SAFETY EVALUATION OF THE BRUNSWICK STEAM ELECTRIC STATION UNITS 1 AND 2

Docket Nos. 50-324
50-325

U.S. ATOMIC ENERGY COMMISSION
DIRECTORATE OF LICENSING
WASHINGTON, D.C.

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SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

CAROLINA POWER AND LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT

UNITS 1 AND 2

DOCKET NOS. 50-324 and 50-325

November 1973

ABBREVIATIONS

a-c	alternating current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AEC	United States Atomic Energy Commission
AISC	American Institute of Steel Construction
ANS	American Nuclear Society
ANSI	American National Standard Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AWS	American Welding Society
BNL	Brookhaven National Laboratory
BSEP	Brunswick Steam Electric Plant
$BTU/hr\text{-}ft^2$	British thermal units per hour per square foot
BTU/lb	British thermal units per pound
BWR	boiling water reactor
cfm	cubic feet per minute
cfs	cubic feet per second
Ci/sec	curies per second

CP&L	Carolina Power and Light Company
CRD	control rod drive
CSC	core standby cooling
CSS	core spray system
DBA	design basis accident
DBE	design basis earthquake
d-c	direct current
ECCS	emergency core cooling system
ft	feet
°F	degrees Fahrenheit
FAA	Federal Aviation Administration
FSAR	Final Safety Analysis Report
g	acceleration, 32.2 feet per second per second
GDC	AEC General Design Criteria For Nuclear Power Plant Construction Permits
GE	General Electric Company
gpm	gallons per minute
HEPA	high efficiency, particulate, absorber
HPCI	high pressure coolant injection
IEEE	Institute of Electrical and Electronics Engineers
in	inch
k_{eff}	effective multiplication factor (for the nuclear fission process)
$\Delta k/k°F$	reactivity ($\Delta k/k$) change per degree Fahrenheit

kV	kilovolts
kW/ft	kilowatts per foot
lb	pound
LOCA	loss-of-coolant accident
LPCI	low pressure coolant injection
LPZ	low population zone
MCHFR	minimum critical heat flux ratio
m	meters
mph	miles per hour
mrem	millirem
MSL	mean sea level
MSLIV	main steam line isolation valve
MWD/STU	megawatt days per short ton of uranium
MWe	megawatts electrical
MWt	megawatts thermal
NDT	nil ductility transition
NOAA	National Oceanic and Atmospheric Administration
NPSH	net positive suction head
nvt	neutron fluence, neutrons per square centimeter
NSSS	nuclear steam supply system
OBE	operating basis earthquake
PBF	Power Burst Facility
PHS	Public Health Service

PMF	probable maximum flood
PSAR	Preliminary Safety Analysis Report
psi	pounds per square inch
psid	pounds per square inch differential
psig	pounds per square inch gauge
QA	quality assurance
QC	quality control
R&D	research and development
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
Rem	Roentgen equivalent man
RHR	residual heat removal
RPS	reactor protection system
sec	second
SRB	safety review board
SGTS	standby gas treatment system
UE&C	United Engineers and Constructors
χ/Q	atmospheric diffusion factor (sec/m^3)
w/o	weight percent
10 CFR	AEC, Title 10, Code of Federal Regulations
Part 2	AEC Rules of Practice

Part 20	AEC Standards for Protection Against Radiation
Part 50	AEC Licensing of Production and Utilization Facilities
Part 100	AEC Reactor Site Criteria

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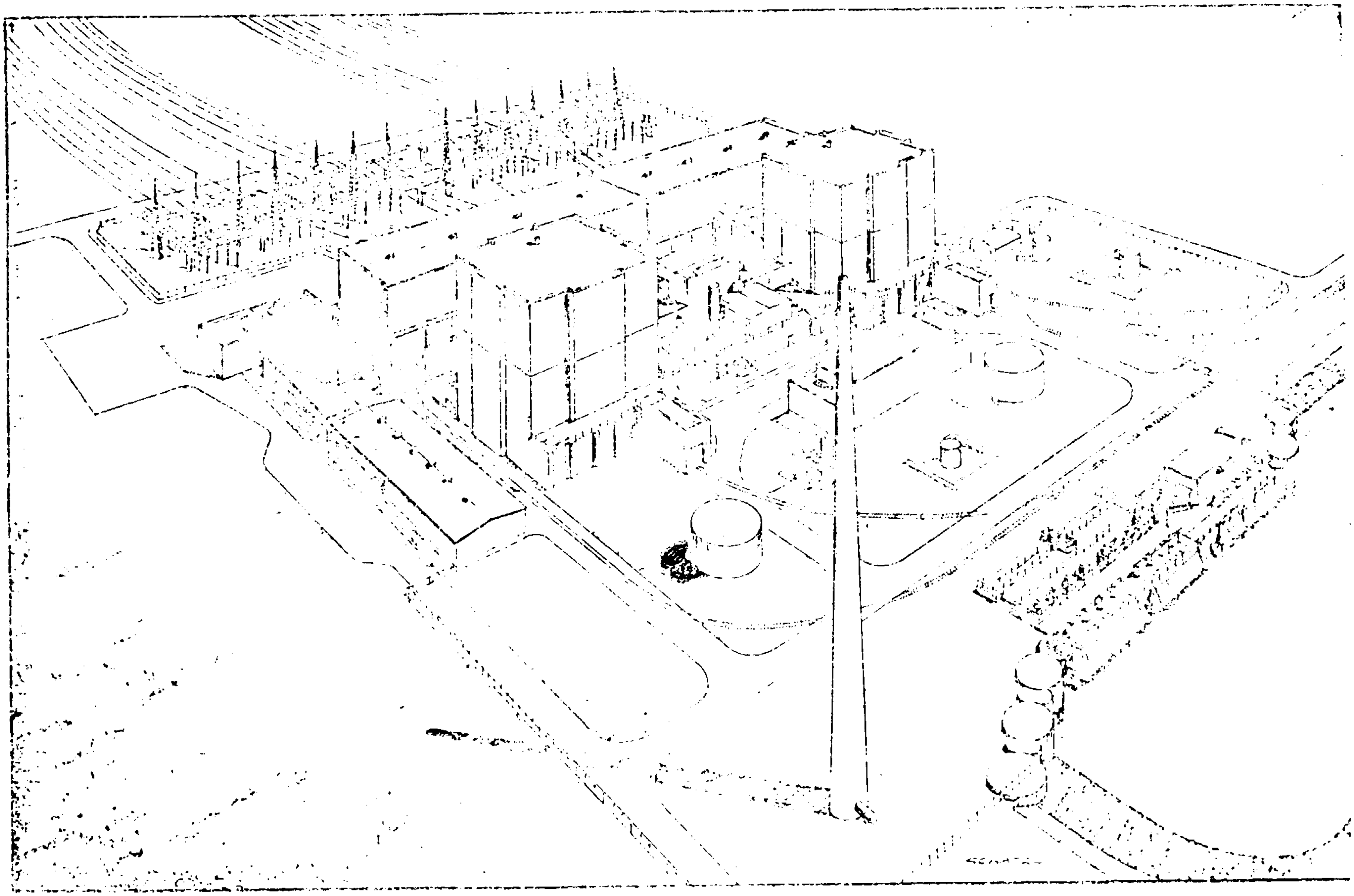

FIGURE 1-1 ARTIST'S CONCEPT OF BSEP PHYSICAL ARRANGEMENT

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

This Safety Evaluation Report was prepared by the Atomic Energy Commission's Directorate of Licensing. It is an evaluation of the Carolina Power and Light Company's (CP&L) application for licenses to operate the Brunswick Steam Electric Plant, Units 1 and 2. CP&L as owner and applicant is responsible for the design, construction, and operation of Brunswick Steam Electric Plant (BSEP) Units 1 and 2.

The application for construction permits was filed by CP&L on July 31, 1968. Following an extensive review by the Directorate of Licensing and by the Advisory Committee on Reactor Safeguards and following completion of a public hearing, Provisional Construction Permits CPPR-67 and CPPR-68 were issued on February 7, 1970 for BSEP Units 2 and 1 respectively. On October 3, 1972, the applicant filed Amendment No. 12, to the application, which is the Final Safety Analysis Report (FSAR).

The BSEP consists of two boiling water reactors located on a 1200 acre site in Brunswick County, North Carolina. The reactors are designed to operate at rated core power levels of up to 2436 MWt. BSEP Unit 2 (CPPR-67) is the same as Unit 1 (CPPR-68) except for the turbine bypass capacity. Unit 2 will have a 100% bypass capacity while Unit 1 will have a 25% bypass capacity. Unit 2 is scheduled to be completed and ready for fuel loading in August 1974. Unit 1 is approximately one year behind the Unit 2 schedule. Figure 1-1 is an artist's concept of the BSEP physical arrangement.

A technical safety review of Units 1 and 2 has been performed by the staff of the AEC's Directorate of Licensing based on the applicant's FSAR and Amendments No. 13 thru 23. During our review of this application, we requested the applicant to provide additional information for use in our evaluation. This additional information was provided in amendments, design reports, and specific responses by letter to staff positions. We also held numerous meetings and conference telephone calls with the applicant to discuss and clarify the technical information submitted. As a result, we requested a number of changes to be made in the design and planned operation of both Units 1 and 2. These changes or modifications are described in Amendments (No. 13 thru 23) to the FSAR and letter responses to various stated requirements of the Directorate of Licensing. The FSAR and its amendments, and letter responses by the applicant have been made available for review by members of the public at the Atomic Energy Commission's Public Document Room (PDR) at 1717 H Street, N.W., Washington, D.C. and at the local PDR located in the Southport-Brunswick County Library at 109 W. Moore

Street, Southport, North Carolina 28461. The applicant has submitted its Industrial Security Plan and certain design information on the nuclear fuel, instrumentation and electrical drawings, and air ejector off-gas treatment system as proprietary documents. We have determined that these documents may be withheld from public disclosure under the Commission's Rules and Regulations, 10 CFR Parts 2.790(d) and 9.5(a)(4). Accordingly, these documents will be withheld from public disclosure in accordance with the provisions of Section 9.10 of 10 CFR Part 9.

A chronology of the review by the Regulatory staff since the application was filed on July 31, 1968, is included in Appendix A of this report.

1.2 General Plant Description

Units 1 and 2 of the BSEP will each have a nuclear steam supply system (NSSS) which includes a boiling water reactor. Each NSSS will have twenty jet pumps supplied by two recirculating water lines, four main steamlines, and two feedwater lines. Fuel rods for the reactor will contain slightly enriched uranium-dioxide (UO_2) in sintered ceramic pellets. Some of the fuel rods will have ceramic fuel pellets that contain gadolinium-oxide (Gd_2O_3) in a mixture with the uranium-dioxide. These fuel rods will contain gadolinium in both full and partial length sections. The gadolinium serves as a "burnable poison" designed for power pattern and reactivity control and permits better fuel economy and elimination of the boron curtain neutron absorbers found in older plants. The fuel pellets are enclosed in Zircaloy-2 cladding tubes which are evacuated, backfilled with helium, and sealed by welding Zircaloy end plugs in each end. A fuel channel will enclose a bundle of 49 fuel rods in a 7 x 7 array; the channel is made of Zircaloy-4. Water flowing through the core serves as both a moderator of neutrons and as a coolant. Movement of water and a two phase water-steam mixture through the core is accomplished by the driving force from the 20 jet pumps (10 per recirculation line) and 2 recirculation pumps and from convective forces. Steam from the boiling process in the reactor core is demoisturized and dried, then vented through the four main steamlines to the turbine-generator system where its energy is converted into electricity. The steam then exhausts to a condenser located beneath the turbine where the condensate is collected and ultimately returned through a clean-up system for recycling through the reactor vessel and core. The cooling water for the turbine steam condenser is supplied by a once through system that take water from the Cape Fear River and discharges the water via a discharge canal to the Atlantic Ocean.

An off-gas treatment system consisting of a recombiner, condenser, moisture separator, and cryogenic system provides for retention of noble gases for decay to acceptable concentration levels prior to release with the plant's exhaust from the 100 meter stack.

The reactor coolant pressure boundary includes the reactor vessel, the two recirculation lines, and main steamlines, feedwater lines, and branch lines to their outermost isolation valves. Enclosing the reactor system is the primary containment structure of welded, inspected, and pressure-tested steel lined reinforced concrete in a light-bulb configuration called the "drywell." Beneath and around the base of this "drywell" structure is the steel lined reinforced concrete torus shaped "wetwell", constructed to the same standards as the drywell. The wetwell is connected to the drywell via downcomers and vents to permit the passage and condensation of any steam (vapor suppression) that may be accidentally discharged into the drywell, thereby limiting the pressure buildup below the containment maximum design pressure of 62 psig. Piping restraints have been designed and installed within the containment to limit the movement of piping during its postulated post-rupture movement (pipe whip). A hydrogen control system for containment atmosphere dilution (CAD) with nitrogen is provided for the normal operational containment inerting and for any post-LOCA (loss-of-coolant accident) needs. Isolation of the primary containment occurs automatically whenever there exists a potential for the uncontrolled release of radioactivity. For instance, the primary containment and the nuclear steam supply system are isolated and shut off, respectively, for the unusual conditions of low water level in the reactor vessel, high radiation level in main steamline, main steamline high flow or low pressure, primary containment high pressure, and many other conditions described in Section 7 of the FSAR.

The reactor protection system (RPS) provides the means to protect against conditions that may cause fuel failures or a breaching of the nuclear system process barrier, thereby limiting uncontrolled releases of radioactivity. The RPS initiates a reactor scram following an abnormal operational transient or pressure pulse, or following a gross failure of fuel or the nuclear system process barrier. The RPS is a reliable system designed to meet the standards specified in IEEE-279. Limits for RPS function are set forth in the Technical Specifications.

Normal reactivity control or rapid scram (shutdown) of the reactor is achieved by the bottom-entry cruciform-shaped control rods (neutron absorbers) that are moved vertically in the spaces between fuel assembly channels by a hydraulic mechanism; water is the hydraulic fluid, and for rapid insertion, nitrogen under pressure in an accumulator provides the driving force. Each control rod is independent of the other rods and has its own control and hydraulic system. A rod worth minimizer (RWM) is available to control positive reactivity insertion over a certain power range. To limit the effect of the reactivity insertion following a postulated control rod drop accident, the applicant will install the rod sequence control system (RSCS) or other method finally prescribed and approved by the Regulatory staff. A standby liquid control system is also available for use in injecting a boron solution into the reactor for emergency, long-term reactivity control.

Engineered safety features provide the capability to isolate containment, shut down the reactor, restrict radioactivity releases to acceptable levels, provide for heat removal for long-term core cooling, and condense steam within the primary containment. Details on these engineered safety features are presented elsewhere in this Safety Evaluation.

The reactor building (RB) encloses the reactor and its pressure-suppression type primary containment system. The reactor building houses the refueling and reactor servicing equipment, fuel storage areas, auxiliary equipment, core standbv cooling system, reactor cleanup filter demineralizer system, standby liquid control system, control rod drive system, the RPS, electrical equipment, heating and ventilation equipment, and the standby gas treatment system (SGTS). Operation of the SGTS will produce a negative internal pressure after building isolation such that the RB atmosphere is filtered and discharged via the SGTS and plant stack. Other structures such as the turbine building, the control building, the administration building, pump house, the intake structure and pumping facility, and 100 meter stack are described in varying detail in this evaluation but are also amply covered in appropriate sections of the FSAR and its amendments.

1.3 Interaction Between Units 1 and 2

The BSEP is a two unit plant which has been designed for both units to share certain facilities. Both units will include an 821 MWe reactor design of the BWR-4 class and will be designed and constructed as a dual unit nuclear plant.

The applicant has listed the systems and structures which are shared between Units 1 and 2 in Appendix B of the FSAR. We have reviewed the safety implications of the shared structures, systems and equipment and conclude that plant safety is not compromised by the sharing of those items indicated in Appendix B of the FSAR. All systems, equipment, or structures that require Unit 1 equipment to provide redundancy are being constructed with Unit 2 and will be completed prior to startup of Unit 2. The specific critical shared equipment, such as the shared onsite emergency diesel generator, are discussed in appropriate sections of this report.

Criteria are set forth in the FSAR that provide for physical separation to prevent radiation exposure of construction personnel working on Unit 1 and to prevent unauthorized personnel from entering key operating areas of Unit 2. Additional criteria provide for separation of electrical, ventilation, and water systems to assure that the continuing Unit 1 construction activities will not affect Unit 2 operations. We find the procedures and commitments for separation to be acceptable.

1.4 Comparison with Similar Facilities

Many features of the design of BSEP Units 1 and 2 are similar to those we have evaluated and approved previously for other nuclear power plants now under final phases of construction or already in operation. To the extent feasible and appropriate, we have made use of our previous evaluations during our review of those features which are substantially the same as previously approved facilities. Where this has been done, the appropriate sections of this evaluation will include the identification of the other facilities involved. Table 1.6-1 in the FSAR provides a comparison of the principle design features of the BSEP with the Brown's Ferry Units 1, 2, & 3, Cooper, and Hatch 1 nuclear plants. Our Safety Evaluations for these other facilities are published and are available for public inspection at the AEC's Public Document Room at 1717 H Street, N.W., Washington, D.C.

1.5 Identification of Agents and Contractors

General Electric Company is furnishing the nuclear steam supply system for the BSEP Units 1 and 2 including the first fuel loadings and the turbine-generator for the station. For those items of the plant within its scope of work, General Electric has acted as procurement agent.

United Engineers and Constructors, Inc. (UE&C), is the architect-engineer firm for both units. In this capacity, UE&C has designed and provided the balance-of-plant systems.

Brown and Root, Inc. is the constructor for the BSEP Units 1 and 2.

Other consultants used by CP&L to perform or verify design concepts for the BSEP are listed in section 1.1.1.6 of the FSAR.

Based upon our discussions at meetings and on the responses to our information requests, we conclude that CP&L in conjunction with its contractors has a technically competent and safety-oriented engineering organization for the management of the design, construction, and operation of the BSEP Units 1 and 2.

1.6 Summary of Principal Review Matters

Our technical review and evaluation of the information submitted by the applicant considered the principal matters summarized below.

We reviewed the population density and use characteristics of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology and hydrology to determine that these characteristics had been determined adequately and had been given appropriate consideration in the plant design, and that the site characteristics were 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.

We reviewed the design, fabrication, construction, testing criteria, and expected 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 guidelines (i.e., Safety Guides and Regulatory Guides) and other appropriate codes and standards, and that any departure from these criteria, codes and standards have been identified and justified.

We considered the response of the facility to certain anticipated operating transients and postulated accidents. We judged that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered credible. We performed conservative analyses of these design basis accidents to determine that the calculated potential offsite 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.

We evaluated the applicant's plans for the conduct of plant operations (including the organizational structure and the general qualifications of operating and technical support personnel), the measures taken for industrial security, and the planning for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that the applicant will be technically qualified to operate the plant and will have established effective organizations and plans for continuing safe operation of the facility. In this evaluation, we also considered the effects of the Unit 1 construction activities on the operation of Unit 2.

We evaluated the design of the systems provided for control of the radioactive effluents from the plant to determine that these systems can control the release of radioactive wastes from the plant within the limits of the Commission's regulations (10 CFR Part 20) and that Technical Specifications assure that the facility will be operated in such a manner as to reduce radioactive releases to levels that are as low as practicable.

We evaluated the financial qualifications of the applicant, and the protection and indemnity agreements for the plant.

During the review numerous meetings were held with representatives of the applicant, its contractors, and its consultants to discuss the facility and the technical material submitted. Members of the Regulatory staff visited the site on several occasions for purposes of determining that the construction of the facility was in accordance with the provisions of the construction permit. A chronological listing of the meetings and other significant events is given in Appendix A to this evaluation. During the course of the review, either the applicant proposed or we requested a number of technical and administrative changes. These changes are described in various amendments to the original application and where significant, are discussed in appropriate sections of this report.

Many features of the design of BSEP Unit 1 and 2 are similar to those we evaluated and approved previously for other nuclear power plants now in the final phases of construction or in operation. The application, as amended, together with the PSAR and FSAR, as amended and supplemented, and other pertinent documents are available for public inspection at the U.S. Atomic Energy Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Southport-Brunswick County Library, 109 W Moore Street, Southport, North Carolina.

The applicant analyzed the safety systems for a reactor design power of 2550 MW rather than the rated power at 2436 MW. Our evaluation of the safety systems was also based on the 2550 design power, however, we expect the licensed power to be 2436 MW.

The topics of geology and seismology were previously reviewed by our consultants as a portion of our construction permit review for Units 1 and 2. We judged that it was not necessary that these matters be addressed again by our consultants. We have, however, addressed the matter of site geology and seismology in this report.

Based on our evaluation of the application for licenses to operate the plant, subject to satisfactory resolution of those items identified herein, we conclude that the Brunswick Steam Electric Plant Units 1 and 2 can be operated as proposed without endangering the health and safety of the public. Our detailed conclusions are presented in Section 22.

This review and evaluation of the BSEP facility near completion of construction is one of the several phases of our continuing review process. The applicant's preoperation test program and the start-up of the reactor will also be under the surveillance of the Regulatory staff. Following initial operation, surveillance will continue by the Regulatory staff to assure that the facility is operated in accordance with the provisions of the licensee.

This report considers both single unit and dual unit operation of the plant. Since Unit 1, the second unit to go into service, will not be completed until about a year after Unit 2 is completed, the conclusions at this time apply only to Unit 2. We expect to prepare a supplement to this report extending the conclusions to Unit 1 and to dual unit operations when Unit 1 is ready for licensing. We will report on any changes in facility design and will update this Safety Evaluation Report in a supplement prior to licensing Unit 1 for operation.

1.7 Facility Modifications Required as a Consequence of Regulatory Staff Review

During our review of the CP&L application for an operating license for BSEP, several areas of equipment design and systems design had to be modified for the staff to conclude that these areas were acceptable and in conformance with the Commission's General Design Criteria, Quality Assurance Criteria, and Regulatory Guides. Eleven of the principal areas in which the staff required design modification are listed below along with a reference as to where these matters are discussed in more detail in this report.

1. Turbine Building Ventilation System (see Section 11.1)
2. Rod Sequence Control System (see Sections 4.3.1.4 and 7.6)
3. Quality Assurance Plan for Operations (see Section 17.0)
4. Flood Protection for Safe Shutdown of Both Units (see Sections 2.4.5, 7.12 and 10.3)
5. Combustible Gas Control System (see Section 6.2.5)
6. Safe Shutdown Capability from Outside Control Room (see Section 6.4)
7. Removal of Engineered Safety Features "Blocking Circuits" on the Onsite Emergency Power Source (see Section 7.3.1)
8. Environmental Qualification of Equipment (see Section 7.9)
9. Fuel Shipping Cask Handling Crane (see Section 9.1.2)
10. Fuel Densification Effects (see Section 4.3)
11. Testing of Diesels for Reliability and Capacity (see Section 8.3).

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

The Brunswick Steam Electric Plant is located in Smithville Township, Brunswick County, North Carolina, about 2 1/2 miles north of the community of Southport. Figure 2-1 depicts its location with respect to the immediate environs, and Figure 2-2 shows the location of the plant within the plant exclusion area. The applicant has defined an exclusion area which is the area within 3000 feet of the plant gaseous effluent release points. That area is within property owned by the applicant and on that basis we conclude that the applicant has authority to determine all activities within the exclusion area, and that the designated exclusion area meets the requirements of 10 CFR Part 100 with respect to control.

The applicant has selected a low population zone (LPZ) radius of 2 miles. The nearest population center with more than about 25,000 people is Wilmington, North Carolina, and its nearest boundary is 16 miles from the plant. The distance to the nearest boundary of the population center is therefore at least one and one third times the low population zone radius, in compliance with 10 CFR Part 100 guidelines. Based upon the population within the LPZ and the applicant's Emergency Plan which provides for evacuation of the LPZ if required, we conclude that the 2 mile LPZ radius is acceptable.

2.1.2 Demography

The applicant has provided estimated 1974 population as a function of distance from the plant, as well as projected population figures for 10 year intervals to the year 2014. Figure 3 shows the applicant's estimated 1974 population surrounding the Brunswick Steam Electric Plant out to a distance of 50 miles. For reference, the cumulative population corresponding to a moderately populated area of 400 people per square mile is also shown. Comparison of the curves shows that the population around the plant is well below the reference curve indicating that the site is not heavily populated.

Wilmington, North Carolina, (the nearest population center) had a 1970 population of 46,169, a gain of 5% over the 1960 population of 44,013. Southport, the nearest community, grew from 2,034 to 2,220 in the 1960-1970 decade, an increase of 9%. Brunswick County's 1970 population of 24,223 represents a growth of 20% from the 1960 population of 20,278.

Recreational use of the seashores during the summer months is the principal cause of transient population in the vicinity of the plant. According to the applicant's figures, this increase amounts to about 10,000 people within 20 miles of the plant as a result of seasonal attractions.

2.1.3 Uses of Adjacent Lands and Waters

The Brunswick Plant is located in a predominantly rural area. Southport, N.C., with a 1970 population of 2,220, is located 2.5

miles south of the site. Major land use in the area is in agriculture with the land which is not devoted to farming consisting of undeveloped non-utilized marshes and woodlands. According to the 1969 Census of Agriculture, less than 16% of the land in Brunswick County is actually in farms, a 2% decrease from the 1964 census. These statistics also show a decrease in the number of milk cows in the county from 404 in 1964 to 363 in 1969. The main farm products in Brunswick and surrounding counties are corn, soybeans, tobacco, poultry, truck farm and dairy products.

The applicant reports that only 2.2% of the area within a three mile radius of the plant is planted in crops. The nearest dairy farm (45 cows) is located 11 miles to the north-northeast. A farm located near the exclusion area boundary, .75 miles from the plant, is the location of the nearest residence and the nearest family cow.

Major water uses in the vicinity of the plant are for navigation, fishing and recreation. The Cape Fear River is used primarily for ship traffic to Wilmington, N.C. Consumable water is generally taken from wells. The applicant estimates that recreational use of the lower Cape Fear area results in a population increase of about 10,000 people during the summer months. It further estimates that about 40,000 bushels of oysters are taken from the lower Cape Fear annually for transplanting to other areas. Statistics for the year 1965 show that

2.2 million pounds of fin-fish were landed in Brunswick county and other seafood, mostly shrimp, amounted to 1.9 million pounds.

2.2 Industrial, Transportation, and Military Facilities

There are no intensive uses of land near the site for industrial purposes. The applicant states that there are fertilizer, chemical, and synthetic fiber industries located some 22 miles north of the site. There are no highways, railroads, or navigable waterways which traverse the exclusion area. The nearest highways and secondary roads, shown on Figure 2, are about 4000 feet from the from the plant.

The Sunny Point Army Terminal, located north-northeast of the site, trans-ships munitions by transfer from trucks and railroad cars to ocean going ships. The closest point in the Cape Fear River channel where munitions could be shipped is about 2 miles from the plant and the closest distance from the Sunny Point Exclusion area line is about one mile. We evaluated the interaction of the army terminal on the Brunswick Plant during our review of the construction permit application. No new information has developed to modify our conclusion that there is not likely to be an adverse effect on the plant as a result of operations at the Sunny Point Army Terminal.

The Brunswick County airport, a small county owned field, is located about 4 miles southwest of the plant. Its single 3200-foot turf runway is used by private planes. There are no air taxi or military operations from the field. A recent check with the Federal Aviation Administration shows estimated annual operations of 22,000 from this airport.

We have reviewed the applicant's evaluation of the probability of an aircraft striking the plant as given in Amendment 24, and found some of its assumptions, such as the use of air carrier rather than general aviation data, inadequate. Nevertheless, our independent analysis indicates that the probability of a damaging aircraft strike on the plant is less than 10^{-6} $year^{-1}$, based on an accident probability (between 4 and 5 miles from the airport) of 1.2×10^{-8} $mile^{-2}$ $-year^{-1}$ for general aviation aircraft.[1] We, therefore, conclude that the probability of an aircraft crashing into the Brunswick Steam Electric Plant and causing offsite radiological consequences is so small that it does not present an undue risk to the health and safety of the public.

2.3 METEOROLOGY

2.3.1 Regional Climatology

The climate of the tidewater section of southeastern North Carolina is maritime in character, influenced to a large extent by the proximity of the Atlantic Ocean. Winters are relatively short and mild, while summers are quite warm and humid. The daily range of temperatures is not as great as that found at more inland locations.

[1] Testimony on Zion/Waukegan Airport Interaction by Darrel G. Eisenhut (USAEC). Docket 50-295.

Maritime tropical air masses predominate over the area during the major part of the year. In the winter, however, outbreaks of cold, continental polar air moving southward from Canada occasionally affect southeastern North Carolina. The cold air is often warmed to some extent before reaching the Carolina coast by the crossing of the Appalachian Mountains and the descent of the eastern slopes, and by the relatively long trajectory from the air masses' source region. High air pollution potential (atmospheric stagnation) may be expected on 2 days during the year. Atmospheric diffusion conditions are expected to be near the average for all sites in the United States.

2.3.2 Local Meteorology

The plant site is located about one and three quarters miles west of the Cape Fear River, 16 miles south of Wilmington, North Carolina. The site is in a rural area of generally flat terrain, interspersed with swamps and marshes. The Atlantic Ocean lies to the east and south of the plant at a distance of about six miles. During the period 1955-1967, 8 tornadoes have been reported within the one degree latitude-longitude square containing the site, giving a mean annual tornado frequency of 0.6, and a computed recurrence interval of 2200 years. The site lies within the potential Atlantic Hurricane track and also within the potential Probable Maximum Hurricane track for the Atlantic Coast. Climatological records for the period 1886 to 1966 indicate that the mean annual frequencies of tropical storms and hurricanes passing within 50 miles of the plant site are 0.6 and 0.3 respectively. The prevailing wind flow over the site is from the southwest.

2.3.3 Onsite Meteorological Measurements Programs

An onsite meteorological measurements program was initiated in September 1970. The program consisted of the instrumentation of and measurements from a 364-foot tower which is located about 1500-feet north-northeast of the Unit 1 reactor. The tower has wind instruments at the 44- and 350-foot levels and temperature instruments at the 35-, 200-, and 340-foot levels. Since one full year of data with a recovery rate of at least 90 percent was not available, the applicant submitted a composite year of data record in joint frequency form similar to that suggested in Regulatory Guide 1.23, to provide a basis for the staff's evaluation of atmospheric diffusion conditions. The composite year of data provided by the applicant consisted of data collected during the following months for winds at the 44-foot level; 10/1/72 - 1/5/73, 1/6/71 - 8/23/71 and 8/24/72 - 9/30/72. For winds at the 350-foot level, the composite year consisted of data collected during the periods 9/25/70 - 12/3/70, 12/4/72 - 1/5/73, 1/6/72 - 5/14/72, 5/15/71 - 9/10/71 and 9/11/72 - 9/24/72. For evaluation of building and vent releases, the joint frequency distribution of wind direction and speed measured at the 44-foot level (reduced to represent wind direction and speed measured at the 44-foot level and to represent wind speed at the 33-foot level), and vertical temperature difference (ΔT) between the

200- and 35-foot levels were used. For evaluation of releases from the plant's 100 meter (328-foot) stack, the joint frequency distribution of wind direction and speed measured at the 350-foot level and vertical temperature difference (ΔT) between the 340- and 30-foot levels were used. The data recovery rate for the composite year of record was at least 90 percent.

2.3.4 Short Term (Accident) Diffusion Estimates

In our evaluation of the diffusion of short term (0-2 hour at the site boundary and 0-8 hour at the LPZ) accidental releases from the plant's buildings and vents, a ground release model with a building wake factor, cA, of 800 m^2 was assumed. The relative concentration (χ/Q) which is exceeded 5% of the time was calculated to be 1.0×10^{-3} sec/m^3 at the minimum site boundary distance of 914 meters. This relative concentration is equivalent to dispersion conditions produced by Pasquill type F stability with a wind speed of 0.5 meters/second. The relative concentration which is exceeded 5% of the time at the outer boundary of the low population zone (3220m) was calculated to be 2.8×10^{-4} sec/m^3. The estimated relative concentration at the LPZ for the 8-24 hour period was 3.1×10^{-5} sec/m^3.

In our evaluation of accidental releases from the 100-meter (328-foot) stack, an elevated point source, modified for terrain height, was assumed. As described in Regulatory Guide 1.3, fumigation conditions were assumed to exist during the first four hours of the accidental release period. The 0-4 hour relative concentration

(χ/Q) was estimated to be 6.0×10^{-5} sec/m^3 at the nearest site boundary (914m), assuming fumigation conditions with a wind speed of 2 meters/second. At the low population zone distance, a relative concentration of 2.5×10^{-5} sec/m^3 was calculated assuming fumigation through a depth of 100 meters and a wind speed of 2 meters/second for the first four hours after an accident. For the time period from 4 to 8 hours after the accident, the relative concentration from an elevated (100m) release which is exceeded 5% of the time was calculated to be 2.0×10^{-6} sec/m^3. The estimated relative concentration for the 8-24 hour period was 7.4×10^{-7} sec/m^3, for the 1-4 day period was 2.2×10^{-7} sec/m^3 and for the 4-30 day period was 6.2×10^{-8} sec/m^3.

The applicant's relative concentration estimates for short time periods are generally less conservative than those of the staff by a factor of two or less. However, in the case of the 0-4 hour release from the 100 meter stack under fumigation conditions, the staff's estimate is greater by a factor of three. This difference is due to the applicant's assumption of a wind speed three times greater than that assumed by the Regulatory staff. Other differences may be attributed to different meteorological and mathematical assumptions used by the applicant and the staff.

2.3.5 Long Term (Routine) Diffusion Estimates

Computations of annual average offsite relative concentration for the stack release, considering plume rise as a function of wind speed and topography, showed a maximum value of 4.7×10^{-8} sec/m^3 northeast of the stack at the site boundary. The highest offsite annual average relative concentration of 6.4×10^{-6} sec/m^3 for vent releases occurred at the 914 meter site boundary distance south-southeast of the reactor complex.

The applicant's relative concentration estimates were about fifty percent less conservative than those calculated by the staff. The differences may be atrributed to different models used by the applicant and the staff.

?.6 Conclusions

The staff concludes that the meteorological data presented in the FSAR provide an acceptable basis for making conservative estimates of atmospheric diffusion for accidental and routine gaseous effluent releases from the plant.

2.4 Hydrologic Engineering

2.4.1 Hydrologic Description

The site is located in southeastern North Carolina near Cape Fear. The plant is on high ground about two miles north of Southport, North Carolina, and is between the Cape Fear River estuary on the east, and the Intracoastal Waterway on the south. Saline cooling water for the plant will be provided at a rate of about 1500 cubic feet per second per unit via an intake canal approximately 3.0 miles long from the Cape Fear River estuary to the east. The bottom of the intake canal is 170 feet wide at elevation +0.5 feet above mean sea level (msl).

Discharge is via a discharge canal approximately 5.5 miles long, a siphon under the Intracoastal Waterway, a pumping station on Caswell

Beach, and ocean discharge conduits which are approximately 3000 feet long. The discharge canal between the plant and the siphon is at elevation +5.0 feet msl at the plant and elevation + 4.5 feet msl at the siphon. It is noted that this cooling water system is the subject of environmental concern and may require future modification.

Plant grade is at elevation + 19.5 feet msl. Safety-related service water pumps and related equipment are contained in the Service Water Pump House adjacent to the outdoor circulating water pumps and traveling screens near the upstream end of the intake canal.

Water supply in the site region is taken from shallow aquifers. Low demand users draw from surficial ground water aquifers, while greater demand is satisfied from the slightly deeper regional Castle Hayne limestone source. Southport, 2 miles to the south, uses five wells at depths of 100 feet or more for municipal water supply, and the Sunny Point Army Terminal, 3 to 6 miles to the north, has five wells at depths of over 170 feet for the largest demand in the area of 1288 gallons per minute. In addition, numerous small private shallow wells exist in the vicinity of the plant, some of which have been deepened or relocated by the applicant as a result of plant construction.

2.4.2 Flooding

The primary sources of potential flooding have been considered in reviewing the safety of the plant; local heavy precipitation, hurricane-induced surges and wave action, and tsunamis.

At the request of the staff, the applicant analyzed the local flooding potential from precipitation of probable maximum severity (as defined by the National Oceanic and Atmospheric Administration). Grading around safety-related facilities is toward the intake and discharge canal. The applicant states that safety-related structures are protected against water levels to elevation 22 feet msl. The applicant has concluded, that runoff around safety-related structures should not pose a flood threat. In analyzing the runoff capability of roof drainage systems for safety-related structures, however, the applicant determined that the design of all buildings except the parapeted roof of the Reactor Building could either safely pass and/or store precipitation as intense as the probable maximum without a failure and resultant threat to safety-related equipment. To assure adequate drainage during periods of severe precipitation on the roof of the Reactor Building, the applicant has added four scuppers through the parapet walls.

Water level rises due to historical tsunamis (earth crustal movement-induced water waves) along the East Coast have been of very low magnitude and frequency. Consideration by the staff of the site location, historic tsunamis, the Atlantic Ocean potential for severe crustal movement, and the magnitude of the hurricane-induced design bases water level condition (as discussed below) indicate tsunamis should not pose a safety problem.

The design basis hurricane-induced high and low stillwater levels were established during the construction permit (CP) review at elevations 22.0 feet msl and -7.5 feet msl, respectively. These levels are based upon the estimated water levels, exclusive of wave action, that would occur during passages of a probable maximum hurricane (PMH)[1] to the south and north, respectively, of the plant. At the request of the staff, the applicant analyzed the wave conditions on safety-related facilities that could accompany the 22 foot msl surge level. The results of these analyses indicate the most severe wave action would be restricted to the canal, and that high ground levels would limit wave heights in the vicinity of exposed safety-related buildings, except the service water intake, to 1.6 feet. For the intake, the applicant has estimated waves 3 feet high. The resulting wave runup levels were estimated to reach a maximum elevation of 28.3 feet msl on the intake, and 25.6 feet msl on other exposed buildings.

2.4.3 Low Water Considerations

Safety-related water supply is taken via the intake canal to pumps in a pumphouse located adjacent to the outdoor circulating water traveling screens and pumps. The source of water is the Cape

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[1] A PMH is considered to be the worst hurricane reasonably possible of occurrence.

Fear River estuary and the adjacent Atlantic Ocean. As discussed in section 2.5, blockage of the wide intake canal sufficient to threaten the safety-related water supply is not considered physically possible. Similarly, blockage of the discharge sufficient to threaten safety-related water supply is also not considered credible because of the physical arrangement and elevations of discharge facilities. The applicant has provided for sufficient submergence on safety-related water supply pumps to prevent a PMH - caused loss of suction (minimum low water level of -7.5 feet msl).

2.4.4 Ground Water

Ground water at the site is contained in the surfical deposits, and in the deeper artesian Castle Hayne limestone formation. The surfical aquifer is discontinuous in the site area and is the primary water supply source for local residents. The deeper Castle Hayne aquifer is the major regional water supply such as used by the town of Southport, some 2.5 miles to the south.

Construction of major plant facilities (buildings and the canals) has resulted in puncturing the aquiclude between the two aquifers. The permeability of backfill around major structures was investigated at the request of the staff to determine whether a direct hydraulic connection with the lower Castle Hayne aquifer had been created adjacent to major plant buildings. The applicant's analysis of the canals indicates such a connection may have been created in the discharge and intake canals. The applicant's analysis of the backfill material around major plant structures indicates horizontal permeabilities of about 1.0 - 5.4 x 10^{-4} cm/sec. The applicant has concluded that the backfill does not create a hydraulic connection with the lower Castle Hayne aquifer. Because of the relatively high permeability estimates provided by the applicant, the staff believes the contrary; a hydraulic connection does exist such that a surface spill or subsurface leakage can reach the Castle Hayne aquifer.

In paragraph 11.5 of this report, a description is provided of the liquid radwaste storage tanks, which are designed as seismic Category I tanks. Based upon the design standards for the liquid radwaste tanks and the applicant's well water monitoring program, we conclude that adequate provisions are made to protect against liquid radwaste reaching the Castle Hayne aquifer through the hydraulic connections cited above.

2.4.5 Technical Specifications Necessary for Hydrologically-Related Events

The staff has taken a position that it would be prudent to shut the plant down before water could reach plant grade during severe hurricanes. The applicant has maintained that design of safety-related facilities includes provision for protection. However, the staff believes the implementation of emergency procedures is required in the event of severe hurricanes to assure the watertightness of exterior doors, to minimize the possible equipment failure which could occur during such an event, should the applicant's single water barrier design provisions not be adequate, would be extremely difficult from a practical standpoint. The staff, therefore, will require a provision in the plant's Technical Specification requiring a flood alert, referring to emergency procedures, when water levels exceed elevation 15 feet msl. In the case of PMH, this would allow a minimum of about 4 hours before water would cross plant grade (some six hours before maximum water levels would be reached) to implement emergency action. Examples of required action are: assuring all exterior accesses are closed and sealed, adequate diesel fuel oil supplies are protected, sandbagging of vulnerable areas may be undertaken, and any necessary emergency equipment is available and operational. The weather conditions during such a situation would be severe (high winds, rain, the likelihood of tornados in the area, etc.), but implementation of outdoor emergency procedures are considered reasonable if accomplished before maximum storm conditions occur.

The applicant has installed a control room water level alarm that is activated when the water level in the intake canal reaches elevation 17.5 feet msl. The staff will require the same technical specification to necessitate an orderly plant shutdown upon activation of the alarm. The requirement is prudent in view of the single line of defense inherent in the water barriers installed by the applicant. Failure of such barriers with the reactor at or near operating levels would allow a very limited time, during extreme weather conditions, for plant operating personnel to prevent a major accident. No other technical specification provisions are considered necessary for hydrologically-related events.

2.4.6 Conclusions

With adequate provisions for implementing flood emergency action and shutting the plant down during severe hurricanes, the staff believes adequate flood protection is assured. The staff also believes an adequate water supply is assured via the wide intake channel and low submergence requirements of safety-related service water pumps.

2.5 Geology and Seismology

At the conclusion of the construction permit (CP) application review, based on reports included in the Safety Evaluation Report (SER) by our advisors, the U. S. Geological Survey (USGS) and the Seismology Division of the Coast and Geodetic Survey (the Seismology Division is now a part of the USGS), we concluded that the applicant's appraisal of the geological and seismological aspects of the site were adequate. The staff also concluded that site foundation conditions, including the proposed engineered conditions were favorable for the construction and operation of the nuclear plant.

The foundation engineering aspects of the facility were reevaluated based on data obtained since construction began. As a result of this reevaluation it is the staff's opinion that there is no reason to change its conclusions stated in the SER following the CP review. Therefore, we feel that the geological, seismological, and foundation engineering aspects of the site are favorable for the operation of the Brunswick Steam Generating Plant, Units 1 and 2.

?.5.1 Basic Geologic and Seismic Information

..5.1.1 Regional Geology

This site is located on the southeastern coast of North Carolina within the Coastal Plain Physiographic Province approximately 100 miles southwest of the Fall Line, the topographic boundary between the Coastal Plain and Piedmont Provinces. The Piedmont Province consists of metamorphic and igneous rocks of Paleozoic age, the surface of which slopes toward the southeast. Southeast of the Fall Line the Piedmont rocks are overlain by the unconsolidated to semi-consolidated sediments of the Coastal Plain Province ranging in age from Cretaceous to Recent. These sediments range in thickness from 0 at the Fall Line to approximately 1500 feet at the site.

Within the Piedmont Province, and more than likely also present beneath the sediment in the Coastal Plain, are basins that were formed due to down faulting of great blocks during the Triassic Period of the

Mesozoic Era. The Triassic Basins are filled with sedimentary rocks that have been intruded by igneous dikes and sills. The nearest known Triassic Basin is the Deep River Basin, which is located approximately 120 miles northwest of the site.

Regional structures within the Piedmont Province generally trend in a northeast-southwest direction parallel to the trend of the province. Many of the structural characteristics of the Coastal Plain, however, are significantly different from those of the Piedmont Province. The dominant regional structures within the southeastern Coastal Plain, the Cape Fear Arch, and the southwest Georgia Embayment have major axes perpendicular to the northeastern trend of the Province. Furthermore, these structures appear to have been active through the Tertiary Period while activity in the Piedmont is thought to have ceased in early Mesozoic. The Brunswick site lies on the northeast flank of the Cape Fear Arch.

The deeply buried structural geology beneath the Coastal Plain, other than the broad features mentioned above, is not well known. Of particular significance to any site in the Southeastern United States is the structural geology that is responsible for the seismic activity in the vicinity of Charleston, South Carolina, including the very large 1886 Charleston earthquake. This activity is believed to be associated with a specific structural anomaly that is confined to the area in the vicinity of Charleston. Evidence, though limited, seems to indicate that the numerous earthquakes that have occurred in the Charleston vicinity are localized along the deepest part of the northwest trending Southeast Georgia Embayment. The Charleston area is approximately 150 miles south southwest of the site.

2.5.1.2 Site Geology

The Brunswick Steam Electric Plant is located about 2-1/2 miles north of Southport and 1-1/2 miles west of the Cape Fear River in southeastern North Carolina. The local terrain is flat at an elevation slightly exceeding +20 msl. The site is underlain by 5 to 20 feet of Pleistocene Pamlico fine sands and clays; 65 feet of Miocene Yorktown formation which consists of two units, the upper being predominantly a very plastic clay with some fine sand extending to a depth of about 50 feet, and the lower being essentially a medium to coarse, dense sand; Oligocene clay and sand with discontinuous beds or lenses of limestone from 80 to 115 feet depth; and the Eocene Castle Hayne Limestone from 114 feet depth to about 230 feet. The Peedee formation underlies the Castle Hayne limestone to a depth of about 600 feet. Older Cretaceous rocks are continuous to a depth of about 1550 feet where they overlie the basement.

During the CP review, to determine the extent of solution activity in limestone beneath the site, the applicant drilled a total of 36 additional borings through the Castle Hayne limestone at the locations of all Category I structures. Representatives from the staff and the U. S. Geological Survey (USGS) visited the site and examined surface geological features, rock cores, and surface exposures of the Castle Hayne limestone. The staff concluded at that time "that the solutioning process at the Brunswick site will not adversely affect the structural adequacy of the foundation in the event of a design basis earthquake, and that cavities do not exist beneath the plant area which would cause ground displacement." Since that time the applicant has stated that "Additional borings drilled during construction have confirmed our e earlier reported findings that there are no cavernous conditions in the underlying formations underneath the plant." Based on the available data we believe that our conclusion arrived at during the CP review is still valid.

As a result of its review of the PSAR, the USGS concluded that: "There are no known faults or other active geologic structures in the area that might localize seismicity in the immediate vicinity of the site. Structural details of the crystalline rocks that underlie the site, however, are very poorly known. Available data suggest that regional structural trends in these older rocks are to the north-paralleling those in the adjoining Piedmont Province to the west. Superimposed on these older northeastward trends are younger broad regional structures that trend northwestward. Although the site is located near the axis of one of these broad structural features, the Cape Fear Arch, there is no evidence to indicate that the structure has been tectonically active since about mid-Miocene time, or for the past several million years. Available evidence indicates that the numerous earthquakes that have occurred in the vicinity of Charleston are localized along the deepest part of the axis of the northwest trending Savannah (Southeast Georgia) Basin." The staff concurs with this conclusion.

2.5.2 Vibratory Ground Motions

During the CP review our consultant, the US Coast and Geodetic Survey concluded: "As a result of this review of the seismological and geological characteristics of the area around the plant site, the Coast and Geodetic Survey recommends that an acceleration of 0.08g resulting from an Intensity VI earthquake would be adequate for representing earthquake disturbances likely to occur within the lifetime of the facility. The Survey also recommends that an acceleration of 0.16g resulting from an Intensity VII earthquake would be adeqate for representing the ground motion from the maximum earthquake likely to affect the site. It is believed that these values would be adequate for designing protection against the loss of function of components important to safety." The staff concurred. At the present time there is no basis to change that conclusion.

2.5.3 Surface Faulting

The staff concludes based on the available information that there is no potential for surface or near surface displacement at the site.

2.5.4 Stability of Subsurface Materials

The site was investigated by numerous borings and laboratory tests prior to and during construction. The investigations have been adequate to define foundation conditions with a relatively high level of confidence.

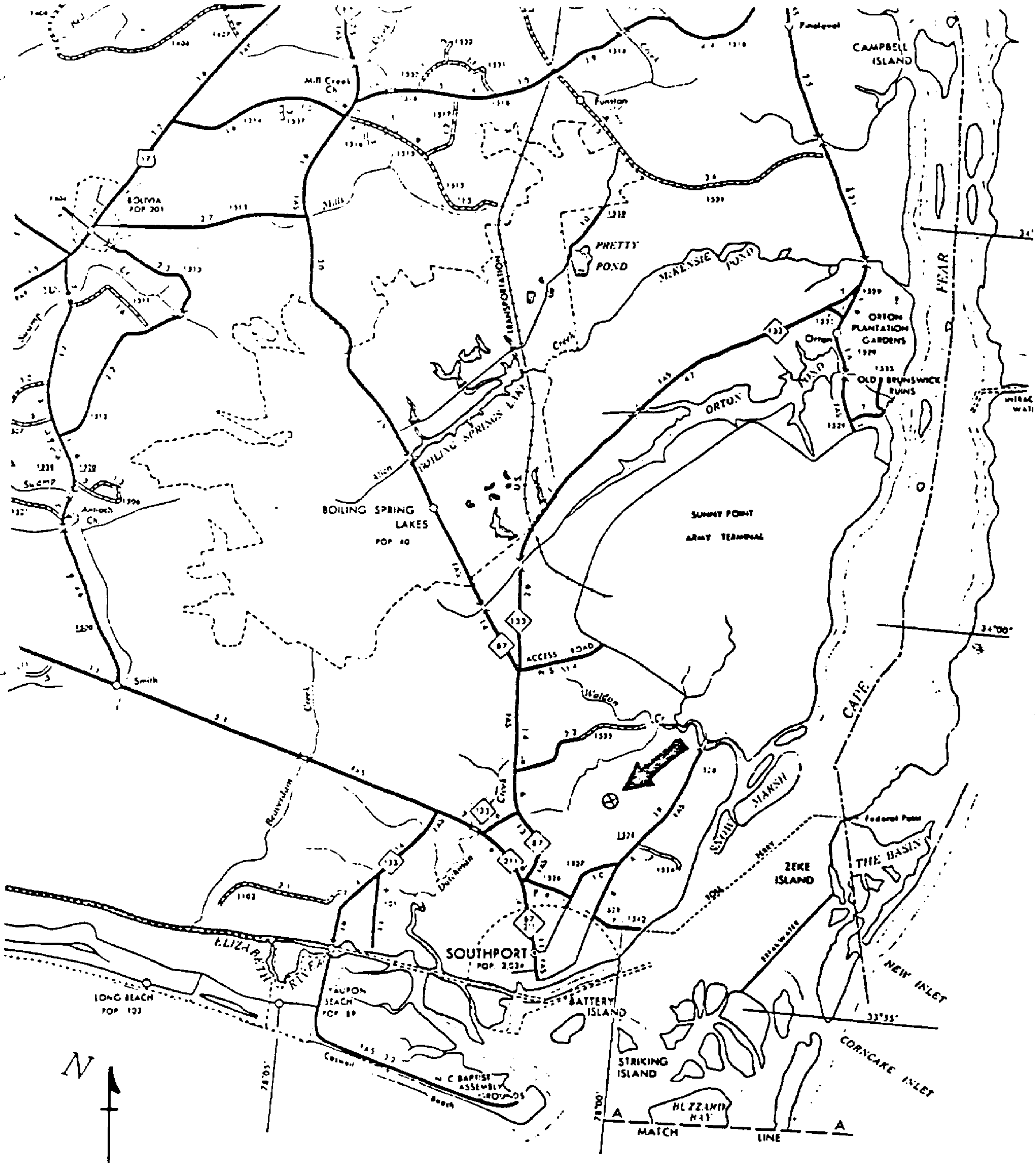
The site area was excavated to about -25 msl, which is within the dense Miocene sands of the Yorktown formation. The reactors' foundation mats were placed at that level. The other structures are founded on structural backfill sand compacted to relative densities as described in Appendix A of the PSAR. The staff concluded in its Safety Evaluation Report (SER) following the CP review that the in place foundation soil after excavation, and the engineered backfill were adequate to support the plant structures. We have completed our review of the results of the applicant's quality control program, which included close checking of borrow material selection, the testing of the adequacy of its placement and compaction, and final testing after completion of backfilling operations. We have also reviewed the results of the applicant's settlement monitoring program following construction of the major plant structures. Actual settlement agrees well with that which had been predicted. The applicant drilled additional borings after excavation to verify that there were no cavity or cavernous conditions within the Oligocene and Eocene limestone formations beneath the plant site. Based on these data, the staff concludes that the design requirements have been achieved and there is no reason to change our conclusions stated in the Safety Evaluation Report for the construction permit review.

2.5.5 Slope Stability

Service water is provided to the BSEP by the Intake Canal which extends from the Cape Fear River to the intake structure. Based on surveys, and borings drilled at 500 foot intervals along the canal route from the Cape Fear River to the intake structures, and from the discharge weir to the ocean outfall pipe, the applicant has divided the intake and discharge canal routes into marsh and high ground sections. Stability analyses were made at the most critical areas, which were determined to be those adjacent to ground elevations of +3 and +38 in the swamp areas and high ground areas respectively. These sections were analyzed using the following extreme conditions (1) Probable Maximum Hurricane (low water - 7.5), (2) the Operating

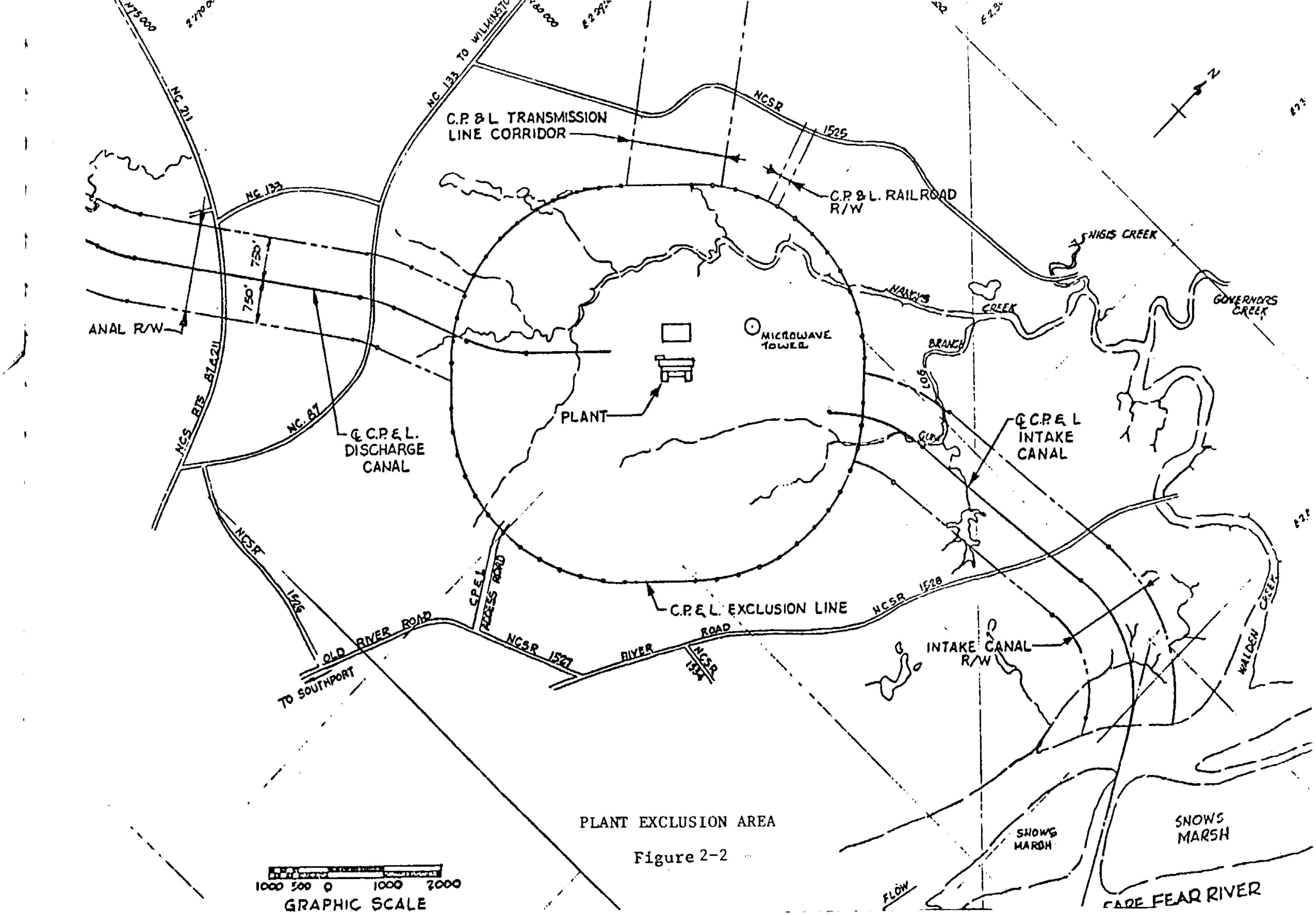
Basis Earthquake (.08g) + the Maximum Probable Hurricane, and (3) the Design Basis Earthquake (formerly DBE - now SSE). The analyses indicated that slope failure would occur under the PMH condition in the swamp area; under the OBE + the PMH in both the high ground and the swamp areas; and under SSE conditions, as much as 43 inches in the swamp areas and 14 inches in the high ground areas. Amounts of displacement were calculated using the technique described in Newmark, N.M., "Effects of Earthquakes on Dams and Embankments," Geotechnique, Vol. XV, No. 2, 1965. The analyses indicated that failures of this magnitude would not significantly affect the capability of the Intake Canal. In Appendix H of the SER for the Construction Permit, our consultants Dr. Newmark et al, stated that "These analyses indicate possible motions of as much as several feet; however, in view of the conservatism believed to exist in the calculations and recognizing that even movements of this magnitude would reduce the capacity of the canal to only a nominal extent, it must be concluded that this does not pose a problem of major significancies." It is believed that, because of its width (170' at the bottom and greater than 300' at the surface), the canal could accommodate a larger slope failure than those determined from the analyses. The cellular cofferdams adjacent to the intake structures were designed to remain stable under SSE and combined

OBE & PMH conditions. We conclude, based on the available data, that the design of the Intake Canal contains adequate margins of safety to preclude the loss of emergency cooling water to the Brunswick Steam Electric Plant.



PLANT LOCATION

Figure 2-1



PLANT EXCLUSION AREA

Figure 2-2

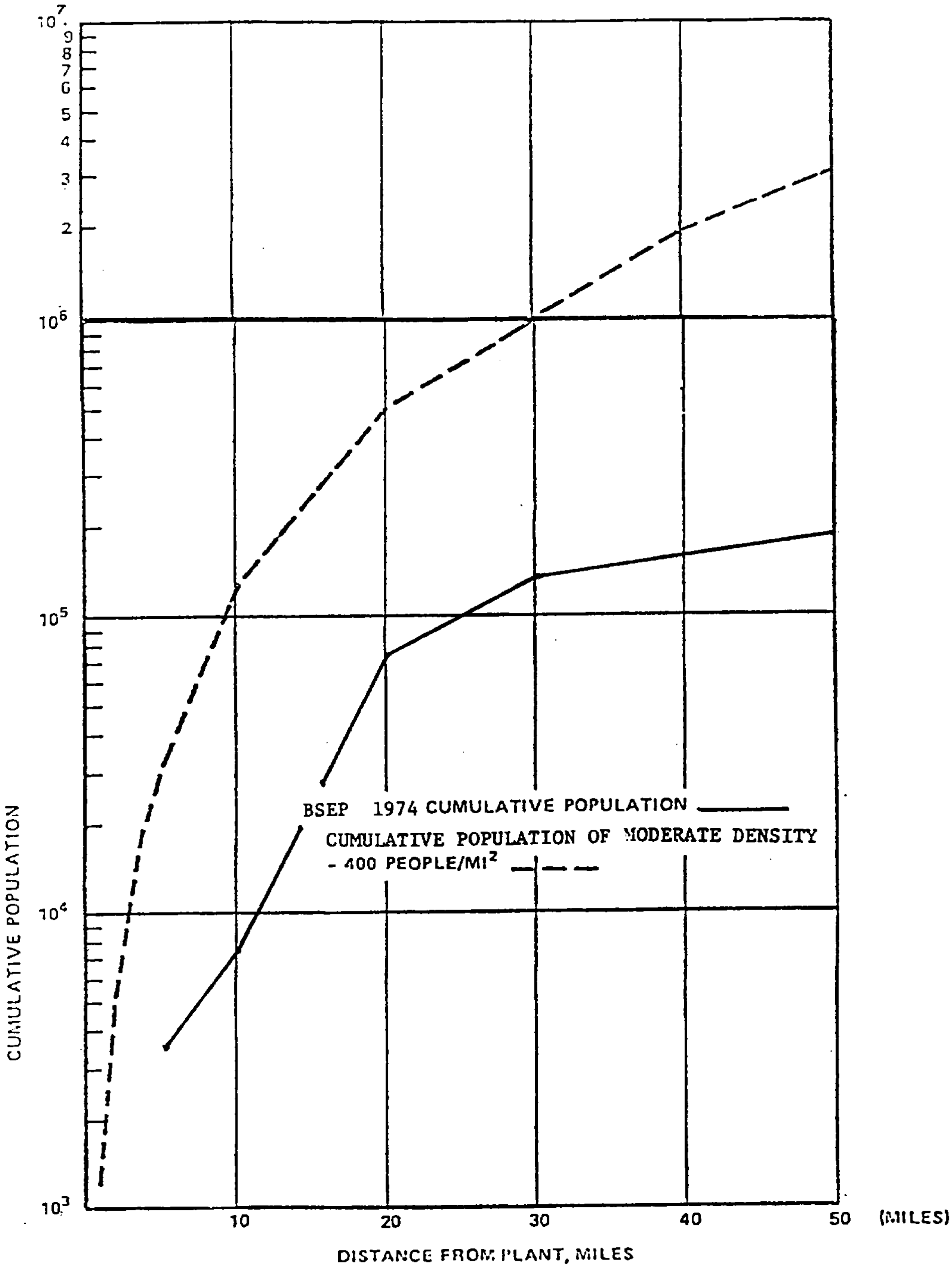


Figure 2-3 CUMULATIVE POPULATION DISTRIBUTION

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 Conformance with AEC General Design Criteria (GDC)

The plant design was reviewed for construction under the "General Design Criteria for Nuclear Power Plant Construction" issued for comment by the AEC in July 1967. The applicant provided an evaluation, Appendix F of the FSAR, of the design bases considering the GDC effective May 21, 1971 as amended July 7, 1971. Based on our evaluation of Appendix F and of the design of the plant we concluded that there is reasonable assurance that the intent of the GDC, published in the Federal Register on May 21, 1971 as Appendix A to 10 CFR Part 50, will be met.

3.2 Classification of Structures, Components, and Systems

3.2.1 Seismic Classification

The applicant has identified in Appendix C those Seismic Category 1 structures, systems and components important to safety that are designed to withstand the effects of the Safe Shutdown Earthquake and remain functional. 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.

All other structures, systems and components that may be required for operation of the facility are designed to Seismic Category II requirements. Included in this classification are those portions of Category 1 systems which are not required to perform a safety function. Structures, systems and components important to safety that are designed to withstand the effects of a Safety Shutdown Earthquake and remain functional have been identified in an acceptable manner. It is concluded that the design of these items in accordance with Seismic Category 1 requirements provides reasonable assurance that the plant will perform in a manner providing adequate safeguards to the health and safety of the public.

3.2.2 System Quality Group Classifications

The applicant has applied a Quality Group Classification System to those water and steam 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 with the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintenance in the safe shutdown conditions, and (3) to contain radioactive material.

For those fluid systems identified in the applicant's Classification Groups IA, IB, IIB, IIA and II, we and the applicant are in general agreement on the application of the Quality Group

Classification System. The applicant has identified in Appendix A those fluid systems or portions of fluid systems important to safety and the industry codes and standards applicable to each pressure-containing component in the systems.

Piping and Instrumentation Diagrams identify the boundary limits of each classification group within the fluid systems. Pressure-retaining components in fluid systems within the boundaries of the applicant's Quality Group Classifications will be built to meet the requirements of the applicable codes. Conformance with such codes is an acceptable basis for meeting the requirements of GDC and provides reasonable assurance that the plant will perform in a manner providing adequate safeguards to the health and safety of the public.

3.3 Wind and Tornado Loadings

The design wind velocity for the Category I structures is based on a recurrence interval of 100 years. With gust factors included, and considering the site to be a coastal area, the following wind velocities were computed by the applicant:

Height (ft)	Gusted Wind Velocity (mph)
0-50	130
50-150	150
150-400	180

The selection of the design wind velocity has been done on the basis of plant location. Velocity distribution and the gust selection have been made on the basis of the ASCE Paper #3269.

The wind pressures have been established by the applicant in accordance with ASCE Paper No. 3269. The same paper has been utilized to determine the loads acting in the structures. This procedure is standard in the industry, and has been used on previously licensed plants.

The design tornado for the Category I structures has, at all elevations, a 300 mph rotational velocity at the periphery and a translation velocity of 60 mph. The simultaneous atmospheric pressure drop is 3 psi for a duration of 3 seconds.

Tornado pressures and resulting forces have also been established on the basis of ASCE Paper No. 3269. For the tornado loads, the Category 1 reinforced concrete structures have been designed with a load factor of 1.0. For the tornado effects the steel framed structures have been designed on the basis of 150% of normal AISC allowables.

The use of these loading criteria provides reasonable assurance the structural integrity and safety function of Seismic Category I structure will not be impaired by the specified environmental forces. Conformance with these criteria is an acceptable basis for satisfying the requirements of AEC General Design Criterion #2.

3.4 Water Level (Flood) Design

The facility is designed for the highest flood level of about two feet above ground elevation (See Section 2.4.2). The hydrostatic pressure and buoyant forces have been computed in the usual way. The effect on the structure of the lateral forces has been considered.

The use of these design loading criteria provides reasonable assurance that the Seismic Category I structures can be expected to withstand the specified environmental forces without impairment of their structural integrity and safety function. Conformance with these criteria is an acceptable basis for satisfying the requirements of AEC General Criteria #2 and #4 as related to environmental design basis for structures.

3.5 Missile Protection

The plant's Category I structures have all been designed for protection against missile effects. The missiles considered are generally of two categories:

1) the externally generated tornado type, and
2) the equipment generated type.

They are in accordance with missile spectra used on previously licensed plants. The method used to assess the damage caused by missile impacts for the Category I structures is that of Amirikian in "Design for Protective Structures," which is based on elastic impacts. In general, the Category I structural portions that are designed to be missile resistant are proportioned to allow limited yielding with the deformations checked to ensure structural integrity.

The criteria used in the design of Seismic Category I structures provide a conservative design basis for determining the forces on the structure to assure that such impact forces will not penetrate structures and shields beyond acceptable limits as governed by the strength and resistance offered by such structures and shields.

3.5.1 Missile Protection Criteria

Potential impact damage from a spectrum of tornado-borne an internally generated missiles was considered by the applicant and the Regulatory staff in the design of essential facility structures and of essential equipment. We find that there will be no loss of function of essential systems or of seismic Category I structures from the effects of the spectrum of missiles considered.

The applicant initially considered four tornado borne missiles: a corrugated sheet siding, a bolted wood decking, a 4000 lb. vehicle, and a cedar fence post, in the design of the facility. In response to our request, the applicant added to include in his missile spectrum, certain additional items normally found at the site or which could be dislodged from structures by tornadic winds, and become missiles. These items included a 35 ft. long utility pole, a 1" solid steel pipe 3 ft long, a 6" schedule 40 pipe 15 ft long, and a 12" schedule 40 pipe 15 ft long.

The applicant's analysis assumed tornadic winds having a maximum tangential velocity of 300 mph. Using a missile characteristic parameter according to Characteristics of Tornado Missiles, Westinghouse WCAP 7897, by Paddleford, dated April 1969, and a horizontal acceleration according to Tornado Protection for the Spent Fuel Pool, General Electric APED 5696, by Miller, dated November 1968, the applicant concluded that none of the above cited missiles would impair the function of equipment essential for safe shutdown located in the Reactor Buildings, Diesel Generator Building, Control Building and Service Water Intake Structure. Missiles could penetrate the Spent Fuel Storage Buildings. However, it was concluded that the spent fuel storage racks would withstand an impact of 9000 ft-lb over a 3" or larger diameter without incurring damage to the fuel. None of the missiles analyzed when assuming a 20 ft drop through pool water after impact, would have this amount of associated energy, i.e., 9000 ft-lb of energy.

Based on the above analysis, we conclude the plant design for tornado missiles is adequate.

We requested the applicant to evaluate (for all tanks outside containment which contain gas under pressure) the stored energy, plant arrangement with respect to potential missiles and possible missile trajectory. Outside storage tanks were evaluated and the Modified Petry Formula used to determine the structural wall thicknesses needed to prevent missiles from striking equipment essential to safe shutdown. Where required, missile barriers were installed, e.g., missile barriers were installed near the diesel generator air receivers. We conclude that the analyses are acceptable and that the required remedial measures taken are adequate.

The applicant has provided a missile probability study based on Probability of Turbine-Generator Rotor Failure Leading to Ejection of External Missiles, a J. E. Downs Memo report, General Electric Company, February 22, 1971. It was observed that probabilities for missile generation and impact range from 10^{-10} to 10^{-13} per turbine-year, for striking critical areas such as the radwaste building, diesel generator building, service water intake building, control building, fuel pool, and even an open reactor vessel. We and the applicant will be studying this matter further and will follow technological developments in the area of in-service inspection techniques and will consider implementation of a suitable program when one becomes available. This action is consistent with that taken on other similar design-year OL reviews, e.g., Duane Arnold Energy Center.

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

The design of piping restraints as applied to the reactor coolant pressure boundary and to related systems of piping and components important to safety within containment provides adequate protection of the containment structure, the unaffected reactor coolant system components, and those systems important to safety which are either interconnected with the reactor coolant system, or in close proximity to the reactor coolant pressure boundary in which postulated pipe failures are assumed to occur as a design basis loss-of-coolant accident. The systems which were considered, the locations and types of piping breaks which might occur, and the protection measures against pipe whip provided are consistent with Regulatory Guide 1.46 "Protection Against Pipe Whip Inside Containment." The method of analysis used adequately accounts for the dynamic loadings that are associated with the pipe rupture postulate and will provide adequate assurance that the containment structure, unaffected system components, and those systems important to safety which are in close proximity to the systems in which postulated pipe failures are assumed to occur, will be protected.

These provisions for protection against the dynamic effects associated with pipe ruptures and the resulting discharging coolant provide adequate assurance that, in the event of the occurrence of the combined loadings imposed by an earthquake of the magnitude specified for the Safe Shutdown Earthquake and a concurrent single pipe break of the largest pipe at one of the design basis break locations, the following conditions and safety functions will be accommodated and assured:

1) the magnitude of the design basis loss-of-coolant accident can not be aggravated by potentially multiple failures of piping,
2) the reactor emergency core cooling systems can be expected to perform their intended function,
3) the containment structure's leak-tight integrity can be expected to be maintained in order to contain any radioactive materials released from the discharging coolant into the containment atmosphere.

The methods used for formulating the hydro-dynamic forcing functions induced by pipe rupture and the dynamic analysis for the pipe whip motion provide an acceptable basis for restraint design. The criteria used for the identification, design, and analysis of piping systems where postulated breaks may occur constitute an acceptable design basis in meeting the applicable requirements of AEC General Design Criteria 1, 2, 4, 14, 15, 31 and 32.

3.7 Seismic Design

3.7.1 Seismic Input

The seismic design response spectra indicate amplification factors of 3.6 between the period range of 0.15 to 0.5 seconds and of greater than 1 in the period range of 0.03 to 0.17 seconds for 2% damping.

The structure and equipment damping is in accordance with the damping factors which have been accepted for recently licensed plants.

We conclude that the seismic input criteria proposed by the applicant provide an acceptable basis for seismic design.

3.7.2 Seismic System Analysis and Subsystem Analysis

Modal response spectrum multi-degree-of-freedom and normal mode-time history methods from the bases for the analyses of all major Category I structures, systems, and components. Governing response parameters are 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 has been used for in-phase closely-spaced frequencies.

Two components of seismic motion are considered; one horizontal and one vertical. The total response is obtained by the absolute sum of the responses to the two components.

Floor spectra inputs to be used for design and test verification of structures, systems and components are generated from the normal mode-time history method. A vertical seismic-system dynamic analysis has been employed for all structures, systems and components where analyses show significant structural amplifications in the vertical direction. The system and subsystem analyses have been performed on an elastic basis. The effects on the floor response spectra of expected variations of structural properties and damping are accounted for by widening the response spectra peaks by ±10%.

We conclude that the seismic-system dynamic methods and procedures proposed by the applicant provide an acceptable basis for the 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 correspond to the recommendations of Safety Guide 12.

Supporting instrumentation will be installed on 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 is acceptable.

3.8 Design of Category I Structures

3.8.1 Concrete Containment

Each reactor and its cooling system is enclosed in a separate reinforced concrete containment structure. The containment systems for the two units are identical. They are described in Section 5.0 of the FSAR.

The concrete containment structure has the shape of a vertical "light bulb" with a torus, with a flat slab base and elliptical steel dome. A steel liner is attached to the inside of the containment vessel.

The containment is designed in accordance with applicable sections of the ACI-318 code for concrete and the pertinent sections of the ASME Pressure Vessel Code, Section III, Division I, for the liner. The containment is designed for dead, live, DBA, OBE, DBE, and environmental loads. Its structural design loads and design criteria are very similar to those used in previously approved license applications.

Stresses in the shell, penetrations, and foundation resulting from static and dynamic loads were calculated by means of well-known methods of analysis for shells and plates.

The liner design is typical for this type of containment. 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.

The stresses computed by the applicant are below the code allowables.

Materials, construction methods, quality assurance and quality control measures are adequately described in the FSAR and in general are similar to those used for other recently reviewed facilities.

The criteria used in the analysis, design and construction of concrete containment structures, are in conformance with acceptable codes, standards and specifications.

The use of these design criteria provides reasonable assurance that these Category I containment structures will withstand all the specified design loads (including those due to earthquakes and various postulated accidents) without impairment of their structural integrity and safety function. Conformance with these criteria constitute an acceptable basis for satisfying the requirements of AEC General Design Criteria #2, 4, 16 and 50.

3.8.2 Concrete and Structural Steel Internal Structures of Steel or Concrete Containment

The internal structure consists of a sacrificial shield around the reactor, the reactor pedestal and other interior compartments and floors. A description of the internal structure is presented in Appendix "C" of the FSAR.

The interior structure is designed in accordance with the ACI-318 Code for concrete, the AISC specifications and the pertinent provisions of the ASME Pressure Vessel Code, Section III.

The applicant has considered all the loads which may act on the structure during its lifetime, such as dead and live loads, accidents loads (pressure and jet loads), seismic loads, etc. The load combinations cover all cases likely to occur and include all loads which may act simultaneously.

The structure was designed by using well-established procedures. The working stress design method and the plastic or ultimate strength design method were used for design. The interior structure is designed in accordance with pertinent codes, indicated above, following well-established design methods. There are no special surveillance requirements for the interior structure.

The criteria used in the analysis, design and construction of the containment internal structure, are in compliance with acceptable codes, standards, and specifications.

The use of these design criteria provides reasonable assurance that the Category I containment internal structure will withstand all the specified design loads (including those due to earthquakes and various postulated accidents occurring within the containment) without impairment of the structural integrity and safety function. Conformance with these criteria constitutes an acceptable basis for satisfying the requirements of AEC general Design Criteria #2, #4, #16 and #50.

3.8.3 Other Category I Structures

The Category I seismic structures, listed in the FSAR, Section C.1, are in general similar to Category I seismic structures approved for previously licensed facilities.

The structures were built from structural steel and reinforced concrete members. In general, the structures are designed as continuous systems. The structural components integrated into the continuous structures consist of slabs, walls, beams and columns.

The design method for reinforced concrete followed that of ACI-318 Code for Concrete Structures with the use of specific loading combinations applicable to nuclear power plant design conditions. The applicable Safety Guides, Nos. 10 and 15, were considered by the applicant. For structural steel, the AISC Specifications were followed.

The loading combinations provide for the design of the structures to resist normal operating, normal shutdown, accident, operating basis earthquake plus normal operating, operating basis earthquake plus normal shutdown, accident plus operating basis earthquake, design basis earthquake plus operating, design basis earthquake plus normal shutdown and accident plus design basis earthquake loads. They are similar to the loads considered on previously licensed facilities.

For all reinforced concrete Class I seismic structures, the principal methods of analysis have been the Working Stress and Ultimate Strength design methods as defined in ACI-318 Code.

For steel structures, the design methods defined in the AISC Specifications were used. The analyses were based on elastic analysis.

The design for the reinforced concrete structures allows the reinforcing steel to reach $0.90f_y$ with the concrete being required to remain at or below $0.85f'_c$. The structural steel meets the allowable stresses outlined in the AISC Specifications (1963 and 1969). Under missile and jet loads or the design basis earthquake some local yielding is permitted in these structures if the resulting strain induced deformations will not result in a functional failure.

The design criteria, loads and load combinations, and the design methods used by the applicant are similar to those used in previously licensed plants.

The stresses in Category I seismic structures are below the code allowables.

The applicant provides in the FSAR a detailed description of materials and quality control. They are similar to those used in previously licensed plants and are acceptable.

No special testing and inservice surveillance procedures are required for Category I structures.

The criteria used in the analysis, design and construction of Seismic Category I structures, are in compliance with acceptable codes, standards, and specifications.

The use of these design criteria provides reasonable assurance that these Seismic Category I structures will withstand all the specified design loads (including those due to wind, tornadoes, earthquakes and various postulated accidents) without impairment of their structural integrity and safety function. Conformance with these criteria constitutes an acceptable basis for satisfying the requirements of AEC general Design Criteria #2 and #4.

3.8.4 Foundations and Concrete Supports

All Category I seismic structures have been designed in accordance with pertinent codes and are supported on competent, compact, firm sand. During the construction permit review a complete evaluation of the foundation conditions was made and the foundations were found structurally adequate to carry the applied loads. No new facts have been uncovered during construction which would affect the previous conclusion.

All structural members supporting Category I equipment have been designed as Category I structures.

The criteria used in the analysis, design and construction of the foundation and concrete support structures, are considered to be in compliance with acceptable codes, standards, and specifications.

The use of these design criteria provide reasonable assurance that these Category I structural foundations and supports will withstand all the specified design loads (including those due to wind tornadoes, earthquakes and various postulated accidents) without impairment of their structural integrity and safety function. Conformance with these criteria constitutes an acceptable basis for satisfying the requirements of AEC General Design Criteria #2 and #4.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

3.9.1.1 Vibration Operational Test Program

The applicant has specified a preoperational vibration dynamic effects test program to check the vibration performance of piping important to safety, including all piping from the reactor vessel to and including the first isolation valve external to the drywell. The vibration due to pump trips and/or valve closures will be checked during plant preoperation and start-up testing procedures. Portions of this test program will be supplemented by predictive analysis including the effect of earthquake loads, e.g., the response of the main steam line to turbine stop valve closure and to relief valve lifting. The effects of any supports and restraints whose addition may be indicated by test will be checked to assure that no adverse system influences will occur.

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. Compliance with this program constitutes an acceptable basis, in part fulfillment of the requirement of AEC General Design Criterion 2.

3.9.1.2 Analysis and Tests of Mechanical Equipment

The applicant has submitted procedures which use acceptable dynamic testing and analysis techniques to confirm the adequacy of non-pressure retaining mechanical components (such as fans, pump drives, etc.) which are Seismic Category I to function during and after an earthquake of magnitude up to and including the SSE and that equipment supports are adequately designed to withstand seismic disturbance. Subjecting the equipment and its 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 as identified in the FSAR will continue to function during and after a seismic event, and the combined loadings imposed on the equipment and its supports will not exceed the specified code allowable design stress and strain limits.

Implementation of these dynamic testing and analysis procedures, constitutes an acceptable basis for satisfying the requirements of AEC General Design Criteria 2 and 14 of 10 CFR Part 50.

3.9.1.3 Preoperational Vibration Assurance Program for Reactor Internals

The preoperational vibration assurance program as planned for the reactor internals must provide an acceptable basis for verifying the design adequacy of these internals under test loading conditions that will be comparable to those experienced during operation. The applicant has designated the James A. Fitzpatrick Nuclear Power Plant as the prototype reactor and has committed to performance of a confirmatory vibration test in accordance with Section D.3 of Regulatory Guide 1.20. In the event that a prototype test on the

Fitzpatrick plant is not successfully concluded in time to qualify Brunswick 1 & 2, the applicant has committed to performing a prototype test on Brunswick 2 of such a nature as to be consistent with the criteria of Regulatory Guide 1.20. The combination of tests, predictive analyses, and inspections will provide adequate assurance that, the reactor internals may be expected, during their service lifetime, to withstand the flow-induced vibrations of reactor operations without loss of structural integrity. The contained integrity of the reactor internals in service is essential to assure the retention of all reactor fuel assemblies in their place as well as to permit unimpaired operation of the control rod assemblies in order to permit safe reactor operation and shutdowns. The conduct of the preoperational vibration tests constitute an acceptable basis for demonstrating design adequacy of the reactor internals in satisfying the requirements of AEC General Design Criteria 2 and 14, and Regulatory Guide 1.20.

3.9.1.5 Analysis Methods Under LOCA Loadings

A dynamic analysis was made of the reactor internal components being acted upon by the computed blowdown forces resulting from a postulated LOCA. Response of the internals to the specified seismic input was also studied using a lumped mass model of the internals coupled to a similar model of the building. Dynamic flow pressure effects of the postulated LOCA upon the remaining piping system were determined to be minor. The analyses performed provide adequate assurance that the combined stresses and strains in the components of the reactor coolant systems, and reactor internals will not exceed the allowable design stress and strain limits for the material of construction, and that the resulting deflections or displacements at any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The assurance of structural integrity of the reactor internals under LOCA conditions for the postulated most adverse loading event provides added confidence that the design may be expected to withstand a spectrum of lesser pipe breaks and seismic loading events.

3.9.2 ASME Code Class 2 and 3 Components (Quality Group B and C Components)

Pressure-retaining components in mechanical fluid systems within the boundaries of the AEC System Group Classifications A, B and C are designed and constructed in accordance with rules consistent with the codes specified in Regulatory Guide 1.26 in conformance with applicable portions of Section 50.55a of 10 CFR 50. Compliance with these rules provides reasonable assurance that the resulting component quality level, is adequate to safely withstand the plant loading conditions, and combination of design loadings which the systems may experience over their service lifetime, without loss of structural integrity. Conformance to these rules constitutes an acceptable basis for satisfying the requirements of AEC General Design Criteria 1, 14 and 30.

3.9.2.1 Plant Conditions and Design Loading Combinations

The specified design loading combinations for all normal and postulated plant conditions and the corresponding limit stress allowables for components that may be classified as ASME Code Class 2 and 3 are consistent with acceptable criteria including those derived from industry codes such as ASME B & PV Code Sections III and VIII.

The procedures used in the design of the safety-related ASME Code Class 2 and 3 pressure-retaining components in systems classified as Seismic Category I provide reasonable assurance that in the event (a) an earthquake should occur at the site, or (b) 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 combination of loading events without gross loss of structural integrity. The design load combinations and associated stress and deformation limits specified for ASME

Code 2 and 3 components constitute an acceptable basis for design in satisfying the General Design Criteria 1, 2 and 4.

3.9.2.4 Component Operability Assurance Program

The applicant has described and has implemented a program of stress and deformation analysis of valves operators and motors which provides adequate assurance of the capability of active valves in Seismic Category I systems including those which may be classified as ASME Code Class 2 and 3 to withstand postulated seismic loads in combination with other significant loads without loss of structural integrity, and to perform the "active" function (i.e., valve closure or opening) when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated. The specified component operability assurance procedures constitute an acceptable basis for implementing the requirement of General Design Criteria #1 as related to operability of ASME Code Class 2 and 3 active valves.

3.9.2.5 Design and Installation Criteria, Pressure Relieving Devices (Class 2)

The relief valve stations in the residual heat removal and high pressure coolant injection systems are analyzed in accordance with the methods and procedures of ANSI B31.1.0. Time history dynamic analyses are performed, using fluid momentum forcing function time histories and a lumped mass model of the piping system.

The criteria used in developing the design and mounting of the safety and relief valves of ASME Class 2 systems provide adequate assurance that, under maximum 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 ASME Class 2 Systems constitute an acceptable design basis in meeting the applicable requirements of AEC General Design Criteria 1 & 2, 4, 14 and 15.

3.9.3 Components Not Covered by the ASME Code - Mechanical Design of Fuel Assemblies - Mechanical Design of Control Rod Drives

The design procedures applied to the fuel and control rod assemblies and control rod drives are based upon technology attained in connection with other comparable boiling water reactors such as the Cooper Nuclear Station and the Edwin I. Hatch Nuclear Plant, Unit 1. The use of this technology provides reasonable assurance that the fuel and control rod assemblies and control rod drives 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 their structural integrity or impairment of function. Compliance with these design criteria fulfills the requirements of AEC General Design Criteria 2 and 14 as these criteria relate to fuel and control rod assemblies, and control rod drives.

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

The seismic qualification testing program for Seismic Category I instrumentation and electrical equipment identifies the safety related elements and the postulated occurrence loads and is in compliance with IEEE Std. 344, "IEEE Guide for Seismic Qualification of Class IE Electric Equipment for Nuclear Power Generating Stations". The program as implemented provides adequate assurance that such equipment may be expected to function properly during the excitation and vibratory forces imposed by the safe shutdown earthquake under the conditions of post-accident operation.

Operability of the instrumentation and electrical equipment is essential to assure the capability of such equipment to initiate protective actions in the event of a safe shutdown earthquake (SSE) as necessary for the operation of engineered safety features and standby power systems. This program constitutes an acceptable basis for satisfying the requirements of AEC General Design Criterion 2.

4.0 REACTOR

4.1 Summary Description

The nuclear steam supply system is a General Electric single cycle, forced circulation boiling water reactor with a rated thermal output of 2436 MWt. The reactor generates high pressure steam for direct use in the steam turbine-generator. Figure 4-1 is a schematic diagram illustrating the principal features of the Brunswick Steam Electric Plant, Units 1 and 2 (BSEP 1 & 2). The design of BSEP 1 & 2 is similar to Browns Ferry Nuclear Plant, Units 1, 2 and 3 (Docket Nos. 50-259, 260 and 296) and Cooper Nuclear Station (Docket No. 50-298) which have been reviewed by the Regulatory staff.

The fuel, the heat source, consists of slightly enriched uranium dioxide pellets contained in sealed Zircaloy-2 tubes about one-half inch in diameter. These fuel rods, which are over 12 feet long, are assembled into individual fuel assemblies, each consisting of 49 rods in a 7 x 7 array, within a square open-ended Zircaloy-4 channel box. Five hundred and sixty of these fuel assemblies form a roughly cylindrical core of 160 inch equivalent diameter.

The core is supported in a domed cylindrical shroud inside the reactor vessel, and the reactor vessel is supported by a concrete pedestal. The steam separators are mounted above the fuel assemblies in the domed cylindrical shroud. The steam-water mixture emerging from the core passes upward through the separators to the steam driers above and then leaves the reactor vessel through four 24 inch steam outlet nozzles. The separated water, together with feedwater pumped into the vessel through four 12 inch feedwater lines, moves downward in an annular region between the reactor vessel wall and core shroud.

Within this annulus are two groups of 10 jet pumps, with each group arranged in a half ring. The 20 jet pumps accelerate the annular flow into the lower plenum before entering the core in an upward direction where water becomes a steam-water mixture. The high velocity water from the jet nozzles entrains and imparts energy to the additional water in the annular region. The flow to each jet pump group is provided from one of two external, high capacity, motor driven, variable speed recirculating pumps.

Reactor power is controlled either by movement of the cruciform-shaped control rods or by the variation of flow rate through the core. Individual hydraulic drives permit the control rods to be axially inserted to any degree desired or to be fully and swiftly inserted upon receipt of a trip signal (scram). A standby liquid control system is provided as a backup system for reactor shutdown and operates by pumping a sodium pentaborate solution into the reactor.

4.2 Mechanical Design

4.2.1 Fuel

The design of the fuel is similar to the design of the fuel for Peach Bottom Units 2/3 (Docket Nos. 50-277 and 50-278) and Duane Arnold Energy Center (Docket No. 50-331) which were previously reviewed and found acceptable. The fuel design is also similar to the design of the fuel in currently operating reactors, but differs in that the clad thickness is greater. A hydrogen-getter material is used within the tube, and urania-gadolinia fuel pellets are used.

The reactor employs Zircaloy-2 fuel tubes which contain slightly enriched uranium dioxide (UO_2) pellets. Some pellets in some of the fuel rods also contain gadolinium-oxide (Gd_2O_3) which is used to control the neutron flux distribution and reactivity. Groups of 49 fuel rods in a square array within a square Zircaloy channel box form fuel assemblies. Three types of fuel assemblies with varying distributions of U-235 enrichments and gadolinia concentrations are used. The addition of gadolinia serves as a burnable poison and supplements the control rods in flattening the power distribution of the core. The applicant submitted a proprietary description of the gadolinia placement in Design Report 14. The fuel in the initial core is designed for an average burnup of 16,700 MWd/t.

Urania-gadolinia pellets are being used in the Quad Cities operating reactors and all General Electric reactors currently under review. The differences in fuel damage limits due to the reduction in thermal conductivity and melting point of urania-gadolinia mixture as compared with urania was evaluated during the operating license review of the Quad Cities reactors.

The design of the fuel which will be used in BSEP 1 & 2 has been evaluated on the same basis and meets the same criterion as previously reviewed and accepted for other boiling water reactors. This criterion is that no fuel cladding damage should occur during normal operation or in the event of anticipated transient conditions. Fuel damage can result from overheating, excessive expansion or collapse, or corrosion of the clad.

Overheating will not occur if the mode of heat transfer remains in the nucleate boiling regime. Although heat transfer effectiveness would decrease if departure from nucleate boiling occurred, the resultant increase in clad temperature would be approximately 500°F and would not necessarily result in failure of the clad. Therefore, a conservative damage limit is defined as the critical heat flux (CHF) at which the departure from nucleate boiling occurs. Evaluation of the CHF is discussed in the section on thermal hydraulic design (Section 4.4).

Excessive expansion is defined as greater than 1% strain.

Expansion of the clad is caused by expansion of the fuel pellets and is a function of both fuel burnup and temperature. Therefore, a second fuel damage limit is defined as the value of linear heat generation rate, as a function of burnup, that will produce a clad strain of 1%. For rods with urania pellets, this limit is calculated to be 28, 26.5 and 24 KW/ft at burnups of zero, 20,000, and 40,000 MWd/T, respectively. For urania-gadolinia pellets, the limits are approximately 3 kW/ft less than the urania pellet limits given above.

Collapse of the cladding can occur due to the effect of densification of the fuel pellets and the creep of the clad. This phenomenon has been observed in some reactors and its causes and effects are described in the Staff's "Technical Report on Densification of Light Water Reactor Fuels," which was issued November 14, 1972. The applicant has responded to our concern in this area by reference to a GE topical report NEDM-10735 and its Supplements 1, 2, 3, 4, 5, 6, 7, and 8, "Densification Considerations in BWR fuel Design and Performance." The staff has made a detailed review of BWR fuel densification and has provided the essential elements to be used by the applicant to account for the effects of fuel densification in the BSEP 1 & 2 cores. The applicant has provided the necessary analyses and relevant data for determining the consequences of densification and the effects on normal operation, anticipated transients, and accidents, including the postulated loss-of-coolant accident. The staff's evaluation of fuel densification effects is given in Section 4.3 of this report.

Corrosion of the cladding due to local formation of hydrides on the inner clad surfaces has occurred in several reactors and caused clad failures and higher than desired off-gas activity. Water vapor present in the rods after their assembly was presumed to be the cause. The fuel rod manufacturing process has been modified and a hydrogen-getter has been added in the rod as means of assuring that moisture is not present or will not contribute to internal hydriding. The getter material is zirconium alloy chips which are loosely packed in a stainless steel tube placed in the plenum areas of the fuel rod. This material is used to absorb hydrogen. Since the temperature of the hydrogen-getter is low due to its position in the core, it will not react with the cladding or reduce its integrity in any way during normal, abnormal, and accident conditions. In Amendment #19 the applicant stated that getter-cladding reaction has not occurred in out-of-pile tests, and concluded that the presence of the getter in the fuel rod will not result in an increase in the number of rod perforations in the core during normal, abnormal, or accident conditions.

The increase in clad thickness is a design change made to improve the performance of the fuel during normal operation by further reducing the potential for cladding failures and the consequent increase in radioactive off-gas release rate. Since the increase in clad thickness has an insignificant effect on the fuel rod thermal properties, the effect on post-accident temperature transient is negligible.

We find the above changes acceptable; nevertheless, the Regulatory staff is monitoring the effectiveness of these design and manufacturing modifications.

Although fuel clad failures may still occur from the above or other causes, the safe operation of the core will not be affected. There is no evidence, either experimental or analytical, to indicate that there is a threshold for sudden and catastrophic release of fission products due to failure of cladding during normal operation or during anticipated transients. Past experience has indicated that any increase in cladding failure has been detected by an increase in coolant or off-gas activity, and before the failures become excessive, the Technical Specifications limiting conditions for operation have restricted plant operation. These Technical Specifications, when required, minimized the radioactive releases from the plant and kept releases well within acceptable limits when the plant operation was appropriately modified.

4.2.2 Reactor Vessel Internals

4.2.2.1 Material Considerations

We have reviewed the selection of materials for the reactor vessel internals required for reactor shutdown and components relied upon for adequate core cooling. All materials are compatible with the reactor coolant, and have performed satisfactorily in similar applications. Undue susceptibility to intergranular stress corrosion cracking will be prevented by avoiding the use of sensitized stainless steel by methods which are essentially in conformance with the requirements of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

The use of materials proven to be satisfactory by actual service experience, and avoidance of sensitization by the methods recommended in Regulatory Guide 1.44 will provide reasonable assurance that the reactor vessel internals will not be susceptible to failure by corrosion or stress corrosion cracking.

The applicant has described the measures that will be taken to ensure that deleterious hot cracking of austenitic steel welds is prevented. All weld filler metal will be of selected composition, and welding processes will be controlled to produce welds with at least 5% delta ferrite, in conformance with the recommendation in Regulatory Guide 1.31, "Control of Stainless Steel Welding." Following these recommendations will provide reasonable assurance that no deleterious hot cracking will be present that could contribute to loss of integrity or functional capability.

4.2.3 Reactivity Control Systems

Reactor power can be controlled either by movement of control rods or variation in reactor coolant recirculation system flow rate. The fuel rods will contain both full length and partial length sections containing gadolinium oxide, a burnable poison, to supplement the moveable control rods in controlling the core reactivity throughout the core life. A standby liquid control system is also provided as a backup reactor shutdown system.

Control rods (137 in number) are used to bring the reactor through the full range of power (from shutdown to full power operation), to shape the reactor power distribution, and to compensate for changes in reactivity resulting from fuel burnup. Each control rod drive has separate control and rapid insertion (scram) devices.

The drives have a common supply pump (and one paralleled spare pump) as the hydraulic pressure source for normal operation and a common discharge volume for scram operation. On the basis of our review of the drive system design and the supporting evidence accumulated from operation of similar systems in other General Electric reactors, we conclude that the installed system will meet the functional performance requirements in a safe manner.

Reactor power can also be controlled through changes in the primary coolant recirculation flow rate. The recirculation flow control system can automatically adjust reactor power level to station load demand whenever the reactor is operating between approximately 65% and 100% rated power. The recirculation flow control system is designed to allow either manual or automatic control of reactor power. This method of reactor power control has been satisfactorily demonstrated in other reactors.

The standby liquid control system is available to pump sodium pentaborate into the reactor vessel. This system is designed to bring the reactor to a cold shutdown condition from the full power steady-state operating condition at any time in core life, independent of the control rod system capabilities. The injection rate of the system is adequate to compensate for the effects of Xenon decay.

During operation at power levels between zero to ten percent of the rated power, control rod reactivity worths are limited by the rod worth minimizer (RWM). This device utilizes a computer to restrict control rod patterns such that the total worth of any in-sequence rod that can be moved will be no more than 1% $\Delta k/k$. For reactor power levels in excess of 10% of rated power RWM operability is not required.

The staff has been evaluating this system and operating experience indicates that this design is not reliable, and from its concept the RWM has not been a single failure proof system. Therefore, a reevaluation of the consequences of the postulated Control Rod Drop Accident indicates that modifications are required to augment the RWM so that the probability of occurrence of the postulated accident is negligibly low and/or that the consequences are consistent with the guidelines of 10 CFR 100. The approach being considered by the applicant is the installation of a Rod Sequence Control System (RSCS) with the RWM as a backup. The RSCS is designed to prevent the operator from moving an out-of-sequence rod and thus limit the consequences of the postulated rod drop accident.

Two mechanical safety features that will prevent rapid control rod motion out of the core are the control rod velocity limiter consists of two conical discs attached to each rod that will limit the downward free fall velocity of a rod to less than five feet per second but will not retard the upward scram action of the rod. The control rod drive housing support is a structure of removable beams beneath the reactor vessel that will limit the motion of a control rod should the rod drive housing fail and reactor pressure eject the drive from the vessel. Rod ejection travel will be limited to 3 inches by the support structure. This will limit the reactivity insertion so that the transient will not cause fuel failure. It should be noted that the velocity limiters mitigate the consequence of a rod drop accident while the housing supports mitigate the consequence of a rod ejection accident.

Each of the above design features with the exception of the RSCS is similar to the corresponding features provided in plants we have previously reviewed. On the basis of our previous review of similar designs and of satisfactory operating experience with similar systems in other operating BWR's, we conclude that the mechanical design aspects for the reactivity control features of BSEP 1 & 2 are acceptable.

4.3 Fuel Densification Effects

A detailed discussion of the causes and effects of densification including the results of observations of irradiated fuel in both test and power reactor fuel, an investigation of the possible mechanisms and evaluation of the controlling parameters, is presented in the staff's, "Technical Report on Densification of Light Water Reactor Fuels," dated November 14, 1972. At this time the only clear conclusion that can be drawn is that under irradiation fuel pellets can shrink and decrease in volume with corresponding changes in pellet dimensions. Four principal effects are associated with the dimensional changes resulting from densification. A decrease in length of pellets could result in the formation of axial gaps in the column of fuel pellets within a fuel rod. Two effects are associated with axial gaps. First, if relatively large axial gaps form, creepdown of the cladding later in life may lead to collapse of the cladding into the gaps. Second, axial gaps produce a local increase in the neutron flux and generate a local power spike. A third effect, which results from a decrease in pellet length, is a directly proportional increase in linear heat generation rate.

A decrease in pellet radius could result in the increase in the radial clearance between the fuel pellet and the fuel rod cladding. A fourth effect, which results from a decrease in pellet radius, is decreased pellet-clad thermal conductance (gap conductance). Decreased conductance would increase the fuel pellet temperature and stored energy and decrease the heat transfer capability of the fuel rod. Each of these four effects has been considered in evaluating the total effect that fuel densification might have on normal operation, transients and accidents.

Based on experimental evidence that no collapse has been observed in BWR fuel rods and on the results of calculations performed independently by the staff and GE, the Regulatory staff has concluded that typical BWR fuel will not collapse during the first cycle of operation. GE has also calculated the creep collapse of fuel in later cycles using a model which includes the modifications specified by the staff. The results of these calculations for fuel in residence up to more than 5 years are reported in Supplement 6 of the GE report "Fuel Densification Effects on General Electric Boiling Water Reactor Fuels," NEDM-10735 and indicate that clad collapse will not occur. The staff has reviewed the GE calculations and performed independent calculations, which also predict that collapse will not occur. Based on the calculations and experimental evidence, the staff concludes that creep-collapse need not be considered as affecting normal operation, transients or accidents.

The increase in linear heat generation rate (LHGR) resulting from contraction of the fuel is offset by compensating factors. Although pellets with initial densities less than the mean initial density will contract more than the average pellet, such pellets also contain correspondingly less fuel and produce less power in a given neutron flux. Therefore, only contraction from an initial mean pellet density need be considered in determining the LHGR. This contraction is offset by thermal expansion, as shown by calculations summarized in Table 3-1 of Supplement 6 of the GE topical report NEDM 10735. Since the increase in fuel column length due to thermal expansion was not considered in the original design calculations or transient and accident analyses, and since the effect of thermal expansion offsets the effect of densification on LHGR, it is appropriate to use the design LHGR in the analyses of normal operation, transients and accidents when considering the effects of densification. This was done in all the analyses presented by GE in Supplement 6 of the topical report NEDM 10735.

Calculations by GE of power spikes resulting from possible axial gaps in the fuel take into account the peaking due to a given gap, the probability distribution of peaking due to the distribution of gaps, and the convolution of the peaking probability with the design radial power distribution. Based on an examination of the methods used, comparison with requirements and approved models given in the staff densification report, and check calculations performed for the staff by Brookhaven National Laboratory, the staff concluded that, if appropriate gap assumptions are made regarding sizes, the GE calculational method is acceptable. The results of calculations of power spikes using acceptable gap sizes are summarized in Figure 3-6 of Supplement 6 of the GE topical report NEDM-10735. During normal operation there is a 95% confidence that no more than one rod would have a power spike greater than approximately 4% at the top of the fuel. At the midplane the corresponding power spike would be approximately 2%. When the reactor power is low and there are no voids, the spike could be greater. Under these conditions, there is a 95% confidence that no more than one rod would have a power spike greater than 5% at the top of the fuel.

Pellet-clad thermal conductance is a function of gap size and linear heat generation rate. The staff has reviewed the experimental data and analyses that GE has submitted to justify their correlation of gap conductance, examined the uncertainties in the data, and performed independent calculations with a fuel thermal performance computer program. The pellet-clad thermal conductance correlation used by GE is depicted in Figure 3-10 of Supplement 6 of the GE topical report NEDM-10735. It is based on experimental data and predicts with a 95% confidence that 90% of the total population of the pellet clad conductances exceed the prediction. The staff concludes that this correlation when used with a gap size adjusted for the effects of densification is acceptable.

4.3.1 Evaluation of Effects of Densification

4.3.1.1 Normal Operation

The design limits affected by fuel densification are the design values of linear heat generation rate (LHGR) and minimum critical heat flux ratio (MCHFR). The power spike resulting from axial gaps is considered in limiting operation of the reactor. The Technical Specifications will require that the LHGR in any rod at any axial location be less than the design value of 18.5 kw/ft by a margin equal to or greater than the power spike calculated using the accepted model. As discussed previously, this power spike penalty will assure at the 95% confidence level that no more than one rod will exceed the design value LHGR. Since the random occurrence of local power spikes will have no effect on coolant flow or quality, the uncertainty in calculation of the critical heat flux is unchanged. Therefore, if the calculated MCHFR is maintained above the steady state design limit of 1.9 and the margin to the design value of LHGR is also maintained, the probability of reaching a MCHFR of 1.0 is essentially unchanged from that calculated in the FSAR.

4.3.1.2 Transient Performance

The key transients for evaluation of BWR performance are those associated with overpressurization, which might imperil the integrity of the primary coolant pressure boundary, and with reduction of coolant flow, which might adversely affect the integrity of the fuel clad. The transient resulting from a turbine trip without opening the bypass valves is representative of transients that might result in overpressurization. The transient resulting from the simultaneous trip of both recirculation pump drive motors is representative of transients that result in a rapid reduction of core flow.

Following isolation of a BWR, such as would result from closure of the turbine stop and bypass valves, stored and decay energy from the core increases the coolant temperature and pressure. Since densification might reduce the pellet-clad conductance and increase the stored energy, densification could effect the peak pressure following a transient. GE has calculated the increase in heat flux, fuel temperature, and peak pressure in the primary coolant system following a turbine trip transient without bypass using gap conductances as low as 400 Btu/hr-ft^2-°F. A conductance of 400 Btu/hr-ft^2-°F is representative of the average fuel rod and its use is appropriate since the average fuel rod stored energy is the appropriate parameter to use when evaluating coolant system pressure. The calculated peak pressure is increased only 5 psi and is not significantly greater than the system pressure calculated using the value of 1000 Btu/hr-ft^2°F for gap conductance. Using a conductance of 400 Btu/hr-ft^2-°F increased the calculated fuel temperature 13°F and the heat flux 1%. These increases are also insignificant.

Following a rapid reduction in core flow, such as would result from simultaneously tripping both recirculation pump motors, the

MCHFR will decrease. A MCHFR of 1.0 is taken as a design limit for fuel damage. The slower thermal response of rods with densified fuel can result in a lower MCHFR following a rapid flow reduction. GE has calculated that the heat flux at the time of MCHFR would increase less than 5%, even if the gap conductance were as low as 400 Btu/hr-ft^2-°F. This conductance is representative of the lower bound of the conductance expected at the axial location where MCHFR occurs.

Based on these calculations, the staff concludes that changes in gap conductance resulting from fuel densification would affect the course of flow and pressure transients. However, the pressure and MCHFR limits would not be exceeded.

4.3.1.3 <u>Refueling Accident</u>

Since fuel densification does not affect any parameters used in the evaluation of the refueling accident, the consequences of this accident are unchanged.

4.3.1.4 <u>Control Rod Drop Accident</u>

A generic evaluation by the staff of the control rod drop accident has been underway for the past several months. General Electric has submitted topical reports revising the techniques for analyses of the control rod drop accident including, among other features, a change in the method for modeling the rate of negative reactivity insertion. These topical reports and revised analyses are under review. However, the parameters important to the analysis such as gross power distribution, delayed neutron fraction and the reactivity changes produced by the dropped rod, the scram insertion of the other rods and Doppler feedback are not significantly affected by densification. The parameters affected by densification are initial stored energy and heat transfer. These factors are not important for the control rod drop accident at low reactor power which results in the largest energy deposition, since the analysis assumes low power and adiabatic fuel pins and therefore no stored energy and no heat transfer.

At low initial power, the effect of densification that is important to the analysis of the rod drop accident is the local perturbation of the power distribution resulting from axial gaps in the column of fuel pellets. This power spiking effect would be very localized and affect only a few fuel rods. The peak enthalpy will occur in the upper region of the core and, as discussed previously, the magnitude of the power spike will be less than 5% even at the top of the core.

The Rod Sequence Control System (RSCS) is designed to preclude the movement of out-of-sequence control rods below 30% reactor power. The peak enthalpy resulting from the dropping of an in-sequence control rod with the maximum worth is calculated to be never greater than 230 cal/g. Therefore, even if the dropped rod were in a region with a 5% power spike, the calculated peak fuel enthalpy would be well below the 280 cal/g design limit.

At high initial power, the most important effect of densification is the high initial stored energy due to decreased gap conductance. General Electric has submitted parametric results on the effect of reduced gap conductance on the initial stored energy for the control rod drop accident. In a letter (J. A. Hinds to V. Stello dated August 30, 1973), the General Electric Company shows that in the hottest fuel rod the maximum initial stored energy at full power, with reduced gap conductance due to densification, is typically 167 calories per gram. Based on low power rod drop accident calculations, GE estimates that the maximum energy added due to the drop of a control rod at high initial power is only 50 calories per gram. Since the 50 calories per gram does not occur in the same bundle as the peak initial enthalpy, according to the GE calculations, the peak enthalpy of any fuel rod could not exceed 217 calories per gram.

An independent calculation performed by the Brookhaven National Laboratory for the Regulatory staff indicates that the maximum energy added due to the rod drop accident while at high initial power is less than 75 calories per gram, and this maximum amount of energy addition does not occur in the same fuel bundle as that having the peak initial enthalpy. Thus, based on the staff further analysis the peak enthalpy of any fuel rod is less than 242 calories per gram. This is well below the 280 calories per gram acceptance limit.

The radiological consequences of the rod drop accident depend on the number of fuel rods that might experience clad damage as a result of the accident. In Table 15-2 of this report it is indicated that a postulated rod drop accident while at low initial power could result in 600 fuel rods experiencing clad perforation. Similar rod perforation estimates were not provided for the rod drop accident postulated to occur while operating at high power. Based on the available information, the staff cannot conclude that the number of damaged fuel rods at high initial power is less than the 600 damaged fuel rods previously calculated for the low initial power case. Consequently, we are extending our generic review of the BWR postulated rod accident to include consideration of the high initial power case.

We have requested the General Electric Company to furnish the necessary information for our use in this generic evaluation. Pending receipt of this information, we have determined the number of fuel rods that would have to fail before the guideline doses of 10 CFR Part 100 are exceeded. We find that if 9,000 fuel rods were to experience clad perforations due to a rod drop accident while operating at high power levels, the resulting two-hour dose at the exclusion distance (915 meters) would be 15 rem-whole body and 300 rem-thyroid.

While the staff is not able to conclude, at this time, that the number of rod perforations for this postulated accident is less than the 600 estimated by General Electric, we can conclude that it would not be anywhere near the 9,000 rods cited above. We, therefore, conclude that, although we are continuing our evaluation of the matter, the dose consequences of the rod drop accident, were it to occur while operating at high power levels, are within the 10 CFR Part 100 criteria and that the peak enthalpy is less than our acceptance limit of 280 cal/gm. These consequences are, therefore, acceptable.

4.3.1.5 Main Steam Line Break Accidents

As in the analysis of transients, the effect of reduced gap conductance resulting from densification is an increase in stored energy and transient heat flux. However, calculations demonstrate that a reduced conductance does not result in departure from nucleate boiling during the transient. As in the calculation presented in the FSAR (gap conductance equal 1000 Btu/hr-ft^2-°F) no clad heatup is predicted to occur and consequently the main steam line break accident is unaffected by densification.

4.3.1.6 Loss-of-Coolant Accident

Small Break

As in the analysis of a transient, the effect of reduced gap conductance resulting from densification is an increase in stored energy and transient heat flux. A higher initial stored energy, when transferred to the coolant during blowdown, maintains the pressure, and increases the break flow rate resulting in a quicker actuation of the Automatic Depressurization System. Therefore, the reactor is depressurized sooner and the low pressure emergency core cooling systems refill the vessel sooner. Since all stored energy is removed during the initial phase of the blowdown, only the decay heat, which is the same in both cases, affects the clad temperature. The net effect is a reduction in peak clad temperature following a small pipe break. Therefore, densification does not adversely affect a small pipe break accident.

Design Basis LOCA

Following a postulated break of a recirculation pipe, densification can affect the hydraulic response of the reactor as calculated by the blowdown analysis and the thermal response of the fuel as calculated by the heatup model. The effect on the blowdown is much less significant than the effect during the heatup.

As discussed in the review of the transient analysis, the effect of densification is a reduction of gap conductance and a corresponding increase in stored energy and transient heat flux. The increased energy and heat flux result in a slightly modified hydraulic response following the LOCA. However, as shown in Figures 4-7 and 4-8 of Supplement 6 to the GE topical report NEDM-10735, the flow rates are not significantly changed and the time of departure from nucleate boiling is unchanged. Therefore, the convective heat transfer coefficients are not significantly changed as a result of densification.

The heatup of the fuel is, however, significantly changed primarily as a result of increased stored energy. Although the formation of axial gaps might produce a local power spike, as discussed previously the spike would be approximately 2% at the axial midplane. As discussed in the staff's "Technical Report on Densification of General Electric Reactor Fuels", dated August 23, 1973, it is improbable that more than one spike of significant magnitude would occur at any axial elevation and that a 1% power spike would result in only a 4°F increase in peak clad temperature. Therefore, the effect of power spikes can be neglected in the heatup analysis.

The peak clad temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate and stored energy of all the rods in a fuel assembly at the axial location corresponding to the peak of the axial power distribution. GE has calculated that expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ±20°F relative to the peak temperature for a typical fuel design. Staff calculations show that variations in individual gap conductances and therefore stored energy within an assembly result in peak clad temperatures approximately 20°F higher than temperatures calculated using only the conductance of the average rod to represent all the rods.

The stored energy is dependent on the LHGR and the pellet-clad thermal conductance. As discussed, the conductance is based on a correlation which underpredicts 90% of the data with a 95% confidence for a selected gap size. The gap size is calculated as specified in the AEC Fuel Densification Model assuming that the pellet densified from the initial density to 96.5% of theoretical density. Since peak clad temperature is primarily a function of average stored energy, the density of 48 rods is taken as the two standard deviation lower bound on the measured initial "boat" pellet density. For the most critical rod, the two standard deviation lower found on initial density of individual pellets was assumed. The result of calculations of peak clad temperature are presented in Fig. 4-10W of Supplement 6 & 7 to the GE Topical report NEDM 10735. The staff concludes that limitation of the average linear heat generation rate of all the rods in any fuel assembly at any axial location to the values of the curve labeled "γ" in Figure 4-9W of Supplement 6 & 7 to the GE Topical report NEDM 10735 will assure that calculated peak clad temperatures will not exceed 2300°F.

4.3.1.7 Conclusions

The Regulatory staff has reviewed the General Electric Co. report, "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," NEDM-10735 (Supplements 1 through 7) for its applicability to Brunswick Units 1 and 2. The staff concludes that the following changes in the operating conditions for Brunswick Unit 1 & 2 are necessary in order to assure that the calculated peak cladding temperature of the core following a postulated LOCA will not exceed 2300°F taking into account fuel densification effects: (1) the control of steady-state power operation so that the average linear heat generation of all the rods in any fuel assembly, as a function of planar exposure, at any axial location, shall not exceed the maximum average planar linear heat generation rate defined by the curve in Limiting Condition for Operation, Figure 3.5.1, of Section 3.5.I of the Technical Specifications and (2) that during steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated using the equation for maximum LHGR provided in Limiting Condition for Operation, Section 3.5.J of the Technical Specifications.

4.4 Thermal and Hydraulic Design

The thermal and hydraulic characteristics of Brunswick Steam Electric Plant Units 1 and 2 reactor are similar to those for Cooper Nuclear Station (Docket No. 50-298) and Edwin L. Hatch

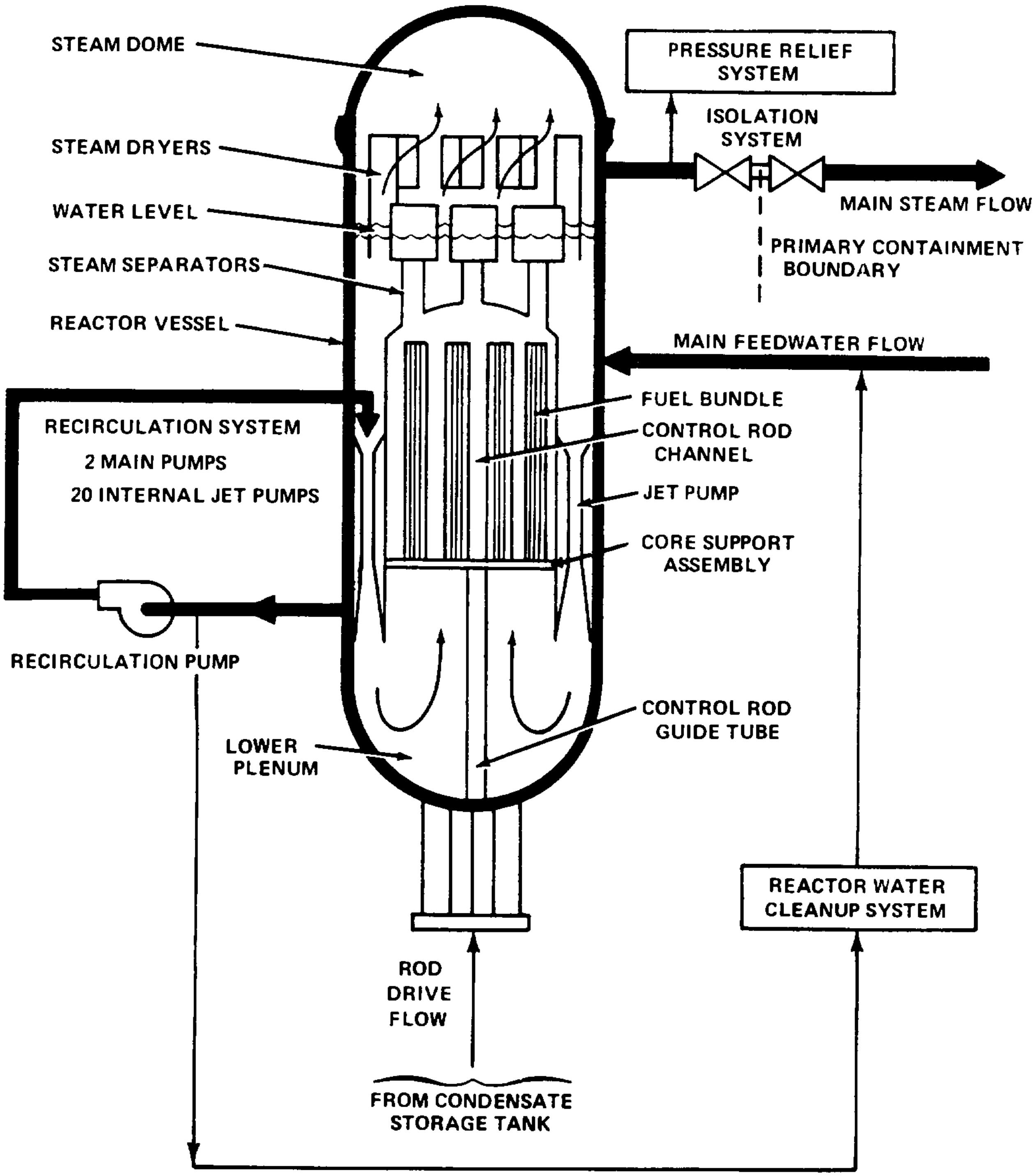
Nuclear Plant Unit 1 (Docket No. 50-321). For BSEP 1 & 2 our evaluation was made on the same basis as the reviews for these other plants.

The core thermal and hydraulic design bases are formulated to limit the local power density and coolant flow within the core to values such that the fuel damage limits, as described in Section 4.2.1, are not exceeded during normal operation or operational transients. One damage limit is the critical heat flux. The present critical heat flux limits are calculated using the correlation reported in the GE topical report APED-5286, "Design Basis for Critical Heat Flux Conditions in Boiling Water Reactors." This correlation is based on experimental data taken over the range of conditions representative of BWRs. The minimum critical heat flux correlation value at the corresponding fluid conditions to the actual maximum calculated heat flux occuring at a given point in the fuel assembly at any time during operation, including anticipated transients. A MCHFR greater than 1.0 conservatively assures that cooling of the fuel rod is maintained in the nucleate boiling heat transfer regime.

The current GE design basis for normal operation is that the MCHFR calculated for any point is greater than 1.9 during normal operation and greater than 1.0 during anticipated transients. These limits provide considerable margin between expected conditions and those required to cause fuel clad damage since the critical heat flux correlation presented in APED-5286 is conservatively based on a limit line drawn below nearly all of the available experimental data points. General Electric has submitted new CHF data based on extensive rod bundle testing. This is now under review by the staff. The design value of linear heat generation rate during normal rated power operation is 18.5 kW/ft, corresponding to a MCHFR 1.9. Analysis of anticipated operational transients shows that the lowest MCHFR, a value of 1.2, occurs following a loss of a feedwater heater.

A second fuel damage limit is the linear heat generation rate (LHGR) which produces a clad strain of 1%. The LHGR producing a strain of 1% is more than 24 kW/ft during normal operation. The maximum LHGR that may be attained by fuel rods during steady-state operation is 18.5 kW/ft. Although higher peak powers occur during anticipated operational transients, fuel temperatures and the resulting expansion are not sufficient to produce the 1% clad strain.

We have reviewed the methods used to calculate the thermal and hydraulic limits, the experimental basis for the calculations, and the applicant's analyses of normal operation and anticipated transients for this plant and previously reviewed reactors, and conclude that the design provides adequate margin to protect the core against fuel damage.



PRINCIPAL FEATURES OF REACTOR AND PRIMARY COOLANT SYSTEM
Figure 4—1

5.0 REACTOR COOLANT SYSTEM

5.1 General

The principal equipment and systems discussed in this section are the reactor pressure vessel, the reactor recirculation system, the main steam and feedwater lines and the pressure relief system. These items form the major components of the reactor coolant pressure boundary (RCPB). The pressure boundary also contains portions of the residual heat removal system and the reactor water cleanup system, and other piping that extends from the reactor vessel to the second outermost isolation valve.

5.2 Design of Reactor Coolant Pressure Boundary Components

5.2.1 Compliance with 10 CFR Part 50, Section 50.55a

The ASME and ANSI Code components within the reactor coolant pressure boundary will be designed, fabricated, and inspected in accordance with the requirements of the applicable codes. The applicable codes, code editions and addenda used by the applicant comply with the rules of 10 CFR Part 50, Section 50.55a, Codes and Standards.

5.2.2 Applicable Code Cases

The specified ASME and ANSI Code Cases, whose requirements will be applied in the construction of pressure-retaining components within the reactor coolant pressure boundary (Applicant's System Quality Group Classification IA), are acceptable to the Commission. Compliance with the requirements of these Code Cases is expected to result in component quality level consistent with the acceptable level intended by the requirements of GDC 1.

5.2.3 Design Transients (Plant Conditions, Loading Combinations and Stress Limits)

The specified design loading combinations for all normal and postulated plant conditions and the corresponding limit stress allowables for RCPB components are consistent with acceptable criteria including those derived from industry codes such as ASME B & PV Code Section III, ANSI B 16.5, ANSI B 31.0, and ANSI B 31.7. The procedures used in the design of the reactor coolant pressure boundary components including the application of the 40 year life predicted transients to the design of the reactor vessel provide reasonable assurance that in the event (a) an earthquake should occur at the site, or (b) a system upset, emergency or faulted 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 stresses and strain limits for the materials of construction. Limiting the stresses and strains under such loading combinations provides an acceptable basis for the design of the system components to withstand the most adverse loading events which have been postulated to occur during the service lifetime without gross loss of the system's structural integrity and for satisfying General Design Criteria 1, 2 and 4.

5.2.4 Operation of Active Pumps and Valves

The applicant has identified the valves in the RCPB which must be capable of reliable operation during the postulated Design Basis Earthquake. Stresses in the valve components due to seismic loads in combination with the other significant loads to which the valves are subjected do not exceed, by analysis, the allowables of the ANSI B 31.1.0 Code. The corresponding valve operators and motors are environmentally and seismically qualified by test.

The recirculation pumps have been designed using methods and stress allowables which are consistent with those of the ASME B & PV Code Sections III and VIII for those plant load conditions wherein their operational integrity is required, including normal and upset, the latter condition including the effects of seismic forces.

The conduct of component operability assurance programs as applied to reactor coolant pressure boundary active valves and pumps, as implemented by industry code analytical predictive methods and seismic testing of valve operators provide adequate assurance that the capability of such active components (a) to withstand the imposed loads associated with Normal, Upset, Emergency and Faulted plant conditions, as applicable to each particular component, without loss of structural integrity, and (b) to perform the "active" function (i.e., valve closure of opening, pump operation) is confirmed under conditions and combinations of 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 specified component operability assurance program constitutes an acceptable basis for implementing the requirement of Criterion 1 of the AEC GEneral Design Criteria as related to the operability of ASME Code Class 1 active valves and pumps.

The safety valves provide protection against overpressure of the nuclear system and discharge directly into the drywell. The two safety valves are set to actuate at 1240 psig and have a combined capacity of 12.2% of rated steam flow.

The safety relief valves which discharge into the suppression pool perform the following functions: (a) limit overpressure and reduce the frequency of spring safety valve opening; (b) augment spring safety valves by opening (self-actuated operation only), and (c) depressurize the primary system following small breaks to allow LPCI and/or CSS operation. The safety-relief valves have a combined capacity of 65.8% of rated steam flow for Unit 1 and 73.2% of rated steam flow for Unit 2. The overpressure protection capability is based on the pressure rise assuming the following: a) the plant is operating at the turbine-generator design condition, 105% rated steam flow, vessel dome pressure of 1020 psig, and a reactor thermal power of 2550 MW; (b) the reactor experiences the worst pressure transient, closure of all main steam line isolation valves, and (c) an indirect high pressure scram. The analysis showed that the peak vessel bottom pressure was 50 psi below the ASME allowable (110% x 1250 = 1375 psig).

The analysis also showed that the above transient with a failed valve (safety or safety/relief) yielded a pressure of at least 25 psi less than the ASME code allowable pressure.

5.2.5 Overpressure Protection

The pressure relief system prevents overpressurization of the reactor coolant pressure boundary (RCPB) under the most severe transients and limits the reactor pressure during abnormal operational transients. In addition, the automatic depressurization feature provides rapid depressurization for small breaks of the primary system to allow effective operation of the low pressure coolant injection (LPCI) and Core Spray System (CSS). This automatic depressurization feature is a backup to the high pressure coolant injection (HPCI) system described in Section 6.3.

The pressure relief system consists of two safety and nine safety-relief valves for Unit 1 (Unit 1 has 25% bypass capability), all of which are located on the main steam lines within the drywell, between the reactor vessel and the first isolation valve. For Unit 2 the pressure relief system consists of two safety valves and ten safety-relief valves (Unit 2 has 105% bypass capability and a 0.2 second scram delay with select rod insert system, thereby requiring an additional safety-relief valve).

5.2.6 Mounting of Pressure-Relief Devices

The relief valve stations on Brunswick main steam piping are analyzed in accordance with the methods and procedures of ANSI B 31.1.0. Combination of moments from different loadings and stress calculations are consistent with the methods specified in the ASME B & PV Code Section III, Winter 1972 Addenda.

A dynamic time-history analysis is performed, using fluid momentum forcing function time-histories and a mathematical model of the piping system. The effects of postulated simultaneous operation of the valves are considered to determine the most adverse loading conditions for design.

The criteria used in developing the design and mounting of the safety and relief valves of the reactor coolant pressure boundary provide adequate assurance that, under discharging conditions, the resulting stresses are expected to remain below 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 without impairment of the overpressure protection function.

The criteria used for the design and installation of overpressure relief devices in reactor coolant pressure boundary constitute an acceptable design basis in meeting the applicable requirements of AEC General Design Criteria 1, 2, 4, 14 and 15.

5.2.7 Reactor Vessel Material Surveillance Program

The toughness properties of the reactor vessel beltline material will be monitored throughout its service life with a material surveillance program that will comply with the intent of Appendix H, 10 CFR 50, and which is consistent with programs that have been found acceptable for other BWR plants. The maximum neutron fluence at 40 years for this reactor vessel is only 8.5×10^{17} nvt. The program is acceptable with respect to the number of capsules, number and type of specimens and retention of archive material. While the withdrawal schedule is adequate, a revision should be made in the Technical Specifications to show a withdrawal schedule that conforms with the requirements of Appendix H, 10 CFR 50.

The surveillance program constitutes an acceptable basis for monitoring radiation induced changes in the fracture toughness of the reactor vessel material, and will satisfy the requirements of AEC Design Criterion 31, Appendix A of 10 CFR, Part 50.

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 service performance. The applicant has stated that the possibility of intergranular stress corrosion in austenitic stainless steel used for components of the reactor coolant pressure boundary will be minimized because sensitization will be avoided, and adequate precautions will be taken to prevent contamination during manufacture, shipping, storage, and construction. The plans to avoid sensitization are in general conformance with Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and include controls on compositions, heat treatments, welding processes, and cooling rates.

The use of materials with satisfactory service experience, and conformance with Regulatory Guide 1.44, "Control of Sensitized Stainless Steel" provide reasonable assurance that austenitic stainless steel components will be compatible with the expected service environments, and the probability of loss of structural integrity is minimized.

.2.8 Fracture Toughness

Compliance with Code Requirements

We have reviewed the materials selection, toughness requirements, and extent of materials testing proposed by the applicant to assure 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. Acceptance testing for the ferritic materials of the reactor vessel was performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III (1965 Edition and with Addenda through Summer 1967). Drop weight NDT data were obtained for all reactor vessel materials opposite the core.

The fracture toughness tests and procedures required by Section III of the ASME Code 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 pressure retaining components of the reactor coolant boundary.

The reactor will be operated in a manner that will minimize the possibility of rapidly propagating failure, in accordance with Appendix G to Section III of the ASME Boiler and Pressure Vessel Code, Summer 1972 Addenda. The use of Appendix G of the Code as a guide in establishing safe operating limitations, using results of the fracture toughness tests performed in accordance with the Code and AEC regulations, will ensure adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and AEC regulations, constitute an acceptable basis for satisfying the requirements of Criterion 31 of the AEC General Design Criteria, Appendix A of 10 CFR Part 50.

5.2.9 Water Chemistry Control

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, and conductivity requirements have been shown by tests and service experience to be adequate to protect against corrosion and stress corrosion problems. The Technical Specifications for water chemistry control will be required to be in accordance with the provisions of Regulatory Guide 1.56.

We have evaluated the proposed requirements for the external insulation used on austenitic stainless steel components, and conclude that it will be in conformance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

The controls on chemical composition that will be imposed on the reactor coolant, and the use of external thermal insulation in conformance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," provide reasonable assurance that the reactor coolant boundary materials will be adequately protected from conditions that could lead to loss of integrity from stress corrosion.

5.2.10 Control of Stainless Steel Welding

We have reviewed the controls proposed to prevent hot cracking (fissuring) of austenitic steel welds. These precautions include control of weld metal composition and welding processes to ensure adequate delta ferrite content in the weld metal. The proposed methods comply with Section III of the ASME Code, and are in essential conformance with Regulatory Guide 1.31, "Control of Stainless Steel Welding." The use of materials, processes, and test methods that are in accordance with these requirements and recommendations will provide reasonable assurance that loss of integrity of austenitic stainless steel welds caused by hot cracking during welding will not occur.

5.2.11 Inservice Inspection Program

To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be inspected periodically. The applicant has stated that the design of the reactor coolant system incorporates provisions for direct or remote access for inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. Remote access methods are under development to facilitate the inspection of those areas of the reactor vessel not readily accessible to inspection personnel.

The conduct of periodic inspections and hydrostatic testing of pressure retaining components in the reactor coolant pressure boundary in accordance with requirements of ASME Section XI Code provides reasonable assurance that evidence of structural degradation or loss of leaktight-integrity occurring during the service life will be detected in time to permit corrective action before the safety function of a component is compromised. Compliance with the inservice inspections required by this Code constitutes an acceptable basis for satisfying the requirements of Criterion 32 of the AEC General Design Criteria.

.12 RCPB Leakage Detection System

Adequate provisions have been made to detect leakage of reactor coolant to the containment. The major components of the system are: containment atmosphere particulate radioactivity monitors, radiogas monitors, halogen monitors, and level indicators on the containment sumps. The system has sufficient sensitivity to measure small leaks, will identify the leakage source to the extent practicable, will include suitable control room alarms and readouts, and is in conformance with the provisions of Regulatory Guide 1.45.

In addition, indirect indications of leakage can be obtained from the containment humidity, pressure, and temperature indicators.

The leakage detection systems will have detection capabilities in conformance with the provisions of Regulatory Guide 1.45, and will provide reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions. This constitutes an acceptable basis for satisfying the requirements of Criterion 30 of the AEC Design Criteria.

.13 Reactor Vessel Integrity

We have reviewed all factors contributing to the structural integrity of the reactor vessel and we conclude there are no special considerations that make it necessary to consider potential vessel failure for the Brunswick Steam Electric Plant, Units 1 & 2.

The bases for our conclusion are that the design, material, fabrication, inspection, and quality assurance requirements will conform to the rules of the ASME Boiler and Pressure Vessel Code, Section III, all addenda through Summer 1967, and selected Code Cases in effect prior to October 1968.

The fracture toughness requirements of the ASME Code, Section III, 1965 Edition have been met. Also, operating limitations on temperature and pressure have been established for this plant in accordance with Appendix G, "Protection Against Non-Ductile Failure," of the 1972 Summer Addenda of the ASME Boiler and Pressure Vessel Code, Section III.

The integrity of the reactor vessel is assured because the reactor vessel: (1) will be designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and pertinent Code Cases listed above; (2) will be 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 the reactor vessel design conditions will not be exceeded during normal reactor operation and that the vessel will not fail under the conditions of any of the postulated accidents; and (5) will be subjected to monitoring and periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under the service conditions.

5.3 Thermal and Hydraulic Design

5.3.1 Analytical Methods and Data

The analytical methods and thermodynamic and hydrodynamic data used are similar to those of other BWR plants and are acceptable to the staff. These are presented in Section 4.4 of this report.

5.3.2 Reactor Recirculation System

The system consists of two loops external to the reactor vessel, within the primary containment, that provide automatic load following capability over the range of 65 to 100 percent of rated power. The loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each loop contains one high capacity (variable-speed) motor-driven pump and two motor-operated gate valves. In each loop, subcooled water leaves the vessel in a 28 inch suction line and enters the suction of the recirculation pump, which is below the vessel water level.

The water is discharged at a head of 530 feet and at a flow rate of 42,200 gpm. The flow control range varies from 50% to 100% to allow a 35% power range, normally from 65 to 100 percent of rated power. The water flows to 20 (10 per loop) jet pumps which are located inside the reactor vessel and accelerate a portion of the flow in the annulus. Water not accelerated by the jet pumps returns to the recirculation pump through the suction lines. There are various system interlocks that provide protection against inadequate suction head.

In our review of this system we asked the applicant to determine the potential for missile generation due to pump overspeed during a postulated blowdown. In Amendment #19, the applicant stated that motor protection can be provided with the use of a decoupler, and pump overspeed protection studies were submitted in topical report NEDO-10677 "Consequences of Recirculation Pump Overspeed in a Typical General Electric Boiling Water Reactor," dated October 1971. The topical report concludes that protection against pump missiles can be provided by the use of additional pipe supports and restraints at specific locations. This report and various decoupler devices are presently under review by the staff. The applicant has stated that decouplers will be available in time for installation in BSEP at the first refueling outage. We find this approach acceptable.

5.3.3 Main Steam Line Flow Restrictors

Each steam line is provided with a venturi-type flow restrictor within the primary containment vessel (between the reactor vessel and the first main steam line isolation valve). The restrictors limit flow in the venturi throat to 200 percent of the flow rated should a main steamline break occur outside the primary containment. The purpose of the restrictor is to limit the coolant blowdown loss prior to isolation valve closure, to reduce the probabilities and consequences of fuel failure in addition to reducing the forces on the reactor internal structure during blowdown.

5.3.4 Main Steam Line Isolation Valves (MSLIV)

Rapid acting isolation valves are located on each steamline on each side of the primary containment. On various signals from the plant protection system, these valves close and isolate the reactor coolant system from other portions of the plant. When such isolation occurs various backup and emergency systems automatically function as described in Section 6.3 of this report.

The analysis of a sudden, complete steam line break outside the primary containment shows the fuel barrier is protected if the valves close in 10.5 seconds or less. However, we require the valves to close within 5 seconds and leakage through these valves shall be within the Technical Specification limit of 11.5 SCFH per valve as determined by a surveillance of the MSLIV leak rates, to assure that even if fuel failures occurred the released fission products would not escape the reactor or the containment.

5.3.5 Reactor Core Isolation Cooling System (RCIC)

The RCIC system is a backup, high pressure source of reactor coolant that will operate independently of the normal plant A-C power supply. Its operational purpose is to provide an alternative source of reactor coolant to the vessel and to provide sufficient cooling to remove residual heat following reactor shutdown and loss of feedwater flow without requiring depressurization of the reactor. It is not designed to operate under accident conditions (pipe breaks in primary system) and is not considered an engineered safety feature. The RCIC consists of a pump driven by a steam turbine, taking steam from the reactor. The pump takes suction from either the condensate tank or the suppression pool and discharges it to the reactor vessel through the feedwater line. The pump may also take suction from the RHR Heat Exchanger operating in its steam condensing mode. It can be activated manually, or automatically by a low water level signal from the reactor vessel.

5.3.6 Residual Heat Removal System (RHRS)

Cooling must be provided to cope with residual heat generated in the reactor following shutdown or scram of the reactor. While insertion of control rods or injection of nuclear poison into the core will rapidly cause the heat generated by the fission process to diminish to a negligible amount, heat continues to be generated due to the radioactive decay of various unstable isotopes produced during the fission process. Initially, this decay heat amounts to about 6 percent of the normal reactor power level but diminishes substantially with time following a reactor shutdown.

The residual heat removal system (RHRS) is provided to meet the cooling needs of a shutdown reactor. It is a low pressure water-cooling system that has both normal and safety related modes of operation. Its normal function is to remove decay and residual heat from the nuclear system so that refueling and nuclear system servicing can be performed. One of the safety modes is the introduction of emergency coolant following a pipe break while operating as an engineered safety feature, as described in Section 6.3. Other functions of the RHRS are: (1) condensation of reactor steam when the reactor is isolated from the main condenser in the shutdown mode, (2) introduction of a spray into the containment drywell or torus, and (3) removal of heat from the suppression pool when the suppression pool is being used as a heat sink.

The major equipment of the RHR system consists of two heat exchangers with steam condensing ability and four pumps. The RHR service water system, described in Section 9.2 provides cooling water to the heat exchangers. The RHR pumps are sized on the basis of the required flow during the accident mode of operation. The heat exchangers are sized on the basis of the required heat load during the shutdown cooling mode.

The service water pumps are sized to cause the pressure at the cooling water outlet of the RHRS heat exchangers to be greater than the pressure of the reactor coolant at the inlet of the heat exchangers during normal conditions to assure no leakage of reactor coolant from the shell side to the service water. With this as the design criterion, heat exchanger tube leaks will not contaminate the service water with reactor coolant water.

Each loop, consisting of one heat exchanger, two RHR pumps in parallel and ancillary equipment, is physically separated from the other. However, a cross connection by a single header makes it possible to supply either loop from the pumps in the other loop. Provision also exists for pumping RHR service water either directly into the containment or into the reactor if necessary. The RHRS operational modes are described briefly below.

During reactor isolation, the RHRS can be operated in the condensing mode to condense reactor steam; hence, the RHRS operates in conjunction with the reactor core isolation cooling system (RCICS). With the reactor isolated, reactor steam normally is directed to and condensed in the suppression pool via the relief valves and the RCIC turbine exhaust piping. However, the suppression pool temperature under these conditions is limited in order that the water temperature rise due to a postulated, subsequent design basis loss-of-coolant accident would not cause the pool temperature to exceed 170°F during the reactor blowdown. The condensing mode of RHRS operation relieves the burden on the suppression pool by transferring a portion of the decay heat; i.e., steam energy, directly to the 4 service water. Reactor steam is taken to the shell side of the RHRS heat exchangers and transfers heat to the service water system through the tubes. The condensate is either dumped to the suppression pool or returned to the reactor vessel through the suction of the steam-turbine driven, RCIC pump. Shortly after shutdown, both heat exchangers are used to handle the decay heat. After about 2 hours, the capacity of one heat exchanger is adequate and the other may be transferred to the suppression pool cooling mode.

The suppression pool cooling mode utilizes the RHRS heat exchangers to cool the suppression pool water by transferring heat to the RHR service water. This mode can be used in conjunction with the condensing mode or to provide long term suppression pool cooling following a loss-of-coolant accident blowdown.

The shutdown cooling and reactor vessel head spray mode is operated during normal shutdown and cooldown. Reactor water is diverted from one of the recirculation loops, through the RHRS pumps and the RHRS heat exchangers (shell side) where heat is transferred to the RHR service water (tube side); then the cooler reactor water is returned to the reactor vessel via a recirculation loop. Part of the cooled reactor water flow is diverted to a reactor head spray nozzle where it maintains saturated conditions in the vessel head volume by condensing the steam generated by the hot vessel walls and the reactor internals.

The containment spray mode of operation is initiated manually after the LPCI requirements are satisfied and aids in reducing post-LOCA drywell pressure. The RHR pumps transfer water from the suppression pool through the RHRS heat exchangers where it is cooled by the RHR service water. The cooled water enters the containment through headers and spray nozzles in the drywell and above the suppression pool and reduces the drywell pressure by condensing existing steam. The spray water will collect in the bottom of the drywell until it overflows into the drywell vent lines and drains back to the suppression pool.

We conclude that the design of the RHRS as described above is acceptable.

5.3.7 Reactor Coolant Cleanup System

The reactor coolant cleanup system continuously cleans the reactor water to limit chemical and corrosive action on heat transfer surfaces. This system is designed to provide for the discharge of reactor water during startup, shutdown and hot standby conditions; limit the loss of heat and fluids from the nuclear system; and maintain the conductivity level (purity) of the system's effluent. Water is removed from the suction line of each reactor recirculation pump. The processed water is: (1) returned to the nuclear system via the feed-water line; (2) directed to the main condenser hotwell; or (3) directed to the radwaste system.

The major equipment is located in the reactor building and includes pumps, regenerative and non-regenerative heat exchangers and two filter demineralizers. The system is protected against overpressurization by relief valves and can be automatically isolated to protect the core from low water level in case of a break in the cleanup system. It is also automatically isolated when the boron injection system is actuated. From a safety standpoint the principal function of the cleanup system is to provide a means for reducing the concentration of radioactive and corrosive materials in the primary coolant system. We conclude that the design of this system is acceptable.

6.0 ENGINEERED SAFETY FEATURES

6.1 General

The purpose of the various engineered safety features is to provide a complete and consistent means of assuring that the public will be protected from excessive exposure to radioactive materials should a major accident occur in the plant. In this section, we discuss the reactor containment systems and the emergency core cooling systems. Descriptions of other engineered safeguards are provided elsewhere as related to the systems they directly serve. As will be seen, certain of these systems have functions for normal plant operations as well as serving as engineered safety features.

Systems and components designated as engineered safety features are designed to be capable of assuring safe shutdown of the reactor under the adverse conditions of the various postulated design basis accidents described in Section 15 of this report. They are designed, therefore, to Category I standards and they must function even with complete loss of offsite power. Components and systems are provided in sufficient redundancy so that a single failure of any component or system will not result in the loss of the capability to achieve safe shutdown of the reactor. The instrumentation systems and emergency power systems are designed for the same seismic and redundancy requirements as the systems they serve. These systems will be described in Sections 7 and 8, respectively.

6.2 Containment Systems

The containment systems consist of the primary containment, a secondary containment which encloses the primary containment, containment cooling systems, isolation valves, a standby gas treatment system, and a combustible gas control system. This plant is the first BWR with a Mark I type containment, which utilizes a reinforced concrete type of construction. In addition, the shape of the drywell has been modified from the previous steel "lightbulb" drywells to a combination right circular cylinder-truncated cone geometry. For containment performance purposes, however, this plant was analyzed in a similar manner as the steel lightbulb-torus designs such as Brown's Ferry Units 1, 2 and 3 and Hatch Unit 1.

6.2.1 Containment Functional Design

6.2.1.1 Primary Containment

The primary containment is a pressure suppression system consisting of the drywell, the pressure suppression chamber, a vent system connecting the drywell and suppression chamber, and a vacuum relief system. The drywell is a steel-lined, reinforced concrete pressure vessel in the shape of a modified lightbulb and the suppression chamber is a torus-shaped, steel-lined, reinforced concrete pressure vessel located below and encircling the drywell. The drywell houses the reactor vessel, the reactor coolant recirculating loops and other branch connections of the reactor primary system.

The pressure suppression chamber contains 655,000 gallons of demineralized water. It serves both as a heat sink for postulated transients and accidents and as the source of cooling water for the core standby cooling systems. In the case of transients that result in a loss of the main heat sink, energy would be transferred to the pressure suppression chamber by the discharge piping from the reactor pressure relief valves. In the event of a design basis LOCA, the drywell vent system is the energy transfer path for all energy releases to the drywell.

Of all the postulated transient and accident conditions, the instantaneous circumferential rupture of the reactor coolant recirculation piping (area 4.28 sq. ft.) represents the accident with the most rapid energy addition to the suppression pool and is considered the design basis loss-of-coolant accident. The applicant calculated that the peak pressures that might be reached as a result of design basis accident to be 49.4 psig in the drywell and 26.5 psig in the suppression chamber. Both the drywell and suppression chamber are designed for an internal pressure of 62 psig. The analytical methods used in the analysis are similar to those used on other recently reviewed BWR plants.

We have performed an analysis of the containment pressure response using the CONTEMPT-LT computer code. The peak pressures resulting from this analysis are in agreement with those calculated by the applicant. Based on the applicant's use of the General Electric model, described in NEDO 10320, "The General Electric Pressure Suppression Containment Analytical Model", and our verification of the analytical results, we conclude that the applicant's analysis of the short-term containment response is acceptable.

Since the primary containment is designed for an external pressure not more than 2 psi greater than the concurrent internal pressure, vacuum relief is provided. Vacuum in the torus is relieved by two sets of valves, each set consisting of a swing check valve in series with a manually operated butterfly valve, which connect the reactor building and torus atmospheres. Vacuum in the drywell is relieved by ten swing check valve vacuum breakers located on the drywell-torus vent header. The torus-drywell vacuum breakers have redundant position switches which indicate in the main control room.

To minimize the potential for the existance of possible leakage paths between the drywell and suppression chamber air space which would result in steam bypassing of the suppression pool following loss-of-coolant accidents, the applicant has proposed to: (1) perform operational testing of the torus-to-drywell vacuum breakers once each month; (2) perform a leakage test of the drywell-to-torus vent system at the end of each refueling outage; and (3) provide redundant position indication in the main control room for each torus to drywell vacuum breaker. Similar programs have been approved on other recent BWR applications. Based on our review of the above provisions, we conclude that the applicant's program will reduce the potential for bypass leakage and is acceptable.

The applicant has performed an analysis of pipe breaks within the reactor vessel sacrificial shield annulus. The results of this analysis indicate that the double-ended rupture of a recirculation line results in the largest differential pressure across the shield. As this resultant differential pressure is below the design value for the shield, we conclude that the sacrificial shield design is adequate.

6.2.1.2 Secondary Containment

The secondary containment system consists of the reactor building, which will be discussed in this section, and the Standby Gas Treatment System which is discussed in Section 6.2.3. The reactor building encloses the reactor and the primary containment system and houses the new and spent fuel storage facilities, the core standby cooling systems, and other reactor auxiliary protection systems. The reactor building is designed to provide protection from all postulated environmental events, including tornadoes, for those systems located within the building which are required for safe shutdown of the plant. In the unlikely event of a design basis accident, the secondary containment is designed to prevent a ground level release of airborne radioactive materials. It provides a means for controlled elevated release of the building atmosphere so that offsite doses from a design basis fuel handling or loss-of-coolant accident will be within the radiological dose criteria of 10 CFR Part 100.

6.2.2 Containment Heat Removal

Containment heat removal capability is provided by a drywell fan-cooler system during normal operation and by the containment cooling mode of the Residual Heat Removal (RHR) System for post-accident cooling. The drywell cooling system utilizes four fan coil units with cooling water supplied from the reactor building closed cooling water system.

The containment cooling mode of the RHR System acts to limit temperature and pressure in the drywell and torus following a loss-of-coolant accident and to remove heat from the suppression pool. The RHR System consists of two heat exchangers, with steam condensing capability, and four pumps. One heat exchanger and two pumps in parallel form a loop and each loop is physically separated and protected to minimize the potential for single failures causing the loss of function of the whole system. The RHR System equipment, piping and support structures are designed to Category I seismic criteria.

Operating in the containment cooling mode, the RHR pumps take suction from the suppression pool, pump water to the RHR heat exchangers and direct the cooled water either back to the suppression pool or to the drywell and suppression chamber sprays.

The applicant has provided analyses of the long-term post-accident containment response assuming various combinations of containment cooling availability. The results of the analysis indicate that long-term containment pressures and suppression pool temperatures are within allowable limits for each case that was considered.

The applicant has provided an analysis of the suppression pool temperature response during plant shutdown with a loss-of-offsite power concurrent with a turbine trip. This condition results in minimum availability of cooling of the suppression pool during plant shutdown. The applicant demonstrated that suppression pool temperatures are limited so that effective condensation of the reactor steam can be ensured and that there is adequate NPSH available to the shutdown cooling pumps.

6.2.3 Standby Gas Treatment System

The Standby Gas Treatment System (SGTS) provides a means for minimizing the release of radioactive material from the containment to the environs by filtering and exhausting the atmosphere from the reactor building. Primary containment vent exhaust can also be directed to the SGTS for processing prior to release. Discharges from the SGTS are exhausted to the main stack for elevated release.

The system consists of two identical, parallel air filtration trains; each train having 100% capacity and consisting of a demister (moisture separator), heating element, prefilter, high efficiency particulate absorber (HEPA), charcoal filter, HEPA filter, and blower. The SGTS and its enclosure are designed to seismic Category I criteria including the underground discharge pipe leading to the main stack.

Each exhaust fan has a 3000 cfm design flow rate which is capable of reducing and maintaining the reactor building at 1/4 inch water negative pressure. The SGTS will start automatically upon receipt of various signals or it can be manually started from the main control room. Each SGTS train receives power from separate diesel generators and SGTS isolation valves fail in the open position on loss of electrical power. Check valves are included downstream of the blowers to prevent backflow. The operation of all active components is indicated and the failure of the system to perform satisfactorily is annunciated in the main control room.

Based on our review of the SGTS, we conclude that it is acceptable.

6.2.4 Containment Isolation System

The purpose of the containment isolation system is to provide the necessary containment integrity between the reactor coolant system pressure boundary, or the containment atmosphere, and the environs subsequent to various postulated accidents. The applicant has tabulated information on all penetrations of primary containment and the associated isolation valving arrangements.

The applicant has specified the design criteria and isolation valve arrangements used for isolation of primary containment penetrations. As a safety system, the isolation valves have been reviewed to assure that no single accident or failure can result in a loss of containment integrity.

Instrument line isolation capability was reviewed and found to be consistent with the guidelines of the supplement to Regulatory Guide 1.11. Instrument lines connected to the containment atmosphere have two isolation valves, one manual and one remote manual, both located outside the containment. Those instrument lines penetrating primary containment and connected to the reactor coolant pressure boundary have an automatic isolation valve, actuated by an excess flow switch, which is located outside containment. The remote-manual and automatic isolation valves have position indication in the control room.

A break in the portion of the instrument line between the containment and the isolation valve would result in an unisolatable leak path directly to the secondary containment. The applicant has included 1/4 inch diameter orifices in each of these lines inside the primary containment. The applicant performed an analysis of an instrument line break in the reactor building which demonstrated that the integrity and functional capability of the secondary containment would be maintained. Based on our review of the design, we conclude that the isolation valves and instrument line orifices are adequate and meet the intent of the supplement to Regulatory Guide 1.11.

6.2.5 Combustible Gas Control

Following a loss-of-coolant accident (LOCA), a) hydrogen gas could be generated inside the primary containment from a chemical reaction between the fuel rod cladding and steam (metal-water reaction), and b) both hydrogen and oxygen would be generated as a result of radiolytic decomposition of the recirculating coolant solutions. Criterion 41 of the AEC General Design Criteria requires that systems to control hydrogen, oxygen, and other substances which may be released into the primary containment, be provided as necessary to control their concentrations following postulated accidents to ensure that containment integrity is maintained.

In accordance with the guidelines of the supplement to Safety Guide 7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident", the applicant has proposed a Containment Atmosphere Dilution (CAD) system. The proposed nitrogen dilution system is a seismic Category I system and is designed such that no single active failure can compromise its function. This system is basically the same that was used on similar BWR plants such as Brown's Ferry, Hatch 1, and Fitzpatrick. We find that the CAD system satisfies the supplement to Regulatory Guide 1.7.

During normal operation, the oxygen concentration in the containment will be maintained at or less than 4%. This would provide at least 12 hours after an assumed design basis loss-of-coolant accident before the initiation of the CAD system would be required. During the post-LOCA period, the CAD system maintains an oxygen deficient (inert) containment atmosphere by addition of nitrogen gas from an external nitrogen makeup and supply system. To ensure that the containment repressurization pressure is limited to a value substantially below design pressure, the applicant has proposed a repressurization limit of 31 psig. Based on an assumption of zero leakage from the primary containment and the assumption indicated in Safety Guide 7, a containment pressure of 31 psig would be reached approximately 22 days after occurrence of the postulated loss-of-coolant accident.

Mixing associated with periodic operation of the containment spray system is relied on to ensure a uniform concentration of hydrogen within the containment. Instrumentation and sampling stations will provide the reactor operators with the necessary information as to the radioactivity levels, the radioisotopes, the hydrogen and oxygen concentrations, and local meteorology to assure that venting operations will be carried out safely.

We have reviewed the design of the proposed system and conclude that the system is acceptable for combustible gas control following the postulated design basis loss-of-coolant accident and the system meets the intent of Criterion 41 of the AEC General Design Criteria and the provisions of Regulatory Guide 1.7.

6.2.6 Containment Leakage Testing Program

The containment design includes the provisions and features planned to satisfy the testing requirements of Appendix J, 10 CFR Part 50. The design of the containment penetrations and isolation valves permits individual periodic leakage rate testing at the pressure specified in Appendix J, 10 CFR Part 50. Included are those penetrations that have resilient seals and expansion bellows, i.e., airlocks, emergency hatches, refueling tube blind flanges, hot process line penetrations, and electrical penetrations.

The proposed reactor containment leakage testing program complies with the requirements of Appendix J to 10 CFR Part 50. Such compliance provides adequate assurance that containment leaktight integrity can be verified periodically throughout the service lifetime on a timely basis to maintain such leakages within the limits of the Technical Specifications.

Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through leak paths will not be in excess of acceptable limits specified for the site. Compliance with the requirements of Appendix J constitutes an acceptable basis for satisfying the requirements of AEC General Design Criteria 52, 53, and 54, Appendix A of 10 CFR Part 50.

6.3 Emergency Core Cooling Systems (ECCS)

The ECCS subsystems provide emergency core cooling during those postulated accidents where it is assumed that mechanical failures occur in the primary coolant system piping, resulting in a loss-of-coolant from the vessel at rates greater than the available coolant makeup capacity using normal operating equipment. The ECCS subsystems are provided in sufficient number, diversity, reliability, and redundancy that, even if any active component of the ECCS fails during a loss-of-coolant accident (LOCA), adequate cooling of the reactor core will be maintained.

The ECCS consists of two high pressure systems and two low pressure systems. The former are the high pressure coolant injection (HPCI) system and the automatic depressurization system (ADS). The latter are the core spray system (CSS) and the low pressure coolant injection (LPCI) system, which is one mode of the residual heat removal (RHR) system. The ECC system for BSEP 1&2 is functionally similar to other General Electric 1967 product line facilities such as Browns Ferry and Hatch 1. Figure 6-1 is a schematic drawing showing the above described elements f the Core Standby Cooling System.

All the ECCS subsystems are initiated by a high drywell pressure signal or a reactor vessel low water signal, except for the ADS. Initiation of ADS requires coincidence of both of these and a third signal, discharge pressure from at least one low pressure ECCS pump. The ECCS is designed to provide adequate core cooling and to limit the peak fuel rod cladding temperature for the complete accident spectrum up to and including the design basis loss-of-coolant accident.

The ECCS can operate independently of any offsite electrical power using power from the onsite diesel generator and battery systems. All evaluations have been made assuming that only onsite electrical power is available. In addition, ECCS performance capability has been shown to be adequate assuming a failure of any active component with the ECCS. This single failure criterion has been applied coincident with the assumed loss of offsite power.

The applicant indicated, in Amendment 16, that provisions are incorporated into the design to keep the ECCS pump discharge lines full of water to prevent water hammer damage.

6.3.1 High Pressure Coolant Injection System (HPCIS)

The HPCIS consists of one 100% capacity, steam turbine driven, constant-flow pump which injects water through one of the feedwater lines into the reactor vessel. The pump can supply 4250 gpm over a pressure range of 1120 psig to 150 psig. Steam for the HPCIS turbine is supplied from one of the main steam headers in the drywell and the exhaust steam is discharged to the suppression pool. Initially, the HPCIS pump takes suction from the condensate storage tank. Should this supply be at a low water level, suction is automatically transferred to the suppression pool.

The HPCIS can provide unassisted core cooling for a loss-of-coolant (LOCA) resulting from a small break (.12 ft^2 for liquid, 0.8 ft^2 for steam). For intermediate sized liquid breaks (between 0.2-0.4 ft^2) the HPCIS must depressurize the reactor so that water can be injected by the low pressure cooling system. For large size liquid breaks (>0.4 ft^2) the HPCIS is not required since the high fluid flow rate and energy loss through the break cause rapid, unassisted vessel depressurization so low pressure cooling systems can operate.

6.3.2 Auto-Depressurization System (ADS)

The auto-depressurization system uses seven of the dual-purpose safety-relief valves of the Pressure Relief System described in section 5.2.2 of this report. Each valve has a capacity of 800,000 lb/hr at 1125 psid and can operate over a pressure range of 1125 to 50 psid. Automatic opening of these seven relief valves requires coincident signals of reactor vessel low water level at two levels, primary containment (drywell) high pressure, and discharge pressure indication of any core spray system or LPCI system pump. After a receipt of the initiation signal, a timer delays operation of the relief valves for two minutes. If an operator determines that the initiation signal is false or depressurization is not required, the timer may be recycled.

The ADS does not itself provide cooling, but depressurizes the reactor so that the low pressure core cooling systems can operate. The ADS is redundant to the HPCIS and is only required if the HPCIS cannot maintain the reactor water level following a loss-of-coolant accident. Similar to the HPCIS, the ADS is not required for large breaks.

6.3.3 Core Spray System (CSS)

The CSS consists of two 100% capacity subsystems, each with an electric motor-driven pump which can spray water drawn from the suppression pool onto the top of the core. Each pump is designed to pump 4700 gpm at a pressure of 113 psid and operates over a pressure range of 265 to 0 psid.

The system can be powered by either offsite power or the onsite diesel generators. Each subsystem is powered by a separate diesel-generator. No single failure of any component can affect both systems.

The CSS provides cooling water following all loss-of-coolant accidents, except those due to small breaks. These accidents can be controlled by the HPCIS. The core Spray System is redundant to the Low Pressure Cooling Injection System (LPCIS) and can provide adequate core cooling independently of the LPCIS.

6.3.4 Low Pressure Coolant Injection System (LPCIS)

The LPCIS mode is one mode of operation of the four Residual Heat Removal System (RHRS) pumps. Each pump is designed to deliver 7700 gpm at a pressure of 20 psid, over a pressure range of 202 to 0 psid. The LPCI system injects suppression pool water into the vessel plenum below the core through the nozzles in the jet pumps of the unbroken recirculation loop to reflood the core. The LPCI control system determines which recirculation loop is broken by measuring the pressure differential between the loops, aligns the valves to direct the flow into the unbroken loop, closes the recirculation pump discharge valves and opens the injection valve after the reactor pressure has fallen below the LPCI system design pressure. The system can be powered by either offsite power or the onsite diesel generators.

The LPCIS provides cooling water following all loss-of-coolant accidents except those resulting from small breaks that can be controlled by the HPCIS. Although the LPCIS is redundant to the CSS, its capability to provide adequate core cooling independently of the CSS is not evaluated, since no single failure can prevent operation of both subsystems of the CSS. Thus, there should always be at least one functional subsystem in the CSS.

6.3.5 Functional Performance

In Section 6.7 of the FSAR, the applicant provided an analysis of the performance of the ECCS using the assumptions and calculational techniques described in the Commission's Interim Policy Statement dated June 19, 1971, titled "AEC Adopted Interim Acceptance Criteria for Performance of ECCS for Light-Water Power Plants." The assumptions established by the above cited criteria were applied without deviation.

Failure of the HPCIS in addition to various other single failures was assumed to determine the situation that resulted in the maximum calculated fuel clad temperature. These other single failures included failure of the LPCIS injection valve, failure of an ADS valve and failure of the most critical diesel-generator. Previous analyses have shown that the complete severance of a recirculation pump suction pipe results in the highest fuel clad temperature and greatest amounts of metal-water reaction, and is therefore designated as the design basis loss-of-coolant accident. Breaks in any other pipe in the reactor coolant system at any location including a main steamline, a HPCIS steamline or any of the ECCS cooling water injection lines, results in lower fuel clad temperatures and less metal-water reaction.

Analysis of the design basis loss-of-coolant accident results in a calculated peak fuel clad temperature of 2121°F and a calculated fraction of clad reacted with water of 0.12%, assuming failure of the LPCIS injection valve and a break area of 4.3 ft^2 (which is the sum of the areas of ten jet pump nozzles, a cleanup line and a 28 inch diameter recirculation line). Calculated peak clad temperature and fraction of clad reacted decrease as the break area decreases to 0.5 ft^2 and then increase to 1480°F and 0.06%, respectively, at a break area of approximately 0.10 ft^2 (equivalent to a 4 inch ID pipe). At these intermediate break areas the highest calculated values of temperature (1480°F) and metal-water reaction result from analyses which assume failure of the HPCIS, one ADS valve and the most critical diesel generator which results in the loss of one core spray system and one LPCIS pump. Calculated temperature and metal water reaction decrease as the break area decreases below 0.1 ft^2.

The fuel heatup following the design basis loss-of-coolant accident is arrested within two minutes by reflooding of the core with water from the ECCS. Since the fuel cladding would be at elevated temperatures for only a short period, no significant amount of cladding would be embrittled. Therefore, fragmentation of fuel rods would not occur and the core geometry would be preserved. Some swelling or ballooning of the clad might occur, but tests have demonstrated that the ballooning is limited, can be defined and would not significantly affect the cooling of the core.

Reflooding the core with water from the ECCS terminates the clad temperature transient and reduces the clad temperature to near the saturation temperature. After the core has been recovered, the ECCS valves are realigned to direct the water through the RHR heat exchangers. The core will continue to produce fission product decay heat for an extended period of time. This heat is removed by using the low pressure core cooling pumps to circulate water through the core and the RHR heat exchangers. The low pressure core cooling systems are designed so that even if any single component of the ECCS fails, adequate long term cooling of the core will be maintained.

The two core spray subsystems are independent of each other and no single failure can prevent operation of both subsystems. The RHR system can be operated as two independent subsystems, each with one heat exchanger and two pumps. These subsystems can be isolated from each other such that no single failure could prevent operation of both subsystems. The pumps of the redundant systems are separated and protected such that flooding from a broken pipe will not affect all systems. Two watertight rooms in the reactor building each contain one core spray pump, two watertight rooms each contain two RHR pumps, and a separate watertight room contains a HPCI pump. Flooding of one room will affect only one of the two core spray systems or two of the four RHR pumps or the HPCI pump and not affect the other ECCS pumps. Each suction pipe to the ECCS pumps has a motor operated isolation valve close to the torus so that any pipe break can be isolated.

The applicant analyzed the availability of adequate net positive suction head (NPSH) for all ECCS pumps in conformance with Safety Guide No. 1 which requires that there be no reliance on calculated increases in containment pressure.

The analysis of the performance of the ECCS was done with no deviations from the evaluation model described in Appendix A, Part 2 of the AEC Interim Policy Statement. The design meets the requirements of the AEC Interim Acceptance Criteria. These criteria require that the consequences of a loss-of-coolant accident are such that (a) the calculated maximum fuel rod cladding temperature does not exceed 2300°F, (b) the amount of fuel rod cladding that reacts chemically with water or steam does not exceed 1% of the total amount of cladding in the reactor, (c) the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling and before the cladding is so embrittled as to fail during or after quenching, and (d) the core temperature is reduced and decay heat removed for an extended period of time.

6.4 Habitability Systems

The applicant proposes to meet Criterion 19 of the AEC General Design Criteria by use of adequate concrete shielding and by installing redundant 2000 cfm recirculating charcoal filters in the control room ventilation system.

Upon receipt of an accident signal or high radiation reading, the control room ventilation system is automatically placed in the emergency mode of operation. The fresh air inlets are closed isolating the control room. At the same time, the charcoal filter unit is placed into operation, recirculating the control room air to minimize contamination build-up in the occupied areas. At the operator's discretion filtered make-up air may be admitted to the control room to pressurize the control room. Under this mode of operation 1000 cfm is supplied as filtered make-up and 1000 cfm is recirculated through the charcoal filter.

We have calculated the potential radiation doses to control room personnel following a LOCA. The resultant doses are within the guidelines of Criterion 19. On this basis, we conclude that the design of the control room ventilation system is acceptable.

The staff has recently completed an analysis of potential chlorine releases on reactor sites that store substantial amounts of chlorine for water treatment purposes. As a result of this analysis several recommendations have been made involving the protection of control room personnel from chlorine releases, and the following provisions or their equivalent will be required for the Brunswick plant.

Adequate protection of the control room against an on-site chlorine release will be achieved if provisions are included in the plant design to automatically isolate the control room to limit the potential build-up of chlorine within the control room and if equipment and procedures are provided to assure immediate use of breathing apparatus by the control room operators.

The automatic isolation requirements results in a need for quick-response chlorine detectors located in the fresh air inlets to the control room. These detectors must be able to detect and signal a step increase in chlorine concentration within a time period not to exceed 3 seconds. The detectors should be capable of signaling a step increase from zero to 15 ppm of chlorine of by volume or greater. These detectors must be so placed, and the detector trip point so adjusted, as to assure detection of a leak or container rupture. Detector trip signals must cause automatic isolation of the control room and must provide an audible alarm to the operators. The means used to initiate automatic isolation must meet single active failure and seismic criteria.

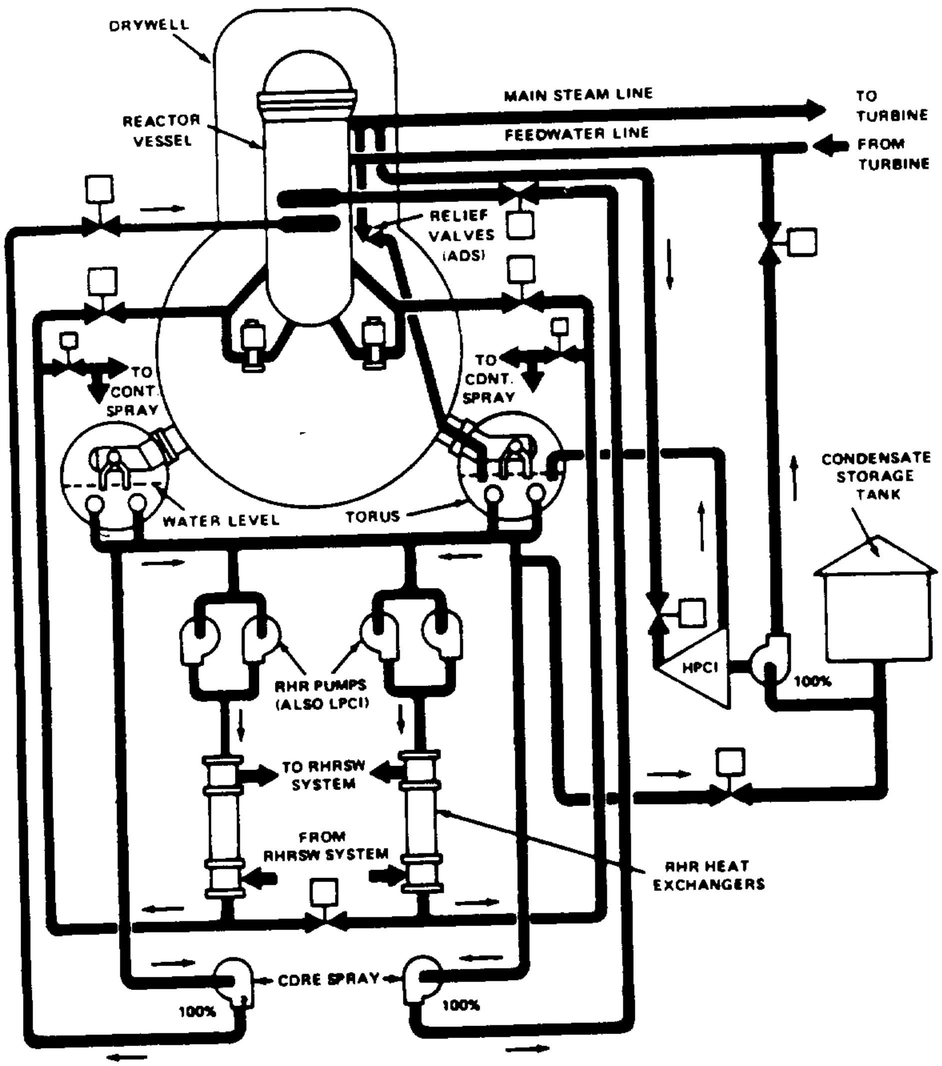
Control room isolation must be accomplished within about seven seconds after detector trip. Adequate isolation requires all openings to the control room to have low leakage characteristics. This would include doors, dampers, and penetrations. Total infiltration into the isolated control room should be less than 25 cfm assuming a 1/8" water gage pressure differential across all openings and the maximum operating differential across the isolation dampers upstream of recirculating fans. Normal fresh air make-up should be limited to no more than 1 to 1 1/2 air changes per hour. An administrative procedure specifying that all doors leading to the control room be kept closed when not in use is required.

Under certain meteorological conditions control room isolation may not be sufficient by itself to limit chlorine concentrations to levels below those which cause physical discomfort or disability. Therefore, the use of self-contained breathing apparatus must be considered when developing a chlorine release emergency plan. Calculations indicate that rapid increases in chlorine concentrations are possible. Emergency plan provisions and rehearsal of these provisions for immediate donning of breathing apparatus on detection of a chlorine release are necessary. Storage provisions for breathing apparatus and procedures for use should be such that operators can begin using the apparatus within two minutes after an alarm. Donning of breathing apparatus should be mandatory prior to the determination of the cause of an alarm.

A toxic environment may be present for several days or longer if a chlorine leak cannot be fixed or the leaking container removed; in any event adequate bottled air capacity (at least six hours) must be readily available onsite to assure that sufficient time is available to locate and transport bottled air from offsite locations. This offsite supply should be capable of delivering several hundred hours of bottled air to members of the emergency crew.

Isolation and air supply equipment relied on should accommodate a single active failure and still perform the required function. (In the case of self-contained breathing apparatus this may be accomplished by supplying one extra unit for every three units required.)

The applicant has committed in Amendment 23 to provide the protection necessary to cope with the above described chlorine accident. The finalized system will be reviewed by the staff and a supplement to this report will be written regarding this matter. We find this approach acceptable for the Brunswick Plant.



CORE STANDBY COOLING SYSTEM

Figure 6-1

7.0 INSTRUMENTATION AND CONTROLS

7.1 General

The protection and control systems and the engineered safety feature circuits have been evaluated against the Commission's General Design Criteria as published July 1971 and against IEEE Std. 279-1968, "IEEE Criteria for Protection Systems for Nuclear Power Generating Stations", and Regulatory Guides for Water Cooled Power Plants served, where applicable, as the basis for establishing the adequacy of these designs.

The evaluation of Brunswick 1 & 2 was accomplished by comparing the designs of specific systems with those of the Vermont Yankee, Cooper, Hatch 1 and Peach Bottom 2 & 3 on a selective basis. Our review was concentrated on those areas of design which are unique to Brunswick 1 & 2 for which new information has been received, or which have remained as continuing areas of concern during this and prior reviews of similarly designed plants.

During the course of our evaluation, we had several meetings with the applicant, Carolina Power and Light Company and its contractors. In addition, an engineering drawing review was conducted at the site on September 10-12, 1973 for the purpose of ascertaining that the design criteria were being properly implemented in the installation of the Instrumentation, Control and Engineered Safety Feature circuits.

7.2 Reactor Trip System (RTS)

The reactor trip system design as described in the FSAR is essentially identical to that of the referenced plants of the same General Electric product line (Vermont Yankee, Cooper, Hatch 1, and Peach Bottom 2 & 3), which has been reviewed in the past and found acceptable. Our review concentrated on those areas of design unique to Brunswick 1 & 2 and those which have remained as continuing areas of concern during this and prior reviews of similarly designed plants. Specifically, these areas are:

a) Average Power Range Monitor (APRM) Reactor Trip in Startup Range;

b) Use of Low Condenser Vacuum to Trip the Main Steam Isolation Valves;

c) Rod Block Monitor System; and

d) Flow Biased Flux Scram.

The above cited areas are discussed in the following sections.

7.2.1 Average Power Range Monitor (APRM) Reactor Trip in Startup Range

The applicant documented the changes necessary to modify the APRM channels to extend their effectiveness down to the startup range and to include an APRM trip at 15% power. In older BWR designs, the APRM channels were made effective only in the "Run" mode. We have reviewed this design change and have concluded that it meets the requirements of IEEE-279 and is acceptable.

7.2.2 Use of Low Condenser Vacuum to Trip the Main Steam Isolation Valves

The direct reactor trip function derived from loss of condenser vacuum had not been included in the previously licensed BWR designs. In a number of the more recently reviewed plants (Cooper, Hatch 1), loss of condenser vacuum initiates closure of the main steam line isolation valves, which in turn initiates a reactor trip. This design approach has also been used in Brunswick 1 & 2 and we find it acceptable.

7.2.3 Rod Block Monitor System

The rod block monitor system was upgraded to safety system standards. Either of two RBM channels can inhibit control rod withdrawal whenever the RBM output exceeds its setpoint. The system design criteria provide for redundancy, separation, and independence, and meet the requirements of IEEE-279 except for some relatively minor aspects of the design. The applicant has identified and justified these exceptions in the FSAR. We conclude that the design is acceptable.

7.2.4 Flow Biased Flux Scram Set Point

The APRM flux scram set point is varied as a function of reactor recirculation loop flow. Each APRM channel receives two independent redundant flow signals representative of total core flow, each derived from a separate pair of instrument lines. This trip has been designed to meet the requirements of IEEE-279 and we find it acceptable.

7.2.5 Anticipated Transient Without Scram

The applicant proposed to implement a design that provides for a trip of the main recirculation pumps in the event of high reactor pressure, which could occur during a transient with a postulated concurrent failure of the control rods to scram. The design includes pressure sensors and relays chosen from a manufacturer other than the supplier of the reactor protection system devices, thus achieving a degree of equipment diversity for this function. This wiring will be installed using the criteria of IEEE-279. The balance of the installation will be of standard (non-safety) design. The staff agrees that the addition of the recirculation pump trip as proposed by the applicant represents a substantial improvement in protection of the reactor for anticipated transients without scram; however, the staff has recently issued Wash 1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," dated September 1973. A letter was sent to CP&L requesting a commitment to meet the staff position given in Appendix A of the above cited report applicable to the BSEP by January 1, 1974.

7.3 Initiation and Control of Engineered Safety Feature Systems (ESFS)

7.3.1 Initiation Logic for ESFS

The basic design of the ESF systems' initiation logic is similar to those for the reference plants cited earlier. The Brunswick

1 & 2 design as proposed originally, incorporated certain features related to the sharing of the onsite emergency power systems.

The onsite emergency power system for Brunswick 1 & 2, discussed in more detail later in this report, is comprised of four 4160 V buses each fed by a 3500 kW diesel-generator (D-G) set. All four buses are shared by Units 1 & 2, and in many respects this design is similar to those for Brown's Ferry 1 & 2 and Peach Bottom 2 & 3. The present design of the ESF initiation logic for Brunswick 1 & 2 includes portions that are intended to block accident signals (actual or false) from the second unit experiencing an occurrence that would initiate such a signal. During our review we questioned the need for these blocking circuits for Brunswick 1 & 2. Having compared the ESF loads required for the Brunswick Units to those required for Cooper, a very similar plant design of the 1967 G.E. product line, it appeared the capacity of the D-G sets was sufficient to allow for the removal of the blocking circuits from the ESF initiation logic. We recommended that the applicant re-evaluate the present design for the purpose of minimizing interaction between units. The applicant has agreed to have all these blocking circuits removed, thus providing a design that will allow for the accommodation of a DBA in one unit preceded, coincident, or followed by a false or spurious signal in the other, assuming loss of offsite power to both units, and a failure of any one diesel-generator set to start. Subject only to diesel-generator verification tests, we consider this a marked improvement over the present design. The verification tests of the D-G sets are discussed later in this report. Subject to implementation of the above cited changes and successful completion of the verification tests, we find the ESF initiation logic acceptable.

7.3.2 Standby Gas Treatment Systems (SGTS)

The SGTS consists of two separate and redundant full capacity filter/absorber/fan units. This system is provided to maintain a small negative pressure (0.25 inches of water) in the reactor building under isolation conditions to minimize ground level release of airborne radioactivity.

Our review revealed that the two mutually redundant trains were started in sequence. We have asked, and the applicant has agreed to change the starting logic to provide for simultaneous startup of both trains. Subject to implementation of this change, we find the SGTS acceptable.

7.3.3 Automatic Depressurization System (ADS) Permissives

The Automatic Depressurization System requires that the following initiation signals occur for its actuation: (a) Reactor vessel low water level; (b) Primary containment (drywell) high pressure; and (c) Low pressure cooling available. The third requirement is satisfied by determination that pump discharge pressure from at least one LPCI or core spray pump is available.

The applicant has also added a second low reactor water level permissive to the ADS logic to confirm that the reactor water level is low. The second low reactor water level (confirmatory) signal has been included in the redundant ADS logic channels.

This confirmatory signal was added to prevent inadvertent actuation of the ADS as a result of a single process line failure. We conclude that this design is consistent with other recently licensed plants of similar design, and is acceptable.

7.4 Systems Required for Safe Shutdown

The design of instrumentation and control systems provided for safe shutdown have been reviewed. The applicant has agreed to provide indications and controls outside the control room to place and keep the plant in a safe shutdown condition in the event that access to the control room is restricted or lost. We have concluded that the design in this regard is acceptable.

7.5 Safety Related Display Instrumentation

The BWR reactor protection and engineered safety feature instrumentation channels generally use blind sensors, and, therefore, do not provide continuous readout in the control room of the parameters being monitored. The neutron monitoring and main steam line radiation monitoring systems are exceptions. The other vital parameters, however, are monitored by instrument channels associated with control systems. As such, these information readout channels are not designed to satisfy protection system criteria and have not been included in the Technical Specifications.

Information readout channels are required by the operator to assess plant conditions during and subsequent to an anticipated operational occurrence or accident, in order that he may determine whether to intervene in the operation of the Automatic Depressurization System (ADS), or to initiate the Containment Spray System (CSS), or take other action as necessary. The applicant has a list of redundant channels that read-out and record in the control room. This list is consistent with that of the Cooper design and we find it acceptable.

With regard to the annunciation of safety related system bypasses, our review of the design revealed that annunciation of the bypass of a safety related system resulting from a deliberate operator action was not included. The applicant was advised that we do not consider administrative controls and bypass light indicators at the component level as an effective and adequate means to identify these bypasses, nor do they satisfy our interpretation of the requirements of IEEE-279. We will require that the applicant provide the capability of initiating control room annunciators, not necessarily automatically, whenever operating and maintenance personnel actions result

Subject to the MSIV position switches qualification, the Category I equipment and circuits inside the drywell are considered acceptable in terms of environmental qualifications to perform their safety functions under normal and accident conditions.

The applicant has orally committed to submit additional information to clarify differences in the environmental qualifications made for Cooper and those for Brunswick and other plants. We do not anticipate any problems in this respect.

7.10 Separation Criteria

The applicant's separation criteria have been incomplete in some areas. One of these areas concerns the cable routing and cable tray separation for both units in a common cable spreading area. Although we will not require the applicant to provide separate cable spreading rooms for each unit, we have asked and the applicant has agreed to make design changes to provide a greater degree of independence and separation. This area of concern will be reviewed during a site visit and the results reported in the supplement to this evaluation report.

7.11 Containment Isolation

Our review revealed that the applicant's design does not provide for full and effective coverage of the entire length of steam pipe runs (or water in the case of the Reactor Water Cleanup System) for the purpose of detecting a pipe break at any point along these pipe runs. The systems that require isolation in case of a pipe break and have not been fully covered are:

a) Main Steam Supply up to the turbine;

b) High Pressure Coolant Injection (HPCI);

c) Reactor Core Isolation Cooling (RCIC);

d) Residual Heat Removal (RHR), that part of the steam line used in condensing mode of operation; and

e) Reactor Water Cleanup System (RWCU).

The design of this part of the containment isolation will be submitted for our review and we will determine its adequacy at that time. We will require that full protection for these pipe breaks be provided prior to initial fuel loading.

During preoperational testing of Browns Ferry 1, it was discovered that isolation of the Traversing-in-Core Probe (TIP) subsystem was defeated by failure of the limit switch providing the TIP withdrawal signal supplied to close the subsystem isolation valve. The failure was caused by increased drywell pressure simulating a DBA. General Electric has analyzed the problem and determined that it represents a generic problem for BWRs of similar designs, and a generic solution was promised for the near future. We will review this solution when submitted for our review, as it applies to Brunswick 1 & 2. We do not anticipate any problems in this regard.

in the loss of a safety function, or reduction in system redundancy. We will review the applicant's design in this regard when it is documented. We anticipate no problem in this regard.

7.6 Rod Sequence Control System (RSCS)

In order to mitigate the consequences of the design basis rod drop accident, General Electric Company is proposing the implementation of a hard wired control system referred to as Rod Sequence Control System (RSCS). The proposed system for Brunswick 1 & 2 would employ an RSCS with a Group Notch Control (GNC) feature that will keep any rod in a group within ±1 notch of all other rods in the same notch group. The preliminary General Electric design would provide for 34 such groups. The design of the RSCS has not been finalized yet by General Electric. We will review it when submitted as a generic item by General Electric, and report the results of our review in a supplemental safety evaluation report. We will require this protection against a rod drop accident to be implemented prior to initial fuel loading.

7.7 Seismic Qualification

Comments on the adequacy of the seismic qualification tests are reported in Section 3.10 of this report. The list of electrical and instrumentation equipment that were tested is not complete yet. We will require that the applicant complete the seismic qualification program and provide appropriate documentation prior to commercial operation of the plant. We expect no difficulty in this regard.

7.8 Radiation Qualification

The applicant has stated that the electrical and instrumentation equipment and cabling has been qualified to withstand up to 10^8 rads for a DBA case or 6×10^7 rads for normal 40 year plant operation. We find this qualification acceptable.

7.9 Environmental Qualification

Instrumentation, equipment and cabling for safety related systems that are located inside the drywell have been, or in a few cases, will be qualified by appropriate environmental qualification tests to withstand the hostile environment following a DBA.

Furthermore, all Category I equipment and circuits outside the drywell have been evaluated against the design basis environment that exists during normal operating and accident conditions.

The applicant maintains that the Main Steam Isolation Valves (MSIV) position switches and related equipment and cabling need not be qualified to withstand a DBA environment, because they perform their safety function immediately following a DBA and before the attendant hostile environment develops. We maintain that this may not always be the case. Furthermore, the possibility of failures within the Protection System raise the serious question of adverse interactions with the rest of the system. Therefore, we will require that the applicant environmentally qualify these MSIV position switches and related equipment and cabling for DBA conditions.

7.12 Internal Flooding of Safety Related Structures and Equipment

In response to our concerns expressed subsequent to the Quad-Cities 1 flooding incident, the applicant has analyzed the possibility of internal flooding of safety related structures and equipment. The results of this analysis show that there are no safety implications of such flooding other than the loss of offsite power. Notwithstanding this finding, the applicant has installed leak detection instrumentation at the condenser pit which will shut down the circulating water pumps immediately upon indication of a significant line break. We intend to review this area of concern during a site visit, and, if necessary require that the instrumentation provided for recirculation pump tripping meet the requirements of IEEE-279. We do not anticipate any problems in this area, and we will report the results in a supplement to this report.

8.0 ELECTRIC POWER SYSTEMS

8.1 General

The Commission's General Design Criteria 17 and 18, Regulatory Guides 1.6 and 1.9 (formerly Safety Guides 6 and 9 respectively), and IEEE Std. 308-1971 "IEEE Standard Criteria for Class IE Electric Systems for Nuclear Power Generating Stations", served as the primary bases for evaluation of the emergency power systems.

8.2 Offsite Power

The offsite power system for Brunswick 1 & 2 will be fed by eight 230 kV transmission lines connected to the Carolina Power and Light Company's (CP&L) network. Four of the eight transmission lines will serve Unit 2, which is scheduled to be placed into operation about one year before Unit 1, and the other four transmission lines will serve Unit 1. The four transmission lines for each unit will be connected to two 230 kV switchyard buses through double feeder breakers. On loss of a unit's generator, offsite power is supplied by the two 230 kV switchyard buses fed from the four offsite transmission lines. The applicant has stated that the stability studies performed show that tripping of one Brunswick unit will not cause a trip of the 230 kV lines. The Brunswick units will be the largest single units on the CP&L distribution grid. The applicant will submit the results of an analysis to show that the offsite transmission lines routing and associated towers will not mutually jeopardize availability of offsite power to each unit as required by GDC-17. We do not anticipate any problem in this area.

Control and protection power for the 230 kV feeder breakers is provided from two separate 125 Vdc distribution panels fed from each unit's batteries located in the control building. Each 230 kV breaker in the switchyard is equipped with two trip coils and breaker failure protective relaying. We find this design acceptable.

Testability of both the offsite and onsite a-c power systems is provided under conditions as close to design as practical. The applicant will submit for our approval a series of preoperational tests aimed at verifying this capability. Subject to successful completion of these tests, we find that the electric power systems design meets the requirements of GDC-18.

8.3 Onsite Power System

A-C System

Onsite standby power is provided by four diesel-driven generators each feeding a critical 4160V bus. The Brunswick 1 & 2 design is similar to that for Browns Ferry 1 & 2 and Peach Bottom 2 & 3. All four 4160V critical buses are shared by Units 1 & 2. As it was discussed in Section 7.3.1 of this report, the D-G set capacity of 3500 kW continuous rating permits certain improvements in the Brunswick design. The removal of certain blocking logic portions of the ESF initiation logic represents a marked improvement aimed at eliminating a major concern relating to interactions between the two units. The adequacy of the applicant's design is predicated on certain conditions aimed at verifying by tests that the D-G sets have sufficient capacity margins to accommodate the worst loading combination without the blocking logic circuits. The size and model of the D-G sets to be used in Brunswick 1 & 2 has not been utilized as a standby power source in any of the earlier nuclear power plants. Therefore, we have required the applicant to submit a qualification test program and data to demonstrate that the start and load reliability of the D-G sets is at least 0.99 at 95% confidence level. Our testing requirements will be consistent with those established during the review of the Zion 1 & 2 and Cooper applications. Subject to the successful completion of these tests by the applicant, we find the qualification of the D-G sets acceptable.

Each D-G set is housed in a separate room together with its respective auxiliary systems. The rooms housing the D-G sets are designed as Seismic Category I buildings.

The onsite distribution system, other than those parts associated with the four 4160V critical buses are not shared by Units 1 & 2. They follow a unitized, divisionally separated design concept. No automatic transfers are to be performed with the exception of certain loads related to LPCI injection and recirculation loop selection. We will require that the applicant remove all other loads from the automatic transfer feature of the present design. Subject to the design changes discussed above and in Section 7.3.1, we find the onsite a-c power system acceptable.

D-C Systems

The 125/250 Vdc power system consists of two battery pairs and four battery chargers per unit and associated buses and electrical circuitry required for the operation and surveillance of the system.

The batteries are of the lead antimony grid construction and have a 1200 ampere hours rating, sufficient to power the plant's critical loads for two hours without the chargers being available. The battery chargers are of the solid-state rectifier type, capable of working independently. Each charger provides 125 Vdc during normal operations, keeps its associated battery fully charged, and recharges the battery after a discharge.

On loss of power to the charger, the battery supplies all required loads.

Each charger is sized utilizing the largest combination of steady state loads with a charging capacity of restoring the battery from the design minimum charged state to the fully charged state in approximately 8 hours under full load condition.

All critical loads are supplied by redundant batteries and load switching between buses is performed manually. Each battery system is housed in a separate and independently ventilated room.

It was discovered during our drawing review that the control logic power buses A & B for the ESF logics supplied from the d-c power systems were inadvertently mixed in the process of interfacing of the systems supplied by the NSSS supplier and those supplied by the architect-engineer. We expect a written report from the applicant defining the extent of the problem and specifying the corrective actions taken. Subject to the satisfactory correction of this problem, we find the standby d-c power system acceptable.

9.0 Auxiliary Systems

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

The new fuel will be stored in a dry storage vault equipped with storage racks. The racks are seismic Category I by design and are located in the Category I spent fuel pool area of the reactor building. The racks are designed to maintain a subcritical array for a maximum inventory of one third of a core load, even if the racks were inadvertently flooded with water. In addition, the access plug to the vault, located on the operating floor is 2 ft thick and is capable of withstanding the effects of the postulated internal and external missiles. In the event of vault flooding, the floor slopes to an open drain. We conclude that the new fuel storage facility is adequate.

9.1.2 Spent Fuel Storage

The spent fuel storage pool is designed to seismic Category I criteria and is located within the reactor building. There is sufficient storage rack capacity to accommodate spent fuel from one and one third cores. The Category I designed racks assure the maintenance of a subcritical configuration for the spent fuel assemblies. Maintenance of the pool's water level over the top of the spent fuel is assured by eliminating any pipe connections below a level 3 feet above the top of the spent fuel.

The spent fuel shipping cask storage area is bounded on two sides by concrete walls which are lined with a stainless steel liner; the other two sides are formed by a stainless steel framework.

The design of the Brunswick fuel storage pool conforms to all the provisions of Regulatory Guide 1.13 Fuel Storage Facility Design Basis, except that provision on protection from a fuel cask drop over the loading area or from the operating floor area. The applicant has proposed a "redundant" crane system to resolve the concern in this regard; this crane system is currently under review by the Regulatory Staff. We conclude that the Brunswick fuel storage design is acceptable, but that, until the above cited review is completed, the installed crane system would not be used for fuel cask handling. We believe the staff's concern in this area can be resolved prior to the first refueling.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

Each of the Brunswick reactors has a separate spent fuel pool cooling and cleanup system designed to maintain the water quality and clarity and to remove the decay heat generated by the spent fuel assemblies that are stored in the pool. The system is cross-connected to the residual heat removal system by a seismic Category I arrangement of double valves to assure the capability for system isolation in the event of a failure of one of the interconnecting valves.

We evaluated the Category I water makeup supply in accordance with the provisions of Regulatory Guide 1.13 - Fuel Storage Facility Design Basis. The residual heat removal system used for supplemental heat removal includes piping designed to Category I requirements, extending up to and including the embedded penetrations through the fuel pool liner. Water source redundancy is satisfied by the residual heat removal system and by a Category I connection to the service water system which can, if needed, provide seawater to the fuel pool.

Reactor operation would be prevented during the interval that the residual heat removal system is interconnected with the spent fuel pool cooling system for supplementary cooling purposes.

We conclude that the system design along with the restriction noted above, is acceptable.

9.1.4 Fuel Handling System

The safety implications of dropping the spent fuel cask have been discussed in Section 9.1.2. We have accepted the results of an applicant study on reactor vessel servicing, which concludes that if the vessel head were dropped inadvertently on to the reactor vessel, there would be no release of radioactivity or other unsafe condition. Therefore, except for the condition regarding redundancy in the crane system as stated in Section 9.1.2 of this report, we conclude that the fuel handling system design is acceptable.

9.2 Water Systems

9.2.1 Service Water System (SWS)

The service water system removes heat from the residual heat removal system (RHR), RHR pump seals, core spray coolers, RHR pump rooms, emergency diesel engines, reactor building cooling system, turbine building cooling system, and circulating water pump bearings. The SWS also serves as a back-up system for core flooding.

There are separate service water systems for each reactor plant but the service water pumps themselves are housed in a shared service water intake structure. The emergency power source for the service water pumps, in the event of a loss of offsite power, is provided by a shared emergency diesel generator system.

The SWS has two redundant headers. One is identified as the conventional header, while the other is termed the nuclear header. During normal plant operation, the conventional header serves the non-vital systems, the reactor building and turbine building closed cooling systems, and the circulating water pump bearings. Both headers, however, are designed to seismic Category I criteria. Where essential piping of Category I design inter-connects with piping for non-vital cooling systems, isolation capability is assured by seismic Category I double valves and connecting piping. Both headers normally operate independently; in the event one of the headers were to fail, alarms will alert the control room and the cross connecting valves can then be operated either remotely or locally to isolate the failed header. Both of the headers and the pump discharge valving are sized so that there will be adequate capacity for the emergency condition, even if one of the headers were to fail.

To detect a release of high activity levels into the SWS, the applicant has provided a number of multi-channel radiation monitors. One is placed in the common discharge line of the two reactor buildings closed cooling water heat exchangers, one in the common discharge line of each of the two RHR Service water heat exchangers, and one in the common line of each pair of RHR pump seal cooling exchangers. A radiation monitor is also located in the service water discharge line to the discharge tunnel. Continuous recordings of radioactivity levels are made and annunciation is provided in the control room for both high radiation level and equipment failure.

Features of physical separation, isolation from lesser quality systems, remote valve control, and improved radiation monitoring, are the bases for our conclusion that the system is acceptable.

9.2.2 Reactor Building and Turbine Building Closed Loop Cooling Water Systems

The Reactor Building and Turbine Building Closed Loop Cooling Water Systems are two separate systems neither of which are shared between Units 1 and 2. These systems have not been designed to seismic Category I Criteria outside of the drywell since none of the equipment served by the systems outside containment is essential for safe reactor shutdown. However, the piping and equipment of the Reactor Building Closed Loop located inside the drywell are designed to seismic Category I criteria for protection against any potential flooding of vital equipment in their vicinity.

We have reviewed the plant arrangement and have determined that any failure in the non-seismic portion of these systems will not degrade the plant's capability for achieving and maintaining a safe shutdown condition.

We conclude that the system design is adequate.

9.2.3 Demineralized Water Make-up System

The demineralized water make-up system is shared by both units. The system capacity in terms of pumping and storage is consistent with that of presently operating reactors of this type. Each reactor unit has its own condensate water storage tank served from one demineralized water storage tank. Our review of the design determined there were numerous closed gate valves and check valves in the system so that any contamination in one unit will not spread to the other unit or to the demineralized water storage tank. We believe the potential for contamination of the demineralized water storage tank is sufficiently remote and therefore conclude that the system is adequate.

9.2.4 Potable and Sanitary Water System

The potable and sanitary water system is served from a single well, which provides drinking water, sanitary service water, and water for plant demineralized makeup. Our review concentrated on the potential for contamination of the potable system and raw water well source by radioactivity. Numerous check valves exist in the potable water system. Even in the event that some contamination leaked past the check valves, it will not contaminate the well since the well water inlet to the potable water storage tank is located up in the air space level, which is above the water levels to be maintained by the level control system. We conclude that the system design and system precautions against radioactive contamination are adequate.

9.2.5 Ultimate Heat Sink

The ultimate heat sink consists of brackish water provided from the Cape Fear River estuary, routed by an open canal and channeled to a Service Water Intake Structure shared by both units. Within the structure, individual service water pump and header systems are separated for each unit and are designed to seismic Category I criteria.

The heated service water is routed by an open canal and through a diffusion facility and then is discharged into the Atlantic Ocean.

The heat removal system satisfies the provisions of Regulatory Guide 1.27, Ultimate Heat Sink, and we conclude that it is adequate for the service intended.

9.2.6 Condensate Storage Facilities

Separate condensate storage facilities are provided for each unit; but they are provided with a cross connect line equipped with two normally closed valves. Adequate separation of facilities is thus provided, yet the capability exists for interchange of supply when required. The system has not been designed to seismic Category I criteria except for those portions of the piping which are located within the reactor building. These Category I portions supply the HPCI, RCIC, and CRD systems, the test connection to the core spray system, and the RCIC/HPCI full flow test return line.

In the event of failure of the non-seismic portion of the storage facilities, the HPCI and RCIC systems have a Category I designed connection to the suppression pool, which has been designed to seismic Category I criteria.

In the event of piping failure, storage tank drain-down oof the water level would be annunciated by the computer in the control room. In this event, the water in the other storage tank is available, as a backup. Manual valves are supplied to serve the transfer pumps both for maintenance, and for the usage of the backup storage tank.

We conclude that the system is adequate.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

Each unit is equipped with a compressed air system. This system, through special valving is divided into the instrument air system, which is dry and oil-free, and the service air system which is piped to numerous hose connections for miscellaneous plant uses. Although each unit has a separate system, a cross-connect is provided between air receiver manifolds which is equipped with a locked-closed valve. Sharing, though possible, is not a normal mode of operation.

Within the instrument air system are two independent loops, one termed interruptible and the other split into two headers which are termed noninterruptible. The two noninterruptible headers serve the Reactor Building systems and enter at separate locations. They supply control air to the main steam isolation valves, scram valves, scram volume vent and drain valves, safety relief valves, and control rod flow regulators. In the event of air supply failure, the inboard and outboard main steam isolation valves are operated by individual accumulators with a capacity for five valve actuations. This accumulator type capability is also provided in seven of the nine safety relief valves. The interruptible instrument air loop provides air to all other pneumatic controllers.

Those valves equipped with accumulators are essential for achieving and maintaining the safe shutdown condition. The capability for valve closures from the seismic Category I design accumulator system is provided in the event of system failure.

We conclude that the system is adequate.

9.3.2 Process Sampling System

Our review of this non-safety related shutdown system was concerned with the adequacy of the placement and number of sample points, as well as the purpose of each sample for plant diagnostic evaluation and determination of equipment performance. We reviewed the design for consistency with previously approved plants of similar type.

We conclude that the system is adequate.

9.3.3 Equipment and Floor Drainage System

There are two separate drainage systems including the contaminated drains system and non-contaminated drains system. The reactor building, turbine building and radwaste building are served by the contaminated drains system. The contaminated floor drains system also serves those portions of other buildings where a contaminated leakage is possible. Roof drains and drains in certain machinery space areas of the turbine building are served by the non-contaminated floor drains systems. Each system has its own separate sump, with contaminated liquids being routed to the radwaste systems, and non-contaminated liquids being routed to the discharge canal.

We have reviewed a tabulation of all reactor building sumps, their size, capacity, sump pump capacity and probable filtrate. We have also reviewed the same information for the radwaste building, turbine building, and filter house sump. We conclude the system design is adequate for the expected drainage. In the event of severe rain storms at the site, operating precautions will be observed in the event the roof drains and normal gravity flow are inadequate to accept the resultant run-off. These precautions are stated in Section 2.4, Hydraulic Engineering of this report.

9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

9.4.1 Control Room

The single Control Building, which serves both reactor units, has a separate ventilation and summer cooling system which services only the Control Building. Rooms which receive this ventilating air are the computer rooms, electronic workrooms, battery rooms, mechanical equipment room, cable spreading room, and illuminating and unit substations. The air is routed first from the clean areas, i.e., control and computer rooms and then to the other areas such as the battery and mechanical equipment rooms. The Control Building is tornado proof and designed to seismic Category I criteria. The ventilation equipment, controls and ductwork supports are also designed to seismic Category I criteria. The fans are redundant, and have redundant power supplies connected to the emergency bus. Separate ducting and fans condition the air for the control room, computer rooms, and electronic workrooms. The remaining areas are serviced only by the ventilation system and not by summer air conditioning system.

Should a radiation incident or an accidental release of noxious gas occur, detectors for radiation, smoke, and chlorine gas will provide an alarm signal. The dampers can then be closed and conditioned inside air recirculated with 100 cfm of outside air entering the activated charcoal-HEPA filter train, which is capable of removing any odors, smoke, or airborne radioactivity. This feature is intended to maintain habitability of the control room, computer room, and electronic workrooms during an abnormal condition.

We conclude that the system is adequate.

9.4.2 Auxiliary Building

The ventilation systems for the auxiliary buildings serve both units. The only safety related auxiliary building is the diesel generator building which is a separate structure having its own ventilation system. The diesel generator building has provisions for heating, but not for air conditioning. Combustion air for each diesel will be supplied through a separate system, drawing air directly from outside. Loss of heating and cooling will not prevent operation in either summer or winter periods.

The ventilation systems are similar to previously approved BWR ventilation systems. We requested a study of the vertical and horizontal distances from the inlet of each building to the other buildings. We have determined that recirculation problems have been adequately minimized.

We conclude that the systems are adequate.

9.4.3 Radwaste Area

The radwaste building ventilation system has two 50% capacity fans delivering air first to the established clean areas, and then routed to the less clean areas. The discharges from two exhaust fans is routed through roll-type roughing filters, intermediate high efficiency filters, and HEPA filters and then through ducting to the plant stack. The exhaust is continuously sampled and monitored by a stack radiation monitoring system. The output from this monitor is observed in the Control Room. Backdraft dampers are placed in each ventilating duct, providing air to each major area of the radwaste building so that any spread of contamination can be minimized.

The system is similar to that of previously approved BWR plants. We conclude that the system is adequate.

9.4.4 Turbine Building

All outside air enters the building via a fan room through roll-type filters and across steam heating coils in winter and evaporative coolers in summer. Air flow is routed to three basic areas, i.e., the areas of the electric motors, controls, and instrumentation; the areas of tanks, turbine driven pumps and non-temperature sensitive equipment; and the areas of non-heat releasing equipment. The areas of feed pumps, air ejectors and other equipment, which potentially could be radioactive, have their own exhaust system, exhausting through roll-type filters. Both the supply and the exhaust ducts which serve compartments containing combustibles, have fire dampers. Control room readout is provided for the fans that service both the air supply fan duct and the air exhaust fan duct, of those systems which serve potentially radioactive areas.

We conclude that the system is adequate.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

The fresh water fire protection system encircles both units and is a shared system. Yard hydrants are placed at 250 foot intervals and are located 40 feet from building walls. Hose connections, wrenches, and rubber lined hose are housed at each station. Water protection extends into the turbine building, reactor building, radwaste building, auxiliary boiler structure, and shop and warehouse. The fire water is supplied by two deep well pumps, a 300,000 gallon storage tank, a motor driven pump and a diesel driven emergency fire pump.

Fire protection systems for the reactor building, the diesel generator building, and the radwaste building are designed to seismic Category I Criteria. As part of our consideration of the potential for flooding of safety related equipment, we requested the applicant to evaluate the potential for flooding. The applicant confirmed that the water main extending into the reactor building will be normally closed and will have a closed post-indicator located outside the reactor building. As a result of our review, the fire water system within the diesel generator building was changed to a dry system to eliminate the potential for flooding. Any flooding in the radwaste building would not present a safety problem.

The fixed water spray system is actuated automatically in the event of a fire; and area, annunciation is provided in the control room.

Deluge spray is provided for a) the main transformers, b) startup auxiliary transformer, c) unit auxiliary transformer, d) Caswell Beach transformer at plant site, e) hydrogen seal unit, f) reactor feed pump oil consoles, g) turbine lube oil reservoir and, h) reactor recirculation pump motor-generator set fluid drive coupling and oil pump sets. Each unit operator has available in the control room a) annunciators for each sprinkler system, b) fire system pressure, c) fire tank level, d) fire pump status, and e) alarms for areas protected by the fixed carbon dioxide system.

Fixed carbon dioxide protection is provided for the reactor building and high pressure coolant injection pump room. A personnel alarm alerts anyone who may be in the area of the HPCI pump room to evacuate the area.

Portable carbon dioxide, dry chemical, and stored pressurized, wall type, water extinguishers are located throughout the plant. We requested the applicant to furnish the results of an evaluation of the potentially toxic effect of the discharge from the portable carbon dioxide extinguishers in the areas they are located. Areas where carbon dioxide concentrations could exceed 3% are provided with detectors that are annunciated in the control room; but these spaces are not spaces which must maintain habitability in order to mitigate the consequences of an accident or to place the plant in a safe shutdown condition.

We conclude that the fire protection system is adequate.

9.5.2 Diesel Generator Fuel Oil Storage and Transfer Systems

The standby a-c power system consists of four diesel generators, shared by both reactor units. The standby power capacity is sufficient to provide for the post accident emergency power requirements of one reactor following a design basis accident, and of the other unit which will be in the process of achieving and maintaining a safe shutdown mode, assuming a complete loss of offsite power for both units.

The diesel generator building, together with the physically separated diesel units are designed to seismic Category I criteria as is all the connecting piping to the diesel oil storage tanks which are located in separate compartments in an underground vault.

The underground vault (seismic Category I) is adjacent to the diesel generator building with top cover slabs at ground level. Fuel oil (diesel No. 2) makeup for these storage tanks is provided from a nearby ground level, large storage tank which also supplies oil to the auxiliary boiler. This large storage tank has not been designed to seismic Category I Criteria. The ground level tank has a dike enclosure to limit any spread of flammable liquid.

We informed the applicant of our concern that the underground storage tanks held only a four day supply of fuel oil for cycled-service diesels where three of the four diesels must be continuously on-line to meet the vital power needs. The applicant responded by providing plans for obtaining diesel fuel prior to the four day limit stating that fuel can be delivered by truck, rail, or barge. State Route NC #211 from the west, NC 87 from the northwest, and NC 133 from the north are alternate truck routes. Rail service can receive oil shipments from a junction at Leland, N.C. to a U.S. Army spur track coming direct to the site. A barge from Wilmington can navigate the intake canal and also serve the site. In addition, the applicant will place priority demand on their nearby power plants at Sutton, Lee, Weatherspoon, and Robinson which would also have large quantities of fuel oil on hand.

We believe the fuel oil storage and transfer system meets the intent of IEEE-Std-308-1971 which requires a seven day Category I storage capacity. We acknowledge the fact that construction on the Brunswick Plant began prior to establishment of the standard and believe the applicant has taken proper measures to comply with the intent of the above cited standard.

9.5.3 Diesel Generator Cooling Water System

The cooling system of each diesel engine originates at redundant take-off points from separate service water headers. The entire system up to the jacket water cooler discharge point is designed to seismic Category I Criteria.

The engine cooling system is a closed system with its own surge tank. This tank is connected to a separate make-up demineralizer system for periodic make-up. The closed system also serves to cool the lube oil in an engine mounted lube oil cooler, which is cooled by jacket water.

The service water system provides cooling to the jacket water cooler on each diesel unit. Motor operated valves, automatically open on engine startup according to a pre-selected mode. In the event of low pressure on the service water line which is in use, a redundant system will automatically open.

When not in use, electric heating elements maintain jacket cooling water at approximately the normal engine temperature.

We conclude that the engine cooling water system is adequate.

9.5.4 Diesel Generator Starting Air System

Each diesel engine has two, air-cooled, motor driven starting air compressors connected to two interconnected air receivers. Each receiver is connected to a different bank of 8 cylinders on the 16 cylinder Vee engines(s). Either bank is capable of starting the engine through the direct injection of starting air through timing sequence valves. Each compressor has controls arranged to maintain a constant receiver pressure to crank the engine for 30 seconds. This system is capable of performing three to four successive starts. Compressor power is from an associated emergency bus.

We conclude that the system is adequate.

10.0 Steam and Power Conversion System

10.1 Summary Description

The steam and power conversion system is of conventional design, similar to previously approved direct cycle BWR plants. There is one difference between the units in that Unit No. 2 has the condenser design capacity and turbine bypass system to accept 105% of full load steam flow. Unit No. 1 is provided with 25% bypass capacity coupled with a select rod insertion system and a delay in scram. The feature in Unit No. 2 enables the applicant to reject full steam flow with minor reactor manipulation. In the event of electrical load rejection, and after the fault is cleared, the Unit No. 2 reactor can supply steam to the turbine without significant loss of time. The feature is an owner-option.

10.2 Turbine-Generator

Each turbine is an 1800 rpm tandem-compound, four-flow, two-stage reheat unit, with 43 inch last-stage blades. It has one high pressure double flow cylinder, and two double flow low pressure cylinders. The turbine-generator is similar to those in previously approved BWR plants.

10.3 High Energy Line Rupture Outside Containment

The main steam and feedwater lines are physically isolated from structures, systems, and components important to safety by a reinforced concrete piping tunnel. The two masonry filled tunnel blowout vents have been sized to preclude a pressure buildup greater than 40 psig in the pipe tunnel by blowing out the masonry and venting the steam into the turbine building.

If a feedwater line should rupture and the condensate and condensate booster pumps do not trip, a substantial amount of water could be discharged into the piping tunnel. Failure of a feedwater line can result in the flooding of the northwest core spray room via the floor drainage system. Under the worst conditions, water would fill this room to less than half full and disable one of the two core spray systems; however, the remaining core spray system would be unaffected. The remaining core spray system is a redundant system located in the southwest room on the opposite extreme from the affected room.

Whipping of a failed HPCI steam line could result in a failure of a demineralized water line. Should this event occur, flooding could take place in the northeast RHR room. Even if the entire demineralized water storage tank were emptied into one RHR compartment, however, only that half of the RHR system would be damaged. The redundant southwest RHR system would not be affected.

A compartment pressure transient analysis of a break in the HPCI room yielded a differential pressure which would have been excessive. Additional venting area has been provided by replacing concrete plugs in the roof of this room with gratings.

Failure of an RCIC steamline cannot result in any direct failure to any safety system piping, electrical component, or instrumentation important to safety. Over-pressure considerations are less severe than those due to the HPCI steam line break. Failure of the RCIC line will not result in the rupture of any other lines which could cause flooding.

Because of the inherent mechanical separation within the reactor building, failure in any portion of the reactor water cleanup system (RWCU) will not result in the loss of function of any mechanical, electrical or instrumentation component important to safety. An over-pressure analysis has been conducted which indicates that sufficient vent area is available in the RWCU system compartment to assure that unacceptable overpressure conditions will not exist.

Failure of the RHR steamline cannot cause direct damage to or loss of function to any mechanical, electrical, or instrument component important to safety. A break in this line will result in similar but less severe consequences as those delineated for an HPCI steamline break. Failure of this line will not result in rupture of any other lines which could cause flooding.

Failure of any high pressure sampling and instrument sensing lines cannot result in a loss of function of any mechanical, electrical or instrument components important to safety since flow through either of these lines is very small and it would not be possible to create an overpressure condition within the reactor building.

We have evaluated the effects of a high energy line rupture outside containment in accordance with Addendum "A" to the letter, A. Giambusso to the applicant, dated December 15, 1972, and conclude the design to be adequate.

10.4 Other Features of Steam and Power Conversion System

Two deaerating, divided waterbox, single pass condensers will maintain turbine back pressure for all normal operating conditions. The hot well capacity will store enough condensate to provide at least a two minute retention time for the required amount of radioactive decay.

Normal water level in the condenser hot well is maintained by the condensate makeup and surge systems. The makeup system connects the condenser to the 500,000 gallon condensate storage tank. Automatic valves operate to maintain condenser water level. Should the amount of water within the condensate storage tank decrease to 100,000 gallons, the condensate storage tank will be automatically isolated. This 100,000 gallons of water ensures a reserve capacity to supply the RCIC, HPCI and core spray systems.

We conclude that the design of the main condensers and condensate storage tank is acceptable.

Two, two-element, two-stage air ejectors with separate inter-condenser and separate aftercondenser are provided to withdraw non-condensible gases from and maintain a vacuum on each condenser. The ejectors use main steam, reduced in pressure by a regulating valve. The air ejector exhaust is routed to the off-gas removal system. The gases are taken through the off-gas processing equipment and then to the stack. We conclude that the main condenser evacuation system is acceptable.

The steam bypass system provides a bypass around the turbine for steam which is not accepted by the turbine. During startup, hot standby service and physics testing, the same steam bypass system can be manually actuated from the pressure controller to effect a simulated load on the reactor plant. We conclude that the steam bypass system is adequate.

11.0 Radioactive Waste Management

11.1 Design Objectives and Criteria

The radioactive waste management systems will be designed to provide for the controlled handling and treatment of radioactive liquid, gaseous and solid wastes. The applicant's design objective for these radwaste systems is to restrict the amount of radioactive material released to the environment to levels that are as low as practicable.

The Technical Specifications that will be issued as part of the operating license will require the applicant to maintain and use existing plant equipment to achieve as low as practicable releases of radioactive materials to the environment in accordance with the requirements of 10 CFR Part 20 and 10 CFR Part 50. The applicant will also be required to establish a radiation protection management system that will assure that radiation exposures to inplant personnel and to the general public are as low as practicable in conformance with the requirements of 10 CFR Part 20.

Our evaluation of the design and performance of the radioactive waste management system is based on the design objectives for liquid, gaseous wastes and solid wastes as detailed below.

Liquids

1) Provisions are made to treat radioactive liquid waste, to limit the expected releases of radioactive material in liquid effluents to the environment to less than 5 Ci/yr/unit, excluding tritium and dissolved gases.

2) The calculated annual average dose to the whole body or any organ of an individual at or beyond the site boundary are not to exceed 5 mrem for expected releases.

3) Concentration of radioactive materials in liquid effluents are not to exceed the limits in 10 CFR Part 20, Appendix B, Table II, Column 2, for the expected and design basis releases.

Gaseous Wastes

1) Provisions are made to treat gaseous waste to limit the expected release of radioactive materials in gaseous effluent from principal release points so that the annual average dose to the whole body or any organ of an individual at or beyond the site boundary does not exceed 5 mrem.

2) Provisions are made to treat expected radioiodine released in the gaseous effluent from the principal release points so that the annual average dose to the thyroid of a child at the nearest grazing area through the pasture-cow-milk pathway will be less than 15 mrem or the applicant uses state of the art technology to reduce iodine releases, coupled with an extensive environmental monitoring program.

3) Concentrations of radioactive materials in gaseous effluents are not to exceed the limits in 10 CFR Part 20, Appendix B, Table II, Column 1, for the expected and design basis releases.

Solid Wastes

1) Provisions are made to solidify all expected liquid radioactive waste prior to shipment to a licensed burial ground

2) Containers and method of packing are selected to meet the requirements of 10 CFR Part 71 and applicable Department of Transportation regulations.

The following sections present our evaluation of the liquid, gaseous and solid radioactive waste systems; the radiation monitoring of process effluents and of inplant areas; and radiation protection management. The liquid, and solid radioactive waste systems are designed to accommodate the waste produced during simultaneous operation of Units 1 and 2. The gaseous radioactive waste system has separate treatment facilities in each unit for the drywell purge, air ejector offgas, and gland seal offgas.

We find that: (a) the proposed liquid and solid radioactive waste treatment systems; (b) the design codes and quality assurance criteria; (c) the proposed radiation monitoring of process effluents and of inplant areas; and (d) radiation protection management, are acceptable. However, the calculated thyroid dose to a child at or beyond the site boundary through the pasture-cow-milk pathway was determined to be excessive in the Draft Environmental Statement. However, the applicant has committed in its letter dated September 18, 1973, to provide for treatment of the turbine building ventilation exhaust to reduce the thyroid dose to a child to the as low as practicable guidelines. The turbine building ventilation exhaust will be passed through charcoal adsorbers that are no less than six inches deep with an exhaust flow that provides a residence time in the charcoal of seconds. We find this to be acceptable.

11.2 Liquid Waste System

11.2.1 System Description

The liquid radioactive waste treatment system consists of equipment and instrumentation necessary for the collection, monitoring, processing, storage, and disposal of radioactive liquid wastes. The radioactive liquid wastes from Units 1 and 2 will be collected and processed in four subsystems that are common to both units. These subsystems are the high purity, low purity, chemical, and detergent subsystems. The terms high purity and low purity refer to the conductivity of the liquid and not the radioactivity levels. The applicant has provided a surge subsystem that will be used for collection and processing of large quantities of high purity wastes that result fron non-routine operations. The surge subsystem has the capability of processing the liquid wastes through a filter and a demineralizer. Table 11.1 of this section lists the major components and process data for these subsystems.

The radioactive liquid waste treatment subsystem have the capability to process the waste by evaporators, demineralizers, filters and a reverse osmosis unit. Treatment of the waste will be dependent on the source, activity, and composition of the liquids. Cross connections between the subsystems will provide flexibility for processing by alternate methods. Prior to release, samples will be analyzed to determine the quantity of the radioisotopes in a processed batch. Based on the results of the analysis, the wastes will be released under controlled conditions to the Atlantic Ocean on retained for reuse or for further processing. During release of the processed liquid wastes if the concentration of radioisotopes exceeds specified limits, a radiation monitor will alarm in the control room and automatically terminate the release.

11.2.2 Low Purity Waste

The low purity subsystem will collect and process liquid wastes from the drywell, reactor building, radwaste building and turbine building floor drain sumps. Low purity liquid radioactive wastes contain impurities that require extensive treatment prior to reuse in the plant. These liquid wastes are collected in the 12,500 gal floor drain collector tank. The principal treatment for liquid wastes from the floor drain collector tank will be through the floor drain filter and waste evaporator or reverse osmosis unit and waste demineralizer. The liquid waste will be analyzed in one of the two 23,000 gallon floor drain sample tanks. Based on the results of the analysis, the wastes will be discharged to the waste neutralizer tanks or to the waste collector tank. The effluent from the floor drain sample tank may be discharged through a reverse osmosis unit to the respective treatment subsystem. The reverse osmosis process consists of pressurizing the floor drain sample tank effluent and bringing it in contact with a semi-permeable membrane. The permeate flow to the waste collector tank and the concentrate flows to the waste neutralizer tanks.

In our evaluation, we assumed that these liquid wastes will be collected at a rate of 9000 gpd per unit in the 12,500 gal floor drain collector tank. The assumed collection rate provides approximately one day of delay before processing. We assumed that the analysis of the floor drain sample tank would result in 50% of the collected liquid wastes being processed in high purity subsystem and 50% of the collected liquid wastes being processed in the chemical waste system.

11.2.3 High Purity Waste

The high purity subsystem will collect and process liquid radioactive wastes from the drywell, the radwaste and turbine building's equipment drain sumps, the reactor building's equipment drain tank, the floor drain sample tank, and the condensate phase separators. High purity liquid radioactive wastes contain few impurities and require little treatment prior to reuse in the plant. These liquid radioactive wastes are collected in the 38,000 gal waste collector tank. The principal treatment for the liquid wastes from the waste collector tank will be through the waste filter and waste demineralizer. The processed radioactive liquid wastes will be analyzed in one of the two 23,000 gal waste sample tanks. Based on the results of the analysis and coolant requirements for the plant, the processed liquid wastes will be discharged to the circulating water discharge canal or sent to the condensate storage tank for reuse.

In our evaluation, we assumed that the radioactive liquid wastes in the high purity waste treatment subsystem will be collected at a rate of 14,000 gpd per unit in the 38,000 gal waste collector tank. This assumed collection rate along with 50% of the low purity wastes provides approximately one day of delay before being processed through the waste filter and waste demineralizer. We assumed that after processing, 10% of the liquid wastes will be released from the waste sample tank to the circulating water discharge canal and that 90% will be reused in the plant.

11.2.4 Chemical Waste

The chemical waste subsystem will collect and process liquid wastes from the condensate demineralizer regenerants, the decontamination drains, and the laboratory wastes. These are collected and treated in one of four 17,500 gal waste neutralizer tanks. The principal treatment for liquid wastes from the waste collector tanks will be through one of two 20 gpm waste concentrators and the waste demineralizer. The processed liquid wastes will be analyzed in the waste sample tank. Based on the results of the analysis and coolant requirements in the plant, the liquid wastes will be discharged to the circulating water discharge canal or sent to the condensate storage tank for reuse.

In our evaluation, we assumed that the liquid wastes in the chemical waste subsystem will be collected at a rate of 8,000 gpd per unit in the four 17,500 gal waste neutralizer tanks. This assumed collection rate along with 50% of the low purity wastes provides approximately 3 days of delay before being processed through the waste concentrator and waste demineralizer. After treatment, we assumed that 10% of the processed liquid wastes will be released from one of the two 23,000 gal waste sample tanks to the circulating water discharge canal and that 90% will be reused in the plant.

11.2.5 Laundry Wastes

The laundry waste subsystem will collect and process liquid wastes from laundry drains, cask cleaning, and personnel decontamination. These liquid radioactive wastes are collected in one of the two 1,200 gal detergent drain tanks. The principal treatment for liquid wastes from the detergent drain tanks will be through the detergent drain filter to the circulating water discharge canal.

Based on the experience at operating reactors, we consider the potential effluents from the detergent drain tanks to be a negligible portion of the total plant releases of radioactive material in liquid effluents. The applicant estimates that the expected annual release rate from the detergent drain tanks will be 0.07 Ci/yr/unit based on processing four batches per day at an activity concentration of 10^{-5} μci/ml. We consider the applicant's estimate to be reasonable and agree with the estimated release rate.

11.2.6 Conclusion

For our evaluation of the liquid radioactive waste treatment system, we estimate the release rate to be 1.4 Ci/yr, which has been adjusted to compensate for equipment downtime and expected operational occurrences for a total of 3.0 Ci/yr/unit, exclusive of tritium and noble gases. Release of tritium from the plant is estimated to be 20 Ci/yr, based on operating experience at boiling water reactors. For comparison, the applicant estimates a release rate from all liquid waste sources of 4.6 Ci/yr exclusive of tritium and dissolved gases. The applicant has not provided an estimate for the release of tritium from the plant. Our estimated annual release of radioactive material in the liquid effluents is based on use of the orgin code and of our liquid treatment model which was adjusted to apply to this plant. The staff's model uses somewhat different values for the input parameters when compared with those of the applicant. Our calculated effluents are therefore different from those of the applicant. From our evaluation of the liquid radioactive waste releases, we calculate that the whole body and critical organ doses are less than 5 mrem/yr at the site boundary.

Based on the results of our evaluation that the proposed systems are capable of reducing liquid releases to less than 3 Ci/yr/unit, we conclude that the liquid radwaste system will reduce radioactive effluents to as low as practicable, in accordance with 10 CFR Part 50 and will meet the requirements of 10 CFR Part 20, and therefore is acceptable.

11.3 Gaseous Waste System

11.3.1 System Description

The gaseous waste treatment system consists of equipment and instrumentation necessary to process radioactive gases and airborne particulates from the reactor, plant equipment and building vents. The gaseous waste treatment system will process waste gases from the condenser offgas, air ejector offgas, and drywell. Sources of untreated gaseous wastes are the turbine building, reactor building, and radwaste building exhaust ventilations.

The major source of gaseous radwaste during normal plant operation will be the offgas from the main condenser. The offgas from the main condenser will be processed through: (a) a 30 minute holdup pipe to permit decay of short-lived radioisotopes; (b) a catalytic recombiner to reduce the volume of gases to be treated; (c) a condenser to remove the water vapor; and (d) a liquid nitrogen cryogenic distillation column to remove the noble gases. The cryogenic distillation columns will liquify the xenon and krypton gases and concentrate these gases in the liquid oxygen bottoms. Residual nitrogen and traces of argon will be continuously vented. The liquified xenon and krypton gases will be periodically bled off to a radioactive gas recombiner to remove the oxygen. The remaining gases, principally xenon and krypton, will be stored in pressurized cylinders for further decay before releasing to the plant stack.

11.3.2 Condenser Offgas Treatment System

Each unit has a condenser offgas treatment system that is capable of processing the offgas stream from either or both units. The offgas treatment system is designed to seismic Category 1 criteria requirement and is housed in a seismic Category 1 structure. To minimize the possibility of an explosion in the system from ozone and hydrocarbons, the applicant has committed to provide for continuous monitoring by at least two temperature or hydrogen monitors downstream of the feed gas recombiner. The components and piping in the feed gas recombiners and upstream of these recombiners have been designed to withstand hydrogen explosion pressures. The applicant has evaluated the consequences of a detonation. We find the applicant's evaluation reasonable and acceptable.

In our evaluation, we assumed that the air inleakage flow rate to the turbine condenser will be 20 scfm, which after the recombiner is the flow rate through the offgas cryrogenic treatment system. We assumed that the cryrogenic treatment system will provide an activity reduction of 10^4 for both iodine and xenon and 4×10^3 for krypton. We assumed that the delay time for concentration of the waste gases in the liquid oxygen cryogenic distillation column bottoms and hold up in the gas storage tanks will provide 60 days of decay for the radioisotopes in the waste gases.

The ventilation system for the offgas service building is a once through system which is designed to maintain a negative pressure inside the building. The exhaust air flows through HEPA filters to the plant stack. A radiation monitor is provided in the exhaust duct from each offgas treatment system compartment. Each monitor will shutdown the air supply to the respective compartment whenever concentrations of radioactive material in the air exceed a specified level. The applicant does not consider this source to contribute to the expected release of radioactive material from the plant. We consider the applicant's conclusion to be reasonable and agree with that conclusion.

11.3.3 Turbine Gland Seal

The turbine gland seal system will use primary system steam. Therefore, the gases released from the turbine gland seal condenser can be radioactive. The gland seal offgas system will collect the gases from the gland seal condenser and hold them up for approximately 2 minutes before being exhausted to the plant stack. In our evaluation, we assumed that 0.001 of the main steam flow will leak through the turbine gland seals. We also assumed a partition factor of 0.01 for iodine that flows across the gland seal condenser. For our evaluation of the mechanical vacuum pump gaseous processing system, we assumed 16 hours of operation per year.

11.3.4 Plant and Drywell Ventilation

The turbine, reactor, and radwaste building ventilation systems will utilize prefiltered air without recirculation. All ventilation systems are designed to direct the air flow from clean areas to areas having a greater potential for radioactive contamination. Normally the ventilation air in the reactor building will be discharged without treatment to the reactor building vent. The radwaste building air will be passed through a HEPA filter and released through the reactor building vent. The reactor building ventilation system will be equipped with isolation values so that in the event of high radioactivity, the air will be routed through the standby gas treatment system which consists of HEPA filters and charcoal adsorbers in series. The applicant assumed negligible releases from the reactor and radwaste building ventilation exhausts. In our analysis we calculated negligible releases of noble gases from the reactor and radwaste buildings and 0.01 Ci/yr of iodine-131 from the reactor building due to the primary coolant leakage.

The applicant has committed to treating the turbine building ventilation exhaust using a method similar to that being designed for the Hatch 1 plant. The Brunswick turbine building exhaust treatment system will utilize charcoal adsorbers that are no less than six inches deep and an exhaust flow rate that provides a residence time in the charcoal of 0.25 sec. We assumed that 1700 lb/hr of steam leaks into the turbine building.

The primary containment (drywell) is normally a sealed volume. However, during periods of refueling or maintenance it may be necessary to purge the drywell and suppression chamber. When this occurs, the potential exists for the release of airborne radioactivity to the environment. The system will be designed so that the purge exhaust will be directed to the standby gas treatment system. The Technical Specifications will require the use of the standby gas treatment system for all purge operations. We conclude that releases from this source will be negligible.

11.3.5 Conclusion

We estimate that a total of approximately 10,000 Ci/yr/unit of noble gases and 0.11 Ci/yr/unit of I-131 will be released from the plant, based on an activity release rate of 100,000 μci/sec after 30 minutes of delay. The applicant estimates that approximately 5500 Ci/yr/unit of noble gases and that no I-131 will be released from the plant. The applicant's estimate is based on an activity release rate of 25,000 μci/sec after 30 minutes of delay, and on no releases from the cryogenic distillation column bottoms.

We calculate the whole body dose due to noble gases to be less than 5 mrem/yr to an individual at or beyond the site boundary. Although we calculate the dose to be above 15 mrem/yr to a child's thyroid due to the pasture-cow-milk chain with the cow at the nearest grazing areas 0.75 mile southeast of the plant, the applicant is using state of the art technology for treatment of the gaseous effluents.

Based on our evaluation that the gaseous radioactive waste system will process the expected noble gases such that the calculated whole body dose will be less than 5 mrem/yr to an individual at or beyond the site boundary, and the applicant uses state-of-the-art technology to keep the dose to the thyroid as low as practicable, we conclude that the proposed gaseous waste system is acceptable.

11.4 Solid Radioactive Waste Management

The solid waste processing system consists of equipment and instrumentation necessary for disposal of radioactive solids resulting from operation of the reactor joolant clean up system, the condensate clean up system, the liquid waste processing system, and miscellaneous debris arising from normal operation and maintenance of the plant. The system will be located in the radwaste building and is common to both units. The system will consist of an evaporator bottoms handling facility, a demineralizer and resin dewatering facility, a waste mixer, a waste compactor, and a shipping container handling facility.

The evaporator bottoms will be collected in a 20,000 gal concentrate waste tank. A 30 gpm transfer pump and flow rate meter will meter the bottoms to the waste mixer. An absorbent is added and the mixture is transferred to DOT approved containers and stored for shipment. The processing of evaporator bottoms is handled remotely behind shielded walls.

The waste filter demineralizer will be sent to phase separators. The decante from the phase separators is sent to the high purity liquid waste treatment subsystem. The slurry from the phase separators and the spent resins will be processed through centrifuges. The liquid from the centrifuges is sent back to phase separators. The dewatered resins will be packaged in DOT approved containers and stored before shipment.

The waste compactor will be used to compact dry non-process type waste such as protective clothing, rags, paper, tools, etc. These wastes will be collected in containers located in appropriate zones around the plant.

The shipping containers will be water tight vessels designed to permit remote, automatic filling. Container filling will be conducted inside a totally enclosed shielded cubicle. The applicant estimates that 1,265 drums of solid waste with a gross activity of 455 curies will be shipped from the plant per year. Based on BWR plant operating experience, we estimate 1,100 cubic feet of solid waste with a gross activity of 2700 curies will be shipped per year. The system will be similar to those previously reviewed and disposal of solid radioactive waste from Brunswick Units 1 and 2 will conform to AEC and Department of Transportation regulations. We find the proposed system acceptable.

11.5 Design

The proposed radwaste systems are designed to meet the existing codes and standards. The turbine condenser offgas system is designed to requirements of Quality Group C. The gland seal offgas system and liquid radwaste system, except the waste evaporator package, is designed to meet the requirements of Quality Group D. The offgas service building and the radwaste buildings which house the liquid, solid, and gaseous radioactive waste treatment systems, are designed as seismic Category 1 structures. The liquid storage tanks containing high activity and turbine condenser offgas processing system components are also design to seismic Category 1 criteria.

We conclude that the radwaste system is designed to the appropriate codes and standards and are acceptable.

11.6 Process and Area Radiation Monitoring Systems

The process radiation monitoring system is designed to provide information on radioactivity levels of systems throughout the plant, on leakage from one system to another, and on levels of radioactivity released to the environment. The system will consist of a main steam line monitor, an air ejector offgas monitor, a main stack monitor, an air ejector offgas monitor, a main stack monitor, reactor building ventilation exhaust monitors, turbine building ventilation exhaust monitors, turbine building ventilation exhaust monitor, and liquid radwaste system effluent monitor. In addition, a hydrogen monitor in the feed gas recombiner outlet stream will shut the hydrogen injection valve to preclude further hydrogen addition whenever a high hydrogen content is detected. Furthermore, the applicant has provided temperature monitors in the feed gas recombiner outlet that will place the plant in a safe condition whenever the hydrogen concentration becomes greater than 4%.

An area radiation monitoring system is provided which reports radiation fields in various areas. The system will consist of monitors in the control room, stack, offgas service building, turbine building, reactor building and radwaste building. This system will detect, indicate, annuniciate and/or record the levels or field of activity to verify compliance with 10 CFR 20 and keep radiation levels as low as practicable. We conclude that the plant is adequately provided with process and area monitoring equipment.

11.7 Radiation Protection Management

The objective of the radiation protection management system is to ensure that radiation exposure to plant personnel is as low as practicable. The applicant will establish health physics procedures under the direction of the Health Physics Supervisor which will assure that all requirements related to radiation protection are followed by all plant personnel. These procedures will provide rules for personnel monitoring, use of protective clothing and equipment and will require a Radiation Work Permit to be obtained for certain areas having a potential for radiation exposure. Supporting data regarding the effectiveness of the health physics program will be obtained through the collection of bioassay samples, comprehensive medical examinations and film badge or thermal luminescence dosimeter (TLD) data.

All areas within the plant will be identified by different radiation zones in accordance with the expected maximum occupancy. The applicant will provide five areas of radiation control within the plant according to maximum design radiation dose rate, D. These are (1) $0.0 \leq D \leq 0.5$ mrem/hr, (2) $0.5 \leq D \leq 2.5$ mrem/hr, (3) $2.5 \leq D \leq 15$ mrem/hr, (4) $15 \leq D \leq 100$ and (5) $D \geq 100$ mrem/hr. These areas will be identified by radiation caution signs at conspicuous locations.

Personnel monitoring equipment shall be provided for all personnel at the plant. All plant employees will wear film badges that contain neutron and gamma film which will be processed monthly. Records showing the radiation exposures of all personnel at the plant will be maintained by the applicant. The records will contain as a minimum a monthly tabulation of readings from beta-gamma film badges. Periodic whole body counts will be made to determine internal exposures to plant personnel. Protective clothing and respiratory protective equipment will be available for the protection of personnel, when required. Portable radiation monitoring instruments will be available to determine exposure rates and contamination levels in the plant.

The applicant's design objective for radiation shielding for normal operation, is to maintain whole body dose rates for all controlled access areas of the plant to less than 60 mrem per week, considering occupancy of each controlled access area. For areas outside the plant the shielding design objective is to maintain whole body dose rates to less than 0.5 rem per calendar year. The principal shielding material used in the plant is ordinary concrete. Other materials will be used by the applicant for special situations. Equipment, pumps, valves, and pipes that will contain significant levels of radioactive material will segregated into modules by shield walls to minimize radiation exposures from performing maintenance on these items. We conclude that precautions taken for personnel protection satisfy the requirements of existing regulations as pertains to exposure of individuals to radiation, and are acceptable.

11.8 Offsite Radiological Monitoring Program

The applicant has undertaken a comprehensive radiological environmental monitoring program in the vicinity of the plant for the purpose of verifying that the design objectives of the radioactive waste control and monitoring systems are met. The program is designed to monitor both direct and indirect pathways to man and will also serve to monitor changes in environmental radioactivity levels that may result from plant operation. The scope of the program is comparable to that of other nuclear facilities currently in operation or being licensed, and it meets or exceeds the provisions of Regulatory Guide 4.1 and of the Environmental Protection Agency as set forth in "Environmental Radioactivity Surveillance Guide" - ORP/SID 72-2.

Preoperational measurements were started early in 1972 and will continue until plant start-up which should provide more than two years of information to serve as a baseline for evaluating the impact of the plant.

The program was submitted for a thorough staff review, which resulted in a number of modifications. The most significant change was a shift in emphasis to placing sampling locations near the plant. Milk samples from the nearest family cow were added as well as locally grown milk-cow feed. Semiannually, the applicant will survey the area around the plant for the location of milk-producing animals and change sampling locations based on these surveys as necessary. Turnips and collard greens from nearby farms were added to the program. Also sample locations were added for aquatic vegetation and benthic organisms and for shoreline sediments in the vicinity of the plant discharge. Gross radioactivity analysis has been replaced by specific radionuclide analysis for most samples in order to further specify the contribution of the plant to observed environmental radioactivity.

We conclude that the applicant's program will be adequate for monitoring the radiological impact of plant operation on the environs and for verifying predictions of concentrations of specific radionuclides in the environment based on effluent measurements. The program is considered adequate to monitor the control of radioactivity with regard to the health and safety aspects of the release of radionuclides to the environment from the proposed operation of the plant.

11.9 Conclusions

Based on our model and assumptions, we calculate an expected whole body and critical organ dose of less than 5 mrem/yr from both units effluents, and releases of less than 5 Ci/yr/unit from liquids at or beyond the site boundary. We calculate the expected whole body dose from gaseous effluents from both units to be less than 5 mrem/yr to an individual. Even though we calculate a critical organ dose to a child's thyroid from gaseous effluents to be in excess of our 15 mrem/yr guideline, we find the gaseous system for the treatment of iodines to be acceptable because the applicant has used state-of-the-art equipment to reduce iodine releases in accordance with Regulatory Guide No. 1.42. The solid radwaste system has adequate capacity and solid radwaste shipments will be in accordance with AEC and Department of Transportation regulations. Therefore, we conclude that the proposed liquid, gaseous and solid radwaste systems for Brunswick 1 and 2 are acceptable.

We conclude that the proposed system is designed in accordance with acceptable codes and standards, that the process monitoring system is adequate for monitoring effluent discharge paths as specified in Criterion 64 of Appendix A to 10 CFR Part 50, and that the personnel protection systems satisfy the requirements of existing regulations as they pertain to exposure of individuals to radiation.

TABLE 11.1

COMPONENT DATA FOR BRUNSWICK LIQUID RADWASTE SYSTEMS

	No.	Capacity
Low Purity Subsystem:		
Floor Drain Collector Tank	1	12,500 gal.
Floor Drain Sample Tank	2	23,000 gal.
Drywell Floor Drain Sump	1	500 gal.
Reactor Building Floor Drain Sump	2	1,000 gal.
Radwaste Building Floor Drain Sump	2	1,000 gal.
Turbine Building Floor Drain Sump	1	1,000 gal.
High Purity Subsystem:		
Drywell Equipment Drain Sump	1	500 gal.
Waste Surge Tank	1	60,000 gal.
Reactor Building Equipment Drain Sump	1	1,000 gal.
Auxiliary Surge Tank	1	200,000 gal.
Radwaste Building Equipment Drain Sump	1	1,000 gal.
Waste Sample Tank	2	23,000 gal.
Turbine Building Equipment Drain Sump	1	1,000 gal.
Chemical Waste Subsystem:		
Waste Neutralizer Tank	4	17,500 gal.
Waste Concentrator	1	20 gpm
Concentrated Waste Tank	1	20,000 gal.
Detergent Waste Subsystem:		
Detergent Drain Tank	2	1,200 gal.

TABLE 11.1 (cont'd)

COMPONENT DATA FOR BRUNSWICK LIQUID TO SOLID RADWASTE SYSTEMS

	No.	Capacity
Condensate Backwash Receiving Tank	2	11,000 gal.
Cleanup Backwash Receiving Tank	1	3,000 gal.
Spent Resin Tank	1	3,000 gal.
Waste Sludge Tank	1	13,000 gal.
Condensate Phase Separator	4	13,500 gal.
Cleanup Phase Separator	2	4,000 gal.
Centrifuge	2	20 gpm

12.0 RADIATION PROTECTION

This section presents an evaluation of the adequacy of the shielding, the ventilation and the health physics program to control radiation exposures within the requirements of 10 CFR Parts 20 and 50 of the Commission regulations. Because the facility design was essentially complete, emphasis in this review was placed on plans and procedures for radiation exposure reduction.

12.1 Shielding

The radiation shielding provided has been designed with the primary objective of minimizing the radiation exposure of plant operating personnel and of the general public. The requirements of 10 CFR 20 were used as a basis. Standard methods were used by the applicant to evaluate the shield design. Staff calculations at selected locations indicate that the shielding provided will be adequate to meet designated radiation zone requirements.

Information provided in the FSAR as well as observations made during the site visit show that the general principle of shielding compartmentalization for major components which are expected to contain radioactivity has been employed. In general enough room has been provided to allow for maintenance and for placement temporary shielding if necessary. Temporary shielding in the form of lead sheeting, lead bricks and concrete blocks will be provided for special maintenance jobs as required. Only nonradioactive process piping is "field-run" so that all radioactive process piping is routed by the architect-engineer.

Based on the applicant's experience with H. B. Robinson Unit 2, the estimated man-rem exposures from operation of the Brunswick plant will be about 250 man-rem per year. This includes 145 man-rem from one major maintenance event per year.

We conclude that adequate consideration has been given to shielding design to keep exposures within applicable limits and to reduce unnecessary exposures during normal operation of the plant. During startup of the plant, and when full power operation is attained, the plant will be mapped for dose levels and these will be compared with anticipated levels.

12.1.1 Area Monitoring

Area radiation monitors will be provided at thirty locations in each Unit. The locations have been selected so as to monitor both high radiation areas and areas where levels are expected to be low such as locker rooms, shops, and control rooms. The system is designed to detect abnormal conditions and to serve as back-up for process radiation monitors.

The coverage provided by this monitoring system as outlined in the applicant's responses to staff questions (p. M12.13-1) is considered adequate.

12.2 Ventilation

The ventilation system has been designed to move air from clean areas into areas with progressively greater contamination potential. Figure 10.10-5 of the FSAR shows that in the radwaste building, clean air is supplied through the corridors to the individual cubicles and from these through filters to the plant vent. Air flow patterns are designed to remove airborne contamination from its source area through the exhaust duct work. The applicant has conservatively estimated that 80 man-rem per year from inhalation might be accrued from operation of the Brunswick plant but it expects the actual number to be considerably less.

Based on the design description of the ventilation system in the FSAR and the planned monitoring of airborne sources and planned procedures for inhalation exposure control in the applicant's responses to Staff questions, we conclude that the Brunswick plant will be able to operate with inhalation exposures below the applicable limits.

12.2.1 Airborne Radioactivity Monitoring

Four fixed systems monitor airborne activity in each unit, one in the Reactor building vents and three in the containment. Also provided will be 6 movable continuous air monitors which can be located in critical areas to provide alarm functions and a record of activity levels. A total of 8 portable air samplers will be provided and equipment is available to analyze the samples collected.

We conclude that sufficient equipment will be provided to adequately monitor potential inhalation exposures.

12.3 Health Physics Program

The stated objective of the health physics program is to limit human radiation exposures to as low a level as practical and to ensure that pertinent regulations are adhered to. The program will operate on the principle that radiation dose is undesirable at any level and should be avoided or minimized within the limits of practicability.

Personnel protection will be accomplished through administrative controls and procedures, through the use of protective equipment and verified through an extensive personnel monitoring program. Administrative exposure limits and the use of Radiation Work Permits (RWP) enable the Radiation Control and Test Foreman to ensure compliance with 10 CFR 20. The issuance of a RWP allows for pre-job surveillance and specification of protective measures such as protective equipment and radiation monitoring. Personnel training, mockup facilities, video taping and review of personnel exposures following major maintenance events will be utilized in an effort to improve procedures and methods and to assure that the "as low as practicable" objectives will be met.

Special protective equipment includes a full array of protective clothing, temporary shielding, respirators and self-contained breathing apparatus. Personnel decontamination facilities are also provided.

All plant employees will wear TLD dosimeters and neutron sensitive film badges. Pocket chambers or special TLD badges will be issued to personnel working in relatively high radiation areas. Whole body counts will be made on all plant employees annually and in special cases as needed. Bioassays for tritium will be performed on an as needed basis.

Based on the above information obtained from the applicant in response to questions by the Regulatory staff, we conclude that the applicant's plans to implement a health physics program of sufficient scope to maintain in-plant exposures of personnel within applicable limits are acceptable.

13.0 CONDUCT OF OPERATIONS

13.1 Plant Organization and Staff Qualifications

The Brunswick Steam Electric Plant staff will consist of approximately 140 full-time employees. The plant activities are conducted under the onsite supervision of the Plant Manager, who reports to the Manager, Nuclear Generation Section, who in turn reports to the Vice President, Bulk Power Supply Department. The Plant Manager is responsible for the safe and reliable operation of the plant. The plant staff consists of operations (approximately 48 people), engineering (approximatel 20 people), maintenance (approximately 43 people), and administrative support groups (approximately 28 people).

The Operating Supervisor, in charge of operations, directs the day-to-day operation of the facility and is responsible to the Plant Manager for all operating activities at the plant. Reporting to him are the plant operating shifts. The normal shift complement for single unit operation is one Shift Foreman, licensed as a Senior Reactor Operator, two Control Operators, licensed as Reactor Operators and two Auxiliary Operators. The normal shift composition for two-unit operation is one Shift Foreman, licensed as a Senior Reactor Operator, four Control Operators, one of whom will be licensed as a Senior Reactor Operator and three who will be licensed as Reactor Operators, and three Auxiliary Operators. The Maintenance Supervisor is responsible to the Plant Manager for all maintenance activities at the plant and the Engineering Supervisor is responsible to the Plant Manager for nuclear engineering, quality assurance, radiation control, health physics, chemistry and environmental activities at the plant.

The applicant has conducted a training program for operating personnel which consisted of six phases; a basic nuclear course, research reactor experiments, BWR technology, BWR simulator training, operating BWR observation and onsite training. The extent to which each individual participated in this program was based on his job responsibilities and previous experience. Selected members of the plant staff technical support groups completed formal training specifically oriented to their assigned responsibility.

The qualifications of key supervisory personnel with regard to educational background, experience, training and technical specialties have been reviewed, and, except as noted below, are in accord with those defined in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel".

The person assigned to the position of Operating Supervisor does not appear to meet the experience provisions of Section 4.2.2 of ANSI N18.1. However, the applicant has made provisions for providing additional experienced technical support for this position through the completion of the startup program for the second unit, and we find this to be an acceptable alternative.

Technical support for the plant staff is available from the Special Services Department, the Fuel Section of the Bulk Power Supply

Department, and the Principal Engineers assigned to the Manager, Nuclear Generation Section.

We have concluded that the organizational structure, the training and qualifications of the staff for the Brunswick Steam Electric Plant, Units 1 & 2, are adequate to provide an acceptable operating staff and technical support for the safe operation of the facility. Additional technical support during the startup test program will be provided by General Electric.

13.2 Safety Review and Audit

The safety review and audit function for the Brunswick Steam Electric Station will be conducted by the Plant Nuclear Safety Committee and the Company Nuclear Safety Committee. The Plant Nuclear Safety Committee is advisory to the Plant Manager and will review all proposed tests, changes in plant operating procedures and design modifications. The Company Nuclear Safety Committee provides corporate management with a review and audit capability to verify that organizational checks and balances are functioning to assure continued safe operation and design adequacy of the plant. The Company Nuclear Safety Committee will function in accord with ANSI N18.7, "Standard for Administrative Control for Nuclear Power Plants", Sections 4.1 through 4.4.

We conclude that the provisions for the review and audit of plant operations are acceptable.

13.3 Plant Procedures

Plant operations are to be performed in accordance with written and approved operating and emergency procedures. Areas covered include normal startup, operation and shutdown, abnormal conditions and emergencies, refueling, maintenance, surveillance and testing, and radiation control. All procedures, and changes thereto will be reviewed by the Plant Nuclear Safety Committee and approved by the Plant Manager prior to implementation.

We conclude that the provisions for preparation, review, approval, and use of written procedures are satisfactory.

13.4 Emergency Plan

The applicant has established an organization for coping with emergencies. The plan includes written agreements, liaison and communications with appropriate local, State and Federal agencies that have responsibilities for coping with emergencies. The State of North Carolina, Department of Human Resources, is developing a radiological emergency response plan which will provide for the direction of offsite agencies by their Radiological Health Section. Until such planning is completed, additional responsibilities in this regard have been undertaken by CP&L with the concurrence of the State of North Carolina, Department of Human Resources.

The applicant has defined categories of incidents, including criteria for determining when protective measures should be considered and for the notification of offsite support groups. Arrangements have been made by the applicant to provide for medical support in the event of radiological or other emergencies. Provisions for periodic training for both plant personnel and offsite emergency organizations have been included in the Emergency Plan.

We have reviewed the Emergency Plan and conclude that it meets the requirements of Appendix E of 10 CFR 50, and that adequate arrangements have been made to cope with the possible consequences of accidents at the site, and that there is reasonable assurance that such arrangements will be satisfactorily implemented in the unlikely event that they are needed.

13.5 Industrial Security

The applicant has submitted a description of his industrial security plans for the protection of Brunswick Units 1 & 2 from industrial sabotage. The information was submitted as proprietary information and is withheld from public disclosure pursuant to Section 2.790 of the Commission's regulations.
We have reviewed the industrial security program and conclude that adequate security provisions have been made and that the program meets the objectives of Regulatory Guide 1.17.

14.0 TEST AND STARTUP PROGRAM

The Brunswick Plant Manager is responsible for all startup activities at the plant site. The initial startup, including check-out of equipment, functional and system tests will be performed by the regular plant staff. This staff will be augmented during the startup of both units by a Plant Startup Supervisor and Startup and Test Engineers. Technical assistance will be provided by General Electric through a General Electric Startup Group under the direction of the General Electric Operations Manager.

Preoperational test procedures are prepared initially by United Engineers and Constructors. Startup Procedures are prepared initially by General Electric. These procedures are reviewed by CP&L operating and startup personnel and approved in writing by the Plant Manager. Startup test procedures and test results are also reviewed by the Plant Nuclear Safety Committee.

We have reviewed the preoperational startup and test program described by the applicant and conclude that it is in accord with the AEC publications, "Guide for the Planning of Preoperational Testing Programs", and "Guide for the Planning of Initial Startup Programs", and is acceptable.

15.0 ACCIDENT ANALYSIS

15.1 General

The applicant described and analyzed the consequences of various abnormal transients and postulated design basis accidents in Chapter 14 of the FSAR. Such safety analyses are provided to evaluate the capability engineered into the plant to control or accommodate such occurrences. The type of situations analyzed ranged from anticipated operational occurrences (e.g., generator trip due to a line fault) to postulated accidents of very low probability (e.g., sudden failure of a major component built to nuclear code requirements). Our evaluation of abnormal operational transients and of accidents are considered in separate sections below.

15.2 Abnormal Operational Transients

We evaluated the applicant's analysis of the response of the reactor to the possible occurrences of various abnormal operational transients. The applicant has provided a description of the transient responses based on revisions in the reactivity coefficients and in the details of the analytical methods. The events that characterize abnormal operating transients included inadvertent control rod withdrawal, turbine trip, loss of electrical load, loss of condenser vacuum, and equipment malfunctions that could perturb normal plant operation, e.g., the seizure of a recirculation pump. The applicant presented the transient study graphically and summarized the results in terms of the values of the MCHFR and the peak system pressures experienced during the transient. Table 15-1 of this report presents material selected from this study by the applicant that represents those cases that most closely approach the design limitations of the reactor. Pressures are computed on the basis of relief valves functioning as designed and are seen to be below both the design pressure of 1250 psig and the code allowable overpressure of 1375 psig. Values for the MCHFR have a satisfactory margin above 1.0 all cases, except that for recirculation pump seizure. For the pump seizure case, we find the value of 1.02 acceptable on the basis that this transient is self-mitigating and does not require reactor scram for its termination, and that the significant variables were computed to stabilize safely in about 10 seconds. The duration of the 1.02 value for MCHFR was less than 2 seconds. The startup test program as discussed in Chapter 13 of the FSAR will provide an opportunity to verify the study's computational methods and assumptions and in many cases should provide a numerical verification of the peak conditions. Subject to the results of the startup test program, we find the applicant's analysis of the abnormal operational transients acceptable.

15.3 Design Basis Accidents

The primary means for preventing accidental releases of radioactive material to the environs is through correct design, manufacture, and operation of the facility. A detailed and rigorous quality assurance program was required for the facility's construction and a similar program was established to maintain the necessary high integrity of the reactor and associated safety features during operation. We consider, therefore, the actual occurrence of a major accident extremely remote. Nevertheless, the facility was designed to include engineered safeguards to reduce the consequences of radiation exposure to the public in the event a major accident should occur. In order to demonstrate the effectiveness of the design of the safeguards we required the applicant to provide an estimate of radiation doses using assumptions acceptable to the staff for the various design basis accidents. These accidents are representative of the upper limits of a wide spectrum of accidents that are considered credible. We also performed similar computations independently and compared the results with those of the applicant. Our criteria for accepting the applicant's design are that the doses from these postulated accidents (as evaluated by the regulatory staff) will be substantially less than guideline values given in 10 CFR Part 100, "Reactor Site Criteria." The Part 100 guideline values are 25 rem of whole body radiation or a total radiation dose to the thyroid gland of 300 rem.

The three postulated design basis accidents that were evaluated in the course of our review are the: (1) loss of coolant (LOCA), (2) fuel handling, and (3) control rod drop. Table 15-2 presents the staff's estimates of the potential offsite doses due to these three design basis accidents; none of the accidents analyzed resulted in calculated doses approaching the 10 CFR Part 100 guidelines. The assumptions used in our analyses are presented in Table 15-3. Additional discussion of the control rod drop accident and the containment purge dose is provided in Sections 15.3.1 and 15.3.2, respectively.

15.3.1 Control Rod Drop Accident

For the postulated control rod drop accident, it is assumed that a bottom-entry control rod has been fully inserted and has stuck in this position unknown to the reactor operator. It is then assumed that the drive becomes uncoupled and the driver-rod withdrawn. Subsequently the rod falls from the core. A specific amount of reactivity is inserted corresponding to the worth of the rod and the reactor is scrammed on high neutron flux. The worst condition for this accident to occur is during reactor startup when individual rods have their greatest worths for the reasons identified in Section 4.2.3. The probability of such an event actually occurring is extremely low. The accident analysis, however, provides a rigorous means for demonstrating the functioning of and the safeguards associated with the reactivity control system.

Two principal factors determine the consequences of this accident. The worth of the control rod that falls from the core and the rate at which negative reactivity can be reinserted into the core. Independent analysis of both factors determined that General Electric had underestimated the worth of the dropped rod and over estimated the rate of negative reactivity reinsertion. GE revised its analysis of the accident in a topical report, NEDO-10527 entitled, "Rod Drop

Accident Analysis for Large Boiling Water Reactors," and in Supplements 1 and 2 thereto. Supplements 1 and 2 were prepared for the Browns Ferry class of reactors which employ axially distributed gadolinia for power shaping similar to BSEP Units 1 and 2 with the analysis for a fresh core given in Supplement 1 and an exposed core in Supplement 2.

Although these reports continue under review and the applicant has not yet provided an estimate of the consequences of the rod drop accident, interim design and operating modifications were adopted or committed to by the applicant. The rod worth minimizer (RWM) and the rod sequence control system (RSCS) are described in Section 4.2.3. If the RWM becomes inoperable during a startup, administrative control by means of a second licensed operator checking withdrawal sequences is permitted. The RSCS must be operable during any startup. The applicant is also committed to a fuel design that will limit the peak (single pellet) fuel enthalpy in the event of a rod drop accident to 280 cal/gm.

We find these commitments acceptable on the basis of additional studies to be performed prior to licensing the reactor. These studies, which must be presented in a form that is capable of independent verification, must show that the doses from the accident fall significantly below the limits given in 10 CFR Part 100.

15.3.2 Containment Purge Dose, Post-LOCA

We calculated that purging of the containment atmosphere after

22 days of elapsed time in the post-LOCA period will be needed in order to prevent a containment pressure buildup of greater than 31 psig, a value that we consider as the maximum acceptable fraction of the 62 psig design pressure of the primary containment system. We calculated an exclusion area boundary dose of 7 Rem to the thyroid and a negligible amount to the whole body resulting from a continuous purge of containment atmosphere beginning after 22 days, post-LOCA, with purge rate of 7 cfm. We assumed a χ/Q of 2.1×10^{-7} sec/m^3, a thermal power level of 2550 MW, and a filter efficiency for radioiodines of 90% elemental iodine and 70% organic iodines. We, therefore, find the applicant's proposed system for controlling the post-LOCA hydrogen concentration in the containment and the resulting offsite doses to be acceptable.

TABLE 15-1

REACTOR RESPONSE TO ABNORMAL OPERATIONAL TRANSIENTS

Event	Parameter Variations	Scram Signal	MCHFR	Peak Pressure, psig
Turbine trip without bypass (high power)	Pressure increase	MSLIV position switches	>1.60	1225
Main steam line isolation valve (MSLIV) closure	Pressure increase	MSLIV position switches	>1.9	1178
Recirculation pump seizure	Core flow decrease	No scram	>1.02	depressurization
Continuous rod withdrawal at power	Reactivity insertion	High neutron flux	>1.0	no significant increase

TABLE 15-2

POTENTIAL OFFSITE DOSES DUE TO DESIGN BASIS ACCIDENTS

Postulated Accident	Two Hour Exclusion Boundary (915 meters)		Course of Accident Low Population Zone (3220 meters)	
	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)
Loss of Coolant	40	3	39	2
Fuel Handling	2	<1	1	<1
Control Rod Drop(1)	20	1	22	1

(1)Based on interim assumptions. See Table 15-2, D

TABLE 15-3

ASSUMPTIONS USED FOR STAFF ANALYSIS OF DESIGN BASIS ACCIDENTS

A. Loss-of-Coolant Accident (LOCA)

1. Power level of 2550 MWt.

2. Regulatory Guide No. 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors."

3. Primary containment leak rate of 0.5% per day.

4. Standby gas treatment system (SGTS) charcoal filter efficiencies of 70% for organic iodines and 90% for elemental and particulate forms.

5. Meteorological diffusion parameters (χ/Q) based on onsite data as described in Section 2.3.4.

6. The doses if the CAD system were used are lower than those given in Table 15.2.

B. Fuel Handling

1. Accident occurs 24 hours after shutdown.

2. Rupture of 111 fuel rods.

3. SGTS functions with the same efficiency as following the LOCA.

4. Meteorological diffusion parameters based on 0-2 hour period. Elevated (χ/Q) as presented in Section 2.3.4.

5. The Regulatory Guide No. 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" which include:

   a. A 1.5 peaking factor.

   b. All gap activity in the damaged rods assumed to be 10% of the noble gases and 10% of the iodines.

c. 99% of the iodines and none of the noble gases are retained in the pool water.

C. The assumptions for the control rod drop accident are being revised based on a study now in progress. The interim assumptions[(1)] are provided below.

1. The accident occurs due to a 1.4% ΔK control rod drop as limited by the Rod Sequence Control System.

2. 600 fuel rods are damaged.

3. Peaking factor = 1.50.

4. 100% of the noble gases and 50% of the iodines are released from the fuel.

5. A reduction factor of 10 is allowed for iodine passing through the primary system water.

6. A plate-out factor of 2 is allowed for iodine in the turbine and condenser.

7. High radiation is detected in the steamline signaling the vacuum pump to stop and the isolation valves to close. (5 second valve closure time.)

8. Essentially all of the activity is contained by the turbine and condenser.

9. A constant leak rate of 0.5% per day from the turbine and condenser is assumed.

10. The total accident duration is 24 hours.

11. Regulatory Guide No. 1.3 ground level release with credit for a wake factor.

[(1)] These assumptions will be modified in the near future to conform with the results of our study and analysis of the control rod drop accident as described in Sections 4.2.3 and 15.3.2 of this safety evaluation report. Preliminary analysis of the rod drop accident that assumes the rod sequence control system to be operable results in an estimated 600 fuel rods perforation and increases the calculated doses from this accident by a factor of two, which is still well below the 10 CFR Part 100 guideline values.

16.0 TECHNICAL SPECIFICATIONS

The Technical Specifications portion of a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the Regulatory staff. The proposed Technical Specifications for the BSEP Units 1 and 2 contained in Appendix K of the FSAR are similar in scope and content of recently licensed BWR's and are essentially complete. We have held meetings with the applicant to discuss their contents and some modifications to the proposed Technical Specifications have been suggested both by the staff and the applicant to more clearly describe the allowed conditions for plant operation. The finally approved Technical Specifications will be included as part of the operating license. Included are sections covering safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls. On the basis of our review, we will assure that normal plant operation within the limits of Technical Specifications will not result in potential offsite exposures in excess of 10 CFR Part 20 limits and/or our guidance on meeting the "as low as practicable" releases of radioactivity. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features for continued plant operation will be available in the event of malfunctions within the plant.

17.0 QUALITY ASSURANCE

The Quality Assurance Program (QAP) for Operations should include all those activities required to assure that:

(1) the facility will be operated in accordance with design and license requirements; and

(2) maintenance, modification and repair activities are conducted such that the quality of safety-related systems, structures and components is maintained.

To provide this assurance, the QAP must contain the elements of Appendix B to 10 CFR Part 50 for each of the operational phase activities to which they apply. Well defined organizational responsibility and authority for quality matters, to implement the QAP, must be established and audit of procedural and program implementation must be effective and timely.

17.1 General

The description of the Quality Assurance (QA) Program for Brunswick Steam Electric Plant (BSEP), Units 1 and 2, is provided in Section 17 of the FSAR, and its Amendments. We have reviewed and evaluated the organizational and programmatic aspects of the QA Program for Operation including those quality related activities associated with the maintenance, modification, repair, and operating phases of the BSEP to determine compliance with the requirements of Appendix B to 10 CFR Part 50 and AEC Regulatory Guide 1.33.

17.2 Organization

Under the Executive Vice President of the Engineering and Operating Group, there are three organizational departments (Special Services Department, Power Plant Engineering and Construction Department, and Bulk Power Supply Department) who have responsibility for performing QA/QC functions and for assuring that the quality related activities associated with the operation of the BSEP are accomplished in accordance with Appendix B to 10 CFR Part 50. The Manager of the QA Audit Section, reporting to the Vice President of Special Services Department is responsible for conducting independent comprehensive audits of the BSEP to verify that quality related activities are in compliance with the QA Program and to evaluate the effectiveness of the Program. The QA Section Manager for the Power Plant Engineering and Construction Department, reporting to its Vice President, is responsible for providing QA/QC functions for the Brunswick Plant associated with the procurement of safety related structures, systems, and components. These include the proper selection and qualifications of suppliers, the review of procurement documents to assure that the quality requirements are adequately stated and the performance of source inspection and surveillance of suppliers. Within the Bulk Power Supply Department the implementation of the QA Program is the responsibility of the Principal QA Surveillance Engineer who is independent of the Brunswick Plant Manager and the QA/QC Engineers reporting to the

Brunswick Plant Manager. The QA/QC Engineers responsible for inspecting and verifying quality related activities, are independent of the personnel performing the work.

We conclude that the QA organization described in the FSAR provides sufficient independence from the personnel responsible for cost and schedules to properly carry out the QA/QC functions in accordance with the requirements of Appendix B to 10 CFR Part 50.

17.3 QA Program

Based on our review, we have determined that the QA Program provides for controlled written policies, procedures and instructions governing the implementation and control of quality related activities associated with the operation of BSEP which includes maintenance, modification, and repairs. The QA Program requires that the quality inspection and verification of plant maintenance, modification, and repairs be performed by individuals administratively independent of those directly responsible for the work. The QA Program has provisions which assure that inspection and verification holdpoints, are inserted in the control documents for maintenance, modification, and repair of safety related structures, systems, components. The QA Program procedure for record control requires the collection and retention of records that define and attest to the quality of the safety related structures, systems, components during operations, maintenance, modification, and repair. The QA Program also requires indoctrination and training programs to be established and conducted for those personnel performing quality related activities to assure they are knowledgeable in regard to the QA Program, procedures, and requirements and become proficient in implementing these procedures.

We conclude that the QA Program described adequately complies with each of the eighteen criteria of Appendix B to 10 CFR Part 50.

17.4 Fuel Quality Assurance

As part of our review, we have evaluated the quality assurance measures provided to assure the long term integrity of fuel for BSEP. The applicant has described the design and manufacturing measures which are intended to minimize possible fuel failures; these include restrictions of possible moisture and hydrocarbon contaminants in the UO_2 pellets, dished fuel pellets, prepressurized fuel rods with a top void region to accommodate fission gas release, and appropriate QC measures during manufacture to assure product conformance.

These actions represent current state of the art actions that should minimize fuel failures during plant operation. We have concluded the fuel QA Program is acceptable.

17.5 Audit

Provisions have been established in the QA Program requiring comprehensive, scheduled audits to be performed by qualified QA personnel independent of those individuals or groups in the area being audited. The audits are required by the QA Program to be in accordance with pre-established written procedures and shall include the verification and evaluation of procedures and quality related activities to assure they are meaningful and effective. The QA Program requires audit results and corrective actions to be documented and reported to responsible management, including President and the Executive Vice President of the Carolina Power and Light Company (CP&L).

We conclude that the audit system of CP&L as described in the FSAR is in conformance with the requirements of Appendix B to 10 CFR Part 50.

17.6 Conclusions

Based on our review of the description of the QA Program contained in Section 17 of the FSAR and related amendments, we conclude that the BSEP QA Program for Operations is in compliance with Appendix B to 10 CFR Part 50 and AEC Regulatory Guide 1.33 and is therefore acceptable for control of the operation, maintenance, modification, and repair activities associated with this facility.

18.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

18.1 ACRS Construction Permit Letter

In its letters dated May 15, 1969 and October 16, 1969 to the Commission, the Advisory Committee on Reactor Safeguards indicated certain matters would require resolution between applicant and the Regulatory staff prior to and during construction of the Brunswick Steam Electric Plant Units 1 and 2. Each of these matters is discussed in this Safety Evaluation Report. The matters indicated in the ACRS letters and referenced in this report are: (1) Flood protection (see Section 2.4.2), (2) Diagonal steel reinforcing bars for resisting inplane shear loads (see Section 3.8.1), (3) Design of equipment hatches and large penetrations of primary containment (see Sections 3.8.1 and 3.8.4), (4) Inservice inspection of reactor primary system (see Section 5.2.7), (5) Reactor instrumentation (see Section in 7.0), (6) ATWS (see Section 7.2.5), (7) TID-14844 fission product releases (Regulatory Guide 1.3 was used), (8) Post-accident cooling system's corrosion protection (see Section 4.2), (9) ECCS suction lines to torus (see ACRS letter dated October 16, 1969, (10) Hydrogen generation during LOCA (see Section 6.2.5), and (11) Steam lines outside containment design, fabrication and inservice inspection (see Section 10.3).

The above matters were specifically identified in the ACRS construction permit letters. Other problems related to boiling water reactors which have been identified by the regulatory staff and ACRS are covered in the organization of this Safety Evaluation report. The applicant has identified and discussed those matters identified by the ACRS letters in Appendix H of the FSAR.

18.2 ACRS Operating License Letter

The report of the ACRS on this application for the operating license review will be placed in the Commission's Public Document Room and will be published in a supplement to this Safety Evaluation Report.

19.0 COMMON DEFENSE AND SECURITY

The applicant states that the activities to be conducted would be within the jurisdiction of the United States and that all of its directors and principal officers are United States citizens. We find nothing in the application to suggest that the applicant is owned, controlled, or dominated 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 which might become involved in accordance with the regulations. The applicant will obtain fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons, and in the absence of any information to the contrary, we have found 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 financial qualifications for applicants for operating licenses are Paragraph 50.33 (f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50. We have reviewed the financial information presented in the application and Amendment Nos. 12 and 23 thereto. Based on this review, we have concluded that Carolina Power and Light Company possesses or can obtain the necessary funds to meet the requirements of 10 CFR 50.33 (f) to operate the Brunswick Steam Electric Station, Units 1 and 2, and if necessary permanently shut down the facility and maintain it in a safe shutdown condition.

Brunswick Steam Electric Station, Units 1 and 2, will be used to augment the applicant's present electrical generating capacity. Revenues from system-wide sales of electric energy will provide the funds to cover cost of operations. Operation and maintenance expenses (nuclear power generation expenses: nuclear fuel expense, other operating expenses, and maintenance expenses; transmission expenses; and administrative and general expenses: property and liability insurance and other administrative and general expenses) during the first five years of commercial operation of both units (1976 - 1980) are presently estimated by the applicants to be (in millions of dollars) $44.8; $43.0; $39.8; $39.2; and $39.6 in that order. The applicant states that during 1976, the first year both units will be in commercial operation, total costs (or "revenue requirements") applicable to both units are estimated to amount to $154,617,000 or 1.3¢ per kwh assuming a plant factor of 80%. Total costs (or "revenue requirements") include operation and maintenance expenses as described above and the following cost elements: depreciation expense; taxes other than income taxes (property and other taxes); income taxes (Federal and other income taxes); deferred income taxes-net; investment tax credit adjustments-net; and return on invested capital. Revenues from system-wide sales of electric energy are expected to provide funds necessary to cover the total costs estimated to be applicable to both units. The applicant indicated in its Amendment No. 23 to the basic application that during 1972 sales of electric energy amounted to 1.4¢ per kwh compared with the estimated cost of 1.3¢ per kwh for both units in 1976.

The cost of permanently shutting down the facility, using a combination of mothballing and dismantling, is estimated by the applicant at $3.7 million for each unit, a total of $7.4 million based on 1973 dollars and technology. The annual cost of maintaining both units in a safe shutdown condition is estimated by the applicant at $50 thousand, also based on 1973 dollars and technology. The source of funds to cover these costs is expected to be system-wide sales of electric energy.

The applicant states that uranium for the initial cores for both units has been purchased from private sources and will be enriched under toll enrichment agreements with the Commission.

We have examined the financial information submitted by Carolina Power & Light Company to determine whether it is financially qualified to meet the above estimated costs. The information presented in Carolina Power & Light Company's annual report for 1972 indicates that operating revenues totaled $307.1 million. Operating expenses were stated at $236.3 million, of which $27.3 million represented depreciation. Interest on long-term debt was earned 2.6 times. Net income totaled $60.5 million, of which $36.8 million was distributed as dividends to stockholders with the remaining $23.7 million retained for use in the business. As of December 31, 1972, the Company's assets totaled $1,418.8 million, most of which was invested in utility plant ($1,357.1 million). Retained earnings amounted to $90.7 million. Financial ratios computed from the 1972 statements indicate an adequate financial condition, e.g., long-term debt to total capitalization - 53%, and to net utility plant - 50%; net plant to capitalization - 1.04; the operating ratio - 77%; and the rates of return on common equity - 11.5%, on stockholders' investment - 9.8%, and on total investment - 7.2%. The record of the Company's operations during 1970-72 shows that operating revenues increased from $204.8 million in 1970 to $307.1 million in 1972; net income increased from $24.8 million to $60.5 million; and net investment in utility plant from $829.6 million to $1,357.1 million. The number of times interest earned increased from 2.5 to 2.6. Moody's Investors Service rates the Company's first mortgage bonds as A (upper medium grade bonds). The Company's current Dun and Bradstreet rating is 5A1, the highest rating.

A summary analysis reflecting these ratios and other pertinent data is attached as an Appendix B to this report.

21.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

21.1 Financial Protection and Indemnity Requirements

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, licensees for facilities such as power reactors licensed under 10 CFR Part 50.

21.2 Preoperational Storage of Nuclear Fuel

The Commission's regulations in Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also to be the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear materials 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. No license authorizing the ownership and possession, for storage only, of special material at the reactor construction site for future use as fuel in the reactor will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

21.3 Operating License

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from, for example, preoperational fuel storage only) has been executed.

Accordingly, no license authorizing operation of BSEP Unit 2 will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

22.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, and assuming favorable resolution of the outstanding matters described herein, we conclude that:

1. The application for facility licenses filed by the applicant, dated July 31, 1968, as amended (Amendments 1 through 23) 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 1; and
2. The construction of the Brunswick Steam Electric Plant Units 1 and 2 (the facilities) has proceeded and there is reasonable assurance that it will be complete, in conformity with Provisional Construction Permit Nos. CPPR-67 and CPPR-68, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and
3. The facilities 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, assuming satisfactory completion of our review of those items which we have elected to defer, that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Part 1; and

5. The applicant is technically and financially qualified to engage in the activities authorized by the operating license in accordance with the regulations of the Commission set forth in 10 CFR Part 1; and

6. The issuance of operating licenses for these facilities will not be inimical to the common defense and security or to the health and safety of the public.

Prior to final consideration of the matter of the issuance of an operating license to the applicant for the Brunswick Steam Electric Plant Unit 2, a supplement to this Safety Evaluation will be prepared that will deal with those matters relating to the status of 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 level must be verified by the Commission's Directorate of Regulatory Operations prior to issuance of a 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

REGULATORY REVIEW OF CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 & 2

July 26, 1968	Carolina Power & Light Company submits an application to construct the Brunswick Steam Electric Plant (License Application and Vols. I, II, and III of PSAR).
August 12, 1968	AEC-DRL letter transmits application to U.S. Department of the Interior, Fish and Wildlife Service and requests their review. (Amendments No. 3, 4, 5, 7, 8, 9, and 10).
September 3, 1968	Carolina Power & Light Company submits Amendment No. 1 (designated as the first supplement) consisting of supplementary data and analyses of activity associated with Sunny Point Army Terminal.
September 11, 1968	AEC-DRL staff met to discuss preliminary regulatory review on quality assurance, staffing and training, pre-operational and initial startup testing for safety systems, emergency plans, outline of technical specifications, and site meteorology and geology.
October 7, 1968	AEC-DRL staff met to discuss site geology.
October 17, 1968	AEC-DRL staff met to discuss organization, staffing and quality assurance.
November 7 & 8, 1968	AEC-DRL staff met to discuss site geology and seismology, structures and containment, ASME design criteria, potential for munitions explosions, seismic design, instrumentation, control and power, personnel staffing and training, and quality assurance.

December 6, 1968	Carolina Power & Light Company submits Amendment No. 2 consisting of revised financial data.
December 12, 1968	AEC-DRL letter requests additional information on radioactive effluents, personnel staffing and training, Class A pressure vessel design, quality control, containment design capability, instrumentation and control, and seismic design criteria.
December 20, 1968	AEC-DRL staff met to discuss seismic design criteria for main steamline piping and anchor supports.
January 7, 1969	AEC-DRL letter requests additional information on design criteria for certain Class I equipment.
January 17, 1969	Carolina Power & Light Company submits Amendment No. 3 (designated as Supplement No. 2) consisting of additional information in response to DRL's December 12, 1968 letter.
January 27, 1969	Carolina Power & Light Company submits Amendment No. 4 (designated as Supplement No. 3) consisting of additional information in response to DRL's December 12, 1968 letter.
January 28, 1969	Carolina Power & Light Company submits Amendment No. 5 consisting of revised information on quality assurance, plant transmission system, personnel staffing and training, and site seismology.
February 11, 1969	Fish and Wildlife Service submits comments on radiological and nonradiological aspects of proposed Brunswick plant construction and operation.
February 12 & 13, 1969	DRL staff met to discuss seismic design, alternate source of cooling water, spent fuel, design criteria for Class II structures, fire protection systems and A.C. interlock system.

February 26, 1969	AEC-DRL letter transmits comments of U. S. Department of the Interior, Fish and Wildlife Service on radiological and non-radiological aspects of proposed plant.
March 3-5, 1969	AEC-DRL staff visits Brunswick site to review site geology.
March 5, 1969	AEC-DRL letter requests further information on reactor design and related systems as requested in DRL's December 12, 1968 letter and additional information on reactor protection and engineered safety features.
March 12, 1969	Carolina Power & Light Cmpany submits Amendment No. 6 (designated as Supplement No. 4) consisting of additional information in response to DRL's letters of January 7, 1969 and March 5, 1969.
April 9, 1969	AEC-DRL staff met to discuss areas requiring further review prior to ACRS meeting.
April 15, 1969	AEC-DRL letter requests additional information on reactor design, flood protection, fuel failure, cadweld testing program and quality assurance.
April 30, 1969	ACRS Subcommittee visits proposed site.
May 2, 1969	Carolina Power & Light Company submits Amendment No. 7 (designated as Supplement No. 5) consisting of additional information in response to DRL's April 15, 1969 letter and revising routing for discharge canal.
May 6, 1969	Carolina Power & Light Company responds to DRL's February 26, 1969 letter outlining plans for environmental monitoring.
May 9, 1969	ACRS met to discuss technical aspects of proposed site and reactor design.
June 3, 1969	AEC-DRL staff met to discuss containment design.

June 30, 1969	Carolina Power & Light Company submits Amendment No. 8 (designated as Supplement No. 6) consisting of additional information relating to comments by the ACRS in its May 15, 1969 letter.
July 10, 1969	Fish and Wildlife Service submits comments on proposed re-routing of discharge canal.
July 14, 1969	Carolina Power & Light Company submits Amendment No. 9 transmitting revised information on completion dates and estimated construction costs.
July 17, 1969	AEC-DRL letter transmits revised report from Fish and Wildlife Service (dated July 10, 1969).
August 13, 1969	Carolina Power & Light Company submits Amendment No. 10 transmitting revised information on technical qualifications, quality assurance and construction responsibilities of new contractor (Brown & Root, Inc.).
September 4, 1969	ACRS met to discuss change in construction contractor and changes in design criteria in response to ACRS comments of May 15, 1969.
September 25, 1969	Carolina Power & Light Company submits a request for exemption to begin certain construction work.
October 8, 1969	ACRS Subcommittee met to discuss design changes.
October 9, 1969	ACRS met to continue discussion of items previously discussed on September 4, 1969 (ACRS report to Chairman Seaborg issued on October 16, 1969).
October 31, 1969	AEC-DRL staff published Safety Evaluation of proposed Brunswick plant.

November 7, 1969	AEC-DRL staff letter grants exemption request.
November 17-21, 1969	AEC-DRL and Compliance staff conduct quality assurance inspection.
December 2-3, 1969	An Atomic Safety and Licensing Board conducts a public hearing (Initial Decision, dated February 4, 1970 orders issuance of construction permit).
February 7, 1970	DRL issues Provisional Construction Permits No. CPPR-67 and CPPR-68.
February 27, 1970	AEC-DRL staff met to discuss (post-CP items) proposed reduction in containment design pressure, inservice inspection program and design criteria for pipe whip.
March 9, 1970	CP&L letter transmits reports on first two items above (March 25, 1970 letter submits table for containment design report).
April 17, 1970	CP&L letter informs DRL of certain management changes.
April 27, 1970	AEC-DRL letter commenting on proposed inservice inspection program.
May 19, 1970	CP&L letter transmits report on pressure suppression concept.
May 22, 1970	AEC-DRL staff met with CP&L representatives to discuss proposed reduction in containment design pressure.
May 28, 1970	AEC-DRL letter transmits request for additional information regarding request for reduction in design pressure for containment.
July 30, 1970	CP&L letter transmits a report containing additional information on pressure suppression concept.

August 19, 1970	AEC-DRL letter approves withholding from public disclosure report transmitted by CP&L on May 19.
August 25, 1970	AEC-DRL letter in response to CP&L request for reduction in containment design pressure.
October 2, 1970	AEC-DRL staff met with CP&L representatives to discuss post-construction permit matters related to the Brunswick plant.
November 9, 1970	CP&L submits Design Reports No. 4, 5, & 6 on Containment Analysis and seismic and structural design.
December 22, 1970	CP&L letter submits Design Report No. 7 - "Containment Design".
December 28, 1970	AEC-DRL letter requests additional information on reactor design.
February 8, 1971	AEC-DRL letter requests additional information on reactor design.
February 26, 1971	CP&L letter submits Design Report No. 8 "Small Steam Line Break".
April 6, 1971	CP&L submits Amendment 11 which provides revised information on completion dates.
April 7, 1971	AEC-DRL letter requesting that CP&L submit additional information for the matters discussed in Design Report Nos. 7 and 8.
October 18, 1971	CP&L submits statements required by Section E of the Revised Appendix D. These statements support the continued construction of Brunswick Units 1 and 2.
October 28, 1971	Meeting with CP&L staff on review required by Section E of Appendix D.
November 1, 1971	AEC-DRL letter requesting additional information required for review in accordance with Par. 2, section E of Revised Appendix D.

November 4, 1971	CP&L submits Environmental Report required by Section B of Revised Appendix D.
November 9, 1971	AEC-DRL letter requesting additional information in connection with Design Report No. 4, Addendum A relating to the dynamics methods to be used for seismic design.
November 11, 1971	CP&L letter providing additional information requested on Nov. 1, 1971, relating to review in accordance with section E of the revised Appendix D.
November 18, 1971	Order to Show Cause issued (transmitted to CP&L by letter dated Nov. 19, 1971) determining that construction activities involving the offsite portions of the discharge canal and the offsite transmission lines at the Brunswick Station should be suspended pending completion of those portions of the NEPA environmental review. With respect to the construction of the intake canal and the outside portions of the plant, the Order concludes that these activities need not be suspended. "Determination to Suspend ....." (Federal Register Notice) and "Discussion and Findings" also included.
November 24, 1971	"Suspension of Certain Construction Activities ...." published in the Federal Register (Citation 36) Federal Register 22324; filed 11-23-71; Comments due by: 12/28/71.
December 2, 1971	AEC-DRL letter requesting additional information required to facilitate an early consideration of the environmental impact of the discharge canal and the transmission lines.
December 15, 1971	CP&L submits "Answer to Order to Show Cause," consisting of Volumes I, II, and III.

December 22, 1971	CP&L submits Design Report No. 9, "Seismic Analysis of the Control Buildings" and Design Report No. 10, "Seismic Analysis of the Radwaste Building" (letter and both reports enclosed in one binder).
December 30, 1971	Federal Power Commission's comments.
January 4, 1972	CP&L submits additional information requested in AEC letter dated 12/2/71.
January 7, 1972	Federal Register Notice re Environmental Report sent to JCAE.
January 19, 1972	Federal Register Notice re Availability of Environmental Report published (Citation 37 FR 820; filed Jan. 18, 1972)
January 31, 1972	AEC-DRL letter requesting additional information for review of Design Report No. 11.
March 13, 1972	AEC-DRL letter requesting additional information on torus baffles.
March 14, 17 & 24, 1972	ASLB hearings held in Washington, D.C. on applicant's Show Cause (Board decision, dated 4-24-72 and Appeal Board decision dated 5-26-72).
March 22, 1972	CP&L letter transmitting Addendum "A" to Design Report No. 11.
March 30, 1972	CP&L letter transmitting Design Report No. 12 & Design Report No. 13.
April 25, 1972	AEC letter requesting additional environmental information.
May 5, 1972	CP&L letter submits Basic Data for Source Term Calculations.
May 18, 1972	CP&L letter furnishing information regarding steam leakage in Turbine Building.

May 30, 1972	CP&L submits Amendment 12 to application, including FSAR. (Tendered application for "preliminary review") - Docketed October 4, 1972.
June 5, 1972	CP&L submits Amendment #1 to Environmental Report.
July 14, 1972	AEC staff met with CP&L representatives to discuss results of preliminary review.
July 31, 1972	CP&L letter in response to our 10-12-71 letter re LOCA.
August 3, 1972	AEC letter requesting information on flooding which may affect safety related systems.
September 1, 1972	CP&L letter responding to our 8-3-72 letter.
September 1, 1972	CP&L letter submits Amendment No. 2 to Environmental Report.
September 25, 1972	CP&L letter submits Amendment No. 4 to Environmental Report.
September 25, 1972	CP&L letter submits responses to Amendment No. 12 - proprietary Design Report No. 14 (BSEP DR-14) "Nuclear Fuel Design" and Design Report No. 15 (BSEP DR-15) "Electrical Design Drawings".
October 6, 1972	Meeting with CP&L in Bethesda, Md. review of fracture toughness of the Brunswick reactor pressure vessels.
October 13, 1972	AEC letter advising the docketing of CP&L's application for a license to operate the Brunswick Steam Electric Plant, Units 1 & 2.
October 18, 1972	CP&L letter submitting information requested during October 6, 1972 meeting with AEC/L staff regarding fracture toughness of Brusnwick reactor pressure vessels.
October 26, 1972	CP&L response to AEC/L letter of August 3, 1972, re failure of expansion bellows in circulating water line at Quad Cities.

October 30, 1972	AEC letter advising CP&L that we are withholding their Design Reports 14 & 15 as proprietary data per their request of September 25, 1972.
November 3, 1972	Notice of Consideration of Issuance of Facility Operating License and Opportunity for Hearing published in F.R. (signed by Bender 10/27/72).
November 8, 1972	CP&L submits Addendum B to Design Report No. 4 - "Seismic Analysis of the Primary & Secondary Containments" - and responses to DL comments as a result of their review of Design Report No. 9, "Seismic Analysis of the Control Building", and No. 10, "Seismic Analysis of the Radwaste Building".
November 13, 1972	CP&L submits Amendment No. 14 consisting of revised pages for the FSAR.
November 20, 1972	AEC/L letter requesting review of fuel densification.
November 24, 1972	Memo to files re Summary of October 6, 1972 meeting with applicant.
November 27, 1972	AEC/L letter to applicant transmitting Summary of October 6, 1972 meeting.
December 14, 1972	Meeting with applicant regarding Regulatory Staff positions on Brunswick.
December 15, 1972	AEC/L letter requesting review of consequences of postulated pipe failures outside of the containment structure.
December 21, 1972	AEC/L requests additional info.
December 26, 1972	AEC/L forwards Statement of Requirements re Operational Radiological Env. Monitoring program.
January 3, 1973	CP&L submits Amend. #5 to Env. Report.

January 5, 1973	CP&L responds to D/L letter of Dec. 15, 1972, regarding postulated pipe failures outside the containment and advises that their reply will be submitted by March 5, 1973.
January 9, 1973	CP&L responds to AEC/L letter of Nov. 20, 1972, and adopts GE;s topical NEDM - 10735 as their response to fuel densification.
January 9, 1973	CP&L responds to AEC/L's letter of Dec. 21, 1972, requesting additional information and advising info. will be submitted by March 5, 1973.
January 12, 1973	AEC/L forwards Errata Sheet to Dec. 15, 1972 letter regarding postulated pipe failures outside containment.
January 29, 1973	U. S. Dept. of Justice letter to J. Rutbera concerning antitrust matters.
January 29, 1973	CP&L submits Amendment No. 6 to its Environmental Report.
February 6, 1973	AEC/L letter requesting additional information.
February 7, 1973	CP&L forwards letter discussing schedule slippage of two weeks.
February 9, 1973	Establishment of Atomic Safety and Licensing Board - names Elizabeth S. Bowers, Esq., Chairman; John B. Farmakides, Esq., Member; and Dr. Marvin M. Mann, Member.
February 14, 1973	CP&L advises that they will meet the April 2, 1973 deadline requested in AEC/L letter of Feb. 6, 1973 regarding additional information.
February 15, 1973	AS&LB to rule on petitions and/or requests for leave to intervene - published in Federal Register.

February 27, 1973	CP&L submits the Brunswick Steam Electric Plant's Industrial Security Plan.
March 5, 1973	AEC/DL letter regarding the two week slippage schedule and advises that all request to applicant requesting additional information have been and will continue to be under management review to assure that they are essential for AEC's review of license applications.
March 5, 1973	CP&L transmits Brunswick Steam Electric Plant Design Report No. 16 - "Balance of Plant Electrical Drawings."
March 5, 1973	CP&L submits Addendum "A" to Design Report No. 15 - "Electrical Design Drawings". Proprietary Information.
March 5, 1973	CP&L submits Amendment No. 15 to the FSAR. This amendment consists of responses to questions raised in Dr. W. R. Butler's letter of December 21, 1972, and revised pages to be inserted in the FSAR.
March 6, 1973	Memorandum and Order issued to permit intervention and Notice of Hearing on a Facility Operating License. Board designated as Michael Glaser, Esq., (chairman); Mr. Glenn O. Bright, and Dr. J. V. Leeds, Jr. members and Dr. Forrest J. Remick designated as technically qualified alternate and James R. Yore, Esq. as alternate in administrative proceedings.
March 6, 1973	CP&L submits a report titled "Analysis of Postulated Pipe Failures at BSEP" and drawings showing the routing of main steam and feedwater lines.
March 22, 1973	CP&L submits the Environmental Radiological Monitoring Program for the Brunswick Plant.
March 23, 1973	Confirmation of Prehearing Conference for Brunswick Station scheduled for April 24, 1973, 10:00 a.m. at the U.S. District Courtroom, U.S. Courthouse & Federal Building, Wilmington, North Carolina 28401.

April 2, 1973	CP&L submits Amendment No. 16 consisting of revised and additional pages to be inserted in the FSAR and responses to AEC/DL's letter of February 6, 1973 requesting additional information.
April 4, 1973	AEC/DL advises CP&L that their Brunswick Steam Electric Plant's Industrial Security Plan submitted on February 27, 1973 is being withheld from public disclosure.
April 4, 1973	Prehearing Conference to be held on May 2, 1973 in US District Court, U.S. Court House, Federal Building, Water and Princess Streets, Wilmington, N.C. 28401 at 10:00 A.M.
April 6, 1973	AEC/DL letter requesting additional justification for Design Report No. 15 to be classified as proprietary data and withheld from public disclosure.
April 10, 1973	State of North Carolina, Utilities Commission, Raleigh, N.C., writes concerning the one year delay in the Brunswick Nuclear Facility.
April 11, 1973	Department of Justice advises that they do feel there is a need for an antitrust hearing on the operating license for Brunswick.
April 12, 1973	AEC/OA&I letter signed by A. Braitman forwarding CP&L a copy of the Attorney General's advice of April 11, 1973.
April 17, 1973	CP&L submits copies of the Brunswick application for all board members, and Ron Horton.
April 18, 1973	Notice of Receipt of Attorney General's Advice and Time for Filing of Petitions to Intervene on Antitrust Matters published in Federal Register on April 18, 1973. Signed by A. Braitman, C/Office of Antitrust & Indemnity, Direc. of Licensing. Signed April 11, 1973.

April 19, 1973	CP&L submits Amendment No. 17 consisting of revised and additional pages to be inserted in the FSAR.
April 30, 1973	See if TB CP&L advises they will reevaluate the radiological doses calculated by the AEC Staff. CP&L's doses exceeds the low as practicable guidelines approved by the Commission. This reevaluation will take one month to evaluate and the results will be discussed with the AEC Staff.
May 3, 1973	CP&L letter transmitting additional justification to withheld Addendum A to Design Report 15 from public disclosure as proprietary information.
May 9, 1973	AEC/DL letter requesting additional information and asking for a reply by July 3, 1973.
May 15, 1973	AEC/DL letter to CP&L advising that Addendum A to Design Report 15 has been withheld from public disclosure as proprietary data.
May 15, 1973	Memo to V. A. Moore from W. Butler, C/BWR-1 advising that Addendum A to Design Report 15 submitted by CP&L on March 5, 1973 has been classified as proprietary and withheld from public disclosure in accordance with Section 2.790(b) of 10 CFR Part 2.
May 18, 1973	CP&L advises that AEC/DL letter, dated May 9, 1973 regarding supplementary information needed to complete the review of the Brunswick application will be answered by July 3, 1973, as requested.
May 21, 1973	CP&L submits Amendment No. 18 to the FSAR. This Amendment consists of revised pages to the FSAR.
June 7, 1973	CP&L meets with AEC Staff in Bethesda, Md. to discuss applicant's responses to AEC questions.

June 12, 1973	AEC/DL Summary of Meeting held on June 7, 1973 with CP&L to Discuss Applicant's Responses to Staff Questions.
June 12, 1973	AEC/DL to CP&L transmitting Summary of Meeting of June 7, 1973.
June 18, 1973	Draft Environmental Statement related to the Continued Construction and Proposed Issuance of an Operating License for the Brunswick Steam Electric Plant, Units 1 & 2 issued.
June 22, 1973	Notice and Order for Prehearing Conference to convene at 9:00 A.M. on Thursday, July 19, 1973 in the U.S. Court House Federal Bldg., Water & Princess Streets, Wilmington, N.C. 28401.
June 29, 1973	AEC/DL letter requesting CP&L to forward its 1972 Annual Financial Report.
July 2, 1973	CP&L submits a new page 51a and revised pages 34, 35, 36, and 38 to the Industrial Security Plan originally submitted on February 27, 1973.
July 3, 1973	CP&L submits Amendment No. 19 to the FSAR. This Amendment contains responses to the questions raised in the Licensing Staff's letter, dated May 9, 1973.
July 3, 1973	CP&L submits the response to AEC question 3.12 which was raised in DL's letter of May 9, 1973. This response is considered proprietary by the General Electric Company and CP&L requests that it be withheld from public disclosure.
July 9, 1973	CP&L submits Amendment No. 7 to the Environmental Report.
July 10, 1973	Notice of Reconstitution of Board issued by the AS&LBP. Dr. J. V. Leeds, Jr. was a member of the Board as was Dr. Forrest J. Remick an alternate member. Neither Dr. Leeds nor Dr. Remick is able to serve on this board due to conflicts. Dr. Frank F. Hooper is appointed a member of the Board.

July 16, 1973	AEC/DL letter requesting additional information regarding redundant features of the spent fuel cask lifting lugs, lifting rig, and the 125 ton Reactor Building Class I crane.
July 19, 1973	AEC/DL letter requesting the necessary analyses and other relevant data for determining the consequences of densification and the effects on normal operation, anticipated transients, and accidents, etc. using the guidance provided in the enclosure.
July 26, 1973	CP&L submits Amendment No. 20 to the FSAR. This Amendment contains the response to question 5.59 raised by the AEC/DL Staff in a letter signed by V. Moore, dated May 9, 1973 and revised pages to be inserted in the FSAR.
July 26, 1973	CP&L submits Addendum A to the report entitled "Analysis of Postulated Pipe Failures at BSEP". The original report was submitted on March 6, 1973.
July 27, 1973	AEC/DL letter requesting additional information and requesting an adequate reply by August 10, 1973.
August 3, 1973	Prehearing Conference Order issued by AS&LB. Date and place undetermined. The Order directed the parties involved to report to the Board by August 15, 1973 on 6 items.
August 10, 1973	CP&L transmits Amendment No. 21 to the FSAR consisting of responses to questions 9.16 through 9.22 and 10.51 through 10.58 raised by AEC/DL in their letters dated July 26, 1973 and July 16, 1973.
August 10, 1973	CP&L letter requesting that the response to W. Butler's letter of July 27, 1973 containing the answers to questions 9.16, 9.19 and 9.22 be withheld from public disclosure as proprietary information.

August 20, 1973	AEC/DL letter requesting additional financial information for testimony (form enclosed for completion).
August 20, 1973	CP&L responds to AEC/DL letter dated July 19, 1973 regarding fuel densification.
August 24, 1973	AEC/DL letter to CP&L advising that their response to question No. 3.12 of May 9, 1973 is being withheld from public disclosure as proprietary information.
August 24, 1973	Memo from R. Powell, BWR-1 LPM to V. A. Moore, AD/BWRS regarding withholding of information pursuant to Section 2.790.
August 27, 1973	Summary of Preliminary Instrumentation, Control, and Electrical Drawing Review Meeting. Summary written by R. Powell, LPM, BWR-1. This meeting was held on August 17, 1973.
August 30, 1973	CP&L advise that they will respond to AEC/DL letter of August 20, 1973 regarding financial information by October 1, 1973, as requested.
September 4, 1973	CP&L transmit Amendment No. 22 to the FSAR. This Amendment contains responses to questions raised by staff and revised pages to be inserted into the FSAR.
September 10, 11, 12, 1973	Instrumentation and Electrical drawing review was held at the Brunswick site.
September 11, 1973	AEC/DL letter to applicant requesting additional information concerning their report "Shared Diesel Generator System Evaluation".
September 13, 1973	AEC/DL and CP&L meet in Phillips Bldg., Bethesda, Md., to discuss quality assurance program, conduct of operations, Emergency Plan and Industrial Security Plan.
September 19, 1973	AEC/DL request additional information regarding the Brunswick County Airport.

September 28, 1973	CP&L advises that the information regarding the Brunswick County Airport will be submitted by October 26, 1973.
October 1, 1973	Memo to V. A. Moore thru W. R. Butler from R. Powell justifying the withholding of information pursuant to Section 2.790. This memo is in conjunction with CP&L's request for withholding in their July 27, 1973 letter.
October 1, 1973	AEC/DL letter advising that CP&L's responses 9.16,9.19, 9.21 and 9.22 have been approved as proprietary information and withheld from public disclosure.
October 9, 1973	CP&L submits Amendment 23 responding to staff concerns raised in meetings and telephone conferences. Updating of the General Information section of the application was also submitted.
October 12, 1973	AEC/DL letter transmitting the "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors", Wash 1270 dated September, 1973 and requesting CP&L response and commitment by January 1, 1974.

APPENDIX B
Financial Information

	(dollars in millions) Calendar Year Ended December 31		
	1972	1973	1970
Long-term debt	$ 684.1	$ 534.1	$ 399.2
Utility plant (net)	1,357.1	1,059.9	829.6
Ratio - debt to fixed plant	.50	.50	.48
Utility plant (net)	1,357.1	1,059.9	829.6
Capitalization	1,301.4	954.3	744.5
Ratio of net plant to capitalization	1.04	1.11	1.11
Stockholders' equity	617.3	420.1	345.3
Total assets	1,418.8	1,114.1	884.9
Proprietary ratio	.44	.38	.39
Earnings available to common equity	50.9	29.1	20.1
Common equity	443.5	295.7	255.9
Rate of earnings on common equity	11.5%	9.8%	7.9%
Net income	60.5	37.5	24.8
Stockholders' equity	617.3	420.1	345.3
Rate of earnings on stockholders' equity	9.8%	8.9%	7.2%
Net income before interest	102.2	69.1	48.8
Liabilities and capital	1,418.8	1,114.1	884.9
Rate of earnings on total investment	7.2%	6.2%	5.5%
Net income before interest	102.2	69.1	48.8
Interest on long-term debt	39.1	27.9	19.6
No. of times long-term interest earned	2.6	2.5	2.5
Net income	60.5	37.5	24.8
Total revenues	338.5	274.4	218.0
Net income ratio	.18	.14	.11
Total utility operating expenses	236.3	205.3	169.2
Total utility operating revenues	307.1	255.6	204.8
Operating ratio	.77	.80	.83
Utility plant (gross)	1,571.5	1,242.9	991.4
Utility operating revenues	307.1	255.6	204.8
Ratio of plant investment to revenues	5.12	4.86	4.84

	1972		1971	
Capitalization:	Amount	% of Total	Amount	% of Tota
Long-term debt	$ 684.1	52.6%	$ 534.1	56.0%
Preferred stock	173.8	13.3	124.4	13.0
Common stock & surplus	443.5	34.1	295.7	31.0
Total	$1,301.4	100.0%	$ 954.2	100.0%

Moody's Bond Rating: A

Dun & Bradstreet Credit Rating: 5A1