiting value based on initial conditions assumed for the postulated loss-of-coolant accident analyses. Also, the operator must periodically monitor the azimuthal tilt magnitude and the thermal power level computed from a secondary calorimetric balance. Addressable constants in the CPC data base are modified to account for the effect of tilt and power level variation on local power densities. The value of the constants will affect the calculated value of the trip variable for both trip functions.

The software system (Core Operating Limit Supervisory System, COLSS) is provided in the plant monitoring computer to assist the operator in maintaining normal operation within the process limits assumed for the CPCS protective functions. Technical specifications will require more conservative administrative controls when COLSS is not in operation.

Limited fuel damage is not considered in itself a significant safety concern as evidenced by the fact that reactors are permitted to operate with failed fuel provided that activity release to the environment, which is monitored and controlled, does not exceed the safety limits imposed by technical specifications. Such fuel failures arise from cladding material defects and other causes independent of the postulated events which are analyzed to determine the design requirements of the plant protection systems.

The staff is considering failure of the digital trip system to perform its design function. While our review is incomplete and awaiting submittal of some requested analyses from the applicant, it appears at this stage of our review that backup analog trips and/or inherent shutdown mechanisms would limit the consequences of this type of a failure to prevent undue risk to the public health and safety.

D2.2 Design Basis Events

A number of plant transients can be expected to occur one or more times during the life of the plant as a result of single electrical or mechanical malfunctions or due to single operator errors. Such transients could result in violation of specified acceptable fuel design limits if protective action were not initiated.

Anticipated operational occurrences that were used to determine the design requirements for the CPCS trips are:

(1) Insertion or withdrawal of full-length or part-length control element assembly (CEA) groups.

(2) Insertion or withdrawal of full-length CEA subgroups.

(3) Insertion or withdrawal of a single full-length or part-length CEA.

(4) Uncontrolled dilution of the soluble boron concentration, when CEA's are withdrawn.

(5) Uncontrolled axial xenon oscillations.

(6) Change of forced reactor coolant flow including simultaneous loss of electrical power to all reactor coolant pumps at 100 percent power.

(7) Inadvertent depressurization of the reactor coolant system, including actuation of full spray flow without proper performance of any pressurizer heaters.

(8) Decrease in heat transfer capability between the secondary and reactor coolant systems.

(9) Complete loss of alternating current power to the station auxiliaries.

(10) Excess heat removal due to secondary system malfunctions.

Postulated accidents that were evaluated taking credit for the CPCS trips are:

(1) Reactor coolant pump shaft seizure.

(2) Steam generator tube rupture.

In addition, the CPCS is expected to provide the initial trip during certain steam line break accidents with a concurrent loss of alternating current power (which causes a four-pump loss of flow transient).

The criteria used to define core safety limits and limiting safety system settings for protection against fuel damage during anticipated operational occurrences are:

(1) The DNB ratio using the W-3 DNBR correlation shall not be less than 1.3 in the limiting coolant channel in the core.

(2) The peak linear heat rate in the limiting fuel pin shall not be greater than that value corresponding to the centerline fuel melting temperature.

The staff has completed review of the CPCS system design bases as set forth in CEN-44 (A), "Core Protection Calculator Functional Description," January 7, 1977 and finds them acceptable. Our review also included independent calculations by the staff for the four pump loss of flow transients (see Section D.6).

D.3 Protection Algorithms

D.3.1 Axial Power Distribution Synthesis

To synthesize the core average axial power distribution, each CPC utilizes measured data from control rod position sensors, core inlet coolant temperature sensors, and a single tri-level excore detector. These data are employed in digital algorithms developed from reactor simulator calculations which relate excore detector segment responses to the core average axial power shape as a function of control rod positions, reactor coolant temperature and fuel burnup.

The excore detectors (four detectors per reactor, three segments per detector) are located in the air gap between the reactor vessel and the biological shield. In this position, the detector segments are sensitive primarily to the fast neutrons produced in those peripheral fuel assemblies seen by the detector. For an unrodded core the peripheral axial power shape and the core average axial power shape are nearly identical. However, control rod insertion can shift the radial power distribution in the rodded region to produce peripheral axial power shapes significantly distorted from the core average axial distribution. To account for this effect each CPC applies to each detector signal a rod shadowing factor which depends on the number and extent of rod banks inserted. An additional factor is applied to each detector signal to account for the effect of reactor downcomer water density on the neutron attenuation between the core periphery and the air gap.

To construct the core average axial power distribution, the excore detector signals modified for rod and temperature shadowing are transformed into pseudo-signals representing responses which would occur if the detectors were immersed in the core average axial power shape. This is accomplished through shape annealing matrices which relate the integrals of the core axial power distribution over each third of the core to the response in each of the excore detector levels. Since the power produced in each third of the core will produce a response in each of the excore detector levels a 3 x 3 matrix of shape annealing coefficients is required to unfold the three detector responses into three core average axial power integrals. These three integral quantities are then used with four point-parameters, i.e., the core boundary power and the extrapolated zero power point for both the core top and bottom to provide input to a seven point spline-curve fitting algorithm. This analytical function is then used to determine the core average axial power distribution at twenty axial locations for use in DNBR and LPD algorithms.

The factors employed to modify the raw excore detector signals, i.e., rod shadowing, temperature correction, shape annealing matrix (3 X 3), and boundary point powers are pre-calculated values obtained from reactor simulators. These factors will be measured during the start-up test program. In addition, comparisons between the CPC axial power shape and the shape determined by incore detectors will be made to verify the CPC procedure.

The calculational procedures employed to construct the core average axial power shape from the excore detector signals has been reviewed by the staff and found acceptable. However, to ensure that the start-up tests are adequate to verify the accuracy of pre-calculated constants and the overall CPC algorithms, a detailed start-up test plan is required for staff review at least two months prior to the commencement of the testing.

D.3.2 Radial Power Distribution Synthesis

The CPC's utilize pre-calculated information to determine radial peaking factors from measured CEA position. Extensive two and three dimensional power distribution calculations have been performed to evaluate radial peaking factors as a function of all normal operating conditions. Maximum values are selected that bound all of the expected values for each rodded configuration allowed during normal operation. These maximum values are stored in tabular form in the data base of each CPC.

Selection of appropriate peaking factors is accomplished within the CPC's by knowledge of the position of CEA's inserted in the core as obtained from the reed switch position indicating system. The shutdown and regulating CEA's are divided into subgroups where a subgroup consists of four symmetrically located CEA's. (An exception is the subgroup that contains the center rod and thus consists of five rods.) There are twenty subgroups of CEA's in ANO-2. Each of the rods in a subgroup serves as a target rod for one of the four CPC's. The position of this target rod is input to the CPC and represents the position of that subgroup. These twenty subgroup positions are used by the CPC to establish the CEA bank configuration in each of the twenty axial intervals of interest. Determination of the radial peaking factors at any given axial level is accomplished by selecting the appropriate peaking factor from the table of precalculated values stored in the CPC.

The radial peaking factors are valid only if all rods are inserted in proper sequence and are properly positioned. The CPC's also assess the subgroup position information to determine if (1) regulating groups are out of sequence, or (2) there is a deviation in the subgroups from the group average position. The CPC's apply precalculated penalty factors to the radial peaking factors for the axial nodes affected by either of these off-normal conditions.

Deviation in the position of a single control element assembly (CEA) is evaluated by the CEA Calculators (CEAC's). The CEAC's have as input the position of all CEA's and evaluate this information to establish possible deviation of individual CEA's within each CEA subgroup. The CEAC's also utilize precalculated penalty factors to establish the peaking factor change attributable to single CEA deviations. These penalty factors are input to the CPC's where they are utilized to further modify the radial peaking factors for all axial nodes.

The utilization of safety grade rod position information in core protection alogrithms is an improvement over previous designs since CEA misalignment and out-of-sequence bank configuration are automatically detected. Previous reactor designs utilize an indirect method, the azimuthal tilt function, to indicate non-normal CEA configurations which in turn require operator action, i.e., no allowance is made in the safety setpoint determination for undetected rod misalignment. The CPCS system, however, utilizes the direct CEA position information system directly and automatically incorporates the effect of off-normal CEA configurations in the core protection algorithms.

The calculational procedures used to determine radial peaking factors and single rod deviation penalty factors are similar to the techniques approved by the staff for previous Combustion Engineering reactors and are thus acceptable at this stage of the review. However, radial peaking factors and CEAC penalty factors are to be verified by analyses of incore detector data during the start-up test program. To ensure that the start-up tests are adequate to verify these factors, a start-up test plan has been required of the applicant. To insure adequate review time, the staff will require the test plan at least two months prior to test initiation.

D.3.3 Parameters Used in Margin Calculations

The calculations of the core average axial and planar radial power distributions are utilized to determine parameters contained in the algorithms employed to determine the maximum linear power density and minimum Departure from Nucleate Boiling Ratio (DNBR). A pseudo-hot pin power distribution, defined as the axial distribution of the maximum planar linear heat rate, is calculated by combining the core average power distribution with the radial peaking factors at each axial node (including penalty factors for improperly positioned control rods) and further modifying this product by a constant azimuthal tilt penalty factor to account for unanticipated power maldistributions that might exist. From this distribution, the hot channel and core average Axial Shape Indices (ASI is the lower half power minus the upper half power divided by total power), the axially integrated hot pin heat flux, and the integrated single pin radial peak-to-average power are calculated for use in DNBR algorithms. Further, tabulated fuel desification factors are combined with hot pin axial powers for use in the maximum linear power density calculation.

The above power distribution parameters are calculated at time intervals which are sufficiently short to ensure that DNBR and linear power density limits are not exceeded prior to the actual termination of a design basis transient.

To accommodate delays in sensor lag, CPC calculational sequencing, reactor protection system response time and local pin transient heat flux deviations from the core average heat flux, digital filters are employed for both the pseudo hot pin heat flux and linear power density claculations. Two sets of filter constants are employed for transiently increasing and decreasing core power. This is done to ensure that the CPC calculated fuel pin temperature and surface heat flux conservatively lead the actual values for increasing power and lag the actual values for decreasing power.

The filter constants are determined from thermal/hydraulic calculational models which utilize initial reactor core conditions that are possible under the limiting conditions of operation (LCO's) to initiate the limiting transients and accidents. The power distribution LCO's are monitored by the COLSS system described in CENPD-169 which is currently under review.

The verification of the adequacy of the CPC timing intervals and lead-lag filter constants is based on calculational models. Representative cases were submitted to the staff for review and the results of the cases submitted indicate that the CPC calculational timing and digital filter algorithms are acceptable.

The staff with assistance from its consultants at Brookhaven National Laboratory has completed a review of the equations used in the synthesis process (as presented in CENPD-170). We find the methods employed to establish the core average axial power distribution from excore detector signals to be acceptable; the radial power distribution determination is similar to methods approved by the staff for previous reactors.

The specific application of the power distribution algorithms to the dropped off-center CEA transient has been addressed by the applicant. A review of the analysis by the staff had led to the conclusion that the CPC methodology is adequate for conservatively predicting three dimensional power distributions during this transient, thus satisfying the concerns of staff Position 2 (see Table 7.1).

In addition, the staff has reviewed and found acceptable the test procedures of the physics algorithms for Phase I qualification of the software in which the implementation of the separate physics algorithms into CPC software was accomplished. The adequacy of the functional behavior of the combined algorithms was to be demonstrated in Phase II testing. However, the Phase II test results were unacceptable to the staff (see staff position 24, Table 7.1). The applicant has committed to re-test and the adequacy of the functional behavior of the algorithms and our evaluation of the results will be presented in a supplement to this report.

With respect to changes in the algorithm constants the staff position is that changes made in nonaddressable constants must be reported to and evaluated by the staff in the same manner that significant reactor parameters and algorithm modifications are reported in reload applications. In philosophy, changing the value of an algorithm constant is equivalent to effecting a change in the power distribution algorithm and thus has a safety significance equivalent to that of a change in Technical Specifications.

For changes in all addressable constants the staff will require that the details of the test procedure employed to verify modified constants be furnished to the staff for approval and that records of constant changes be available for periodic staff review.

The COLSS and INCA systems will be utilized to verify CPC power distribution algorithms and radial peaking factors, respectively, at start-up and periodically throughout reactor operation. In addition, COLSS will be used to monitor LCO's during reactor operation which are necessary to assure the CPC response is adequate. The methodology and uncertainty analysis for both COLSS and INCA is presently under review.

D.3.4 Determination of Corrected Neutron Flux Power.

The sum of the signals from the detector segments in each excore detector string provides the primary indication of reactor power utilized by each of the CPC's. The relationship of the excore detector responses to core power is affected by the presence of CEA's (CEA shadowing) and excore detector shape annealing. Thus corrections must be applied to the excore detector readings to correct for these effects. This process is identical to that presented in the discussion of the axial power distribution synthesis technique (see Section D.3.1). Following these corrections a calibration factor is applied to the sum of the three segment detector responses to correct this sum to reactor power. This calibration factor is determined periodically by ratioing the total detector response to the steady state thermal power obtained from a heat balance on the secondary system and is input to the CPC's by the reactor operator.

The procedure used by the CPC's to obtain core power from excore detector signals is similar to that approved by the staff for non-CPC plants and has been found acceptable for ANO-2.

D.3.5 Power Distribution Uncertainty

The calculational uncertainties associated with the CPC synthesized local power density have been addressed in CENPD-170 and CENPD-145. The techniques used involved the generation of a large number (in excess of 4200) of power distributions with the three-dimensional core simulators, ROCS and FLARE. The calculations simulated excore detector responses for a variety of static and transient core power distributions typical of first cycle operation of the Arkansas Nuclear One-Unit 2 reactor. Abnormal CEA configurations in which individual CEA's and CEA banks were mispositioned were also considered. The simulated detector signals, together with the CEA positions assumed in generating the power distributions of interest, were then processed by a FORTRAN version of the CPC algorithms to produce, for each case, a value of the maximum peaking factor. The algorithm constants employed by the CPC's, including rod shadowing, calculated by the simulators from the core conditions employed in the simulators. The CPC synthesized peaking factor was then compared with the value produced by the simulator.

The errors between the synthesized and simulator peaking factors were evaluated for all cases involving normal CEA positions.

A maximum uncertainty factor of approximately 1.085 was realized at End-of-Cycle. The uncertainty factor is the factor that must be applied to the CPC synthesized peaking factor to ensure that 95 percent of the true peaking factors are no larger than the calculated values at the 95 percent confidence level.

As noted above, the radial peaking factors used in the analysis corresponded to the calculated values. Thus the impact of the uncertainty on the true radial peaking factor (vs. the calculated value) is not included in the above factors. During start-up, the radial peaking factors used in the CPC's will be verified using incore detector responses processed with the INCA code. Hence, the radial peaking factors will contain uncertainties associated with the INCA procedure reported in CENPD-145. These uncertainties are combined statistically with the errors reported above to arrive at an uncertainty factor of about 1.10 for the CPC processed peaking factors. The precise value to be applied during operation will be based on results of the start-up tests and simulator calculations which accurately model the ANO-2 reactor.

The assessment of the peaking factor accuracy for those cases involving CEA misalignment has been handled separately. In lieu of a statistical uncertainty factor, a sufficiently conservative penalty factor is employed to ensure the total peaking factors produced by the CPC's during misalignment events will always be conservative relative to the actual values. Approximately 600 cases involving CEA misalignment were evaluated to demonstrate the conservatism of the penalty factor by the vendor.

Of those cases presented for review, the CPC's consistently overestimated the total peaking factor relative to the value calculated by the simulators during the time the CEA's are misaligned. Following alignment, some azimuthal xenon oscillations occur that are not accommodated by the CPC's, and, during this time, the CPC peaking factor estimate exclusive of uncertainty factors may be nonconservative. However, this effect is on the order of a few percent for the worst cases and is less than the tilt allowance plus the CPC uncertainty previously described. In addition, those cases for which the CPC calculation was slightly nonconservative can only occur for rod configurations associated with power levels well below the licensed value. Therefore, conditions involving non-normal CEA configurations can be accommodated without further increasing the peaking factor error to be incorporated into the CPC's.

The general method for determining the value of the uncertainty and penalty factors described above has been found acceptable. However, the specific values for the uncertainty and penalty factors depend on simulator constants, e.g., rod and temperature shadowing factors, the shape annealing matrix, and boundary point powers, which are to be either calculated or verified during start-up and pre-operational tests. Although CE has submitted a general start-up test plan, the staff requires a more detailed description of the program to assure that the CPC constants are verified in a manner consistent with the assumptions and applications in the CPC simulator uncertainty analysis. The test plan should include specific reactor configurations to be used in verifying each algorithm constant together with acceptance criteria for the difference between calculated and measured values with alternatives proposed if acceptance criteria are exceeded. In addition, the number and type of cases to be employed to verify uncertainty and penalty factors should be specified. The tests performed should be sufficient to demonstrate the conservatism in the approximations to theory required to implement algorithm constants, in particular shape annealing and rod shadowing factors. The inclusion of the above in the start-up test program will satisfy the requirements of staff Position 1, given in Table 7.1. We will address the resolution of this item upon reviewing the start-up test plan and will report on its evaluation in a supplement to this report.

D.3.6 Uncertainty Assessment

The topical report CENPD-170 describes the methods used in the Combustion Engineering (CE) Core Protection Calculator System (CPCS) to synthesize the three-dimensional peaking factor (Fq). The resulting Fq is used in conjunction with other measured parameters to determine a minimum departure from nucleate boiling ratio (DNBR). The Supplement 1 to the above subject report provides additional discussion of the uncertainties associated with the synthesis of the minimum DNBR. Some of the information in CENPD-170 and its supplement has been superseded by information in the functional description and is under review. Related material has been provided by the response to a series of questions asked by the staff on Arkansas Nuclear One, Unit 2 docket. The evaluation of the topical report was an integral part in the CPCS safety review but its usefulness has been diminished by its replacement with the functional description. A detailed assessment of Supplement 1, CENPD-170 is presented in Section D.5.

The core protection calculators compute thermal hydraulic conditions in the hot channel using a snapshot of both directly monitored and calculated input values. The Combustion Engineering standard design code for computing DNBR is COSMO, an open hot channel thermal margin code. The CPC's use a simplified closed channel fast running version of this code, CPCTH. Since the CPCTH code is derived from and justified with respect to the design code COSMO, review and approval of the topical report on TORC and COSMO is a contingency for acceptance of CPCTH. That report has been reviewed and final approval is contingent upon satisfactory comparison of TORC with data from an operating reactor. However, the staff has concluded that the codes may be used in licensing applications prior to review of the operating reactor date.

Analytical tools and procedures for synthesis of the static departure from nucleate boiling ratio are still under review.

D.3.7 Primary Coolant Mass Flow Algorithm

The primary coolant mass flow algorithm computes a normalized flow rate in each leg of the primary coolant system and in the reactor core. It also computes a projected value of DNBR based on the time derivative of core flow rate. The normalized volumetric flow rates are computed from the speeds of the four reactor coolant pumps and the specific volume of the primary coolant. The flow rate algorithm also outputs the number of pumps running at a specified minimum speed for the purpose of selecting pump-dependent constants and for initiating trip if less than two pumps are running.

The flow algorithm contains a representation of the flow resistance and flow inertia for each flow path in the reactor coolant system. Each pump is represented by a steady state set of head-flow-speed curves. The algorithm solves simultaneous differential equations expressing conservation of momentum around closed flow loops through each pump to obtain core mass flow rate.

Flow resistance coefficients are determined during preoperatioal testing and are entered as addressable constants. When flow reverses in a reactor flow path, the flow resistance is computed using reverse flow resistance coefficients in a reverse flow differential pressure correlation. Preoperational testing must be designed to measure both forward and reverse flow characteristics.

The proper representation of system thermal hydraulic characteristics (loss coefficients) in the CPC's and the functional adequacy of the monitoring technique are items of staff concern which must be verified based by preoperational testing results. Our evaluation of this issue will be presented in a supplement to this report.

D.3.8 Minimum DNBR Algorithm

The static DNBR and power density program computes the static value of DNBR, hot channel quality, primary thermal power, and maximum hot leg enthalpy. The program output establishes the baseline conditions for the DNBR update.

The calculational methods and the uncertainty associated with the DNBR algorithm are discussed in the topical report CENPD-170 (Supplement 1). The staff has reviewed that report and found the methods and procedures described therein acceptable for synthesis of the static departure from nucleate boiling ratio. The applicant has since submitted CEN-44(A), which is a detailed functional description of the final design software. That document remains under review and the staff has outstanding questions. Design modifications are extensive enough, that the total applicability of the CENPD-170 (Supplement 1) review remains in question. The total acceptability of the CPCS software for static DNBR calculations cannot be judged until our evaluation of all documentation and test results is complete.

A partial derivative update is performed to provide estimates of thermal margin changes between static calculations based on changes in the core average mass flow rate, core inlet temperature, primary system pressure, power distribution and core power level. The partial derivatives were determined based on COSMO thermal margin calculations over specified ranges. The most adverse calculated partial derivative for each process variable was defined as the limiting partial for the applicable range. The effect of the change is converted into equivalent overpower margin units and the result is then converted to DNBR and quality units. The change in DNBR (ΔDNBR [t]) is added to the current static DNBR and input directly to the CPC trip decision logic for comparison to the 1.3 limit. In addition, DNBR (t) is compared to a modified DNBR Operating Limit value and the appropriate value is selected for input to the CPC mass flow rate projection logic, which provides protection against loss of flow transients.

The CPC mass flow rate projection determines the time rate of change of mass flow based on the current and previous monitored values. This is multiplied by an axial shape dependent partial derivative of DNBR with respect to flow rate times the projected time required to terminate the DNBR decrease by reactor trip. The resultant DNBR for the projected flow rate is added to the modified DNBR operating limit or to the CPC current calculated value, whichever is greater. This yields the flow projected DNBR which is input to the CPC trip program for comparisons to the 1.3 limit value.

A third value of DNBR is calculated based on a projected depressurization transient at a rate determined from the monitored values of pressure. Thus, three values of DNBR are calculated for comparison to the limit:

(1) The static or updated DNBR based on the most recent scan of input signals,

(2) The projected value of DNBR to the termination of a decreasing flow transient, and

(3) The projected value of DNBR to the termination of a depressurization transient.

A DNBR value of less than 1.3 for any of the above will result in a DNBR trip signal.

Subject to our continuing review of CEN-44(A) and other documentation to be submitted, including test results, there are no outstanding concerns with the design of the DNBR algorithms.

D.3.9 <u>Core Thermal Power Algorithm</u>

The primary coolant system consists of the two steam generator loops, with each of the loops having one hot leg and two cold legs for coolant transport. Reactor inlet temperature is measured in one cold leg of each steam generator loop for each separate CPC channel: each hot leg is monitored for each CPC.

The delta temperature-power is calculated in two portions, a static calculation and a dynamic calculation. The static calculation correlates variations in reactor power to variation in the measured temperature rise across the core and the system mass flow rate. During four pump operation, the average of the two sensed cold leg temperatures and the average of the two hot leg temperatures are used to compute the core enthalpy rise. For the condition in which less than four pumps are running, the maximum cold leg temperature is used to compute the core inlet enthalpy. The product of core mass flow rate and the enthalpy rise monitored for each CPC is multiplied by a thermal power constant to obtain the reactor power. The value of the constant is selected for each CPC to make the computed power equal to the power periodically determined by the plant secondary calorimetric calculation which is also checked with other available power level indicators.

The dynamic power calculation adds a correction term to the static power calculation to account for delays due to fluid transport times between sensors and the core, plenum mixing time and temperature sensor time constants. In the execution of the protection algorithms, the highest value of delta temperature-power versus neutron flux power is selected. The determination of neutron flux power is discussed in Section D.3.4.

The power calibration is dependent on a thermal power constant equating the secondary calorimetric power calculated by COLSS to the primary coolant flow and temperatures sensed by each individual CPC and on power constants obtained from other plants. The technique has an inherent assumption of no temperature or flow maldistribution between the four primary coolant cold legs. The accuracy of the power calculation bears directly on the uncertainty of CPC trip levels. The staff is awaiting additional information to evaluate the overall uncertainty and will require quantification and justification for uncertainty values on power calculations.

D.3.10 Computer Program Structure and Sensor Inputs

The core protection software consists of four interdependent programs and a trip program module that is accessible to all four programs:

(1) Coolant Mass Flow Program (FLOW)

(2) DNBR and Power Density Update Program (UPDATE)

(3) Power Distribution Program (POWER)

(4) Static DNBR and Power Density Program (STATIC)

(5) Trip Sequence Program (TRIPSEQ)

Sampling of the input signals is initiated within the protection programs is consistent with the frequency of execution of the calling program. Program execution intervals are specified in Table 3-3 of CEN-44(A), "Core Protection Calculator Functional Description," January 7, 1977. Each program maintains an output buffer which is revised after each execution.

The following is a listing of the CPC process input signals to each program and indicates the required accuracy for each input.

Program	Inputs Sampled (No.)	Accuracy Required
FLOW	Reactor coolant pump shaft speed (4)	b
UPDATE	Cold leg temperature (2)	$\pm$1 degree Fahrenheit
	Hot leg temperature (2)	$\pm$1 degree Fahrenheit
	Pressure (1)	$\pm$6 pounds per square inch absolute
	Excore neutron flux (3)[a]	$\pm$.005 fraction of rated power
	CEA deviation penalty factors (2)	b
POWER	CEA positions (20)	b
STATIC	None	

NOTES: [a]An excore detector channel is comprised of three axially stacked excore detector subchannels. These subchannels are scaled such that the sum of three is approximately equal to 100 percent at 100 percent rated power.

[b]Additional accuracy requirements are specified in Table 3-1 of CEN-44(A), "Core Protection Calculator Functional Description," January 7, 1977.

Reactor coolant pump shaft speed and the CEA deviation penalty factors are received as digital signals; all other are analog.

The accuracy requirements establish the maximum allowable uncertainty introduced by the conversion of input signals to internal binary format. This inlcudes loading effects, reference voltage regulations, electrical noise, linearity, analog to digital converter power supply sensitivity and quantization.

All inputs to the CPC from sensors are checked to determine if the readings are out of range. The sensor limits are fixed constants stored in protected memory and are not subject to operator modification.

Sensor alarms to indicate out of range readings are annunciated at the station annunciators on the main control board and also on the CPC operator's module. The alarm is automatically reset if the sensor returns within range. However, sensor status words are set to indicate that the sensor has been out of range and hardcopy documentation of the failure is generated by the periodic test program.

When sensors fail or are out of range of the analyzed parameter space which is valid for the user algorithm, the parameter is set to its limit value for algorithm calculations. Operating condition trip signals are generated by out-of-range signals on specified parameters and on detected interval processor faults.

The staff has revived the program structure for input signal processing of the core protection software and concludes that the design is acceptable. However, the system is dependent on the proper specification of process parameter limit values to assure that calculations are performed in valid operating space for the algorithms used. The applicant is committed to provide values of all constants for staff review. Our review of the implementation of the program structure and input signal processing features is incomplete. The total acceptability will be addressed after our review of the constant values to be specified and after review of Phase II test results for the final design software.

D.3.11 Addressable Constants

The CPCS dependence on interaction of the plant operator with the protection computer data base and the operator reliance on obtaining values of data base constants from the COLSS software is of concern to the staff. The applicant has addressed these concerns by modifications to the frozen software to revise the design of the data base constants and to place range limits on the values which can be entered by the operator. Table 3-4 of CEN-44(A) shows range limits placed on the addressable constants and indicates that a validity check must be implemented to reject values outside of the indicated range for each constant. The design appears to be consistent with staff position 15 (see Table 7.1). However, the acceptability of the design implementation remains under review, pending evaluation of the design basis for the range limit values and satisfactory demonstration of the implemented design during performance qualification testing. The revisions reduce the susceptibility of the system to operator error.

The operators module (see D.4.1.5) can be used by the operator to verify the correctness of addressable constants at any time. During the safety evaluation, consideration was given to the lack of hard-copy printout of data entered into the computer memory. Staff concerns were expressed in position 15 and acceptably addressed in a design modification to limit test the value of constants before altering the original memory value. The operators log is considered acceptable hard-copy documentation and a deviation from current operating practices is not needed.

Our review of the CPCS and interfaces with the Plant Computer System (PCS) identified concerns regarding the Core Operating Limits Supervisory System (COLSS) program which runs in the PCS. Specifically, our concerns involved the use of the COLSS program to calculate calibration constants for input by the operator into the CPCS computer programs. Item 7 in Table D-1 documents our evaluation of the potential impact of COLSS and the PCS on the CPCS operation and plant safety. The need for specific policy and criteria for conducting the review of COLSS and the PCS is also noted. In response, Item 8 in Table D-1 defines the scope and bases for the limited review of COLSS and the acceptability of the PCS as a non-safety system.

The staff believes that appropriate administrative controls can be applied to provide an acceptable interface between the plant operator and the CPCS. Appropriate attention will be given to the consideration in our continuing review of the software change implementation and in preparation of the plant technical specification.

D.4 Design and Qualification

D.4.1 Hardware Design

D.4.1.1 General Description

The primary function of the core protection calculator system (CPCS) is to monitor the operational status of the reactor core and provide a trip input to the reactor trip system (RTS) whenever the departure from nucleate boiling ratio (DNBR) or local power density (LPD) reaches a calculated setpoint. The CPCS consists of four redundant digital computers, identified as core protection calculators. They will acquire data from plant process sensors and from two redundant computer-based control element assembly calculators (CEAC's) which provide each CPC with control rod position deviation information. Each CPC provides trip input to one of four redundant and independent reactor trip system (RTS) channels when the calculated variables exceed the trip setpoints contained in the memory of the CPC's. The hardware configuration block diagram of Figure D-1 depicts functionally the main elements of the CPCS and shows its interconnections with the plant computer.

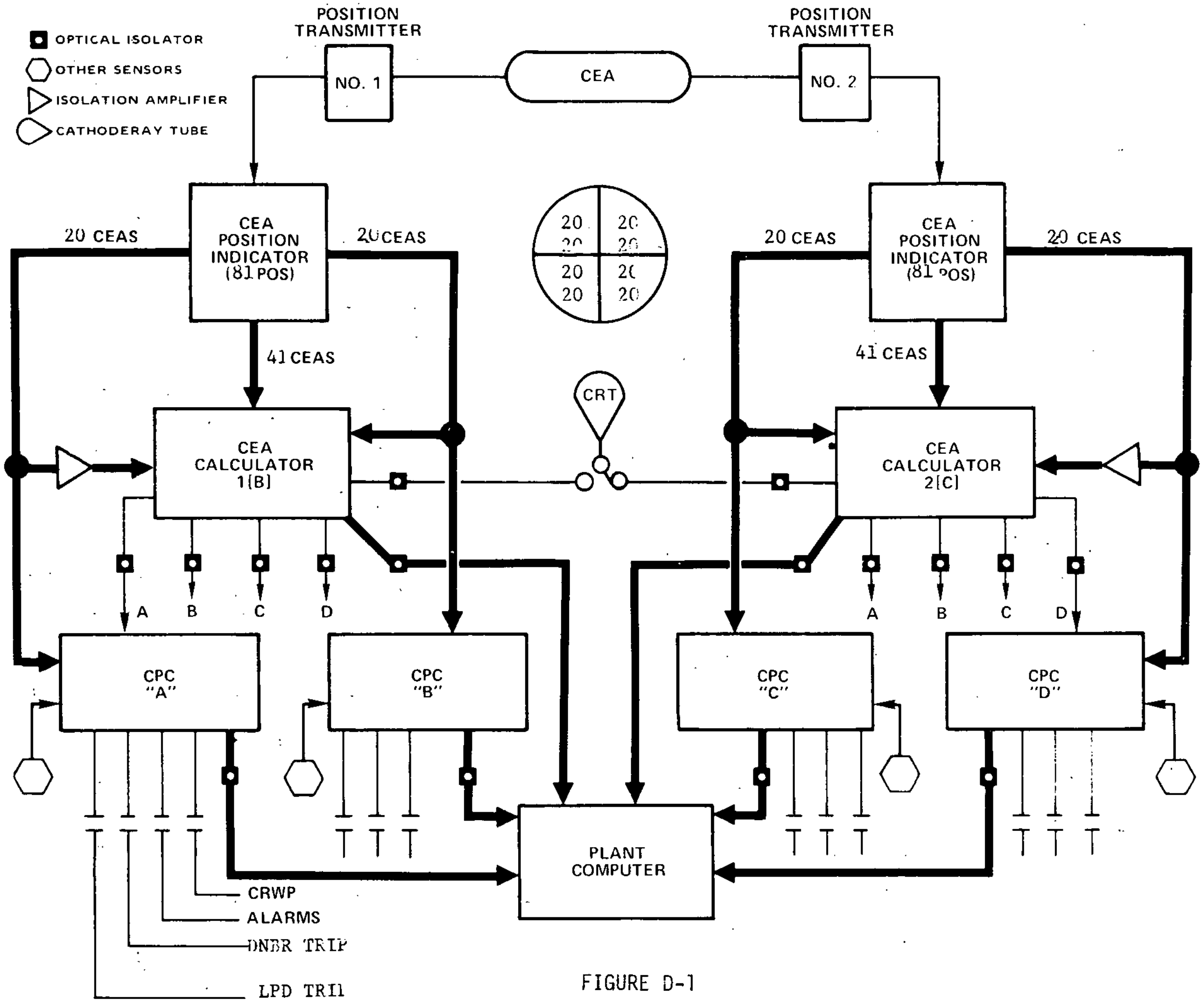


FIGURE D-1

CPS Hardware Configuration Block Diagram

The CPCS is divided into five major functional groups of hardware:

1) Signal generation and processing equipment

(2) Core protection calculators

(3) Control element assembly (CEA) calculators

(4) Calculator operator's module

(5) Permanent mass storage units

Our evaluation, limited to the first four major functional groups of hardware, is presented in the subsequent sections.

D.4.1.2 Signal Generation and Process Equipment for the CPCS

The signal generation and processing equipment converts process parameters, such as neutron flux and pressure, into signals compatible with the calculators input subsystem. Each redundant channel has separate sensors and signal processing equipment. The following parameters are the inputs that interface with each CPC.

(1) Neutron flux (3 per channel)

(2) Hot leg temperature (2 per channel)

(3) Cold leg temperature (2 per channel)

(4) Pressurizer pressure (1 per channel)

(5) Primary coolant pump shaft speed (4 per channel)

(6) CEA positions (20 per channel)

(7) CEA deviation information (2 per channel from the two CEA's)

The following sections address our review and evaluation, the problems areas revealed during our review and the resolutions concerning them.

Process Instrumentation

Our review of the process instrumentation for the CPCS 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-IE wiring, and the susceptibility of the IE circuits to noise or electromagnetic interference from the non-IE circuits. The applicant has committed to perform a noise immunity qualification/susceptibility test to demonstrate the noise immunity of the CPCS to electromagnetic interference, radio frequency interference, and process noise. The test procedures for these tests have been submitted and test results will be provided upon completion of the tests. We will review the test procedure and test results and address our resolution of this item in a supplement to this report.

Isolation Device

Information from the process instrumentation sensors (Items 2, 3 and 4 above) is also used by the non-Class 1E plant computer and the CPC's. A current to current (I/I) isolation device is used to isolate the class IE from non-IE circuits. We have reviewed the applicants qualification, test procedures and test report, and conclude that the isolation device (I/I) is qualified in accordance with the Commission's requirements as referenced in Section 7.1 of this report and is acceptable. Refer to Table 7.1, Position 3, for more detail.

CEA Position

The CEAC inputs from the CEA's in the reactor core are arranged such that there are 20 sets of four symmetrically located CEA's. Each CEA has two reed switch position transmitters (RSPT) to provide position information. This information is used to calculate planar radial peaking factors in the CPC's and to calculate rod deviation in the CEAC's.

Due to 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 CPC. For example, Channel "C" has 61 CEA's, therefore, 366 conductors are non-IE and 183 are Class IE analog signals that are transmitted to the CPCS. Note that all of these conductors are contained in the same raceway, inside containment. Needless to say that the Class IE conductors are dominated by non-IE and the potential for noise susceptiblity does exist.

The applicant has committed to perform a noise immunity qualification/ susceptibility test to demonstrate that a single event in the non safety circuits (e.g., "electrical noise") will not degrade the IE circuits from performing their safety function. Refer to Table 7.1, Position 5, for more information. The test procedures for these tests have been submitted and test results will be provided upon completion of the test. We will review the test procedure and test results and address our resolution of this item in a supplement to this report.

D.4.1.3 Core Protection Calculator

Each CPC is a 16 bit computer which is comprised of the following major components:

(1) Central processing unit (CPU)

(2) Memory

(3) Data input/output system

The following sections address our review and evaluations of the problem areas revealed during our review and the resolution concerning them.

The computer used in the CEAC and CPC is an Interdata Model 7/16 which is a 16 bit general purpose minicomputer. The computer is comprised of two major components: (1) central processing unit (CPU) which manages the resources provided by the memory and the input/output controller, and (2) memory which has a word length of 16 bits plus a parity bit and has a cycle time of one microsecond maximum.

Central Processing Unit (CPU)

The CPU has the capability of detecting parity bit errors, check sum calculations and power loss. Upon detection of an error the CPU generates an interrupt that causes that channel to place its output in the trip condition. In addition, each CPC and CEAC included a watchdog timer utilized for detecting both hardware and software faults in the computer. A timeout of the watchdog timer is immediately indicated by means of a status lamp which seals in and has to be manually reset. Upon observation of the status lamp, the operator is then required to analyze the situation and act accordingly such as bypass or trip of the affected channel. We concluded that dependence on the operator to take appropriate action for a computer channel upon advice of the same channel diagnosis of a hardware and/or software malfunction is not necessary. This action should be accomplished automatically to ensure more accurate and timely response to the watchdog timer which initially determined the malfunction. This requirement would be in compliance with the espoused philosophy of GDC 23. Therefore, we require an automatic trip of the associated protection channel upon timeout of the watchdog timer. Refer to Table 7.1, Position 23, of this report for more information. The applicant has agreed to implement a hardware and software modification to institute an automatic channel trip upon time out of the CPC watchdog timer, and the setting of all bits to "1" in the CEAC data link out of the CEAC watchdog timer. The staff finds this commitment to be acceptable.

Memory

Memory consists of protected and nonprotected areas. The protected area is defined as that portion of the calculator memory from which the CPU may read, but into which the CPU cannot write during normal on-line operations. Protected memory contains program instructions, including those affecting data storage into non-protected memory. The nonprotected memory is the portion of memory which the CPU may write data into or read data from at any time. Nonprotected memory is utilized for storage of the input data base, past history values of input data, intermediate calculated results, executive task and addressable constants. Operations are controlled by the CPU based upon instructions stored in protected memory.

The memory protection hardware causes instructions attempting to write into protected memory to be converted into read instructions. The evaluation of the protected memory feature is discussed in Section D.4.2.1.

The staff did not review the hardware of the Interdata 7/16 but did an extensive reveiw of the input/output controller, multipurpose acquisition and control systems (MACS) and its interface to the Interdata 7/16 computer.

Data Input Subsystem

The functions controlled by the data input/output (I/O) controller are analog input conversion, digital input conversion, digital data link service, and pulse input conversion to signals compatible with the computer. These operations by the data I/O controller are controlled by the CPU upon executing the appropriate stored program in protected memory.

The I/O Controller is known as the multipurpose acquisition and control system (MACS) which is manufactured by Systems Engineering Laboratory (SEL). This system is being utilized for the first time in a reactor protection system.

We have reviewed the design of the MACS including the logic diagrams of selected functional cards within the system and concluded that the design meets the Commission's requirements referenced in Section 7.1 and is acceptable.

Data Output Subsystem

The output from each of the CPC's will provide trip inputs (open contacts) to its associated one of four redundant and independent reactor trip system channels when the trip setpoints for high local power density and/or low DNBR are exceeded. The hardware configuration black diagram in Figure D-2 depicts functionally the various interactions of the CPCS with the plant computer. These data links provide information to permit automatic cross-channel comparison to assist the operator in his surveillance task and to provide documentation and early detection of anomalous operation of a channel.

Another output of the CPC is the CEA withdrawal prohibit (CWP) signals. Each CPC will send a single contact opening to the CWP two out of four logic matrix in the plant protection system. When two out of four coincidence conditions exist for a low DNBR pre-trip, or an LPD pretrip, an inhibit signal is transmitted to the control element drive mechanism control system to prohibit a CEA withdrawal.

The CWP logic does not meet the single failure criteria in this design. However, because of the unique functional application of the CEAC's, a failure of the CWP logic can be tolerated. The CEAC's are responsive to rod deviation and in doing so determine a penalty factor for transmittal to the CPC's. In turn, the CPC's utilize the penalty factor in the evaluation of DNBR and LPD. As the CPC's are adaptive to the positional states of the CEA's, functional diversity for the CWP is achieved.

We have reviewed the design of the output functions within the MACS and the CWP circuitry including their logic diagram and conclude that the design meets the Commission's requirements as referenced in Section 7.1 and is acceptable.

D.4.1.4 <u>Control Element Assembly Calculator (CEAC)</u>

There are two redundant CEA Calculators that provide penalty factor information to each of the four CPC's. The CEAC calculators are separate and independent of each other and each receives 81 CEA positions information that is derived from separate position transmitters. Twenty of the CEA positions are isolated by position isolation amplifiers in order to maintain interchannel independence. The outputs of the CEAC's are isolated from the inputs of the CPC by optical isolators.

Each CEAC is a 16 bit computer which is comprised of a CPU, memory and data input/output system which are identical to those described in the CPC Section D.4.1.3 of this report.

Important criteria used in the review were the requirements for channel independence, single failure during channel bypass, and the adequacy of the design for safety. These requirements were of specific concern in the review of the protection systems functional design, and the implementation of the design in terms of the electrical and physical independence of the system.

The functional design, in terms of the arrangement of the CEAC's and the CPC's also presents concerns in terms of the channel independence requirements. In particular, the use of two CEAC's to provide inputs to four CPC channels is of concern. An analogy to this design would be a single sensor feeding all four protection channels through isolation devices. However, there is a major difference in that the CEAC's only modify the trip set-point by a limited amount and furthermore the CPC's are not the only devices that can generate a trip signal to the reactor trip system.

Due principally to space limitations, there are only two positions sensors (reed switch assembly) per control element assembly. The channel independence requirements of GDC 22 and Section 4.6 of IEEE Standard 279-1971 do not specifically prohibit interchannel communication when stating that:

> "....the same protective function shall be independent and physically separated to accomplish decoupling of the effects of unsafe environmental factors, electric transients, and physical accident consequences documented in the design basis, and to reduce the likelihood of interaction between channel malfunction."

The intent of Paragraph 4.6 of the standard is compromised with the interchannel communication from the two channel CEAC's to the four channel CPC's, even though the communication occurs through isolation devices. However, the CEAC's have been designed to provide a new function in the protection system - automated monitoring of rod position and response to rod malfunctions. As discussed in Section D.3.2, it is the staff's opinion that the system, if properly implemented, the CPCS provides additional safety by providing automatic protection based directly on control rod deviation. This is in comparison to existing analog systems which rely on the operator to respond to misaligned control rods.

The functional design, in terms of the single random failure requirements presents concerns in terms of the operation of the two channels of CEAC's. The concern is removal from operation of one of the CEAC's with a single failure in the remaining CEAC. The requirements of Paragraph 4.11, IEEE Standard 279-1971 contains an exception for one out of two systems, which are permitted to violate the single failure criterion during channel bypass provided the acceptable reliability of operation can be otherwise demonstrated. In addition, removal from operation of one CEAC and the failure of the operational CEAC does not necessarily mean the total loss of the CPCS protection functions. The CPC's are designed to operate automatically with a maximum misalignment penalty factor upon detecting a failure of both CEAC's. Also, the loss of flow protection function does not rely on the misalignment penalty factor. However, a failure of a CEAC can occur which results in a non-conservative penalty factor transmitted to the CPC's. This represents a compromise to safety when the remaining CEAC is being tested or bypassed for repair. We will require that the duration of this mode of operation be limited by technical specification, based on the reliability of the system. For more information on bypass refer to Section D.4.1.6.

In our review, we identified concerns regarding the reliability of the system to justify the added complexity (reference Standard Review Plan Section 7.2, Appendix A, Item 1(d)). We also identified concerns regarding potential adverse affects of interchannel connection. See item 17 in Table D-1. The applicant responded by stating that the system design meets single failure criterion and that use of qualified isolation devices would adequately maintain independence.

The reliability of the system has not been estimated in terms of prediction, calculation, or demonstration. We require the applicant to submit an acceptable reliability analysis of the CPC's. The applicant has committed to provide the staff with a reliability analysis. For more detail refer to Table 7.1, Position 8, and Section D.4.2.1. Furthermore, as this is a new and unique system, we required a system burn-in test. The applicant has responded with a proposed three-month burn-in test in demonstration of the system's reliability. The test procedures for this test have been submitted, reviewed, and found acceptable. The test report documenting the results of the burn-in test is still outstanding. For more detail, refer to Table 7.1, Position 18, and Section D.4.2.1.

It is our intent to impose limiting conditions for operation (LCO) on one out of one operation of the CEACs. The LCO restrictions will be included in the facility technical specifications.

In conjunction with Position 18, the applicant has taken exception to the incorporation of software design changes prior to the burn-in test due to the impact on the ANO-2 schedule. A three-month burn-in test was completed in February 1977, using the "frozen design" software as agreed to by the staff.

However, additional tests to demonstrate and evaluate the integrity of the 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. As the staff has not received a commitment to this requirement, we consider this issue unresolved. We will address the resolution of this item in a supplement to this report.

The following section addresses our review and evaluation of the problem areas revealed during our review, and the resolutions concerning them.

CEAC Separation Criteria

The applicant has not demonstrated how the output of the optical isolator cards within the CEAC will meet the single failure criteria. Card slots 11 through 15 encompass the output cards to channels D, C, plant computer, B, and A respectively. Refer to Figure D-2, "MACS Universal Chasis-CEAC." All five cards are within a 3.5-inch proximity. We required the applicant to identify their design basis events for the CEAC and verify that no single event either internal or external to the CEAC will result in the loss of function. Refer to Table 7.1, Position 4, for more detail.

A failure mode and effects analysis was performed by the applicant, in response to this concern, in accordance with the general guidelines of IEEE Standard 352-1971 and is presently being evaluated by the staff.

We will address the resolution of this item in a supplement to this report.

Isolation Devices

Paragraph 4.7.2 of IEEE Standard 279-1971 permits the use of isolation devices to transmit signals from protection systems for use in non-safety systems. GDC 22 and IEEE Standard 279-1971 neither states nor implied that such devices may be used to couple signals between alleged independent channels within the protection systems. Paragraph 4.6 clearly states that channels shall be independent and physically separated. The cross coupling of two CEAC channels with four CPC channels results in degree of interdependence that exceeds previous accepted design practice. The acceptance of this unique design will be predicated upon the qualification of the electrical isolation devices that are used to maintain the electrical independence of the redundant channels.

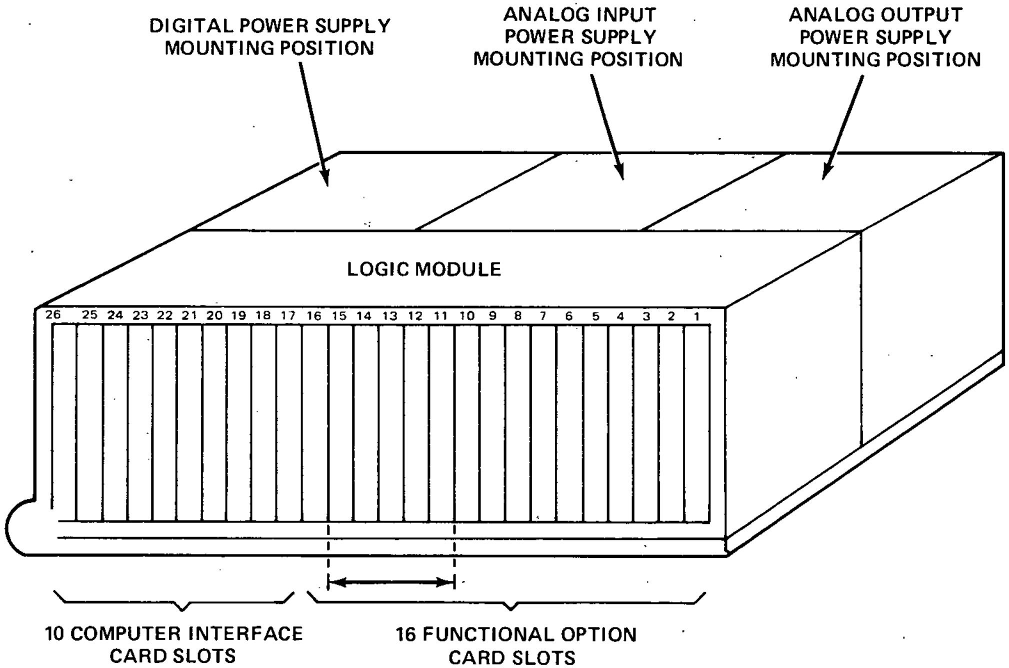
Optical Isolators

We have reviewed the qualification test procedure and test report of the optical isolator and concluded that it did not meet the applicant's criteria stated in Figure 7A.4-23 of the Final Safety Analysis Report. The test did not demonstrate that the application of a credible fault (120 volts alternating current or 125 volts direct volts direct current) across the output of the optical isolator, had no effect on the functional integrity of the isolator input circuit. Refer to Table 7.1, Position 26 for more information.

We will report the resolution of this item in a supplement to this report.

We have also discussed with the applicant our concern on the optical isolator and the effects of exposure to radio frequencies (RF) greater than 100 megahertz. Refer to Table 7.1, Position 12 for more detail.

The applicant has committed to perform a noise susceptibility test of radio frequencies from 50 megahertz through 600 megahertz via a broadband signal generator and antenna system.



<u>CARD SLOT</u>

11. OUTPUT DATA LINK TO CHANNEL D
12. OUTPUT DATA LINK TO CHANNEL C
13. OUTPUT DATA LINK TO PLANT COMPUTER
14. OUTPUT DATA LINK TO CHANNEL B
15. OUTPUT DATA LINK TO CHANNEL A

FIGURE D-2 MACS UNIVERSAL CHASSIS - CEAC

The test results will be provided upon completion of the test. We will review the test results and address our resolution of this item in a supplement to this report.

Position Isolation Amplifiers

The isolation amplifiers are located in Channels A and D which transmit twenty CEA position informations to the CEA calculators in Channels B and C respectively. This configuration is shown in Figure D-2. These isolation amplifiers are used to maintain channel independence.

We have reviewed the test procedures and test report, and conclude that the position isolation amplifiers are qualified in accordance with the Commission's requirements as referenced in Section 7.1 and are acceptable.

D.4.1.5 Operator's Module

The operator's module is designed to permit the operator to monitor system status and performance. The module houses a digital display for operator readout, a keyboard for data entry by the operator, function push buttons enable the operator to obtain information and enter addressable constants as discussed in Section D.3.11. The design of the operators module does not permit the operator to change anything in computer memory except addressable constants. There are four operator modules on the PPS section of the main control board.

Additional information read out is also provided to supplement the operators module. This consists of analog meters for monitoring sensor inputs and calculated variables and annunciators. Similarly the CEA position input signals are displayed on a single monochrome cathode ray tube (CRT) mounted in the main control board, and provides bar charge indication of CEA positions.

This information read out is similar to that which is required for analog hard-wired systems. Consistent with the criteria for analog systems, only those displays required for post accident monitoring (PAM) are qualified for Class IE equipment. PAM is discussed in Section 7.5.1 of this report.

We have reviewed the operator's module design which included logic diagrams and interconnection diagrams and concluded that the operator's module meets the Commission's requirements as referenced in Section 7.1 and is acceptable.

D.4.1.6 Bypasses and Interlocks

The CPCS is designed with an operating bypass and a channel bypass. The operating bypass disables the low DNBR and high LPD trip and pretrip digital output signals from the CPCS to the plant protection system when reactor power is less than 10^{-4} percent power. The bypass is manually enabled under administrative control by a key switch. The bypass is automatically set by a signal from the nuclear flux instrumentation when reactor power is less than 10^{-4} percent. Likewise, the bypass is automatically removed when the nuclear instrumentation signal indicates that reactor power is greater than 10^{-4} percent power. Indication of the normal (non-bypassed status) or the bypass status of each channel is provided by lamps in each core protection calculator operator's module. Annunciation of the bypass is also provided at the main control board. Enabling of this bypass effects only the trip output contacts; the DNBR/LPD trip computations are performed continuously regardless of whether or not the reactor is at power or the bypass is actuated. We have reviewed the design of the operating bypass and conclude that it is acceptable.

The applicant has identified a CPCS design change to provide a CEAC operating bypass. This change will be implemented by modifying the CPC program to detect an operator request for a CEAC bypass and to ignore the penalty factor being transmitted from a CEAC. The details of this CEAC bypass have not yet been submitted for our review. Our review and evaluation of the CEAC operating bypass will be reported in the supplement to this report.

The channel bypass provides the capability to bypass one of the four CPCS channels (i.e., the high LPD and low DNBR trips) for maintenance or testing. The CPCS bypasses are part of the reactor trip system bypass system provided for each plant trip function. The results of the review of the channel bypass design is discussed in Section 7.2 of this report. In addition, the unique design of the CPCS required that the low DNBR and high LPD trip bypass circuits for CPCS channels B and C be designed with interlocks to channels A and D respectively and other features. The purpose of these interlocks is to enable testing of the CEACs and the CEA position isolation amplifiers. These modifications do not affect the bypass and interlock circuits for the six matrix logic networks in the reactor trip system.

As described in Section D.4.1.4 of this report, 20 CEA position signals are shared between CEAC channel 1 and CPC channel B and 20 CEA positions are shared between CEAC channel 1 and CPC channel A. (CEAC channel 2 and CPC channels C and D have the same configuration.) In addition, the output of each CEAC also provides an input to each CPC. As a result of this functional interaction, we are concerned about the potential for initiating false reactor scrams during testing, maintenance and repair of the CEACs and CEA position isolation amplifiers. In this regard, we considered the requirement for an additional CEAC bypass similar to the CPC channel bypasses. In evaluating the need for a CEAC bypass, we determined that the bypassing of CEACs during test and maintenance would compromise the reliability of the CPCs and the ability of the CPCs to meet the single failure criterion. Thus, although the implementation of a CEAC bypass could improve CPCS operational availability, a decrease in CPCS reliability with respect to performing it safety function could also result. Therefore, it is our opinion that the implementation of a CEAC bypass, in addition to the channel bypasses currently provided, would not improve the CPCS capabilities with respect to safety. However, to ensure that the CPCS integrity is sustained during test and maintenance, we will require that the detailed procedures describing test and maintenance methods and administrative controls to be used during CPC, CEAC and CEA position isolation amplifier testing and maintenance be submitted for our review. The applicant has committed to provide this information in his response to Position 9 (Table 7.1). This information has not yet been submitted for our review. Contingent upon our review of these procedures, we conclude that the CPCS channel bypass is acceptable. Our final evaluation of the CPCS channel bypass will be reported in the supplement to this report.

D.4.2 Test, Maintenance, Monitoring and Qualification

D.4.2.1 Operational Testing

Operational testing for the CPCS consists of those tests which are performed during plant operation to verify the continued functional performance of the CPCS and to ensure the operational availability of the CPCS. For the CPCS, three types of operational tests are provided as follows:

(1) Automatic on-line tests,
(2) Periodic tests, and
(3) Response time tests.

The automatic on-line tests consist of hardware and program design features which continuously monitor the computer operation for failures. For the CPCS, the major automatic (or self-test) features include memory parity checks, power fail detection, sensor failure, analog input system reference checks, protected memory check sum tests, and multiply and divide arithmetic checks. In addition, the watchdog timer described in Section D4.1.3 also provides automatic detection of failures which interrupt the normal system operation. Depending upon the type of faults, the automatic tests will act in one of two different ways. For faults which may occur intermittently either due to a gradually weakening component or due to a spurious "gliche" in the computer (e.g., parity errors and arithmetic errors) the trip outputs and/or alarms for the CPC or CEAC affected will be set and the affected computer will automatically attempt to reinitialize and restart. If the restart is successful, the trip/alarm outputs will automatically reset; the cause of the restart will be stored in the computer memory; the computer will continue to operate. If the auto restart is not successful, the computer will stop operation with the trip outputs set. Each computer can store information for up to three auto restarts. The operator can request the auto restart information for display on the operator's console or the auto restart record for the last three auto restarts will be printed automatically during periodic testing. For other types of faults such as check sum errors or input/output (I/O) timing errors the affected computer will stop operation with its trip outputs set or, for CEACs, with its failed bits set.

Section D.4.1.3 describes the protected memory hardware. In our review, we identified the fact that this feature does not include means for alerting the operator to violations of protected memory. As a result, as identified in Table 7.1, Position 7, no safety credit would be given for this feature. The applicant has agreed with our position. Therefore, this issue has been acceptably resolved.

The periodic tests are intended to verify the functional operation of each CPC and CEAC and to reveal failures not detected by the automatic tests. The periodic tests are comprised of a series of tests. Some of these tests are performed with the system operating (on-line); others are performed with the CPC channel bypassed (off-line). The on-line tests consist of interchannel comparisons by the operator of like parameters for sensor transducer signals, intermediate calculated values, the calculated values for DNBR, DNBR margin, LPD, LPD margin, and calibrated nuclear power. The off-line tests are conducted using a recorded test disk and a test program which is included in each CPCS computer. The information on the test disk consists of a predefined data base, the results of calculations using this data base, hardware diagnostic programs and a duplicate image of the protected memory of the computer which is to be tested. A separate disk is provided for each CPCS channel. For Channels B and C, which include a CPC and a CEAC data base, calculated results and memory image are stored for both the CPC and the CEAC. The hardware diagnostic programs, predefined data base, expected calculations and memory are the same for each CPC and for each CEAC. To perform the periodic off-line tests, the channel to be tested is bypassed as described in Section D.4.1.6. After the channel is bypassed, the periodic test data and diagnostic programs are loaded from the disk into the calculator being tested using the disk unit. A send and receive teletype with keyboard and printer is manually connected to the channel to be tested. The keyboard is used to control the execution of the tests and the printer documents the test results. The calculator algorithms are checked by sequentially executing the calculator programs using the defined data base and verifying that the actual calculated values agree with the expected results. Any disagreements are printed. The calculator protected memory is compared with the memory image on the disk and any disagreements are printed. The hardware diagnostic programs are executed to check for proper operation of calculator hardware modules. For the CPCs, after completion of these tests, the bypass is removed and a "dummy" neutron power signal is ramped into the CPC and the output of the pretrip and trip indications for low DNBR and high LPD is observed.

The response time tests are designed to verify the response time characteristics of the CPCS. The response time tests for the reactor protection system including the CPCS are discussed in Section 7.10 of this report.

As noted in Table 7.1, Positions 21, 22 and 23, our review of the CPCS operational tests identified several concerns regarding the design, implementation and adequacy of the automatic on-line tests and the periodic tests. With respect to the automatic tests, we required that the watchdog timer circuits and program be modified as discussed in Section D.4.1.3. The applicant has identified additional program design changes to improve the automatic detection of arithmetic overflow and underflow errors. The final design for the watchdog timer and for the program modifications has not yet been submitted for our review. Based on the information provided to date, we consider that the automatic on-line test features have been implemented without restricting the primary safety functions of the CPCS and that they provide additional capabilities for detecting equipment failures which do not exist in present designs for analog hard-wired systems. Therefore, we conclude that the automatic on-line testing for the CPCS is acceptable, conditioned by our review and acceptance of the detailed design changes. The results of our review of these changes and the final acceptance of the automatic on-line test functions will be reported in the supplement to this report.

In our review of the periodic tests, we questioned the basis for the CPCS time interval of periodic testing. The applicant stated that the time interval of 30 days was based on past experience and the test intervals of analog protection systems. However, the CPCS design represents a new configuration for reactor protection systems. In addition, many of the components in the CPCS (digital computers and I/O interfaces, CEA position transmitters, pump speed sensors and CEA isolation amplifiers) are being used for the first time in a protection system. Several are also first-of-a-kind designs. Therefore, the staff concluded that the past experience with analog protection systems could not be extrapolated and applied to the CPCS. As identified in Table 7.1, Positions 8 and 18, we required additional technical justification of the periodic test interval. In response to our positions, the applicant submitted a test plan for a three- to six-month burn-in test and proceeded to perform a three-month burn-in test of the CPCS hardware exclusive of sensors and signal processing equipment. The applicant also committed to submit, for our review, a test report documenting the results of this burn-in test. In conjunction with this test report, a reliability analysis of the CPCS and an evaluation of the reliability during the burn-in test will also be submitted to justify a periodic test interval of 30 days. We have reviewed and accepted the test plan. Based on the applicant's commitments, we conclude that an acceptable time interval for periodic testing can be established. The final requirements for the time interval for periodic testing will depend upon our review of the burn-in test report and reliability analysis to be submitted by the applicant. The results of our review will be reported in the supplement to this report.

In our review of the periodic tests, we also questioned the adequacy of the test procedures and the off-line tests for periodically checking and verifying the functional operation of the CPCS in accordance with the requirements of GDC 21 and Section 4.10 of IEEE Std. 279-1971. As identified in Table 7.1, Position 9, we require that the periodic test program be modified to include procedures for testing each trip function in each channel from sensor input to the CPCS to trip output to the reactor trip system. The applicant has responded to our position by stating that the proposed periodic tests are based on the overlap testing philosophy and are adequate for verifying the functional operation of the CPCS. However, it is our position that the proposed overlap tests are inadequate and that a functional operation check from CPCS sensor inputs to the trip output is required to adequately ensure that the CPCS is operational. This test should be accomplished by injecting a test signal for each sensor input as close as practical to the sensor and monitoring for proper trip output when the trip setpoint is reached. We will report the resolution of this item in the supplement to this report.

Our review also identified an additional concern regarding the off-line test using the predefined data base and calculated results. As identified in Table 7.1, Position 10, we required that the applicant develop practical techniques and procedures for using the off-line program to verify calculated results after changes to addressable constants. These changes will be designed to reduce the interaction between changes to addressable constants and the periodic test results. Based on this commitment, we conclude that an acceptable method of verifying CPCS program calculations will be provided. The applicant has not provided these changes for our review. The final approval of the program performance tests will depend upon our review and acceptance of these modifications. The results of this item will be reported in the supplement to this report.

Section D.4.1 discusses the unique design of the CPCS and the reliance on many isolation devices (i.e., optical isolators for CEAC to CPC data transfer and CEA position isolator amplifiers for shared CEA position signals) to maintain electrical independence among the protection channels. As noted, the ability of these devices to maintain the isolation among channels is one of the bases for accepting the design of the CPCS. It is our concern that failures of the isolation characteristics of these devices would seriously compromise the ability of the CPCS to function. The current periodic test procedures do not include provision for verifying that the isolation capabilities of these devices have been maintained.

As identified in Table 7.1, Position 27, we will require that periodic tests be performed to verify the isolation characteristics of those isolation devices used to ensure channel independence. Therefore, we will require that the applicant submit for our review and approval, a test procedure for periodically checking the isolation characteristics. We will report the resolution of this item in the supplement to this report.

D.4.2.2 Maintainability

In our review of the CPCS we identified to the applicant concerns regarding the maintainability of the CPCS. Previous experience with nuclear power plants, and other industrial uses of process computer systems has identified several concerns regarding maintainability of digital computer systems over the operating life of the plant. These concerns are summarized as follows:

- Lack of standardization in hardware and software design has led to difficulties in identifying second sources of parts supply.

- The short commercial life cycle of electronic parts compared to plant operating life has resulted in obsolescence of equipment and unavailability of spare parts.

- Suppliers and users lack of experience; trained technicians to maintain equipment.

- Incomplete maintenance and trouble shooting procedures and system documentation has made maintenance difficult.

In addition, IEEE Std. 279-1971, Section 4.21, identifies maintainability as one of the requirements for the reactor protection system. In response to our concerns, the applicant has provided information describing the procedures for diagnosing CPCS failures. These procedures do not consider the maintainability plan of the CPCS for the life of the plant. Therefore, Table 7.1, Position 25, identifies the requirement that the CPCS maintainability plan for the life of the plant be docketed for the regulatory staff's review and evaluation.

The applicant has not submitted his response to our position on maintainability. Therefore, our evaluation and the acceptance of the CPCS maintainability will be reported in the supplement to this report.

D.4.2.3 Plant Computer System Monitoring

The CPCS is designed with a data link and a special program module in each protection computer to service the plant computer system (PCS). The program module which checks for a request by the PCS is scheduled at a high priority. When a PCS request for data is present, the protection computer transmits sensor information, in digital form, from its data base to the PCS. The PCS, upon receiving data from all protection computers, performs an automatic cross-channel comparison of protection system parameters. In addition to these data links and programs, all RPS sensor data (including CPCS sensors with the exception of pump speed and CEA position) are transmitted to the PCS in analog form. The PCS processes these analog data similar to the digital data received via the data links. The PCS also has independent sensor inputs for all CPCS process parameters. As identified in

Table 7.1, Position 20, we required that the PCS data links to the protection computers be removed and that the plant computer service routine be deleted from automatic program scheduling.

In response to our position, the applicant has stated that the independence between the CPCS and the PCS is maintained by using qualified optical isolation devices at both ends of the digital data links and by using qualified current-to-current isolation devices for the analog data links to the PCS. The applicant has further stated that the PCS checks of both the analog and digital data do not provide any safety function. Rather, the PCS monitoring is intended to assist the operator in his surveillance of the reactor protection system and to aid in the detection of reactor protection systems sensor failures.

The limited capabilities of analog protection systems and the convenience of the PCS as an operator's aid for protection system surveillance has, in the past, justified a compromise to GDC 24 for analog protection systems. The additional capabilities which digital computers provide for direct surveillance and presentation of information to the operator (without going through the PCS) eliminate this need to compromise GDC 24 for digital computer based protection systems.

For the CPCS, the per channel surveillance features include the following:

(1) The ability to automatically detect and alert the operator to sensor failure.

(2) The ability to warn the operator of an approaching trip condition by initiating pre-trip alarms.

(3) Continuous, direct indication of the margin to trip at current operating conditions.

(4) The ability to automatically detect central processing unit and MACS failures; to alarm the operator of the condition; and to set the trip outputs of the channel.

(5) The ability to record, display, and print on demand the cause of detected failures.

(6) The ability to display on demand the values of sensor inputs or calculated parameters at any time.

The capability to log data could also be easily implemented in the CPCS on a per channel basis if required. This surveillance information is directly and independently, either continuously or on demand, available to the operator from each channel using qualified safety devices. Therefore, the proposed PCS links would provide little additional assistance to the operator in his monitoring of the CPS.

Thus, the data link to the PCS adds unnecessary complexity to the CPCS design. This imposition of requirements on the protection system design for nonsafety data requirements is an unacceptable encroachment on the protection system safety functions. We have concluded that the PCS data links add complexity with no enhancement to safety nor to protection system operation. Therefore, we will require that the PCS data links to the CPCS be removed and that the PCS service routine be deleted from automatic program scheduling in the CPCS. The resolution of this item will be reported in the supplement to this report.

D.4.2.4 Environmental Qualification

The CPCS environmental qualification program was based upon the guidelines of IEEE Std. 323-1971. In accordance with Section 4.3 of this standard, the CPCS equipment was qualified by type testing of components. The environmental test conditions for the CPCS components were determined by classifying and identifying safety equipment environmental design categories based upon equipment location and functional requirements as described in Section 3.11 of this report. During our review, we requested that the applicant submit documentation to verify the adequacy of the environmental qualification tests for the following CPCS equipment:

(1) CPC central processing unit and MACS modules,

(2) CPC power supply,

(3) CEA position isolation amplifier,

(4) CPC operator's modules,

(5) RCP speed sensor and signal processor,

(6) CEA reed switch position transmitter, and

(7) Current-to-current isolation transmitters.

In response to our concerns on the adequacy of the environmental tests, the applicant has submitted test plans, test procedures and the requested test reports for the equipment listed. In our review of these test plans, we noted that the temperature and humidity test profiles for CPCS equipment in the same environmental design category were different. However, the tests did, in all cases, meet or exceed the maximum and minimum temperature and humidity conditions for the applicable environmental design conditions. Table 7.1, Positions 3, 6, 11 and 13, state our requirements for a performance test of the integrated system. In response to our requirement, the applicant has submitted a test procedure for the thermal test of the process protective cabinet (PPC) which houses the CPCS equipment, exclusive of sensors. This test is to be accomplished after installation of the equipment. The test results will be provided upon completion of the tests.

Section D.4.1.2 of this report discusses our concerns regarding the electromagnetic interference and electrical noise susceptability of the CPCS equipment. In this regard, we required that the applicant perform electrical noise qualification tests for the CPCS. In response to our requirements that electrical noise qualification tests be performed, the applicant has committed to performing a noise immunity qualification test. Test results will be provided upon completion of the tests.

We have concluded that the environmental qualification tests for the CPCS equipment meet the minimum requirements and are acceptable with the exception of the following:

- For the CEA reed position switch transmitter (CEARPST), the applicant has stated that radiation exposure qualification will be demonstrated by analysis. The applicant has committed to providing this analysis for our review. The acceptability of the CEARPSTs is dependent upon our review and acceptance of this analysis.

- The acceptability of the environmental qualification of the CPCS equipment housed in the PPC is dependent upon our review and acceptance of the test results for the thermal qualification tests for the PPC.

- As noted in Section D.4.1.2 of this report, the resolution of our concerns regarding noise susceptability of the CPC is dependent upon our review of the test reports for the CPCS noise immunity qualification and susceptability tests.

- As noted in Section 3.11 of this report, qualification of Class IE equipment operation during and following the main steamline break accident is under review. The acceptability of the qualification of those CPCS sensors (CEARPSTs and RCP speed sensors) located inside the containment is dependent upon the outcome of this review.

The resolution of these outstanding items will be reported in this supplement to this report.

D.4.2.5 Seismic Qualification

The applicant has submitted Topical Report CENPD-182 for seismic qualification of electrical equipment for the nuclear steam supply systems. This report includes the seismic qualification of the CPCS equipment as follows:

(1) Temperature transmitter

(2) Current-to-current isolation transmitter

(3) CPC mass storage unit

(4) CPC MACS modules and central processing unit

(5) CPC power supply

(6) CEA position isolation amplifier

(7) RCP speed signal processor

(8) CPC operator's module

(9) Temperature sensor

(10) RCP speed sensor

(11) CEA reed switch position transmitter

The status of the staff's review of Topical Report CENPD-182 is discussed in Section 3.10 of this report.

In our review, we also requested that the applicant provide additional information on the seismic qualification test configuration and test procedures used to verify the operability of the unique CPCS equipment (i.e., items 3 through 8, 10, and 11 on the immediately proceeding equipment list). Based on our review of these procedures, we have concluded that the test configuration and test procedures were acceptable for verifying the functional operability of this equipment before, during and after the seismic excitation.

As noted in Table 7.1, Position 14, the staff requested that the applicant provide additional information to verify the adequacy of the seismic loads used for testing the CPCS equipment housed in the Plant Protective Cabinet (PPC), i.e., 1 through 7 on the preceeding list. Information to support the ability of the PPC to survive the seismic events was also requested.

The staff has reviewed the analysis for the Process Protection Cabinet (PPC) and the testing for CPC modules. We conclude that the analysis is acceptable to ensure structural integrity of PPC under seismic loads. However, the analysis is inconclusive for verifying the adequacy of testing input motions used for seismic qualification tests of CPC modules mounted on the PPC. The applicant is currently conducting further investigation on methods to determine the required response spectrum at the mounting location on CPC modules. The final acceptance of CPC seismic qualification is pending on our evaluation of additional information.

The resolution of these outstanding items will be reported in the supplement to this report.

D.4.3 Software Evaluation

Computer software is a new concept in the field of protection systems licensing review. It is important, therefore, that the staff perspective of software be understood. Basically, software is regarded as the "blueprints" of the system as conceived by the designer. As in conventional design, these "prints" range in complexity from simple function overviews to detailed point-to-point wiring diagrams. Consequently, when construction is completed, computer programs are not "in" the system just as blueprints are not "in" the hard-wired system. This is an important perspective in the consideration of software reliability.

Software Reliability

The term software reliability is used in this report to mean the degree of certainty achieved by the computer programmer in translating the PPS functional specifications and equations into the computer programs necessary to perform the required function. In conventional protection system design review, this is equivalent to assessing the detailed wiring diagrams for accuracy and completeness. Similarly, as in the design review of conventional systems, the task of assessing software for design accuracy requires a detailed analysis of the final "as built" programs.

Algorithm Implementation

The CPC algorithms are coded in assembly language. This provides the advantage of not having to evaluate the reliability of a higher level language compiler. The programs, however, are functionally based on algorithms developed in FORTRAN using design type codes. This offsets the "lack of overview" problem encountered when programming complex engineering calculations in assembly language.

The overall structure of the software was reviewed for basic conformance to the good engineering practice of:

(1) Using a proven assembler,

(2) Coding conventions, notations, etc.

(3) Descriptive comment fields,

(4) Modularized construction, and

(5) Data based handling and construction.

A detailed audit of several hundred instructions was conducted on selected programs. Comparisons between the instrumentation and the logic flow diagrams revealed a number of discrepancies that resulted in a more detailed audit of the vendors software documentation procedures (see Item M12 of Table D-1).

A detailed audit of the full core dump was conducted to resolve an error produced by two source cards getting reversed. The vendor has since implemented effective procedures to detect out of sequence cards during assembly. It is noteworthy that any changes from the "as built" software are very easy for the reviewer to identify whereas in hard-wire systems it is virtually impossible to determine that no changes have been made.

Documentation Adequacy

During the early stages of this review, the software documentation was inadequate. The vendor had no precedence on which to judge the regulatory requirements for software documentation. This problem has been substantially resolved with the major deficiencies relating to documentation revision.

Automatic Recovery of Computer Protection Systems

The CPCS design has implemented (on a channel basis) many self-checking, fail-safe type automatic diagnostic programs. Consequently, the system will tend to respond to a much larger class of noise-spike type anomalies than will its analog counterpart. Likewise, because there are many thousands more electronic components involved, computers tend to have a higher frequency of "spikes" or gliches as they are called in computer jargon. Not unlike the analog hardware, intermittent behavior of some unknown component cannot be justification for taking a channel out of service for impractical trouble shooting. Experience has shown that continued service is the most practical course of action until the failure is "firm" or the frequency of malfunction sufficiently high to reasonably expect detection and repair. It is for this reason that auto restart is a common design feature for most computers.

The staff's primary concern is any failure to danger that may go undetected, not the failure which causes a momentary channel trip. The primary safeguard to ensure continued system integrity is the frequency and thoroughness of the periodic testing program. An important part of multichannel protection philosophy is the option of operating with a failed channel during the entire test period. The denial of the auto-restart feature is in effect removing this option by requiring the operator to verify that any and every momentary failure has not been a failure-to-danger.

The possibility of some common mode failure must be considered, however, in situations potentially affecting more than one channel. It is planned to limit the use and conditions for operation of the CPCS auto restart feature by plant technical specification but not in a manner to deny its usefulness.

System Failures and Input Data Anomalies Detection of Computer Protection Systems

The CPCS design includes methods for input data anomaly detection and sensor failure alarms. Although these schemes could be based on more exotic algorithms, they are considered acceptable in the redundant channels of the CPCS systems. In comparison to existing analog designs, the proposed detection scheme is much more extensive.

Any requirements for the self-detection of all possible failures are not reasonable. The state-of-the-art of digital computer technology has not developed methods for automatically recording in memory all system failures. The CPCS has incorporated means for failure detection as discussed in Section D.4.2.1.

Quality Assurance

Staff audits of the design, development and qualification testing of the software defined quality assurance deficiencies in these tasks. The staff's concerns were expressed as a safety position (see Table 7.1, Position 16).

The applicant's response to the staff's quality assurance concerns proposed a program, which if properly implemented, will comply with the applicable requirements of 10 CFR Part 50, Appendix B. Design documentation for the final ANO-2 software design is incomplete as of March 25, 1977. The acceptability of the quality assurance plan remains under review and the evaluation cannot be completed prior to the evaluation of software design documentation, performance qualification testing, and qualification of software change procedures. A review of the implementation of the quality assurance program will be contained in a supplement to this report.

Software Review Summary

The ANO-2 CPCS programs are currently being rewritten to correct design deficiencies revealed during the licensing review conducted to date and during the November 1975, Phase II testing. In addition, the designer has elected to restructure the system programs partly in response to staff positions 15, 21, 22 and 23 presented in Table 7.1. The applicant's response to these positions also indicate that the designer intends to more effectively utilize the automatic fault detection capabilities of the computer. While the modifications are expected to enhance the overall software reliability, they are sufficiently extensive to preclude any definitive assessment of their acceptability until completion of Phase I and Phase II testing.

Based on our review to date, the following general conclusion can be made concerning the software review:

(1) The designer has recognized the need for minimizing undue complexity in the executive program structure.

(2) Programs have been structured into modules for ease of testing and comprehension.

(3) The importance of stringent quality assurance practices for maintaining software reliability has been recognized and improved procedures have been adopted.

(4) Conditional on the successful resolution of the outstanding issues the ANO-2 CPCS software can be made acceptable based on the requirements of IEEE Std. 279-1971, experience at other facilities, and good engineering practices.

The resolution of these issues will be given in a supplement to this report.

D.4.4. Software Qualification

Consistent with the definition of software discussed in D.4.3, this section could also be considered "design qualification."

The vendor has developed a two phase test methodology based on individual program modules, whole program units and finally the entire system of programs running in real time with simulated reactor inputs.

An audit of the initial Phase I testing was conducted at the vendors plant (see Item M11 of Table D.1). The filing and formality of data record numbering was considered inadequate, however, greatly improved in a later docketed Phase I test report. This report is to be replaced by a report that will apply to the ANO-2 final software.

The Phase II Test Report, CENPD-222, was reviewed by the staff and found to be unacceptable. The applicant was notified of our concerns in a September 2, 1976 meeting and by Docket No. 50-368 letter, "Issuance of Core Protection Calculator System Position 24 (ANO-2)," dated September 16, 1976. The basic response of the applicant is documented in a letter, dated September 24, 1976, to the Director of NRR, Docket No. 368, "Core Protection Calculators." The applicant has commited to a reperformance of the Phase I and Phase II test series with the final CPC algorithms and executive system software (refer to Item A of Position 24, Table 7.1).

The applicant has attempted to document all software modifications from the "frozen" design, although our continuing review has identified several changes which were not clearly specified (refer to Item B of Position 24, Table 7.1).

The applicant has committed to inclusion of justifications in the Phase II test report on the final software design to satisfy concerns expressed in Items C and D of Staff Position 24, Table 7.1.

The applicant has proposed to perform single variable transient tests on the channel system and on the configured four channel system during Phase II testing to supplement the evaluation of dynamic algorithm implementation in response to the NRC concerns expressed in Item E of Position 24, Table 7.1.

The applicant has agreed to submit a Phase II test plan prior to the Phase II test of the final design software. The staff must review the test plan to determine if the content is acceptable with respect to Item G of our Position 24, Table 7.1.

The applicant has committed to submittal of documentation which will explain the anomalies observed by the staff during testing of the computer protection system (refer to Item H of Position 24).

The applicant has committed to address all CPC error components pertinent to the Phase II acceptance criteria for static tests as required by Item I of Position 24. We will evaluate the acceptability of his response during review of the Phase II test documentation which contains this information.

The applicant takes exception to Item J of Position 24, which indicates that transient analyses for selected dynamic test cases should be used to specify acceptance criteria for reactor trip times. However, the applicant has committed to provide such analyses using codes normally employed for Chapter 15 transient analyses. We find this response acceptable since the submittals will enable the staff to make the desired trip time evaluation.

Software modifications performed by the applicant have addressed our concerns regarding errors in the scaled range of variables as expressed in Item K of Position 24. We will complete our evaluation during review of the Phase II test results.

The applicant has committed to additional analyses to address staff concerns about round off errors during Phase II testing of the "frozen" software. Item L of Position 24 will remain unresolved until the results of these analyses are submitted for staff review.

The applicant has performed software modifications to address staff concerns about recurring automatic restarts that occurred due to out-of-range multiplicative values obtained during Phase II testing of the "frozen" software. Resolution of Item M of Staff Position 24 regarding these concerns will require staff review of the Phase II test plan and test results for the modified final design software.

Phase II dynamic testing will be reperformed on the configured four channel system with all changes from the "frozen design" incorporated. This will satisfy the requirements of Item N of Staff Position 24 and will permit evaluation of the dynamic algorithms in the corrected design software.

The total acceptability of the final design software with respect to Position 24 concerns will be evaluated, based on the submittals and test programs committed by the applicant. We will address the final resolution of these issues in an SER supplement upon completion of our review.

An audit was conducted at the Phase II test site to review the basic data log books and test environment (see Item M7 of Table D.1). During this visit, the staff requested several ad-hoc demonstrations, one of which revealed a design deficiency in the data input program. This same deficiency was also noted during the software audit (Item M12 of Table D.1) and is reflected in Position 22, Table 7.1.

The applicant has provided statements of intent on the Staff Positions 22 and 24 (see Table 7.1) relating to software qualification which, if implemented properly, will be found acceptable.

The CPCS Phase II Test Report, CENPD-222, was reviewed by the staff and found to be unacceptable, including the test procedures and acceptance criteria utilized for the tests. The conclusions from this review were expressed in Staff Position 24. The applicant now proposes to reperform the Phase I and Phase II tests on the final design software. CPC trip times are to be evaluated based on results of design analyses for representative dynamic test cases, as required by Position 17. Since the final design software is to be tested in accordance with Position 24 and consistent with the requirements of Position 17, the response to Position 17 is acceptable. The staff will continue to review the applicant's conformance to the requirements of Staff Position 24 for final design software and for any software changes implemented after completion of Phase II retesting.

The applicant has submitted CEN-39(A), "CPC Protection Algorithm Software Change Procedure" and CEN-40(A), "CPC Single Channel System Verification Tests," in partial response to Staff Position 19. CEN-39(A) provides the required documentation of procedures for specification and implementation of modifications to the CPC algorithms and the CPC data base constants. The staff has reviewed CEN-39(A) and finds it acceptable, provided that the procedures described in the document are successfully qualified by testing.

CEN-40(A) describes the test cases, test configuration, and method of analysis to demonstrate the adequacy of the proposed single channel test system and of the procedures for transfer from the testbed to the plant system. The staff has reviewed this document and finds the proposed test program acceptable as a basis for qualification of the single channel test system. The acceptability of the proposed software change procedures, including procedures for transfer from the test bed to the plant system, will be evaluated based on results of the entire Phase I and Phase II test program in conjunction with the single channel system verification tests.

Results of this evaluation will be reported in a supplement to this report after receipt of the test report and completion of our review. The staff will require a more definitive description of the single channel test system in the test report on the single channel system verification tests.

The applicant has taken exception to Position 19E which states that software design changes and revisions to constants in memory (except addressable constants) are subject to documentation, review, and approval of the staff. The applicant has stated that the software changes will be performed in accordance with the requirements of Section 50.59 of Title 10 CFR Part 50 (Changes, Tests, and Experiments). Section 50.59 states that changes, etc., may be made without prior Commission approval, unless the proposal involves a change in the technical specifications incorporated in the license or an unreviewed safety question. Section 50.59 further describes the considerations made to determine if an unreviewed safety question is involved. It is the staff's opinion, that design changes and revisions to constants in memory directly affect the margin of safety of the protection system and therefore are subject to staff review. We will require that all design changes and revisions to constants in memory be administratively controlled in strict accordance with technical specifications to be issued by the staff.

D.5 Uncertainties of Minimum DNBR Synthesis - Topical Report Evaluation

Report Number:	CENPD-170 (Supplement 1)
Title:	CPC - Assessment of the Accuracy of PWR Safety System Actuation as Performed by the Core Protection Calculators
Report Date:	November 1975
Originating Organization:	Combustion Engineering, Inc.

Summary of Topical Report

The topical report CENPD-170 describes the methods used in the Combustion Engineering (CE) core protection calculator system (CPCS) to synthesize the three-dimensional peaking factor (F_q). The resulting F_q is used in conjunction with other measured parameters to determine a minimum departure from nucleate boiling ratio (DNBR). Supplement 1 of CENPD-170 provides additional discussion of the uncertainties associated with the synthesis of the minimum DNBR. Related material has been provided in the response to a series of questions asked by the staff on the Arkansas Nuclear One, Unit 2 (Docket No. 50-368).

The CPCS has been proposed for use to assure that the specified acceptable fuel design limits on DNB and fuel centerline melt are not exceeded during anticipated operational occurrences and to assist the engineered safety features in limiting the consequences of certain postulated accidents. Primary system variables are measured and processed through algorithms to yield synthesized values of minimum DNBR and local power density. These inferred values are compared with specified trip set points to assure that fuel design limits are not exceeded during any condition in normal operation, including the effects of anticipated operational occurrences. The topical report CENPD-170 and supplementary material deal primarily with the trip functions related to the limitations on departure from nucleate boiling and centerline fuel melt. The uncertainty in DNBR and local power density due to application of the CPC process is determined by comparing the FORTRAN-coded algorithms with standard design codes.

The procedure used to access the core minimum DNBR involves the snythesis of a hot pin and hot channel power distribution which is used in conjunction with values of primary system process parameters to calculate DNBR. The determination of DNBR is done on both a static level and is updated between calculations using dynamic calculations based on conservative values of partial derivatives. The static calculation uses standard thermal-hydraulic correlations and the W-3 correlation for DNBR; the static calculation is the primary analysis used to evaluate the accuracy of the CPC analytical procedure.

Static DNBR Synthesis

A description and assessment of the accuracy of the algorithm used to calculate static DNBR is provided in Supplement 1 of CENPD-170 and is the subject of this evaluation.

The CPC's employ the CPCTH code to compute the hot channel minimum DNBR, and the limiting void fraction. Inputs to the algorithm include the core power, coolant temperature at core inlet, primary system pressure, core average coolant mass velocity, integrated radial peaking factor, and the normalized hot pin axial power distribution. The code is a simplified version of the Combustion Engineering thermal hydraulic code COSMO, and includes bias factors which are applied to the input to force the output to be in good agreement with results from the COSMO code. The bias factors were obtained and evaluated by comparing the calculated over-power margin from CPCTH to that for the same reactor core and coolant conditions as calculated by the BULL code for several hundred cases representing a broad range of process inputs for BOL, MOC, and EOC core conditions. The BULL code and COLSS are discussed in topical report CENPD-169. BULL is a fast running simplified version of COSMO for use in the COLSS software system. CPCTH is a further simplification of BULL to obtain faster running time with some sacrifice in accuracy.

The CPCTH algorithms are fit to nine distinct operating regions for convenience in defining partial derivatives used in update algorithms which are defined on the values of inlet coolant temperature, coolant pressure, axial shape index, and integrated radial peaking factor. The static DNBR calculation is performed as follows:

(1) The region dependent multiplicative power uncertainty factor is used with the equivalent power correlation (EPC), to adjust the value of core power input to CPCTH. The region is determined based on the values of the process variables.

(2) A normalized mass velocity profile is determined as a function of the integrated radial peaking factor and nodal fluid quality. A single valued multiplier (the equivalent mass velocity multiplier, GM) is calculated as a function of the process variables, the core average mass velocity and the axial shape index. The product of the normalized mass velocity profile, the core average flow, and the GM produce the axial mass velocity distribution used in the evaluation of the thermal limits.

(3) Calculation of the thermal limits including evaluation of the W-3 correlation is performed using the mass velocity profile from (2).

The above calculation over 20 axial nodes generates the minimum static DNBR using periodic snapshots of the process parameters. A trip signal will be generated if the margin to one or more of the following limits is calculated to be less than or equal to zero:

(1) DNBR limit (1.3),

(2) Fluid quality limit (W-3 correlation), and

(3) Mass velocity dependent void fraction limits for flow stability.

Other trip signals based on out-of-range values for certain process parameters have been identified in CEN-44(A) "Core Protection Calculator Functional Description," January 7, 1977, but are not discussed in CENPD-170.

A trip is also generated when individual sensor values are beyond the limits for which the corrections and algorithms are valid.

Uncertainty Analysis

The subject report and supplementary information describe the analysis performed to assess the accuracy with which the synthesized DNBR is inferred from the CPC power distribution and thermal margin algorithms. The procedure involves the comparison of the DNB overpower margin inferred by the CPC to the minimum overpower margin as calculated by the design thermal margin codes COSMO and a related code, BULL.

To ensure that the design objective of the high local power density trip is accomplished, the trip setpoint must account for errors associated with measured input, model uncertainties and calculational uncertainties associated with use of the CPC algorithms. Sensor measurement and calibration error are considered separately in the final setpoint determination established during pre-operational testing.

The assessment of the DNB accuracy was performed by comparing the overpower margin calculated by the CPCTH algorithm to that calculated by BULL and/or COSMO for the same range of reactor conditions. The purpose of the analysis was to determine a factor to account for the uncertainty of CPCTH relative to the COSMO design code and a second factor to account for the uncertainty in the power distribution synthesis. These factors are combined to give a net uncertainty factor to apply to the CPC method of synthesizing the DNBR.

The subject report includes the results of an analysis to determine the magnitude of the uncertainty factor to be applied to provide a 95/95 probability/confidence level that the actual DNBR is larger than the CPC synthesized value.

Summary of Staff Evaluation

The Core Protection Calculators compute thermal hydraulic conditions in the hot channel using a snapshot of both directly monitored and calculated input values. The combustion Engineering standard design code for computing DNBR is COSMO, an open hot channel thermal margin code. The CPC's use a simplified closed channel fast running version of this code, CPCTH. Since the CPCTH code is derived from and justified with respect to the design code COSMO, review and approval of the topical report on TORC and COSMO is a contingency for acceptance of CPCTH.

That report has been reviewed and final approval is contingent upon satisfactory comparison of TORC with data from an operating reactor as stated in the NRC Staff Topical Report Evaluation, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core" (CENPD-161), September 7, 1976. However, the staff has concluded that the codes may be used in licensing applications prior to review of the operating reactor data.

The closed hot channel calculation in CPCTH does not take into consideration the turbulent crossflow between the hot channel and the neighboring channels. To account for this in CPC input, an adjustment is made to the mass velocity input to the applicable algorithm. The adjustment to the mass velocity is made such that when all other system conditions are the same, the closed channel DNBR equals the COSMO minimum DNBR.

Regulatory Position

Based on our review of the subject topical report and related information, the staff considers the analytical tools and procedures acceptable for synthesis of the static departure from nucleate boiling ratio.

D.6 Staff Independent Calculations

In connection with the staff review of the Core Protection Calculators software, an independent analysis has been made of the four-pump loss of flow transients for Arkansas Nuclear One-Unit 2 (ANO-2). The staff version of RELAP 3B and COBRA-IIIC were used with design details applicable to the ANO-2 plant design.

The purpose of the calculation was to determine the margin to DNB for a top peaked axial power shape as proposed by Combustion Engineering. The RELAP 3B code was used to evaluate the flow coastdown data. The flow coastdown as calculated by RELAP matched the hot channel flow as predicted by Combustion Engineering.

The COBRA-IIIC code was used by the staff for the calculation of minimum DNBR. The relevant input parameters for the COBRA analysis include a 1.47 to peaked axial flux shape, a radial peak-to-core average power of 1.61 (hot pin), and a coolant inlet temperature of 553.5 F. The COBRA analysis for the top axial peak used the heat flux decay curve supplied by Combustion Engineering as input for surface heat flux. The heat flux decay curve as supplied by Combustion Engineering assumes a total trip delay time of 0.75 seconds.

Figure D-3 compares the staff result from minimum DNBR versus time for the top axial peak with the results of the Combustion Engineering analysis. The staff calculation was initiated at a DNBR of 1.515 and reached a minimum of 1.315 at 2.0 seconds. Based on the staff audit calculations, the Combustion Engineering prediction of minimum DNBR for the top axial peak is acceptable.

Figure D-3

DNBR Transient
Four Pump Loss of Flow
Top Axial Peak

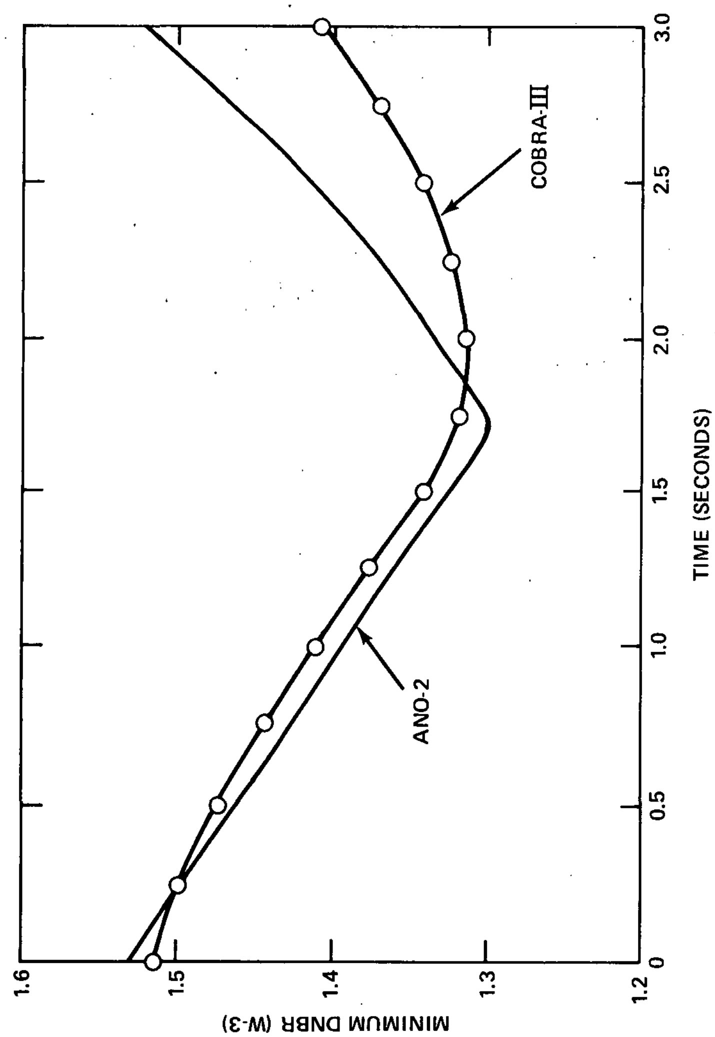


Table D.1

Core Protection Calculator System
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M2. "Minutes of Meeting, Core Protection Calculator Meeting with Combustion Engineering, August 6, 1975," to V. Stello, Jr., Division of Technical Review.

M3. "Minutes of Working Level Meeting, Core Protection Calculators, Combustion Engineering," dated September 17, 1975, to T. A. Ippolito, Division of Technical Review, NRC.

M4. "Trip Report: CPC Review, Combustion Engineering, October 6 and 7, 1975," (Proprietary) dated October 17, 1975, to T. A. Ippolito, Division of Technical Review, NRC.

M5. "Minutes of CPC Test Meeting, October 21, 1975," dated October 28, 1975, to T. A. Ippolito, Division of Technical Review, NRC.

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