NUREG-0717

Safety Evaluation Report
related to the operation of Virgil C. Summer Nuclear Station, Unit No. 1

Docket No. 50-395

South Carolina Electric and Gas Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

February 1981

[redacted] 8103030656

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

Page

APPENDICES

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES (Continued)

1 INTRODUCTION AND GENERAL DESCRIPTION OF FACILITY

1.1 Introduction

The South Carolina Electric & Gas Company (hereinafter referred to as the applicant) filed an application dated June 30, 1971, for licenses to construct and operate the proposed Virgil C. Summer Nuclear Station, Unit 1 (hereinafter referred to as the facility or plant). The proposed facility is located at a site in Fairfield County, South Carolina, approximately 15 miles southwest of Winnsboro, the county seat.

The Atomic Energy Commission, now the Nuclear Regulatory Commission (NRC), reported the results of its review prior to construction in a Safety Evaluation Report dated August 29, 1972, in Supplement 1 to the Safety Evaluation Report, dated January 12, 1973, and in Supplement 2 to the Safety Evaluation Report, dated February 28, 1973. Following a public hearing before an Atomic Safety and Licensing Board, Provisional Construction Permit No. CPPR-94 was issued on March 21, 1973.

The applicant tendered an application for an operating license for the facility by letter dated December 10, 1976. Upon completion of our acceptance review, the application was docketed on February 24, 1977.

Our technical review of radiological safety matters with respect to issuance of an operating license for the facility was based on the Final Safety Analysis Report including 22 amendments thereto, all of which are available for public inspection at the NRC's Public Document Room which is located at 1717 H Street, N.W. Washington, D.C., and at the Richland County Library which is located in Columbia, South Carolina. In the course of our review, we held a number of meetings with representatives of the applicant to discuss the design, construction, and proposed operation of the facility. As a consequence, we requested additional information which the applicant provided in amendments to the Final Safety Analysis Report. A chronology of the principal actions related to the processing of the application is included as Appendix A to this Safety Evaluation Report.

This Safety Evaluation Report summarizes the results of the NRC staff's radiological safety review of the facility.

In accordance with the provisions of the National Environmental Policy Act of 1969, we considered the environmental impact of the proposed operation of the facility in accordance with 10 CFR Part 51. The NRC staff's Final Environmental Statement for the operating license stage of review will be published in NUREG-0719.

Our review and evaluation of this facility for an operating license is only one stage in our continuing review of the design, construction, and operation of the facility. The proposed design of the facility was reviewed at the construction permit stage. Construction of the facility has been monitored in accordance with the NRC's inspection program. At this, the operating license stage, we have reviewed the final design to determine that the NRC's safety requirements have been met. If an operating license is granted, the facility must be operated in accordance with the terms of the operating license and the NRC's regulations and will be subject to the NRC's continuing inspection program.

1.2 General Description of the Facility

The facility utilizes a nuclear steam supply system incorporating a pressurized water reactor and a three-loop reactor coolant system. The reactor core is composed of fuel rods made of slightly enriched uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs that are grouped and supported into fuel assemblies. The mechanical control rods consist of clusters of stainless steel-clad silver-indium-cadmium alloy absorber rods that are inserted into Zircaloy guide tubes located within the fuel assemblies. The fuel assemblies are loaded in three regions, each utilizing fuel of a different enrichment of Uranium-235, with new fuel being introduced into the outer region, moved inward at successive refuelings, removed from the inner region, and transferred to fuel storage.

Water will serve as both the moderator and the coolant, and will be circulated through the reactor vessel and core by three vertical, single stage certifugal pumps, one located in the cold leg of each loop. The coolant will be heated by the core and circulated through the three steam generators, where heat will be transferred to the secondary system to produce saturated steam, and then be returned to the pumps to repeat the cycle. An electrically heated pressurizer connected to the hot-leg piping of one of the loops will establish and maintain the reactor coolant system pressure and provide a surge chamber and a water reserve to accommodate changes in reactor coolant volume during operation.

The steam produced in the steam generators will be utilized to drive a tandem compound four-flow exhaust turbine generator and will be condensed in a twin shell, single-pass condenser with divided water boxes. Cooling water drawn from the man-made Monticello Reservoir will be pumped through the tubes of the condenser to remove the heat from, and thus condense, the steam after it has passed through the turbine. The condensate will then be pumped back to the steam generator to be heated for another cycle.

The reactor will be controlled by a coordinated combination of a soluble neutron absorber (boric acid) and mechanical control rods whose drive shafts will allow the facility to accept step load changes of 10 percent and ramp load changes of five percent per minute over the range of 15 to 100 percent of full power during normal operating conditions. With steam bypass, the facility will also have the capability to accept a 100 percent step load rejection without reactor trip.

Plant protection systems are provided to automatically initiate appropriate action whenever a monitored condition approaches pre-established limits. These protection systems will act to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

Supervision and control of both the nuclear steam supply system and the steam and power conversion system will be accomplished from the main control room.

The emergency core cooling system for the facility consists of accumulators and both high and low pressure injection subsystems with provisions for recirculation of the borated water after the end of the injection phase. Various combinations of these features will assure core cooling for the complete range of postulated coolant pipe break sizes.

The nuclear steam supply system is housed in a large, dry, free-standing steel containment structure within a reinforced concrete shield building. An auxiliary building located adjacent to the containment building houses the radioactive waste treatment facilities, components of the engineered safety features, and various related auxiliary systems. A fuel building also located adjacent to the containment building houses the spent fuel pool and new fuel storage racks.

The facility is supplied with electrical power by independent transmission lines from offsite power sources and is provided with independent and redundant onsite emergency power supplies capable of supplying power to shut down the facility safely or to operate the engineered safety features in the event of an accident.

1.3 Comparison with Similar Facilities

Many features of the Virgil C. Summer Nuclear Station, Unit 1, are similar to those we have evaluated and approved previously for other nuclear power facilities now under construction or in operation. To the extent feasible and appropriate, we have relied upon our earlier reviews for those features that were shown to be substantially the same as those previously considered. Where this has been done, the appropriate sections of this Safety Evaluation Report identify the other facilities involved. Our safety evaluation reports for these other facilities have been published and are available for public inspection at the NRC's Public Document Room which is located at 1717 H Street, N.W., Washington, D.C.

1.4 Identification of Agents and Contractors

Gilbert Associates, Incorporated, has been retained by the applicant as architect-engineers for the entire project including plant layouts, system arrangements, and design of balance of plant equipment. The Westinghouse Electric Corporation is supplying the nuclear steam supply system and technical consultation in such areas as initial fuel loading, testing, and initial startup. The General Electric Company designed and supplied the turbine-generator.

1.5 Summary of Principal Review Matters

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

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

We evaluated the design, fabrication, construction, and testing criteria, and expected performance characteristics of the facility structures, systems, and components important to safety to determine that they are in accordance with the NRC's General Design Criteria, quality assurance criteria, regulatory guides, and other appropriate codes and standards, and to determine that any departures from these criteria, guides, codes, and standards have been identified and justified.

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

We evaluated the applicant's plan for the conduct of facility operation, including the organizational structure and the qualifications of the applicant's management, operating and technical support personnel, the measures to be 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 is technically qualified to operate the facility safely.

We evaluated the design of the systems provided for control of the radioactive effluents from the facility to determine that these systems can control the release of radioactive wastes from the facility within the limits of the NRC's regulations (10 CFR Part 20) and that the equipment provided is capable of being operated by the applicant in such a manner as to reduce radioactive releases to levels that are as low as reasonably achievable within the context of the NRC's regulations (10 CFR Part 50), and to meet the dose design objectives of Appendix I to 10 CFR Part 50.

We are evaluating the financial information provided by the applicant as required by Section 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50 to determine the financial qualifications of the applicant to operate the facility. The results of this evaluation will be presented in a supplement to this Safety Evaluation Report.

We evaluated the information provided by the applicant in response to the additional requirements resulting from the accident at Three Mile Island, Unit 2. We evaluated the applicant's responses to the requirements specified in NUREG-0694 in accordance with the Commission's Statement of Policy dated June 16, 1980.

Subsequently, NUREG-0737 was issued superseding NUREG-0694. We have not completed our review of the applicant's responses to NUREG-0737. We will report the results of that evaluation in a supplement to this Safety Evaluation Report.

1.6 Outstanding Issues

We have identified outstanding issues in our review which have not been resolved with the applicant. We will complete our review of these items prior to issuance of an operating license and will discuss the resolutions of each of these items in a supplement to this Safety Evaluation Report. These items are listed below and are discussed further in the sections of this Safety Evaluation Report as identified below.

1.6.1 Stability of Subsurface Materials and Foundations of the Service Water Pumphouse and Intake Structure (Sections 2.5.4 and 3.7.2)

1. The service water pumphouse and intake structure settled further than predicted, causing a number of cracks over the longer length of the intake structure. The cracks were grouted and the service water pond was filled. We are reviewing the information provided by the applicant to determine whether additional settlement of the soils beneath the pumphouse and intake structure can be expected.

2. Some misalignments of the 36-inch bypass pipe line to the circulating water intake structure and the 30-inch service water pipe lines have been discovered since the rebound of the pumphouse. The applicant had to re-excavate in order to connect the pipes to the pumphouse. We are currently evaluating the significance and the cause of this misalignment.

3. The joint provided for the electrical duct bank at the pumphouse wall is designed to allow for a differential displacement of at least one inch all around. We are currently evaluating the cause and significance of a downward movement of 0.84 inches of the duct bank relative to the pumphouse.

1.6.2 Slope Stability of the West Embankment of the Service Water Pond (Section 2.5.6)

At the west embankment of the service water pond, the as-built conditions are somewhat different from the design. The effect of this on static and dynamic behavior of the west embankment and its effect on the service water pumphouse and intake structure are currently under evaluation.

1.6.3 Seismic System and Subsystem Analysis (Section 3.7.2)

The seismic effects of the activities resulting from the construction of the Monticello Reservoir near the plant are still under review.

1.6.4 Seismic Qualification of Seismic Category I Instrumentation and Electric Equipment (Sections 3.9.2 and 3.10)

We will perform a confirmatory review for the seismic qualification of five selected pieces of equipment and will review the applicant's information regarding our concerns with the seismic qualification program as identified in our trip report.

1.6.5 Environmental Qualification of Mechanical and Electrical Equipment (Section 3.11)

We require that the applicant reassess the qualification documentation for equipment installed at the facility to establish that the qualification methods used and the results obtained are in conformance with the staff positions contained in NUREG-0588.

1.6.6 Preservice Inspection Program (Sections 5.2.4 and 6.6)

The preservice inspection program for the reactor coolant boundary and Class 2 and 3 components is currently under review by the NRC staff.

1.6.7 Reactor Vessel Integrity-Compliance with Appendix G to 10 CFR Part 50 (Sections 5.3.1 and 5.3.2)

1. We require that the applicant identify all of the high strength ferritic welds in the pressurizer and submit the necessary fracture toughness test results for these welds.

2. We require the applicant to submit impact energy data for the ferritic pressure-retaining materials (including base, weld, and heat-affected zone materials) of the steam generator.

3. We require that the applicant submit the required fracture toughness data for the ferritic materials for bolting and other fasteners within the reactor coolant pressure boundary.

1.6.8 Engineered Safety Feature and Reactor Protection System Status Monitoring System (Sections 7.5.1 and 22.2)

We have not reviewed the final results of the applicant's control room design review which is being performed to identify and correct any human factors deficiencies.

1.6.9 Use of a Load Sequencer with Offsite Power (Section 8.3.5)

We require the submittal of a complete design description, reliability analysis, and sneak circuit analysis of the load sequencer.

1.6.10 Fire Protection (Sections 7.4.2, 8.3.4, and 9.5.1)

We have not completed our review of the design of the alternate shutdown system.

1.6.11 Emergency Planning (Section 13.3 and 2.3.3)

We have identified a number of discrepancies in the applicant's emergency plan which will require a revision to the plan.

1.6.12 Quality Assurance (Section 17.5)

We require the applicant to justify certain systems, structures, and components which are not currently under the control of the quality assurance program.

1.6.13 Financial Qualifications (Section 20.0)

To assure that we have the latest information to make a determination of the financial qualifications of the applicant, our review will be conducted during the later stages of our review of this application.

1.6.14 Conformance with NUREG-0737 (Section 22.0)

We will require that the applicant demonstrate conformance with the additional requirements of NUREG-0737.

1.6.15 Joint IE/NRR Audit (Sections 22.2, 22.3 and 22.5)

We will conduct a joint IE/NRR audit of the applicant's management and technical competence.

1.6.16 Inadequate Core Cooling Instruments (Section 22.2 and 22.5)

We require additional information from the applicant in order to complete our review of inadequate core cooling instruments.

1.6.17 Reactor Coolant System Vents (Sections 22.3 and 22.5)

We require additional information from the applicant in order to complete our review of the reactor coolant system vents.

1.7 Confirmatory Issues

There are a number of matters for which we have completed our review and have established positions which are acceptable to us and for which there appears to be no significant disagreement on the part of the applicant. The applicant has been advised of our positions and we are awaiting confirmation of the applicant's commitment to comply with these positions and/or receipt of the appropriate confirmatory information. Further discussion of these items will be reported in a supplement to this Safety Evaluation Report. These items, with reference to the applicable sections of this Safety Evaluation Report, are identified below.

1.7.1 Containment Pressure Test (Section 3.8.1)

Prior to operation of the facility, the containment will be subjected to an acceptance test during which the internal pressure will be 1.15 times the containment design pressure.

1.7.2 Preoperational Flow-Induced Vibration Testing of Reactor Internals (Section 3.9.2)

The preoperational vibration program (tests, predictive analysis, and post-test inspection) will be performed prior to operation of the facility.

1.7.3 Inservice Testing of Pumps and Valves (Section 3.9.6)

The applicant will conduct a testing program which includes baseline preservice testing and periodic inservice testing. The inservice test program will be submitted 30 days prior to fuel loading.

1.7.4 Functional Design of Reactivity Control Systems (Section 4.6)

Preoperational scram time tests will be performed to verify that the control rods will scram within the time requirements identified in the Technical Specifications.

1.7.5 Steam Generator Inspection Program (Section 5.4.2)

The Technical Specifications do not contain the details of the required preservice inspection, and Technical Specification Sections 4.4.5.2.b, 4.4.5.1.b.3, 4.4.5.2.c, and 4.4.5.3.b will be rewritten to convey the same meaning found in the corresponding section of NUREG-0452, Revision 2.

1.7.6 Residual Heat Removal System (Section 5.4.3)

The applicant will establish the applicability of the Diablo Canyon natural circulation tests to this facility and will assure that the phenomenon of voiding in the reactor vessel for high cooldown rates ir properly reflected in the testing.

1.7.7 Emergency Core Cooling System Performance Evaluation (Section 6.3.4)

1. The applicant will incorporate the revised analyses for evaluating peak clad temperature based upon the approved LOCTA computer program into the Final Safety Analysis Report.

2. The applicant will revise Table 15.3-2 and the corresponding figures of the Final Safety Analysis Report for the small-break loss-of-coolant accident.

1.7.8 Emergency Core Cooling System Tests and Inspections (Section 6.3.5)

The applicant will conduct scale-model sump tests to confirm that recirculation sump performance will be acceptable following a postulated loss-of-coolant accident and that undesirable vortex formation will not be experienced.

1.7.9 Engineered Safety Features Atmosphere Cleanup Systems (Section 6.5.1)

The applicant will amend the Technical Specifications on adsorber efficiencies for iodine removal in accordance with Regulatory Guide 1.52.

1.7.10 Potential Design Deficiencies in Bypass, Override, and Reset Circuits of Engineered Safety Features (IE Bulletin 80-06) (Section 7.1.4)

We will review the results of the applicant's testing of engineered safety features systems control circuits with respect to deficiencies in bypass, override, and reset of engineered safety features systems action.

1.7.11 Trip Setpoints and Margins (Section 7.2.3)

We will review the applicant's response to our generic letter on concerns of level measurement errors due to environmental temperature effects on level instrument reference legs in order to make appropriate changes to the Technical Specifications.

1.7.12 Auxiliary (Emergency) Feedwater System (Section 7.4.1)

The applicant will document the modifications (including descriptive information and electrical schematics) to the emergency feedwater system which resulted from our site visit of November 1980.

1.7.13 Sustained Degraded Grid Voltage Position and Offsite/Onsite Power System Interaction (Section 8.?.2)

1. The applicant will provide a comparison table to demonstrate that the time delay chosen in the event of a degraded voltage is less than the maximum time delay assumed in the accident analyses.

2. The applicant will document the analytical method used for calculating voltage at all distribution levels to demonstrate that the transformer tap settings have been fully optimized for the facility, and test plans and test results to demonstrate that the analytical method used for calculating these voltages at all distribution levels is valid.

1.7.14 Onsite Emergency Power System (Section 8.3.1)

The applicant will provide an analysis to demonstrate that at no time during sequencing of safety loads on diesel generators will the starting voltage at the 460 volt level go below 80 percent of the rated voltage.

1.7.15 Onsite Emergency Power System (Section 8.3.1)

Successful preoperational testing of the onsite emergency power systems will be verified by the Office of Inspection and Enforcement.

1.7.16 Emergency Feedwater System (Section 10.4.7)

We have evaluated the preheat model steam generators of the emergency feedwater system for its hydraulic instabilities (water hammer phenomenon potential). Based on the studies, we have established the need for a verification test to demonstrate that no damaging water hammer will occur in the steam generator or the feedwater system. The applicant will conduct the verification test and report the results to the staff.

1.7.17 Solid Radioactive Waste Treatment System (Section 11.2.3)

We will review the applicant's complete process control program as part of our review of the radiological effluent technical specifications.

1.7.18 Input Parameters for Transient and Accident Analyses (Section 15.1.1)

The analyses of the transient and accident analyses assumed a time of 2.3 seconds to reach 85 percent of the control rod travel. This will be verified during the preoperational testing program.

1.7.19 Core Reactivity Insertion Events (Section 15.2.4)

The results of the boron dilution events from hot standby and cold shutdown will be submitted prior to issuance of the operating license.

1.7.20 Decrease in Reactor Coolant Inventory Event (Section 15.2.5)

An event which can result in a decrease of reactor coolant inventory with an expected moderate frequency is an inadvertent opening of a pressurizer safety or relief valve. The applicant has informed us that this analysis is documented in WCAP-9600 and will also document this analysis in Section 15.2.5 of the Final Safety Analysis Report.

1.8 Licensing Conditions

We have taken positions on certain issues requiring implementation and/or documentation after issuance of the operating license. The license will be conditioned as necessary to assure acceptable implementation of our positions. These items are listed below and are discussed further in the sections of this Safety Evaluation Report as indicated below.

1.8.1 Nearby Facilities (Section 2.2.2)

According to the applicant, when the facility is put into operation, the military training route, Route 46, will be relocated to a new location, or abandoned. We will require the applicant to provide written notification from the appropriate military authorities that this will be done, and we will verify that Training Route 46 has been relocated or abandoned prior to the operation of the facility.

1.8.3 Missile Selection and Description - Internally Generated Missiles (Inside Containment) (Section 3.5.1)

The applicant did not consider the reactor coolant pump and motor component to be a credible source of missiles using the Westinghouse analysis presented in WCAP-8163. Further research is being performed by the Electric Power Research Institute and the French Atomic Energy Commission. We are following the development and performance of this program. If the results of these generic investigations indicate that additional protective measures are warranted to prevent excessive pump overspeed or to limit potential consequences to safety-related equipment, we will determine what modifications if any, are necessary to assure an acceptable level of safety.

1.8.4 Thermal Performance-PAD 3.3 (Section 4.2.2)

The improved Westinghouse code described by WCAP-8720 was used to analyze the densification effects on fuel thermal performance. We have not completed our review of this evaluation. We anticipate completion of our review of the Westinghouse evaluation prior to operation at extended burnup.

1.8.5 Mechanical Performance (Section 4.2.3)

The applicant has agreed to examine all fuel rods residing in specific locations for baffle-jetting failure at the first refueling outage. Should damage be observed at that time, corrective action would be taken.

1.8.6 Design Description - Control (Section 4.3.2)

The applicant has performed all analyses concerning reactivity control requirements without part-length rods and therefore, the use of part-length rods will be prohibited.

1.8.7 Steam Generator Inspection Ports (Section 5.4.2)

We require that the steam generator inspection ports be installed prior to startup after the first refueling.

1.8.8 Inservice Inspection Program (Section 5.2.4 and 6.6)

The inservice inspection program will be evaluated after the applicable ASME Code edition and addenda have been determined and before the initial inservice inspection.

1.8.9 Row 1 Steam Generator Tube Plugging (Section 5.4.2)

Unless information is developed to demonstrate that potential cracking in the U-bend region of row 1 tubes can be avoided, we will require the plugging of all row 1 tubes prior to issuance of the full power license.

1.8.10 Residual Heat Removal System (Section 5.4.3)

We require that the applicant comply with ASB 5-1 by installing a switch in the control room that would lock or unlock power to the residual heat removal system suction valves. We require that such a modification be made prior to the first refueling unless an acceptable alternative is provided prior to issuance of the operating license.

1.8.11 Instrument and Control Vibration Tests for Emergency Diesel Engine Auxiliary Support Systems (Section 9.5.4)

We require the applicant to either provide test results and results of analyses which validate that the skid-mounted control panels and mounted equipment have been developed, tested, and qualified for operation under severe vibrational stresses encountered during diesel engine operation or floor mount the control panels presently furnished with the diesel generator separate from the skid on a vibration-free floor area. These modifications must be implemented prior to the first refueling.

1.8.12 Emergency Diesel Engine Lubricating Oil System (Section 9.5.7)

We require that an alarm which indicates failure of the motor-driven pump of the emergency diesel engine lubricating oil system be installed by the first refueling.

1.8.13 Industrial Security (Section 13.6)

The identification of vital areas and measures used to control access to these areas may be subject to changes in the future based on our confirmatory evaluation of the facility to determine those areas where acts of sabotage might cause a release of radionuclides in sufficient quantities to result in dose rates equal to or exceeding 10 CFR Part 100 limits.

1.8.14 Analytical Techniques (Section 15.1.2, 15.2, 15.3)

The analytical techniques for which we have not completed our review are described in the following topical reports and are discussed in the listed paragraphs:

1. WCAP-7907 LOFTRAN Code Description (Sections 15.2.2., 15.2.3, 15.2.4, 15.2.5, 15.2.6, 15.3.1, 15.3.3)

2. WCAP-7908 A FACTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod (Sections 15.2.3, 15.2.4, 15.3.3)

3. WCAP-9227 Ma n Steamline Break Sensitivity Studies (Section 15 3.2)

4. WCAP-9230 Report on the Consequences of a Postulated Main Feedline Rupture (Section 15.3.1)

5. WCAP-7998 BLKOUT Code Description (Section 15.2.2)

6. WCAP-7909 MARVEL Code Description (Section 15.3.2)

If the final approval of these methods indicates revisions to the analyses are required, the applicant will be required to implement the results of such changes.

1.8.15 Anticipated Transients Without Scram (Sections 4.6 and 15.3.5)

The matter of anticipated transients without scram is currently before the Commission for review. The applicant will be required to implement facility modifications in conformance with the Commission's final resolution of this matter.

1.8.16 Post-Accident Monitoring (Sections 6.3.3 and 7.5.2)

We require the applicant to conform with Revision 2 to Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980, or provide adequate justification for any deviations.

1.9 Generic Issues

The Advisory Committee on Reactor Safeguards periodically issues a report listing various generic matters applicable to light water reactors. A discussion of these matters is provided in Appendix C to this Safety Evaluation Report which includes references to sections of this Safety Evaluation Report for more specific discussions concerning this facility.

We continuously evaluate the safety requirements used in our review against new information as it becomes available. In some cases, we take immediate action or interim measures to assure safety. In most cases, however, our initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether our existing requirements should be modified. These issues being studied are sometimes called generic safety issues because they are related to a particular class or type of nuclear facility. A discussion of our program for the resolution of these generic issues is presented in a Appendix C to this Safety Evaluation Report.

2 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

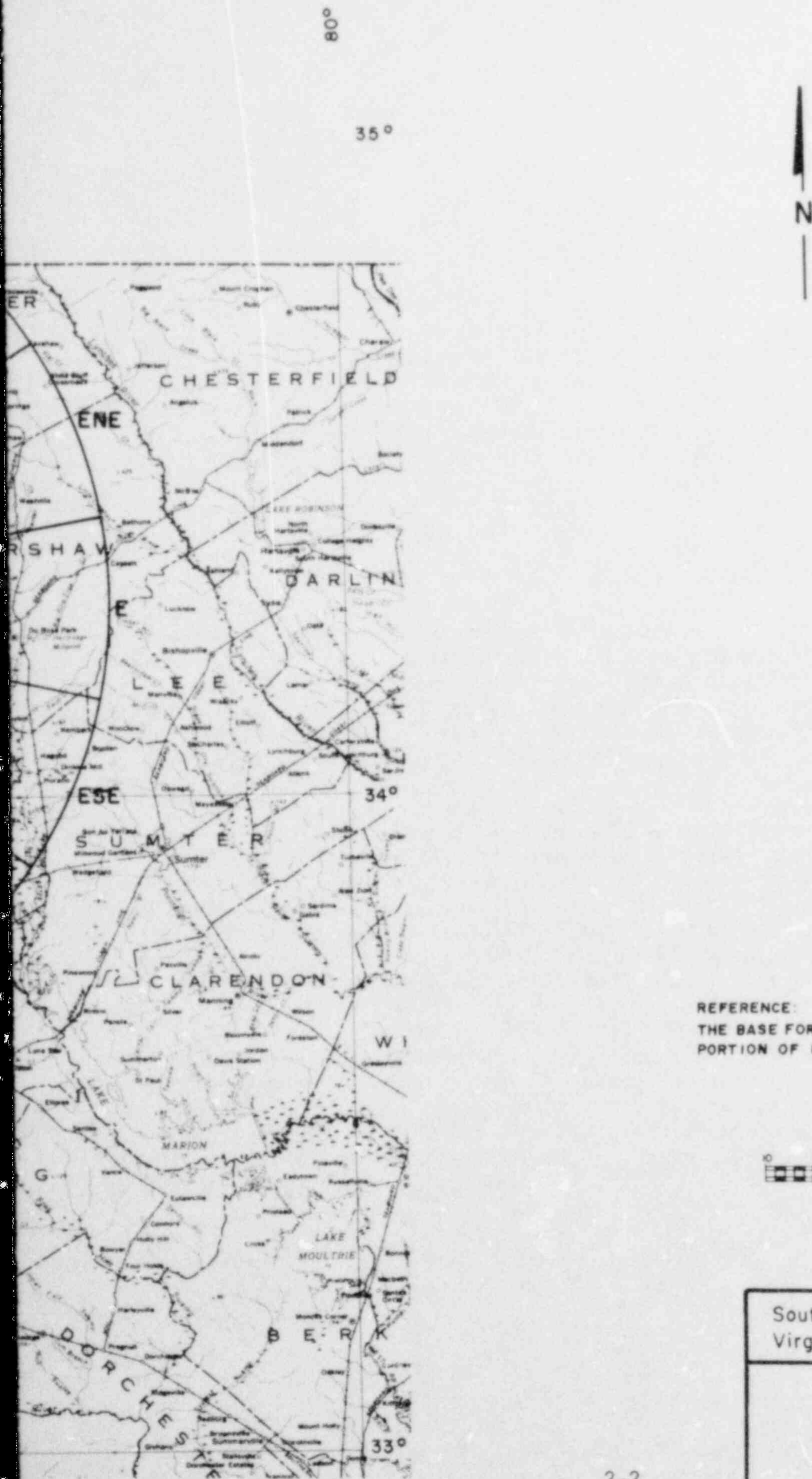
The Virgil C. Summer Nuclear Station, Unit 1 is located in Fairfield County, South Carolina, approximately 15 miles southwest of Winnsboro, South Carolina and 26 miles northwest of Columbia, South Carolina. The site is about 2.5 miles north of Parr, South Carolina. Parr is the location of a decommissioned experimental reactor, the Carolinas-Virginia Tube Reactor. The facility is adjacent to Monticello Reservoir. This reservoir was created by the applicant to provide cooling water for the facility's main condenser and to act as the upper pool of the Fairfield Pumped Storage Facility. The site is shown on a regional map of the area in Figure 2-1 of this Safety Evaluation Report.

2.1.2 Exclusion Area Authority and Control

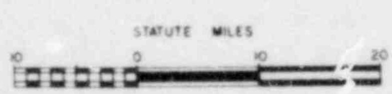
The exclusion area consists of the area within approximately one mile of the reactor building. The minimum distance from the center of the reactor building to the exclusion area boundary is 5,347 feet. The exclusion area includes parts of Monticello Reservoir and the Fairfield Pumped Storage Facility. The exclusion area and principal site features are shown in Figure 2-2 of this Safety Evaluation Report. The applicant owns all of the property within the exclusion area. There may be some limited recreational use of that part of the reservoir within the exclusion area. However, the surface water of Monticello Reservoir, in accordance with South Carolina State law, is in the public domain. The applicant has made arrangements with State and local authorities to control the movement of people on the reservoir in the event of a plant emergency. The applicant has also installed a siren on the circulating water intake structure to immediately warn people on the reservoir in the event of a serious plant accident. Personnel who operate the Fairfield Pumped Storage Facility are employees of South Carolina Electric & Gas Company and therefore are under the applicant's administrative control.

There are no public highways or railroads which traverse the exclusion area. A right-of-way, 68 feet wide, through the exclusion area has been granted to the Duke Power Company for a 115-kilovolt transmission line. Under the terms of the agreement, South Carolina Electric & Gas Company has the authority to exclude or remove personnel and property of Duke Power Company from the exclusion area if necessary. Mineral rights within the exclusion area are jointly owned by the applicant.

We conclude that the applicant has the authority to determine all activities within the exclusion area during normal operation and in the event of an emergency, as required by 10 CFR Part 100.



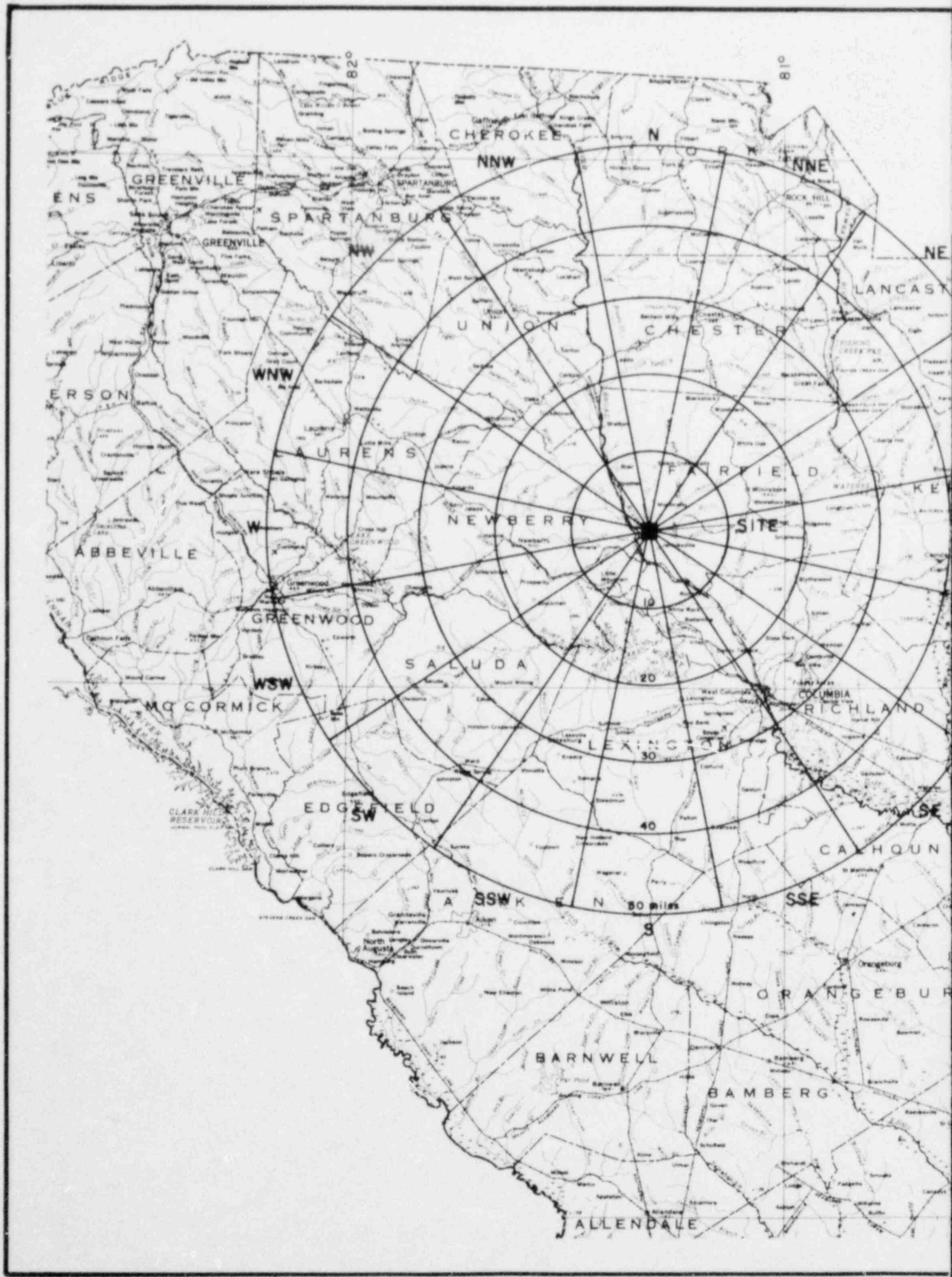
REFERENCE:
THE BASE FOR THIS MAP WAS PREPARED FROM A PORTION OF USGS STATE OF GEORGIA, 1970.



South Carolina Electric & Gas Co.
Virgil C. Summer Nuclear Station

Regional Location Map

Figure 2-1



82°
81°
N
NNE
NE
NNW
NW
WNW
W
WSW
SW
SSW
S
SSE
SE
SITE
10
20
30
40
50 miles
CHEROKEE
YORK
GREENVILLE
SPARTANBURG
UNION
CHESTER
LANCASTER
LAURENS
FAIRFIELD
NEWBERRY
ABBEVILLE
GREENWOOD
SALUDA
MC CORMICK
RICHLAND
COLUMBIA
LEXINGTON
EDGEFIELD
CALHOUN
AIKEN
ORANGEBURG
BARNWELL
BAMBERG
ALLENDALE
CLARK HILL RESERVOIR
ROCK HILL

2.1.3 Population Distribution

The facility is located in a predominantly rural area with generally low population densities. The nearest community with more than 1,000 residents is Winnsboro, South Carolina. Winnsboro is located 15 miles northeast of the facility. In 1970, the population of Winnsboro and the unincorporated community of Winnsboro Mills was 5,723. The largest city within 50 miles of the site is Columbia, South Carolina which is located 26 miles southeast of the facility. In 1970, Columbia had a population of 113,542.

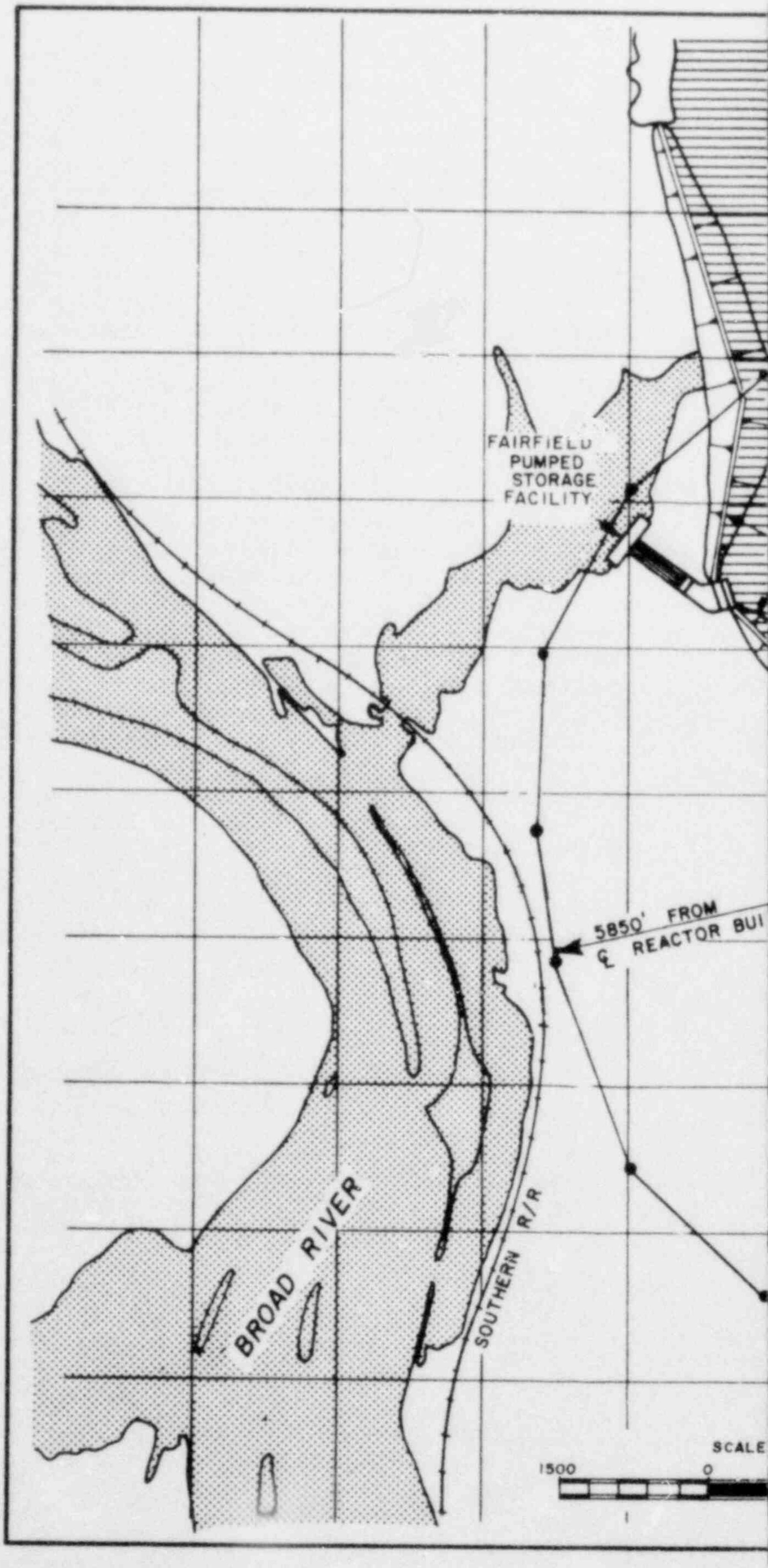
The resident populations within 30 miles of the facility are shown in Table 2-1 of this Safety Evaluation Report. These data were provided by the applicant. Based on 1970 data from the U.S. Bureau of the Census, we made an independent count of the resident population within 30 miles of the facility. Our estimate is in close agreement with the applicant's projections. As shown in Table 2-1 of this Safety Evaluation Report, the population within 30 miles of the facility will increase at a rate of 15 percent per decade between the years 1970 and 2020. The U.S. Bureau of Economic Analysis projects that the population for Economic Area 29, an area comprising the city of Columbia and the surrounding counties including Fairfield County, will increase at a rate of nine percent per decade between the years 1970 and 2020. This indicates that the applicant's population projection for the area within 30 miles of the site is conservative, i.e., higher, compared to the regional population projection made by the U.S. Bureau of Economic Analysis. Based on these comparisons, we find that the applicant's population estimates and projections are acceptable. The most significant source of transient population within 30 miles of the facility is Lake Murray. Lake Murray is a 50,000-acre reservoir located south of the facility. Lake Murray, due to its location northwest of Columbia, attracts a large number of recreational visitors. The creation of Monticello Reservoir is likely to cause some increase in transient population primarily in the area between four and six miles from the facility. We find, based on the information provided by the applicant, that the transient population in the region surrounding the facility does not alter the population distribution to the extent that the boundary distances specified for the facility would be affected.

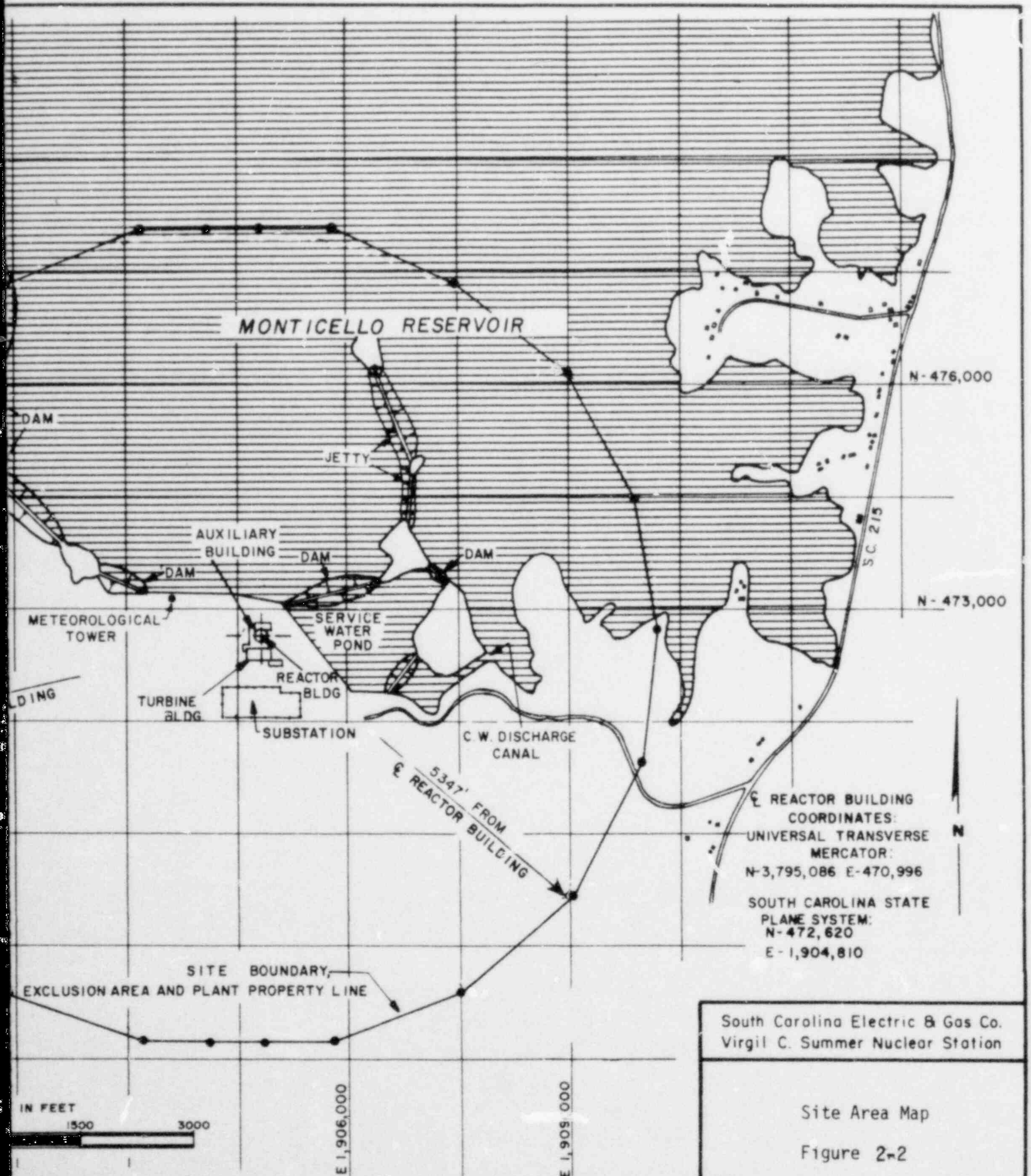
TABLE 2-1

1970 CENSUS AND PROJECTED POPULATIONS WITHIN 30 MILES OF THE SITE

Radius, Miles	CUMULATIVE POPULATION 1970	1980	2020
0-5	1,211	1,295	1,584
0-10	6,370	6,954	8,871
0-20	55,103	62,615	89,768
0-30	352,874	417,714	699,976

The applicant has selected a low population zone with an outer radius of three miles. The resident population within the low population zone as determined


FAIRFIELD
PUMPED
STORAGE
FACILITY
5850' FROM
℄ REACTOR BUI
BROAD RIVER
SOUTHERN R/R
SCALE
1500
0



South Carolina Electric & Gas Co.
Virgil C. Summer Nuclear Station

Site Area Map

Figure 2-2

by the applicant was 365 in 1970. The applicant projects that this population will increase to 470 by the year 2020. For our evaluation of the proposed emergency plans to determine if there is reasonable assurance that appropriate protective measures can be taken in behalf of persons within the low population zone in the event of a serious plant accident, see Section 13.3 of this Safety Evaluation Report.

The nearest population center containing more than about 25,000 persons is Columbia. Columbia's corporate limit is approximately 23 miles southeast of the facility. The applicant estimates that development occurring in the northwestern suburbs of Columbia could bring the boundary of Columbia to within 15 miles northeast of the site over the lifetime of the facility. In addition, the applicant states that it is possible that the area around Winnsboro, 15 miles northeast of the facility, may also grow to a population of 25,000 over the lifetime of the facility. We find the distance from the facility to the current population center, Columbia, or to any other population center likely to develop over the lifetime of the facility, is greater than one and one-third times the low population zone distance of three miles as required by 10 CFR Part 100.

2.1.4 Conclusions

On the basis of our evaluation of the population density and use characteristics of the site environs and the demonstration of acceptable radiological consequences (see Section 15.4 of this Safety Evaluation Report), we conclude that the applicant's specified exclusion area, low population zone and population center distance for the facility meet the guidelines of 10 CFR Part 100 and are acceptable.

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Transportation Routes

South Carolina highways 215/213, the only primary roads close to the site, are located approximately 6,800 feet east and running north and south past the reactor building at their nearest point. County road 311, a secondary road that runs within the exclusion area, is used as an access road, and connects the facility with highways 215/213. The only major highway in the area (I-26) is located approximately 7.5 miles southwest of the site. The Broad River runs generally in a north-south direction approximately 6,050 feet west of the reactor building just beyond the exclusion area boundary. The Monticello Reservoir lies predominantly north of the site, but the southern portion of the reservoir is within the exclusion area.

There are three railroad lines within ten miles of the facility. The closest is located about one mile west of the facility just outside of the exclusion area. A second line approaches within 3.5 miles of the site, and the third line is located 7.5 miles away.

We evaluated the potential hazards that traffic along the highways in close proximity to the site would pose to the facility. Because of the distances between the highways and the facility, there is no danger of blast effects to any of the plant structures from the detonation of the maximum quantity of explosives that may be transported along these routes based on the criteria in Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants."

There are no commercial shipping or other navigation hazards on the Broad River, or on the man-made Monticello Reservoir which is used to supply cooling water to the facility. Both of these bodies of water are used for recreational purposes and present no potential safety problems to the facility.

The closest railroad line to the facility is a branch of the Southern Railway which runs from Alston through Parr and Strother and passes west of the site, about one mile from the reactor building. Two freight trains per day consisting of about 150 cars each are estimated to use this line. No passenger traffic is carried. The applicant presented analyses of a variety of potential accidents along this line, including accidental releases of chlorine, ammonia, propane and methanol. As a result of these analyses the applicant determined, after considering the distance from the railroads to the facility, the atmospheric dispersion of any materials released, the frequency of railroad accidents, and the number of shipments of hazardous materials carried that the probability of an event causing a hazard to the facility was about 10^{-7} per year. Since this value meets the acceptance criteria given in Section 2.2.3 of the Standard Review Plan, the applicant concluded that the probability of such an event is acceptably low and that the railroad line poses no threat to the safe operation of the facility. We have reviewed the applicant's analyses and concur.

2.2.2 Nearby Facilities

There are no airports or military facilities within 10 miles of the site. There are several industrial facilities within five miles of the facility that store or use materials of an explosive nature. Interstate Materials, Inc., has a quarry three miles northeast of the site and stores up to 40 tons of a high explosive known as Torpex (ammonium nitrate-gelignite). An animal feed and fertilizer company located about 3.6 miles northeast of the site carries up to 200 tons of ammonium nitrate as fertilizer in stock. Although ammonium nitrate fertilizer under proper storage conditions is not considered to be detonable, the applicant has postulated the detonation of the total quantity of either Torpex or ammonium nitrate stored at either location and determined that the blast overpressures could be safely accommodated by the facility's safety-related structures. We concur in this determination.

A 12-inch diameter natural gas pipeline leading to the Parr Steam Plant passes within approximately 13,000 feet south of the Virgil C. Summer Nuclear Station Unit 1, at its closest point. Complete rupture of this line, which is buried three feet underground, would not create an explosion or fire hazard to the reactor building or other safety-related structures.

The closest airport, Fairfield County Airport, is located approximately 10 miles east of the site. It is unattended and has a 3200-foot single-strip asphalt runway that can accommodate a twin-engine C-47. Three single-engine planes are permanently based at this airport which has about 3000 to 5000 operations per year. The largest airport in the area is the Columbia Metropolitan Airport located 24 miles southeast of the site. It has two asphalt runways (7550 feet and 5000 feet) and provides 24-hour attendance. Ninety single- and twin-engine, and one small jet aircraft are permanently based at Columbia Metropolitan Airport which can accomodate a C-5 transport. Columbia Metropolitan Airport had about 118,000 operations in 1975. Based on the distances of these airports from the facility as well as previous studies by the NRC staff, we conclude that they will pose no threat to the safe operation of the facility.

There are two military training routes in the vicinity of the facility. Route 157 is about 20 to 35 miles southwest and poses no hazard, but Route 46, controlled by Shaw Air Force Base, passes directly over the site at an altitude of 500-1500 feet. Military regulations restrict aircraft from flying within five miles of, and less than 1500 feet over operating nuclear power plants. According to the applicant, when the facility is put into operation, Training Route 46 will be moved to a new location, or abandoned. We will require that the applicant provide written notification from the appropriate military authorities, and we will verify that Training Route 46 has been relocated or abandoned prior to the operation of the Virgil C. Summer Nuclear Station Unit 1.

2.2.3 Conclusions

Our review has been conducted based upon Criterion 2 of the General Design Criteria and Section 2.2.3 of the Standard Review Plan. We conclude, subject to the relocation or abandonment of Training Route 46, that the facility is adequately protected and can be operated with an acceptable degree of safety with regard to potential offsite accidents occurring as a result of activities at nearby transportation, industrial, or military facilities.

2.3 Meteorology

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

2.3.1 Regional Climatology

The applicant has provided a sufficient description of the regional meteorological conditions of importance to the safe design and siting of the Virgil C. Summer Nuclear Station Unit 1.

The climate of central South Carolina is characterized by cool winters and relatively long and quiet warm summers, as is typical of continental climates in southern regions. The predominant air mass type over this region during the warm half of the year is maritime tropical as influenced by the Gulf of Mexico. During the colder half of the year, continental polar air from Canada alternates with the maritime tropical air over the region. Cold air moving southward into the area from Canada is usually modified and warmed somewhat in crossing the Appalachian Mountains and descending the eastern slopes. Temperatures of 32 degrees Celsius (90 degrees Fahrenheit) or higher may be expected to occur on about 60 days annually, and temperatures of 38 degrees Celsius (100 degrees Fahrenheit) or higher may be expected to occur on about five days annually. Temperatures of zero degrees Celsius (32 degrees Fahrenheit) or lower may be expected to occur on about 60 days annually, but temperatures -18 degrees Celsius (zero degrees Fahrenheit) or lower rarely occur.

Precipitation is well distributed annually and totals about 1170 millimeters (46 inches), occurring mainly as thundershowers in summer, and as rain, or occasionally snow, in the winter. On an annual basis, the relative humidity averages around 75 percent.

Severe weather occurrences in the vicinity of the facility are mainly associated with severe thunderstorms or tropical storms and hurricanes. About once or twice a year, the effects of passing tropical storms are felt by way of strong winds and heavy rains. Forty-nine tornadoes were reported in the period 1953-1974 in a 10,000-square-mile area containing the site, giving a mean annual frequency of 2.2 and a computed recurrence interval for a tornado at the plant site of about 1590 years. There were 14 reports of hail, 20 millimeters (three-quarters of an inch) in diameter or greater, within the one degree latitude-longitude square during the period 1955-1967 and 22 reports of gusts with wind speeds of 25 meters per second (50 knots) or greater. During the period 1871-1977, 45 tropical depressions, storms, and hurricanes passed within 80 kilometers (50 miles) of the site. The maximum "fastest mile" wind speed of 27 meters per second (60 miles per hour) was recorded at Columbia, South Carolina, 42 kilometers (26 miles) southeast of the site in March, 1954. A heavy ice storm accumulating 13 millimeters (one-half inch) or more may be expected to occur in about one year out of five. From 1936 through 1970, there were 84 cases of air stagnation totalling about 340 days; eight of these cases persisted for seven days or more in the area in which the facility is located.

The design basis tornado of 160 meters per second (360 miles per hour) maximum wind speed is based upon a tangential velocity of 130 meters per second (290 miles per hour) and transverse velocity of 30 meters per second (70 miles per hour) and an associated pressure drop of three pounds per square inch and rate of pressure drop of two pounds per square inch per second. The acceptability of the applicant's design capability for tornadoes of this magnitude is discussed in Section 3.3.2 of this Safety Evaluation Report. Based on the maximum "fastest mile" wind speed (27 meters per second or 60 miles per hour) ever recorded at Columbia, South Carolina and the distance of the site inland from the Carolina coast, the operating basis wind speed, defined as the "fastest mile" wind speed at a height of nine meters (30 feet) above ground with a return period of 100 years, of 45 meters per second (101 miles per hour) selected by the applicant is considered to be sufficient.

2.3.2 Local Meteorology

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

Long-term meteorological records from Columbia, South Carolina show that the extreme maximum and minimum recorded temperatures are 42 degrees Celsius (107 degrees Fahrenheit) in June 1954 and -19 degrees Celsius (-2 degrees Fahrenheit) in February 1899, respectively. The maximum 24-hour precipitation amount of record at Columbia, South Carolina is 195 millimeters (7.66 inches) in August 1949. The maximum 24-hour snowfall total of 399 millimeters (15.7 inches) and the greatest monthly snowfall total of 406 millimeters (16.0 inches) occurred in February 1973. On an annual basis, thunderstorms may be expected to occur on approximately 55 days. Freezing precipitation (ice storms) may be expected to occur once per year. Heavy fog with a visibility distance of 0.4 kilometer (one-fourth mile) or less normally occurs on about 30 days annually. Wind data collected on site at the 10.5-meter level during the three-year period from 1975 to 1977 indicates that the predominant wind flow over the site is from the southwest. The mean wind speed at the site during this period was 2.9 meters per second (6.5 miles per hour) at the 10.5-meter level. Data collected at Columbia, South Carolina also show the predominance of southwest winds.

The 9.6 x 10^2 Pa (20 pounds per square foot) estimate, representing the weight of 1016 millimeters (40 inches) of snow, was used by the applicant as the design basis snow load on the roofs of safety-related structures. This design basis is acceptable to the NRC staff.

2.3.3 Onsite Meteorological Measurements Program

Meteorological data collection began on the site in June 1973 when the applicant initiated the measurements program on an instrumented tower approximately one-half kilometer west of the reactor building near the shore of Monticello Reservoir. Measurements of wind direction and speed were made at the 10.5-meter and 61-meter levels on this tower and vertical temperature differences were measured between the 10-meter and 61-meter levels and the 10-meter and 40-meter levels. Drybulb and dew point temperature measurements were made at the 10-meter level, and precipitation and solar radiation measurements were made at the 1.5-meter level near the tower. Additional meteorological measurements (wind direction, speed and dry bulb temperature) were made atop a 10-meter mast located across Monticello Reservoir from the primary meteorological tower.

The applicant has provided meteorological data collected on site during the three-year period from January 1975 through December 1977. The dispersion estimates used in diffusion evaluations were based on the joint frequency distributions of wind speed and direction measured at the 10.5-meter level and atmospheric stability (defined by the vertical temperature difference measured between the 10-meter and 61-meter levels on the primary tower) for the three-year period. The joint parameter data capture rate was 99.7 percent.

The preoperational meteorological measurements program meets the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs" regarding the accuracy specifications of each meteorological sensor and component in the data reduction system. We will require that the operational meteorological measurements program meet the upgraded meteorological criteria associated with emergency response plans and preparedness in accordance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants."

2.3.4 Short-Term Diffusion Estimates

Conservative assessments of atmospheric diffusion conditions for assessing postulated accidental releases of radioactivity from buildings and vents have been based on the meteorological data collected by the applicant for the period 1975-1977 and appropriate diffusion models. In the evaluation of accidental releases from the facility buildings and vents, a ground-level release considering a building wake factor, cA, of 870 square meters was assumed. The relative concentration estimates at the exclusion area boundary and low population zone distances for the various time periods following a postulated accidental release were calculated using the diffusion models described in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and Section 2.3.4 of the Standard Review Plan.

The relative concentration estimate for the 0-2 hour time period, which is exceeded no more than five percent of the time is estimated to be 3.3×10^{4} seconds per cubic meter at the exclusion distance of 1.6 kilometers. This relative concentration is equivalent to dispersion conditions produced by Pasquill type F stability with a wind speed of 0.7 meter per second.

The relative concentration estimated at the low population zone distance (4.8 kilometers) for the various time periods following a postulated accidental release are:

Time period	Relative concentration, seconds per cubic meter
0-8 hours	4.1×10^{-5}
8-24 hours	2.6×10^{-5}
1-4 days	1.0×10^{-5}
4-30 days	2.6×10^{-5}

2.3.5 Long-Term Diffusion Estimates

Estimates of average relative concentration and deposition, used in evaluating the potential effects of routine releases of radioactivity, were based on the meteorological data collected by the applicant for the period 1975-1977 and the straight line diffusion model described in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors." Relative concentration and relative deposition were evaluated at various points of interest for no decay and no depletion as well as with decay and depletion via deposition. The highest undecayed, undepleted values of relative concentration, as well as relative deposition for various points of interest, are given in Table 2-2 of this Safety Evaluation Report.

TABLE 2-2

HIGHEST ANNUAL AVERAGE RELATIVE CONCENTRATION AND RELATIVE DEPOSITIONS

Location	Direction	Distance, kilometers	Relative concentration, seconds per cubic meter	Relative deposition, per square meter
Site boundary	North-northeast	1.6	4.4×10^{-6}	1.8×10^{-8}
Residence	East-southeast	1.8	2.7×10^{-6}	6.9×10^{-9}
Vegetable garden	East	1.9	2.6×10^{-6}	8.2×10^{-9}
Milk cow	North-northeast	7.2	2.1×10^{-7}	4.9×10^{-10}
Meat animal	Southeast	3.5	7.6×10^{-7}	1.5×10^{-9}

2.3.6 Conclusions

The applicant has provided acceptable information concerning meteorological conditions which are of importance to the safe design and siting of the facility. The applicant's onsite meteorological program conforms to the recommendations of Regulatory Guide 1.23 and has produced data which adequately describe site atmospheric dispersion conditions and which we used to make both conservative and realistic estimates of atmospheric dispersion characteristics for accidental and routine gaseous releases, respectively, from the facility.

2.4 Hydrologic Engineering

2.4.1 Hydrologic Description

The facility is located approximately one mile east of the Broad River and three miles north-northeast of Parr Dam. The facility is situated on a hilltop, and plant grade is 435 feet above mean sea level, or about 180 feet above the Broad River floodplain. The facility is adjacent to Monticello Reservoir which provides condenser and emergency coolant water for the facility as well as serving as the upper-level reservoir for the Fairfield Pumped Storage Facility.

The Broad River, the principal hydrologic feature in the vicinity, drains an area of 4550 square miles above the facility. This river basin lies between two southeast-northwest trending ridges stretching from Columbia, South Carolina, to the headwaters in North Carolina about 100 miles northwest of Columbia. The average annual runoff is about 4.1 million acre-feet. Many streams and creeks carry runoff and groundwater drainage into this water course. The important rivers draining into the Broad River are not generally attractive for recreational use and there is no commercial navigation. At Columbia, approximately 28 miles downstream from the facility, the water is a source for municipal and industrial supply. Monticello Reservoir has been formed in the

Frees Creek Valley and receives water from Parr Reservoir through the Fairfield Pumped Storage Facility. Monticello Reservoir has a surface area of about 6800 acres and extends north of the facility for about seven miles. The average depth is 57 feet in Monticello Reservoir, and the deepest parts are about 100 feet deep. During planned operations, the normal drawdown in the impoundment will be about four feet; this represents a change in the reservoir's volume of about 28,000 acre-feet. The design elevation of Monticello Reservoir, 425 feet above mean sea level, will be reached each day by pumping water from Parr Reservoir.

The main dam at the Fairfield Pumped Storage Facility is approximately 180 feet high and 5000 feet long at the crest (see Figure 2-2 of this Safety Evaluation Report). All dams have crest elevations of 434 feet above mean sea level and are of earth fill type. The dams are protected from the forces of storm waves with riprap on the critical faces. In the main dam, a concrete intake channel 400 feet wide connects four 26-foot-diameter, 800-foot-long penstocks to the lower pump storage generating station. No emergency spillways are provided since the structures and reservoir storage are considered adequate to safely contain and eventually pass severe floods originating from the Frees Creek drainage basin. The tailwater of the Fairfield Pumped Storage Facility is at elevation 266 feet above mean sea level, the pool elevation of Parr Reservoir. Columbia Dam is approximately 28 miles downstream from the site on the Broad River. It forms a small reservoir with a surface of about 265 acres.

There are two small impoundments within Monticello Reservoir. The first is a small recreational impoundment in the northern portion that is physically isolated and not subject to water level changes from operation of the pumped storage facility. The second is the service water pond, which is protected by seismic Category I dams and is part of the ultimate heat sink system for the facility.

2.4.2 Flood Potential

Broad River

It has been estimated that the largest recorded flood on the Broad River in the vicinity of the facility resulted in a peak discharge of 228,000 cubic feet per second. This flood occurred on October 3, 1929. An even larger flood occurred in 1916, but no flow rate estimates are available.

The applicant has estimated a probable maximum flood of 960,000 cubic feet per second for the Broad River at Parr Dam. This would result in a maximum flood elevation of 390.5 feet above mean sea level or about 145 feet below plant grade. We consider that the occurrence of such a flood on the Broad River would have no effect on the safety of the facility.

Monticello Reservoir

The applicant analyzed the potential for flooding of the site due to precipitation on the drainage basin of Monticello Reservoir with wave runup and setup on the Monticello Reservoir. The applicant conservatively assumed that there were no releases from the reservoir during a 48-hour probable maximum precipitation and that the reservoir was initially at maximum normal pool elevation of 425 feet above mean sea level. The increase in stillwater level due to the 48-hour probable maximum precipitation was estimated to be 4.1 feet, bringing the level of the reservoir to 429.1 feet above mean sea level. We concur in this estimate. The effects of coincident wave runup and setup were estimated by the applicant and added to the maximum stillwater level. The applicant used an effective fetch of three miles and an ove[illegible]ad wind speed of 50 miles per hour. The applicant estimated that a [illegible] wave runup plus setup would add 7.5 feet to the stillwater level, the maximum level in front of the facility would then be 436.6 feet above mean sea level. The shoreline of Monticello Reservoir in front of the facility, as well as the emergency cooling pond, are protected by a riprapped berm to an elevation of 438 feet above mean sea level.

We have independently analyzed the wave runup and setup coincident with the probable maximum flood in Monticello Reservoir using more recent guides than used by the applicant. The guides we used were U.S. Army Corps of Engineers, ETL 1110-2-227, 1976, and Draft Guide CETA 79, "Wave Runup on Rough Slopes," U.S. Army Corps of Engineers Coastal Engineering Research Center, 1979. We also assumed the same three-mile fetch and a wind speed of 50 miles per hour. We used the 50-mile-per-hour windspeed because it was found to agree closely with the suggested fastest mile values in ANSI-N170-1976, "Standards for Determining Design Basis Flooding at Power Reactor Sites," for coincident wind speeds. We conservatively estimated a wave runup plus setup of about 8.5 feet.

When this is combined with the stillwater level resulting from the probable maximum flood, the total water level is predicted to be 437.6 feet above mean sea level. Since riprapped protection extends to 438 feet above mean sea level, we conclude that the facility is adequately protected from the combined effects of the probable maximum flood wave setup and runup in Monticello Reservoir.

The applicant estimated that the wave runup plus setup from the design basis windstorm would be approximately 12.5 feet. This combined with a maximum normal full pool elevation of 425 feet above mean sea level yields a level of 437.5 feet above mean sea level which is also below the level of protection. We independently and conservatively estimated the runup plus setup to be roughly 12.8 feet for a combined elevation of 437.8 feet above mean sea level. Therefore we conclude that the facility is protected from the effects of wind setup and wave runup from the design basis windstorm in Monticello Reservoir.

Effects of Severe Local Precipitation

The storm drainage system at the facility is conservatively designed to pass seven inches of rainfall per hour with some pounding in the catch basins. A maximum water level at the site was conservatively predicted by the applicant to be 435.4 feet above mean sea level. All structures on the site are protected against a water level of 436 feet above mean sea level. At our request the applicant reanalyzed the effects of extreme local precipitation postulating a total impairment of the subsurface drainage system. The maximum water level is predicted to be 436.15 feet above mean sea level in the immediate vicinity of the facility's structures, and not higher than 435.5 feet above mean sea level on the rest of the site. However, safety-related equipment in the facility is protected to a level of 436.5 feet above mean sea level.

Roof drains are designed to accommodate a maximum rainfall of six inches per hour. Scuppers are provided at various locations and they limit ponding anywhere on the roofs to four inches. The safety-related structures are designed to withstand this loading on the roofs.

We conclude that the severe local precipitation used for evaluation of the facility is conservative and that the effects of this precipitation have been accounted for in the facility design.

Conclusion

We conclude that the design and design bases for the site meet the recommendations of Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants" Revision 2, and 1.102, "Flood Protection of Nuclear Power Plants" Revision 1.

2.4.3 Water Supply

Cooling water for the circulating water system is supplied from Monticello Reservoir. The applicant estimates that heat input from the facility increases the evaporation rate of the reservoir by about 15 cubic feet per second. The source of makeup water to the reservoir are the Broad River basin, runoff from the Frees Creek Basin, as well as direct precipitation onto the reservoir. We conclude that there is an adequate source of cooling water for normal operation.

Service water is supplied from an approximately 44 acre impoundment within Monticello Reservoir formed by seismic Category I dams with riprap protected embankments (see Figure 2-2 of this Safety Evaluation Report). This service water pond is connected to the main reservoir by a pipe. Loss of Monticello Reservoir would not lower the water level in the service water pond to less than 415 feet above mean sea level. The applicant evaluated the performance of the service water pond to supply water to the facility for emergency shutdown. The integrity of the service water pond dams from the effects of wind-generated waves was evaluated and found to be acceptable. Based on a detailed review of the applicant's analysis and independent calculations, we conclude that the design bases for emergency water supply meet the criteria recommended in Regulatory Guide 1.27, "Ultimate Heat Sinks for Nuclear Power Plants," Revision 2.

2.4.4 Groundwater

Groundwater in the region occurs in two types of formations: (1) jointed and fractured crystalline bedrock and (2) the lower zones in the residual soil overburden. Recharge to these formations is by infiltration of precipitation falling in the upland areas. Some of the water infiltrating the surface soils evaporates, transpires from plants, or reemerges at the surface at short distances downslope from the point of infiltration. A small portion of the water percolates to perched water zones in the lower soil and into the water table in the underlying jointed bedrock.

In general, the groundwater table follows the land surface but with more subdued relief. Groundwater discharges as visible seeps and springs and/or precolates through the ground into creeks and steams. Some groundwater is discharged via wells, but the amount pumped is very small because the formations are generally not permeable enough to sustain well yields greater than five to 10 gallons per minute.

Preconstruction groundwater levels varied between elevation 350 feet and 420 feet above mean sea level in the jointed bedrock under the site. The filling of Monticello Reservoir has raised the water table in the area. Operation of the facility does not require groundwater and will not adversely affect local use of groundwater.

The design basis groundwater level for hydrostatic loading was taken as 420 feet above mean sea level.

The applicant considered the potential for radioactive contamination of water supplied from a failure of a waste evaporator waste concentrates holdup tank, which immediately releases its content to the groundwater and subsequently to surface water. The failure of this tank was found to produce the releases of the highest quantity of activity to the environment (see Section 15.4 of this Safety Evaluation Report). No groundwater users exist downgradient of the facility. A highly conservative analysis of the rupture of the waste holdup tank predicted that concentrations in the Broad River would be well below maximum permissible concentrations, as specified by Table II of Appendix B to 10 CFR Part 20.

We independently and conservatively estimated the transport of 5,000 gallons of liquid radwaste instantaneously spilled into the groundwater. Using very conservative values of groundwater and surface water transport parameters, and neglecting any ion exchange, we have calculated a minimum travel time of 11.1 years for radioactivity to reach the nearest user of Broad River water with an attendant dilution of about 1,900,000. All concentrations were well below the limits specified in Table II of Appendix B to 10 CFR Part 20. Even smaller concentrations would be expected if the effects of ion exchange were considered.

2.4.5 Conclusions

Based on our independent review and analysis, we conclude that adequate design bases for flooding have been provided, and adequate water supply can be assured for safety-related purposes, and postulated accidental spills of radioactive liquid will not exceed established criteria.

2.5 Geology and Seismology

2.5.1 Basic Geologic and Seismic Information

Introduction

The geological and seismological details of the site, as presented in the Preliminary Safety Analysis Report were reviewed by the NRC staff and its advisors, the U.S. Geological Survey (USGS) and the National Oceanic and Atmospheric Administration (NOAA), the seismology division of which is now a part of the USGS. We and the USGS concluded in the Safety Evaluation Report for the construction permit review that there are no known faults in the area that might be expected to localize seismicity in the immediate vicinity. We and NOAA concluded that the earthquake design bases for the safe shutdown earthquake and operating basis earthquake were adequately conservative. We reconfirmed these conclusions after the construction permit had been issued when faults were discovered in the excavation. The applicant evaluated these faults in "Report of Supplemental Geologic Investigations." As a result of the staff's review of that document and independent work by our consultants, Dr. R. Hay and Dr. G. Curtis of the University of California, Berkeley, we concluded that the faults were no younger than 45 million years.

For operating license safety evaluation reports we review all new information gathered since the construction permit review relating to the regional and site geology and seismology and reported in the Final Safety Analysis Report. With this new information, the staff has identified the following as the main issues for assessment:

1. The occurrence of extensive microearthquake activity associated with the Monticello reservoir impoundment and its significance to the design bases earthquakes.

2. Recent hypotheses and geologic and seismologic findings relating to the cause of seismicity in the Charleston, S.C. area including a newly discovered low-dipping detachment zone in the Southern Appalachians.

3. The possible projection of the newly mapped Wateree Creek fault into the area of the reservoir-induced earthquakes.

Our advisor, the Los Alamos Scientific Laboratories (LASL), has completed its review of reservoir-induced seismicity at the site. Its report is included as Appendix D to this Safety Evaluation Report. The U.S. Geological Survey has assessed the regional impact of recent studies in the epicentral region of the 1886 Charleston Earthquake. Its report is included as Appendix E to this Safety Evaluation Report.

After careful consideration and review of the new information, as provided and evaluated by the applicant, and the reviews of our advisors, we conclude:

1. There is no reason to alter our conclusions in the Safety Evaluation Report for the construction permit stage that a safe shutdown earthquake of 0.15g horizontal ground acceleration for structures founded on rock and 0.25g for soil foundations and an operating basis earthquake of 0.10g for rock and 0.15g for soil foundations are acceptable. A magnitude (M_L) 4.5 earthquake is an adequately conservative representation of the maximum reservoir-induced event at Monticello. The occurrence of such an event near the site will produce low energy, short duration, high frequency ground motions that may exceed the safe shutdown earthquake at high frequencies. The evaluation of this high frequency, short duration ground motion will be contained in Section 3.7.1 of a supplement to this Safety Evaluation Report.

2. There will be a need to continue monitoring the microseismic activity.

3. We should, along with the applicant, monitor ongoing mapping activities funded by the USGS related to the Wateree Creek fault and its possible extension into the site area to continue to assess any potential significance to the site.

4. The applicant is in conformance with applicable portions of Standard Review Plan (NUREG 75/087) - Sections 2.5.1, 2.5.2 and 2.5.3, Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2, and Appendix A to 10 CFR Part 100 "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

In the following sections we present a brief background of our Summer review, a synopsis of the geology and seismology of the site, and the staff's assessment of the three issues.

Background

At the time of the Safety Evaluation Report for the construction permit, issued August 29, 1972 (AEC, 1972), the staff agreed with the applicant in its evaluation of the regional and site geology, and the design earthquakes.

The USGS and NOAA, as our advisors, concurred in these conclusions. The USGS further suggested that the principal seismic hazard was the proximity to the Charleston Seismic Zone which, because of the continuing earthquake activity, requires conservatism in the choice of the safe shutdown earthquake and operating basis earthquake.

On November 26, 1973, the Applicant reported that several faults were found during the excavation of the reactor complex. Following mapping and a detailed investigation, a "Supplemental Geological Investigations Report" was submitted to the NRC on January 14, 1974. The report concluded that the faults and shear zones were not capable within the meaning of Appendix A to 10 CFR Part 100. This was based on detailed microscope study and isotopic dating of undisturbed post-faulting minerals present in some of the filled shear zones. The faults were shown to have been inactive for at least the last 45 million years and most likely for 150-300 million years.

The staff's evaluation of the Supplemental Report (February 12, 1974, memorandum from H. Denton to R. DeYoung) concurred with the Applicant's conclusions concerning the non-capable nature of the faults and shear zones at the site. However, the presence of through-going structures, which were not recognized prior to the issuance of the Safety Evaluation Report for the construction permit led to the staff's concern that while there was little likelihood that the proposed reservoir would induce renewed movement along some of these or related structures in and around the reservoir, they could localize reservoir-induced seismicity. The staff, therefore, required the applicant to monitor possible microseismic activity in the vicinity of the reservoir before, during and after impoundment.

On October 29, 1975, the applicant notified the NRC of new faults in the excavation of the dam for the emergency cooling pond. This was followed by a detailed study of the new faults and a report submitted on December 10, 1975, as Addendum I of the Supplemental Geologic Investigation Report. These faults were shown to be of the same set and age as the faults previously found and reported on in January, 1974 and therefore not capable faults.

On December 10, 1976, the Final Safety Analysis Report was submitted by the applicant. On January 21, 1977, the staff completed its acceptance review of the Final Safety Analysis Report and recommended docketing of the application and the Final Safety Analysis Report, which was accomplished in February 1977.

Infilling of the reservoir began on December 3, 1977. Three weeks later the seismic network detected microseismic activity under and around the reservoir. Infilling was completed on February 8, 1978. Microseismic activity has continued since that time.

In response to questions submitted on June 20, 1980, by NRC and its advisor, Los Alamos Scientific Laboratories, on the reservoir-induced seismicity, the applicant submitted a Supplemental Seismologic Investigation Report in December 1980. An assessment of that report makes up a large part of this safety evaluation review.

In addition, the applicant presented to NRC reviewers a report of field mapping by Professor Donald Secor, working on a grant from the USGS, which included a newly mapped north-south fault, the Wateree Creek fault, located a few miles south of the site. A staff evaluation of the potential significance of this fault is included in this report.

Finally, continuing interest and controversy in the scientific community over the causes of the Charleston 1886 earthquake and subsequent seismic activity has led to several investigations and interpretations. These are reviewed to determine the impact these considerations may have on the site.

2.5.2 Geology

Regional Geology

The site is located in the Piedmont Physiographic Province about 26 miles northwest of Columbia, S.C. The Piedmont is one of several subdivisions of the Appalachian tectonic belt, which is now believed by most geologists to have evolved through the late Precambrian and Paleozoic Eras (800 million years before present (mybp) - 250 mybp) by stages of plate tectonic processes. Recent investigations by deep seismic reflection profiling (Cook et al, 1979; Harris and Bayer, 1979) indicate that the southeastern part of the Appalachian belt from the Blue Ridge eastward across the Piedmont is underlain by a low-dipping detachment zone or large-scale thrust fault. This implies that everything above this fault, including the Piedmont, is part of an allochthonous (transported) sheet 6 to 15 kilometers thick that has been moved more than 200 kilometers westward from its original position. The possible significance of this deep fault is discussed in Section 2.5.3.3 of this Safety Evaluation Report.

Site Geology

The Piedmont is subdivided into several northeast-trending belts of late Precambrian to Paleozoic rocks distinguished by characteristic lithic sequences, metamorphic grade and/or structural style. In South Carolina, proceeding northwestward, the Carolina Slate belt is succeeded by the Charlotte belt, the Kings Mountain Belt, the Inner Piedmont, the Brevard Zone, and the Blue Ridge.

The Charlotte belt, in which the site is located, is underlain primarily by complexly deformed Paleozoic (600 mybp - 250 mybp) metasedimentary rocks intruded by later Paleozoic (350 mybp - 250 mybp) silicic igneous rocks and Mesozoic (Jurassic?) age (190 mybp - 136 mybp) diabase dikes. These are all cut by two dominant sets of joints and shear structures, one trending northeast, the other trending slightly west of north.

Surface faults and shear zones discovered during construction in excavations of various parts of the nuclear facility were carefully mapped and studied; and it was shown that they could not have moved in the last 45 million years and that most likely they have been inactive for 150-300 million years.

Wateree Creek Fault

The applicant has reported in the Final Safety Analysis Report that, as part of the investigation to determine the geologic factors associated with the induced seismicity of the Monticello Reservoir, a newly-discovered fault has been mapped and reported in the Chapin Quadrangle, which is south of the Summer site (Secor, 1980). This fault, the Wateree Creek fault, trends slightly west of north along the length of the Chapin Quadrangle, dips steeply and offsets the Slate Belt strata which strike roughly east-west across the fault. It has been mapped up to the Broad River, at a distance of eight kilometers south of the plant site. The applicant states that:

1. The only evidence for the fault is the deflection of a lineation close to the fault.
2. It has not been shown that the fault offsets the contact between the Slate Belt and the Charlotte Belt to the north of it in which the site is located.
3. The closest approach of the structure is four to five miles south of the site.

The report by Secor (1980), however, states that, while the fault is not directly observable because of poor rock exposure, evidence for the fault includes rotation of the foliation in the vicinity of the fault trace, areas of silicified fault breccia along the trace, offset of the mapped strata contacts, apparent offset of a Mesozoic (Jurassic?) diabase dike and offset of the contact of the Slate Belt and Charlotte Belt at the north end of the Chapin Quadrangle. Concerning the age of the fault, Secor reports,

> "Our preliminary observations indicate that both silicified breccia zones of probable Mesozoic age and a Jurassic (?) diabase dike are offset by the Wateree Creek fault. The time of latest movement must therefore be more recent than the Jurassic. In a road cut it appears that a layer of surficial colluvium is not offset by the fault; however, the age of the colluvium is not known and so no definite upper limit on the time of latest movement has yet been determined."

The extension of this fault northward to the site area and the reservoir is considered possible by Secor. The applicant's consultants accompanied by Secor conducted a brief reconnaissance investigation to find the fault in the vicinity of the reservoir, but could not find it.

Despite the absence of direct evidence, the NRC staff considers that, based on the available evidence, it is reasonable to consider the extension of the Wateree Creek Fault northward into the Jenkinsville Quadrangle up to and adjacent to the Monticello Reservoir. The evidence includes:

1. A series of topographic linear features coincide with the projected trace of the Wateree Creek Fault the entire length of the Jenkinsville Quadrangle. The linear pattern is formed by elongate, narrow channels of small streams aligned along the projected trace, suggesting a continuous zone of weak, easily-eroded rock which commonly controls the location and courses of streams (Figure 2.5-11 of the Final Safety Analysis Report). The orientation of this postulated fault projection parallels one of the two dominant joint and shear trends in the area.

2. A prominent linear trend of offset aeromagnetic anomaly contours coincides with the projected trace of the fault (Supplemental Seismological Investigation, Appendix V, Figure 3).

3. There is an apparent offset of the Charlotte Belt gneiss/migmatite contact and of the granodiorite/migmatite contact in a right lateral sense along the projected trace of this fault in the site area.

There is no evidence that indicates that the Wateree Creek fault is a capable fault within the meaning of Appendix A to 10 CFR Part 100. Investigations will be continued in the area of the Wateree Creek fault in the Jenkinsville Quadrangle (quadrangle in which the site is located) for the next several years by Secor. Although we don't consider this fault to be a hazard to the site, the staff considers it prudent to remain cognizant of these ongoing investigations. Thus, we have requested the Applicant to stay abreast of this mapping and report to the NRC any findings. A discussion of the possible relationship between this fault and the reservoir-induced seismicity is included in Section 2.5.3 of this Safety Evaluation Report.

2.5.3 Seismology

Vibratory Ground Motion

The safe shutdown earthquake is based on the occurrence near the site of the largest historic earthquake in the southern Piedmont with a resulting site intensity of VII (Modified Mercalli). The applicant assumed the maximum horizontal ground acceleration for the safe shutdown earthquake is 0.15g for rock foundations and 0.25g for soil foundations. The staff agrees that the safe shutdown earthquake as proposed by the applicant is adequately conservative. The applicant's design response spectra differ from the Regulatory Guide 1.60 response spectra. However, the differences between the applicant's modified Newmark and Blume response spectra and the Regulatory Guide 1.60 spectra anchored to 0.15g are small in the frequency range of interest to Seismic Category I structures (two to nine Hertz).

We have reviewed the applicant's evaluation of the ground motion effects of Monticello reservoir-induced seismic events at the site. This evaluation and a comparison of the maximum reservoir-induced earthquake and the safe shutdown earthquake is discussed in Section 2.5.3 and Appendix D to this Safet Evaluation Report. The staff finds that the largest reservoir-induced event which is likely to occur is magnitude (M_L) 4.5. The applicant's proposed ground motion for this event exceeds the safe shutdown earthquake spectrum at high frequencies. The effect of this high frequency energy on the seismic Category I structures will be discussed in Section 3.7.1 of a supplement to this SER.

The operating basis earthquake is based on the recurrence of the 1886 Charleston earthquake (maximum Modified Mercalli intensity X; Bollinger, 1977) in the vicinity of Charleston with a resulting site intensity of VII. For the operating basis earthquake the maximum horizontal ground accelerations used are 0.10g for rock and 0.15g for soil foundations. Although the site intensity is the same for both the safe shutdown earthquake and operating basis earthquake, the lower acceleration values for the operating basis earthquake result from the fact that the earthquake source is located 125 miles from the site. For earthquake sources at large distances, the high frequency energy is attenuated more than low frequency energy as the energy travels between the source and the recording site. Since ground motion acceleration is a relatively high frequency phenomenon, for the same intensity values the ground motion for sources at large distances will have lower accelerations than for sources near the observation point.

During the construction permit review, the staff and its advisors, USGS and NOAA, considered the possibility that earthquakes in the Charleston area might lie on a northwest trending line that could extend near the Summer site. The consultants to the Advisory Committee on Reactor Safeguards (ACRS) suggested that the 1886 Charleston earthquake could occur near the Summer site (Summary Report, 149th ACRS Meeting, September 14-16, 1972). It was the staff position, based on presentations by the USGS and NOAA, that there was sufficient evidence from the spatial distribution of earthquakes and the basement structures for keeping the 1886 Charleston earthquake near Charleston for the purpose of seismic design evaluation. The ACRS agreed with the staff's position, but suggested the need for further seismic research in the Charleston area (Summary Report, 151st ACRS Meeting, November 9-11, 1972).

Most of the earthquakes which have occurred in the Coastal Plain province are centered near Charleston, South Carolina. The recurrence of the largest Charleston earthquake (maximum intensity X) is significant to the determination of the operating basis earthquake for the site as previously discussed. Since 1974 the USGS has conducted extensive geological and geophysical studies in the Charleston area, including seismic monitoring. The results to date have not changed the NRC staff position reached during the construction permit review that there is sufficient evidence, including the distribution of earthquakes and the existence of basement structures, to localize the higher intensity earthquakes near Charleston, as discussed later in this section and in Appendix E to this Safety Evaluation Report.

Reservoir-Induced Seismicity

The site is adjacent to the Monticello Reservoir, which was created as part of a planned electric power generating complex. The Monticello Reservoir stores water for a pumped storage facility, provides cooling water for the nuclear plant, and serves as a makeup source for emergency cooling water.

In its evaluation of the applicant's investigation of the faults discovered in the excavation area at the site, the staff concluded that the impoundment of Monticello Reservoir would not adversely affect the faults exposed in the excavation. However, to confirm the absence of any effects from Lake Monticello on local seismic activity, the staff required microearthquake monitoring during a period extending from six months before to one year after filling of the reservoir.

Prior to filling of the reservoir, the USGS seismograph station at Jenkinsville (three miles east-southeast of the site) had recorded about one local event every six days from 1973 to 1977. In December 1977 a four-station seismic network was installed in the area of Monticello Reservoir by the applicant under the guidance of Dr. Bollinger, Virginia Polytechnic Institute and State University. Teledyne Corporation was contracted to analyze the data. Quarterly reports of seismic activity are submitted to the NRC. Filling of the reservoir began on December 3, 1977 and full pond elevation occurred on February 8, 1978. A strong motion accelerometer was installed by the USGS in February, 1978 on an abutment of Fairfield Dam. In May 1978 the USGS began a six-station seismic monitoring network in the area. In January 1979 Dr. Pradeep Talwani, University of South Carolina, took over the contract to analyze the applicant's seismic data. Dr. Talwani is also contracted by the USGS to analyze seismic data from their network near Monticello Reservoir.

Characteristics of Seismicity at Monticello Reservoir

An increase in seismicity near Monticello Reservoir began during the last week of December 1977 and is most likely related to the filling of the reservoir. Seismicity was observed as several clusters in the reservoir vicinity. The seismicity spread in subsequent months, with most of the spreading occurring during approximately six months following impoundment and over 90 percent during the first year. Since October 1979 there has been no further apparent growth in epicentral area. The applicant has defined five distinct clusters of seismicity (Figure 10, Appendix II, Supplemental Seismologic Investigation Report, South Carolina Electric & Gas Company, 1980, hereinafter referred to as December 1980 Report). The peak activity occurred in February and March 1978, after the completion of filling. In general, seismic activity has decreased since March 1978, interrupted by three swarm episodes during August-December 1978, October 1979 and July-August 1980. To date, the maximum magnitude earthquake associated with the filling of the reservoir was the August 27, 1978 magnitude 2.8 event.

The seismicity extends from 0 to four kilometers in depth. Almost all the events are shallower than two kilometers in depth and most are shallower than one kilometer. There has been no marked increase of focal depth with time. Although the applicant's four-station network gives adequate epicentral locations, the depth estimates are unreliable. Both the applicant's and USGS's data are needed for more reliable depth measurements.

Focal mechanism solutions for the induced earthquakes indicate thrust-type movement as the predominant mechanism. Some events, especially the deeper events (one to two kilometers), also exhibit a component of strike-slip motion. There are two predominant orientations of the nodal planes - north-south and northwest-southeast.

The USGS has drilled two wells, one with a depth of 1100 meters and the other 1203 meters, west and southwest of the reservoir in the epicentral areas of greatest activity. In situ stress measurements in one of the USGS deep wells show that at shallow depth the greatest horizontal principal stress is substantially greater than the vertical stress, suggesting stress conditions that may lead to thrust faulting in approximately the upper 300 meters. The stress measurements show that it is possible that at shallow depths stresses are sufficiently close to failure to cause generation of microearthquakes along preexisting planes of weakness. Therefore, pore pressure increased to hydrostatic levels has probably induced the observed seismicity at Monticello Reservoir.

Maximum Earthquake Associated with Reservoir Impoundment at Monticello Reservoir

The strength of the maximum earthquake associated with the Monticello Reservoir can be defined as either a maximum intensity or a magnitude. Magnitude is a measure of earthquake source size using instrumental recordings of ground motion. Magnitude is a better indicator of earthquake source strength than intensity, which is a measure of observed damage and felt effects. Intensity depends upon the size of the earthquake, its depth, the distance from the earthquake source, the nature of the geologic materials between the source and the point of observation, the geologic conditions at the point of observation itself and differences in structural design. Where there are no instrumental recordings, however, the only source of information on earthquake size is intensity data. For reservoirs in the Piedmont, the largest events associated with reservoir impoundment are maximum Modified Mercalli intensity VI. For a few recent earthquakes at Piedmont reservoirs, the magnitude was also determined. Unfortunately, different magnitude scales, which measure different phases in different frequency ranges, were used. One of the magnitude scales is local magnitude (M_L), which is determined at most eastern U.S. stations as some function of signal duration and epicentral distance. The magnitude scales m_b and m_b(Lg) are determined from amplitudes of P-waves and Lg-waves respectively. The relationship among these magnitude scales has not been determined for the Piedmont.

A number of reservoirs in the Piedmont are believed to have induced earthquakes. At Clark Hill Reservoir on the Georgia-South Carolina border the largest event was m_b (NOAA) 4.3, which occurred on August 2, 1974 (Appendix A, p. 4 and 6). This is a weak case of reservoir-induced seismicity since the event occurred 22 years after impoundment and historic earthquakes have been located near the dam site (Packer et al, 1979). In addition, at Lake Jocassee in northwest South Carolina the largest event was M_L or m_b (Lg) 3.7 which occurred on August 25, 1979, six years after the reservoir reached full pool elevation. Also, at Lake Keowee near Lake Jocassee the largest event was the M_L or m_b(Lg) 3.8 Seneca earthquake on July 13, 1971, three months after Lake Keowee had reached full pool elevation. At Monticello Reservoir, the largest earthquake was the M_L 2.8 event on August 27, 1978, 19 months after Monticello Reservoir reached full pool elevation.

The applicant has proposed an upper bound of $M_L = 4.0$ based on the following two arguments. The first argument is that the effects of the reservoir reflect minor local adjustments and at Monticello only small fault areas (about one kilometer or less) can experience movements in any one earthquake. In addition, it is assumed that the average stress drop for several of the induced earthquakes is a few bars; the maximum observed is 17 bars (Fletcher, 1980); and a maximum assumed stress drop for the induced earthquakes is 25 bars. Using Brune's (1970) model and the relationship between seismic moment and local magnitude developed by Thatcher and Hanks (1973), the applicant calculated an upper bound magnitude of (M_L) 4.0.

In their second argument, the applicant indicates that the largest earthquakes considered to be induced by reservoir impoundment in the Piedmont were maximum Modified Mercalli intensity VI. The applicant derived seismic moment estimates based on the area of Modified Mercalli intensity VI shaking and then obtained a local magnitude from the seismic moment. The earthquake magnitude was calculated to be less than or equal to (M_L) 4.0.

As a basis for concluding that the reservoir-induced earthquakes are local adjustments, the applicant has relied upon the geologic and seismic conditions near the site. The applicant finds that the clusters of seismicity are spatially associated with the boundaries of small shallow bodies (plutons) and the effect of the reservoir impoundment has been to relieve local near-surface remanent stress around the plutons. Also, the applicant finds that the in-situ stress measurements in the two USGS wells indicate variable stress levels vertically and laterally and suggest a stress barrier which will tend to limit the vertical extent of the induced seismicity.

In addition, focal mechanisms of earthquakes have nodal plane orientations generally corresponding to the orientation of fractures observed in the two USGS wells, suggesting that seismicity is occurring along a network of pre-existing fractures. These fractures are not continuous in their spatial extent either laterally or vertically. Well water level and resistivity data indicate significant variations in permeability both laterally and with depth beneath the reservoir area. The limited spacial extent and the overall decline in the rate of seismic activity suggest that the stored strain is being relieved rather than replenished.

Our consultant at LASL, Dr. Carl Newton, estimated the maximum reservoir-induced earthquake at Monticello to be $M_L = 4.5$. Dr. Newton indicates that the reservoir-induced seismicity is possibly associated with small-scale anomalous features, but the evidence is too weak to draw a definite conclusion (Appendix A).

There is no evidence of faults capable of earthquakes of magnitudes greater than (M_L) 4.0. In cases where no fault has been identified, reservoirs like Monticello have not been associated with macroseismicity. Dr. Newton determined that the largest historical earthquake in the Piedmont south of Chesapeake Bay has been $m_b = 4.5$ with an uncertainty of up to a half magnitude. In addition, extrapolation of the magnitude-frequency relationship for Monticello earthquakes results in an estimate of $M_L = 4.45$ as the largest event to occur every 50 years at Monticello.

Dr. Andrew Murphy of the NRC Office of Nuclear Regulatory Research has assisted in the review and indicates that the maximum reservoir-induced earthquake of $M_L = 4.0$ as proposed by the applicant and $M_L = 4.5$ as recommended by LASL may not be sufficiently supported by the arguments currently presented. He indicates that at Monticello Reservoir the maximum induced earthquake should be

M_L = 5.3 until further supporting information is provided. Dr. Murphy's primary concern is that the applicant has not provided sufficient data to establish that the maximum dimension of geological structures within the immediate vicinity of this reservoir is one kilometer or less and that the maximum stress drop is less than 25 bars. In addition, the validity of the applicant's method for relating the area of Modified Mercalli intensity shaking and local magnitude has not been established for the southeastern U.S.

Although the applicant has attempted to show that the size of the area available for rupture can be limited to one kilometer or less, Dr. Murphy is not satisfied with the strength of this argument since sufficient weight may not have been given to the observation that the clusters of seismicity as identified by the applicant are at least as large as three kilometers. The use of the 25-bar stress drop in the Brune model was justified on the basis of an abstract by Fletcher (1980) in which he reported a 17-bar stress drop for the August 27, 1978 earthquake that occurred at Monticello Reservoir. The conservatism gained by the applicant's use of 25 bars over the observed 17 bars may have been invalidated by new calculations of stress drop for the earthquake, which indicate the possibility of about 90 bars.

Although the seismicity may be spatially associated with the surficial boundaries of the plutons, there is no reason that all the seismicity is relieving local stress around the plutons because several clusters of seismicity and the focal mechanisms generally agree with the orientation of a projection of the Wateree Creek fault. Also, the applicant's suggested stress barrier might be better considered a boundary between two stress regimes rather than an impenetrable barrier.

Dr. Murphy further indicates that at this time there is no way of knowing how the level of seismicity is going to vary over the expected life of the facility. If 3.2 kilometers (length of the clusters of seismic activity) is taken as the source dimension and 100 bars as the stress drop, by Brune's model (1970), a magnitude (M_L) 5.3 event is possible in the immediate vicinity of the reservoir.

In summary, Dr. Murphy's recommendation that an event of magnitude 5 to 5¼ occurring in the near-field should be used for the safe shutdown earthquake is based more on inferred flaws in the applicant's arguments rather than on an independent analysis.

We have evaluated the range of values M_L = 4.0 to 5.3 provided by the applicant, LASL and Dr. Andrew Murphy. Although we find the applicant's arguments reasonable, the staff agrees with our consultant (LASL) and has chosen a magnitude (M_L) 4.5 as the largest reservoir-induced event which is likely to occur. We find that the earthquakes may be a result of minor local adjustment; however, all the seismicity may not be associated with the plutons and may have a possible association with a projection of the Wateree Creek fault. We will continue to evaluate the stress drop calculations for the events at Monticello Reservoir, especially those presented by Fletcher (1980), but we find at this time that 25 bars is a conservative estimate for events in the Piedmont because of the preponderance of lower stress drop values for events in the eastern U.S.

We also observe that the largest reservoir-induced earthquakes generally occur up to 10 years after impoundment (Packer et al, 1979). For example, at Lake Jocassee, another Piedmont reservoir, the maximum event to date (magnitude 3.7) occurred about six years after the water level approached full pond. We note that there are no significant geologic differences among Piedmont reservoirs; therefore, there is no reason to assume that the largest earthquake induced by Monticello Reservoir has yet occurred.

After a period of rapid epicentral region growth associated with the initial impoundment, further growth has been extremely limited. Continued spreading would indicate an increase in the maximum earthquake potential; however, since growth has been limited, no substantial increase in maximum earthquake potential is anticipated. The limited spatial extent of the reservoir-induced seismicity suggests that the induced seismicity is a minor local adjustment and the reservoir would not induce a large tectonic earthquake. Based on available evidence, there is no reason to expect that this local perturbation would localize an earthquake equivalent to the maximum historical earthquake at shallow depths beneath the reservoir.

Finally, we observe that world-wide data show that reservoir-induced earthquakes of magnitude greater than about five have occurred in active tectonic environments along faults which exhibited late Cenozoic displacement (Packer et al, 1979); the absence of such faults near Monticello indicates that the size of the largest event expected to occur should be limited.

Possible Association of the Wateree Creek Fault with the Reservoir-Induced Seismicity

There is some indication that the Wateree Creek fault, which the evidence suggests may be present adjacent to the reservoir, may also be, in part, responsible for localizing some of the reservoir-induced microearthquakes. Three of the four major epicentral clusters reported by the applicant, clusters I, II, and IV (Final Safety Analsyis Report, Figure 3.6.14-1) are aligned along the projected trace of the postulated fault. While each cluster shows a non-linear distribution, these three clusters are bisected by the proposed fault trace, and the cluster groups are aligned along the trace.

The fault plane solutions (Final Safety Analsyis Report, Figure 361.14-2) for cluster I, for the granodiorite of cluster II, and the 1-1.5 kilometers depth solution for cluster IV are almost identical, showing thrust movement with a small component of strike-slip. The nodal planes are oriented in the same manner, with the intersections plunging in the same direction. The solutions for the other locations differ from these three in orientation. In addition, seismicity profiles (Technical Report No. 79-4, "Seismic Activity near the V. C. Summer Nuclear Station for the period October-December, 1979," Figures 3 and 5) show the focal depths for clusters II and IV following roughly on an eastward dipping plane which corresponds to the orientation of the Wateree Creek fault, while focal depths for events east of these clusters shown no pattern. There is also some suggestion that the deeper focal depths generally occur to the east of the fault trace, along the western side of the reservoir, while the shallow quakes occur close to the fault trace, suggesting an eastward dipping plane.

The applicant has not considered the possibility of the Wateree Creek fault extending to the reservoir because of lack of field evidence. However, as discussed earlier, we feel that it is reasonable to assume its existence based on available mapping evidence. We conclude that if the Wateree Creek fault extends west of the reservoir, fractures related to that fault zone may be serving to localize stress release. There is, however, no reason to expect an earthquake that would exceed the maximum credible reservoir-induced earthquake recommended for this site because all of the microearthquakes are occurring within or near the boundary of the reservoir, there is no indication that seismicity is propagating beyond this location in the directions of the projected fault trace, and there is no historic seismicity that can be related to the Wateree Creek fault. Focal mechanism solutions and calculated depths of most of the events indicate a shallow dipping plane, whereas the Wateree Creek fault is thought to be a steep angle fault (Secor, 1980). Also seismicity is occurring in other areas of the reservoir where the Wateree Creek fault is not present. Finally, there is no geological evidence for capability of the Wateree Creek fault. Unfaulted colluvium, of unknown age, lies across the fault in Chapin Quadrangle (Secor, 1980).

Ground Motion from the August 27, 1978 Earthquake

The largest earthquake to date at the Monticello Reservoir was the magnitude (M_L) 2.8 event on August 27, 1978. This earthquake occurred about a mile northwest of the plant and at a depth of 100 to 500 meters (Dames and Moore, 1980). The earthquake produced a strong motion record at the USGS strong motion accelerometer which is about 640 meters southeast of the epicenter and is on soil. This record is significant because the peak horizontal acceleration of 0.25g for the 180 degrees component exceeds the operating basis earthquake zero-period acceleration of 0.15g and is equal to the safe shutdown earthquake value of 0.25g for structures founded on soil. The applicant has compared the response spectra produced from this strong motion record and the operating basis earthquake spectra (Dames and Moore, 1979; Dames and Moore, 1980; Final Safety Analysis Report question response 361.13).

The duration of strong ground motion on the strong motion accelerometer record was short (about 0.5 second) and the duration of motion greater than 0.1g for the 180 degree component was about 0.06 second. Response spectra were computed from the 180 degree, 90 degree and vertical components of the strong motion record. These response spectra show the signal to be largely high frequency energy. The earthquake spectra fall below the operating basis earthquake spectra at frequencies below 10 Hertz, but some of the motion exceeds the operating basis earthquake spectra above 10 Hertz.

In the November 2, 1979 submittal (Dames and Moore, 1979), the applicant indicated that from an engineering point of view the response spectra from the August 27, 1978 event did not exceed the operating basis earthquake spectra. The applicant indicated that the structures would respond to motions of significant duration with several cycles of vibration approaching the peak value at each frequency, and not the very short, sharp impulsive motion such as that recorded during the August 27 event. In addition, the frequency of the impulsive motion of the strong motion accelerometer record (greater than 20 Hertz) is greater than the dominant structural frequency of the plant (at or less than 20 Hertz). The ground motion at the accelerometer also may be amplified by site topography. Finally, the applicant indicated that the finite size of large structures such as the plant foundation can attenuate high frequency motion.

In the May 6, 1980 submittal (Dames and Moore, 1980), the applicant presented a realistic and conservative analysis of the probable ground motion at the nuclear station due to the August 27, 1978 event. The applicant considered the following effects:

1. As the energy leaves the source, there is attenuation by geometric spreading, which results in reduction of ground motion. Since the plant site is further from the epicenter than the strong motion accelerometer, motion at the plant site would be less.

2. Ground motion expected at the plant site would be significantly attenuated relative to the strong motion accelerometer record due to material damping in soils and rocks.

3. The finite size of large structures would attenuate high frequencies.

4. The motion has probably been amplified in the 56 foot soil column at the strong motion accelerometer location. Most of the plant, however, has a bedrock foundation.

The applicant calculated a realistic estimate of the level of ground motion at the plant site considering the combined effects of the first three attenuation phenomena. The attenuated response spectra fall below the operating basis earthquake spectra, indicating that the operating basis earthquake was not exceeded at the plant site for the August 27, 1978 earthquake (Dames and Moore, 1980, Figures 3 through 8).

The staff considered the possibility that an earthquake similar to the August 27, 1978 event could occur under the rock foundations at the plant site. The staff asked the applicant (Final Safety Analysis Report Question 361.13) to assume that the earthquake occurred at the recording station and that the station was near the plant. The only attenuation phenomena to be considered was the soil condition at the strong motion accelerometer site. The applicant used deconvolution procedures to infer motion at bedrock below the strong motion accelerometer. The spectra of the deconvolved motion at bedrock are less than those recorded at the surface. This suggests that ground motion has probably been amplified in the soil column at the strong motion accelerometer location. The applicant thus concluded that, with the exception of only one minor excursion, the deconvolved motion spectra for the 180 degree and 90 degree components of the August 27, 1978 event are enveloped by the operating basis earthquake spectrum (Final Safety Analysis Report Figure 361.13-4).

We agree with the applicant that during the August 27, 1978 event the operating basis earthquake was not exceeded at the plant site. The expected ground motion at the plant site would have been attenuated due to geometric spreading and material damping in the rocks along the path between the earthquake source and the plant site. In addition, the applicant has demonstrated that the ground motion was amplified in the soil column at the strong motion accelerometer site. Such amplification would not occur at the plant site, where most of the foundations are bedrock. If this event had occurred under the plant site the lack of amplification due to the soil column would lead to smaller recorded ground motions. The effect of short duration high frequency motion on the plant site will be discussed in Section 3.7.1 of a supplement to this Safety Evaluation Report.

Ground Motion at the Site from the Maximum Reservoir-Induced Earthquake

The Safety Evaluation Report for the construction permit approved a safe shutdown earthquake defined by a modified Newmark and Blume response spectrum anchored at 0.15g for structures founded on rock. In this section we will evaluate that spectrum with respect to ground motion from the maximum reservoir-induced event defined as an earthquake of $M_L = 4.5$. This event is assumed to occur in the volume of seismicity currently defined for the Monticello Reservoir. For design purposes the event would be shallow in depth and near the site.

Determination of ground motion in the near field of earthquakes is a difficult and problematic task. Since the earthquake is assumed to occur in the volume of seismicity currently defined for the Monticello Reservoir and the plant lies in the zone of seismicity, estimates of ground motion from an event near the site can clearly be considered a "near field" problem.

The sources of uncertainty in near-field ground motion estimation are several. First, there has been a relative lack of data recorded close in (less than 10 kilometers) from earthquakes and especially for shallow depth events such as at Monticello. The vast majority of data was recorded at distances greater than 20 kilometers. Secondly, simple extrapolation of the data to close-in distances is not easily accomplished since ground motion further from the source is effected more by gross source strength, geometric spreading, and seismic wave attenuation, whereas ground motion near the source is more sensitive to source geometry and details such as localized stress conditions and direction of faulting. The interpretation of these near-field effects can lead to large differences in the near field.

Recently, a great deal of effort has been placed on theoretical models of earthquake sources and attempts have been made to theoretically predict ground motion at various distances. While these efforts are certainly encouraging they are controlled by assumptions about the physical nature of the earthquake source. Different assumptions such as the size of the stress drop and the effect of local inhomogeneities have a major impact upon ground motion, particularly at those frequencies (greater than two Hertz) of concern to nuclear power plants. As of this time, no consensus with sufficient detail exists within the seismological community that would allow the exclusive use of theoretical models in order to estimate ground motion in the near field.

Since there is so little near-field ground motion data, the applicant has presented theoretical calculations for $M_L = 4.0$ and 4.5 events. In the sections below we discuss the applicant's efforts at predicting ground motion from a theoretical approach and a comparison of their results with data from the August 27, 1978 earthquake. We find that a conservative representation of ground motion expected at the site from the occurrence of the maximum reservoir induced earthquake (that is a $M_L = 4.5$ event close to the site) exceeds the safe shutdown earthquake at high frequencies (greater than about 10 Hertz). We also find the applicant's approach to be conservative.

In Final Safety Analysis Report Question 361.17, the staff asked the applicant to determine peak accelerations and response spectra at the site for a magnitude 4.0 event at distances of 10.0, 3.0 and 1.0 kilometers from the plant. The magnitude 4.0 event was the applicant's maximum possible reservoir-induced event. The applicant used a theoretical model (Brune, 1970, 1971; McGuire and Hanks, 1980) to estimate peak accelerations. This method depends on estimates of stress drop, seismic moment, shear wave velocity, density and attenuation. For a stress drop of 25 bars and a source-to-site distance of 1 kilometer, the applicant calculated a peak acceleration of 0.121 g, which compares well with the digitized peak accelerations of 0.130 g and 0.106 g for the two horizontal components for the August 27, 1978 event. The recorded instrumental peak acceleration for the event, as discussed previously, was 0.25 g.

Response spectra were estimated using the method of Johnson and Traubenik (1978). Spectral amplification factors were determined for acceleration, velocity and displacement for different magnitude ranges recorded at distances less than 20 kilometers on rock sites. Newmark and Hall (1973) have observed that the response spectrum at short periods is proportional to the peak acceleration, at intermediate periods is proportional to the peak velocity and at longer periods is proportional to the peak displacement. Because the applicant was only able to estimate peak acceleration, they used ground motion ratios of Johnson and Traubenik (1978) to estimate peak velocity and peak displacement. Using the peak ground motion values and spectral amplification factors, response spectra were estimated for the postulated reservoir-induced earthquakes. The calculated M_L = 4.0 spectra contain less long period energy, as expected.

In Appendix X of the December 1980 report, the applicant calculated ground motion for a magnitude (M_L) 4.5 event for a stress drop of 25 bars and at a distance of 2.0 kilometers. The zero period acceleration value is 0.22g, which is higher than the zero period acceleration value for the safe shutdown earthquake for structures founded on rock. The response spectrum for this event and for the M_L = 4 event at a distance of one kilometer exceed the safe shutdown earthquake in the high frequency region above 10 Hertz. The applicant argues that the safe shutdown earthquake is adequate because the calculated peak acceleration and the spectrum derived from it are conservative and the return period of the safe shutdown earthquake for tectonic earthquakes is 3100 to 10,000 years (Final Safety Analysis Report Question 361.19), which is conservative. They also indicate that there is no adverse effect on the structures due to the excess conservatism of the original damping values and the use of the artificial time history response spectrum.

Our consultant at LASL, Dr. Carl Newton, found the safe shutdown earthquake to be conservative (Appendix D to this Safety Evaluation Report). He believes that when compared with the results of other techniques, the peak accelerations predicted by the applicant may be overly conservative.

Dr. Newton indicates that if the maximum reservoir-induced Piedmont province earthquake were migrated to the site, as is done for the maximum tectonic earthquake for establishing conservative peak design accelerations, then, assuming the 1974 Clark Hill earthquake was induced, the maximum possible sustained vibratory ground motions so obtained are smaller than the safe shutdown earthquake design levels. In addition, since the applicant received a construction permit, there have been several studies that have found formal relationships among magnitude, maximum intensity, and peak acceleration. The applicant has applied the Brune model and formulas from McGuire and Hanks (1980) to calculate a peak acceleration for the maximum tectonic earthquake. Although the latter was done in response to a question submitted by the NRC, the results are overly conservative. He notes that by applying recent formulas of Nuttli and others to find peak accelerations for all tectonic earthquakes of concern to the site, the applicant's design response spectrum is conservative.

Dr. Newton concludes that the maximum reservoir-induced event is expected to be no greater than the largest tectonic event for which the safe shutdown earthquake was chosen. However, small, near-field earthquakes may generate acceleration spikes that may be twice the safe shutdown earthquake design acceleration. The utility has shown in Appendix X to the Supplemental Seismological Investigation that these acceleration spikes have practically no damageability.

The staff agrees that the applicant's estimates of peak acceleration are conservative. Although we believe that 25 bars is a conservative stress drop for reservoir-induced events in the Piedmont, the 90-bar stress drop proposed by Fletcher (1980) and discussed in a previous section by Dr. Murphy needs to be evaluated by the applicant to determine its effects on the applicant's method. We agree with our consultants, LASL, that small, near-field earthquakes may generate acceleration spikes (events of relatively low energy, high acceleration and short duration). The evaluation of these high frequency acceleration spikes, the damping values, and the use of the artificial time history will be discussed in Section 3.7.1 of a supplement to this Safety Evaluation Report.

Continued Seismic Monitoring

Since we believe there is no reason to assume that the largest earthquake induced by Monticello Reservoir has already occurred, the seismic activity in the vicinity of Monticello Reservoir warrants careful attention. The staff believes it is prudent for the applicant to continue monitoring the seismicity to detect possible changes in seismic activity as a precursor to larger events. The applicant has committed to the following program until the end of 1982, at which time an evaluation will be made to determine if it should be continued:

1. The applicant will continue seismic monitoring at the Monticello Reservoir, analysis of the data, and the quarterly submittal of the data to the NRC. The quarterly reports will maintain their present format.

2. The applicant should inform the Operating Reactor Project Manager by telephone of the location, depth and magnitude of any unusual activity as soon as possible after the event. Unusual activity should be considered as any of the following:

    a. any earthquake larger than magnitude 2.5.
    b. more than 100 events per week
    c. any plans to lower water level in the reservoir below 415 feet or revise it above 430 feet.

3. The applicant is currently in the process of installing two accelerometers in the free field in the vicinity of the plant - one at the ground surface and one at bedrock. If the accelerometer located on bedrock records an earthquake with a peak acceleration of at least 0.10g and with an interval of at least one second where peak accelerations exceed 0.05 g, then a copy of the accelerometer record and its response spectrum and a calculation of its stress drop will be included in the quarterly reports.

4. The applicant will continue to look for correlations between seismicity and other variables, such as water level changes, and report any possible correlation in the quarterly reports.

Conclusion

There is no reason to alter our conclusion presented in the Safety Evaluation Report for the construction permit that the applicant's proposed safe shutdown earthquake and operating basis earthquake are conservative. The staff believes the maximum earthquake which could be expected to be associated with reservoir impoundment at Monticello Reservoir is magnitude (M_L) 4.5. Ground motion expected from this event would be of short duration and possess high frequencies which may exceed the safe shutdown earthquake above 10 Hertz. A discussion of the effects of this short duration high frequency ground motion on the plant structures is contained in Section 3.7.1 of a supplement to this Safety Evaluation Report.

Charleston Earthquake

During the construction permit review, the staff concluded that the weight of the seismologic and geologic information support the interpretation that seismicity in the vicinity of Charleston, SC, including the Modified Mercalli intensity IX-X, 1886 Earthquake was related to structure beneath the Coastal Plain in the Charleston area, and should not be assumed to migrate out of that region. We based this conclusion on the available data, past licensing positions, and advice from the U.S. Geological Survey (USGS) and the National Oceanic and Atmospheric Administration (NOAA) as a result of their review of the Summer site. Recognizing that none of the factors supporting restriction of the Charleston seismic zone taken by itself is definitive, the USGS concluded (October 20, 1972 letter to William Gammill (AEC), from Dr. Henry W. Coulter (USGS)) that the cumulative weight of the following factors support that conclusion:

1. The frequency per unit area of historical earthquakes is much higher than elsewhere in the Eastern United States.

2. Event distribution within the high frequency unit shows no evidence of directional trend or predominant pattern which would suggest lateral migration of activity.

3. The microseismic flux in the Charleston area is higher than that measured elsewhere in the Eastern United States.

4. Seismic refraction and aeromagnetic data suggest atypical basement structures in the Charleston area.

Recognizing the lack of definitive data regarding the structural geology in the Charleston region, and in accordance with a recommendation by the ACRS, the AEC contracted the USGS to perform an extensive geologic and seismologic investigation in the Charleston region. These investigations included a regional earthquake monitoring network, borings, geologic mapping and geophysical studies. The investigations are still underway and to date a wealth of new information on the geologic and seismic characteristics of the region has been accumulated. As the USGS investigation of the Charleston region has progressed, numerous working hypotheses evolved concerning the source mechanism of seismicity in that area. The USGS has summarized its current position on that subject in the December 30, 1980 letter to Dr. R. E. Jackson from J. F. Devine. This document is included as Appendix E to this Safety Evaluation Report.

During the course of our review of the Final Safety Analysis Report, we requested the applicant to reassess the impact of Charleston seismicity on the site in light of the new data obtained from the NRC-USGS funded investigations and the various working hypotheses that have emerged from these studies since the construction permit review. The applicant complied with that request and presented its position and the bases for that position in the December, 1980 report, "Supplemental Seismologic Investigation Virgil C. Summer Nuclear Station Unit 1."

In its analysis, the applicant has categorized most of the working hypotheses regarding the source of the Charleston seismicity into one of three principal mechanisms: décollement reactivation, reactivation along steep basement faults, and stress amplification at the margins of mafic plutons. The applicant points out the merits, unexplained questions, and inconsistencies of these mechanisms. The following is a brief discussion of the applicant's analyses and the staff's assessment of the significance of the new information from the Charleston region to the site.

Décollement Reactivation

Cooke et al (1979), based on COCORP data, describe a major thrust fault that underlies part of the Valley and Ridge, the Blue Ridge, and probably extends beneath the Coastal Plain. Harris and Bayer (1979) suggest that the décollement may extend the entire width of the Appalachian Orogen. The décollement reactivation hypothesis infers that it extends beneath the entire southern Applalachians and the Coastal Plain.

Seeber and Armbruster (1980) present the hypothesis that the Charleston seismicity, including the 1886 event, was caused by back-slip of the décollement. They proposed that current seismicity occurs within the sheet above the detachment zone, and that great earthquakes occur along the zone. Along with comparing these phenomena with the Himalayan detachment, their supporting evidence is the nature of stress across the Applachian Orogen, based on the work of Zoback and Zoback (1980).

Based on multichannel seismic reflection profiles in the meizoseismal area of the 1886 Charleston earthquake and in the nearby offshore area, Behrendt et al (1980) have identified two high angle, northeast striking reverse faults which show evidence of Cenozoic deformation. The western-most fault, the Cooke fault, shows 50 meters of displacement, southeast side down, of a Jurassic-age basalt layer at a depth of 750 meters. The fault lies within the meizoseismal zone and trends into a cluster of earthquake epicenters that were recorded between 1972 and 1978. However, these focal depths are substantially greater than the depth of the fault plane beneath the epicenter cluster. Amount of displacement of the Cooke fault increases with depth below the basalt and decreases above the basalt, indicating recurrent faulting from pre-late Cretaceous into Eocene time or later (100 mybp - 40 mybp). The Helena Banks fault, which is 12 kilometers offshore, offsets strata to within 10 meters of the ocean floor. Most recent movement is interpreted to be post-Miocene or Pliocene (7 mybp - 2 mybp). There is no known seismicity that can be associated with this fault. The offshore data also indicate a subhorizontal surface at a depth of 11.4 ± 1.5 kilometers which is interpreted by Behrendt et al (1980) as evidence for a dęcollement similar to that detected in the COCORP data to the west.

The above data are interpreted by Behrendt et al. (1980) as evidence that the Charleston seismicity, as well as other seismicity in the eastern U.S., may be related to northeast striking, high angle reverse faults; or that movement on the thrust faults (dęcollement) is the primary cause of modern seismicity and movement on other types of faults, including the high angle reverse faults, are a second order effect. The latter hypothesis is equivalent to the dęcollement reactivation theory of Seeber and Armbruster (1980).

The applicant maintains that the relationship between the high-angle reverse faults, continent side up (Behrendt et al., 1980), with respect to the dęcollement have not been explained. Furthermore, lateral shortening due to mid-ocean ridge spreading seems inadequate to reactivate the low-angle thrusts. The stress provinces described by Seeber and Armbruster are highly interpretive because of the mixture of types of information used (focal mechanisms, hydrofracturing, overcoring, etc.) relative to the stress field of North America. Due to the high effective vertical normal stress on the detachment surface at hypocentral depths, and the probable absence of the required excessive fluid pressures that characterize active thrusting, there is great difficulty invoking gravity-induced backslip. The presence of asperities and lateral boundaries represented by aulacogens (Rankin, 1976) would provide resistance to low-angle slip.

Based on our review of the data and our assessment of analyses performed by the applicant, the staff agrees with the conclusion of the applicant that the dęcollement reactivation theory is not a viable basis for determining the seismic design at the site because:

1. Backsliding of a low-angle dęcollement by gravity, considering the lithostatic load and probable absence of excessive pore pressure at the depth of the detachment zone, appears to be highly unlikely.

2. The existence of a continuation of the detachment zone beneath the Charleston area has not been demonstrated, and is controversial.

3. Even if the dęcollement exists beneath the Charleston area, it would probably not behave as a single thrust sheet because of the presence of

irregularities along the detachment zone, secondary geologic structures, possible ancient transcurrent faults normal to the direction of slip, and variations in stress regimes throughout the thrust sheet.

4. Relationship between the high-angle reverse faults and the dęcollement has not been explained. Furthermore, the basalt flows cored at Clubhouse Corner in the Charleston Earthquake epicentral area apparently were derived from the upper mantle (Gottfried et al., 1977). The most likely source for these rocks is by way of high-angle faults as is the case with similar flows in the Newark Basin and other Mesozoic basins in eastern U.S. The dęcollement reactivation theory argues that the high-angle reverse faults are listric to the thrust fault. If the high-angle faults were truncated at the detachment surface, it is not clear where the basalt flows came from.

5. There has been no evidence of Quaternary dislocation found along the western front of the Valley and Ridge Province where the detachment zone crops out, or along any of the major low-angle east dipping Paleozoic thrusts in that region. This is also an area of relatively low seismicity (Appendix D to this Safety Evaluation Report).

Reactivation of High-Angle Basement Faults

A second general mechanism addressed by the applicant is reactivation of steep basement faults. These faults have had different tectonic origins during the history of the Appalachian Orogen, but many scientists believe that they remain the ultimate cause of energy strain release in this region (Rankin, 1978, Wentworth, and Mergner-Keefer, 1980). This hypothesis conflicts with the master dęcollement theory discussed previously in that, according to the latter, the detachment is not displaced vertically by these deep basement faults. The high-angle faults are listric into the dęcollement.

The applicant recognizes two schools of thought regarding this mechanism: (1) strain release occurs along northwest zones of weakness inherited from evolutionary development of the Appalachian Orogen (that is, along failed arm troughs of a triple junction); or (2) vertical tectonics occur along fault-bounded basement blocks or old Triassic border faults.

Sbar and Sykes (1973) recognized trends of seismicity crossing the Atlantic Coastline that appeared to be along westward projections of major ocean fracture zones. One of these trends was a zone of seismicity that crossed South Carolina along a projection of the Blake Fracture Zone. Bollinger (1973) presented evidence for a South Carolina-Georgia seismic zone. Fletcher et al. (1978) show a relationship between their Charleston-Cumberland seismic zone and the Blake Fracture Zone. Rankin (1976) postulated that the northwest trending South Carolina-Georgia seismic zone, including the Charleston earthquake, is related to the reactivation of a Precambrian aulacogen (a fault-bounded intracratonic trough or graben).

Rankin (1976) postulated that the salients and recesses in the Appalachian structural trends were inherited from the initial breakup of proto-North America by the intersection of rift valleys radiating from triple junctions about 820 mybp. Rankin (1978) added that the Charleston meizoseismal area is located in a Late Triassic-Early Jurassic (200 mybp - 170 mybp) rift basin that connects the ancestral Gulf of Mexico with rifts that parallel Appalachian structural trends but are located on the continental shelf. The north boundary of this rift basin trends east-northeast in South Carolina. The Blake Fracture Zone, which originated as a Jurassic transform fault, strikes N55°W and offsets the Atlantic Shelf Edge Magnetic Anomaly about 200 km east of Charleston. Rankin (1978) and Fletcher et al. (1978) suggest that all of this evidence explains earthquakes along the South Carolina-Georgia Seismic zone (Bollinger, 1973). Nishenko and Sykes (1979) suggest that the Blake zone is related to the Georgia-Florida rift zone which it intersects at the Shelf anomaly, and that the Charleston earthquake is related to that intersection.

The second school of thought holds that high angle reverse faults in the Charleston vicinity and along the Fall Zone result in complex differential vertical crustal movements. These blocks are bounded by steep Precambrian to Mesozoic (pre-600 mybp - 65 mybp) faults with various orientations. The faults may have been active in the Cenozoic (post-65 mybp) (Sheridan, 1976). Geodetic findings of Meade (1971), Holdahl and Morrison (1974) and Lyttle et al. (1979), and geologic evidence of Owens (1970), Winker and Howard (1977), Winker (1980), Heller et al. (1980), and Zimmerman (1980) indicate Cenozoic crustal movement. Wentworth and Mergner-Keefer (1980) describe the existence along the Atlantic margin of a domain of northeast trending reverse faults that follow older discontinuities, especially Mesozoic (225 mybp - 65 mybp) normal faults, which they believe are undergoing sporadic movement. They suggest that the 1886 Charleston earthquake and the 1755 Cape Ann earthquake are related to displacements on these types of faults.

The applicant concludes and we agree that most of the evidence for northwest trending structure is circumstantial. There is no direct evidence of a major NW throughgoing structure in the Charleston meizoseismal area even though there has been much effort expended to identify one. The dominant structural trend identified in the area is oriented northeast-southwest. The South Carolina-Georgia seismic zone of Bollinger (1973), which is cited as evidence for such a structure, is diffuse and there are aseismic areas within it.

We have also considered the possibility that Charleston seismicity may be related to high angle northeast trending faults in the vicinity of Charleston (Behrendt et al., 1980, Wentworth and Mergner-Keefer, 1980). These authors suggest that Charleston-type seismicity is possible in other areas where similar structures are present. In its letter report (Appendix E to this Safety Evaluation Report), the USGS states in regard to the Cooke fault, a high angle reverse fault in the Charleston meizoseismal area, that "until further research provides more definitive concepts of southeastern U.S. seismicity and of its fault length and history of movement, the Cooke Fault by virtue of its coincidence of location with the Charleston earthquake should remain as a candidate structure to associate with that earthquake. Consequently, it should be considered as having a potential for generating similar events in the future." The USGS further concludes, "the concentration of seismicity in the Charleston earthquake epicenter both before and after the August 31, 1886, event and the lack of post-Miocene faulting in the Coastal Plain or any evidence for localizing large earthquakes indicate that the likelihood of a Charleston sized event in other parts of the Coastal Plain and Piedmont is very low." The NRC staff concurs with these conclusions.

Mafic/Ultramafic Plutons

McKeown (1975), recognizing a spatial relationship between mafic bodies and seismicity, suggested that high stress conce trations occur within mafic bodies and at their contacts with countr ocks, and that this was a possible source mechanism for high seismic areas. Kane, (1978) based on laboratory tests and regional observations, suggests that most high seismic areas in the United States may be associated with mafic or ultramafic intrusives manifested by high gravity and aeromagnetic anomalies. The idea is that regional stress is concentrated in the crystalline rock surrounding mafic plutons because the mafic bodies are serpentinized and cannot themselves sustain high-stress fields.

In regard to the theory of stress amplification near the boundaries of mafic/ultramafic plutons, the applicant points out several uncertainties which are important to applying this theory to Charleston. (1) The source of the gravity anomalies is not known, as borings have never penetrated them. (2) The state of regional stress near the anomalies is not known. (3) The three-dimensional geometry of the anomalous masses and the boundary conditions of the host rock containing them are unknown. (4) There are regions of eastern North America in which there are positive gravity anomalies which are not spatially correlated with high seism city.

The staff agrees with the applicant based on the reasons given above and our evaluation of this phenomenor relative to the Pilgrim 2 site study (NRC, 1976) that evidence is not yet strc g enough to positively associate areas of high seismicity directly with mafi or ultramafic plutons for purposes of siting decisions.

Another hypothesis to emerge from the Charleston studies, is the interpretation that a different type of basement rock underlies the Charleston Coastal Plain than that exposed in the Piedmont. This hypothesis contains elements of both the high angle fault and mafic pluton mechanisms.

Popenoe and Zietz (1977) interpret the smooth gravity and magnetic fields associated with basement in southeast South Carolina and east-central Georgia as suggesting that the basement is underlain by a deep structural basin filled with Triassic clastic and volcanic material, and has been intruded by a number of Triassic or later mafic plutons. They further suggest that the boundary between this material and the northeast trending aeromagnetic anomalies to the northwest probably reflects a series of major faults which juxtapose metamorphic and nonmetamorphic terrain associated with mafic plutons. The crystalline basement beneath the Charleston-Summerville area is therefore not seen as simply a seaward extension of crystalline rocks exposed in the Piedmont (Ran' n, 1977). This implies that there is a different crust beneath the Charleston area than the Piedmont to the west and could thus be a basis for a tectonic boundary between the Piedmont and Coastal Plain. This hypothesis has been neither confirmed nor disproved, as borings have not penetrated the basement beneath the Mesozoic rocks in the meizoseismal area. This hypothesis was not specifically addressed by the applicant. However, elements of it were addressed in either the discussions of the mafic pluton mechanism or the reactivation of high angle fault mechanism.

Staff Conclusions - Charleston Earthquake

The staff has reviewed the information developed since the construction permit Safety Evaluation Report, particularly the results of the USGS studies in the Charleston region, the working hypotheses formulated as a result of that work, and the analysis of the Charleston region performed by the applicant. Based on our consideration of this information, we conclude that the position presented in the construction permit Safety Evaluation Report is still valid; that there is no basis to assume that an earthquake equivalent to the 1886 Charleston earthquake is likely to occur anywhere but in the general vicinity of Charleston-Summerville, South Carolina. The conclusion is based in part on information provided by our advisor, the USGS, in its December 30, 1980, letter to R. E. Jackson from J. F. Devine included as Appendix E to this Safety Evaluation Report. The rationale for the staff's position includes the following:

1. Seismic evidence supports a local source mechanism for the Charleston earthquake activity. Frequency of earthquakes per unit area is much higher than elsewhere in the eastern United States outside of New Madrid. Event distribution shows no evidence of directional trend or predominant pattern which would suggest lateral migration of activity. The microseismic flux in the Charleston area is higher than that measured elsewhere in the Eastern United States.

2. Although certain similarities with other areas exist, the geologic and seismic investigations in the Charleston area indicate that a wide variety of geologic and tectonic features characterize that area in a manner that is not known to typify other sections of the Piedmont and Coastal Plain of the Southeastern United States. The data indicate that the following are among many significant geologic features that characterize the Charleston area: high-angle post-Cretaceous reverse faults (Cooke and Helena Banks faults); deep (10 to 15 kilometers) thrust faults offshore; deep mafic intrusions (gravity and aeromagnetics); a Triassic basin; widespread, thick Triassic-Jurassic basalt flows; a large basement controlled structural basin (Southeast Georgia Embayment), bounded on the north by the Cape Fear Arch and on the south by the Peninsula Arch; relatively high localized seismicity; major basement surface irregularities; clastic dikes in Coastal Plain soils; and possible nearby aulacogen.

   Charleston seismicity could be related to one of these features, a combination of two or more, or to some feature that has not been identified yet. It is possibly associated with the complex interaction of tectonic structures as, for example, suggested by Nishenko and Sykes (1979). One of the working hypotheses is based on the assumption that the identification or suggestion of a specific tectonic structure within or near the meizoseismal area of the 1886 Charleston Earthquake, requires the transposition of a possible recurrence of that event to all similar structures within the Piedmont or Coastal Plain (Wentworth et al., 1980, Behrendt et al., 1980, and Seeber and Armbruster, 1980). We regard this theory as a serious working hypothesis, but not as being directly applicable without additional supporting data, to the siting of nuclear facilities. Based on the currently available data which shows atypical geologic and seismologic complexity at Charleston, it is not reasonable in our view to select a single tectonic structure, such as northeast trending high-angle faults identified at Charleston and assume that a Charleston earthquake can occur on any similar feature throughout the eastern U.S. The fact that there is sufficient geologic structural complexity observed in the immediate Charleston area to allow for development of several working hypotheses, some of them conflicting, lends strong credibility when combined with seismological observations that there is no strong basis to assume that the Charleston earthquake is likely to occur anywhere but in the vicinity of Charleston, South Carolina.

Therefore, it is our position that the 1886 Charleston, Modified Mercalli Intensity IX-X earthquake, can be reasonably related to complex geologic structure unique to that region; and in consideration of the recurrent seismicity in the Summerville area, should not, in developing the earthquake design basis for the facility, be assumed to occur at the site.

Overall Conclusions

Based on our review of the available data we conclude:

1. The Wateree Creek fault does not represent a hazard to the site. Although the fault zone may be localizing some of the seismicity associated with reservoir loading along fractures related to it, these earthquakes do not define a linear pattern nor is there any evidence that seismicity is propagating along the postulated trace beyond the boundary of the reservoir suggesting throughgoing fault movement. The staff considers it prudent however, for the applicant to continue to monitor the on-going mapping of the Wateree Creek fault.

2. The staff reaffirms its conclusion presented in the construction permit Safety Evaluation Report that the applicant's proposed safe shutdown earthquake and operating basis earthquake are conservative. A magnitude (M_L) 4.5 earthquake is an adequately conservative representation of the maximum reservoir induced earthquake. The occurrence of such an event near the site will produce low energy short duration, high frequency ground motions that may exceed the safe shutdown earthquake at high frequencies. This conclusion is based on our evaluation of the applicant's data regarding site geologic and seismic characteristics, our review of the applicant's analysis of Monticello Reservoir induced seismicity, and our consideration of world-wide data regarding reservoir induced seismicity. The evaluation of these short duration, high frequency ground motions will be contained in Section 3.7.1 of a supplement to this Safety Evaluation Report. We recommend, and the applicant has agreed, that seismic monitoring of the reservoir should be continued for at least two years.

3. We agree with the applicant that the 1886 Charleston earthquake is not the safe shutdown earthquake design event because the weight of the seismic and geologic evidence supports localization of seismicity with structure near Charleston. However, because a clear association between structure and seismicity has not been demonstrated, geological and seismological research should be continued in the Charleston area.

2.5.4 Stability of Subsurface Materials and Foundations

The topography of the general area of the facility is characterized by gently to steeply rolling hills and generally well-drained mature valleys which empty ultimately into the Broad River. The site is situated on a hilltop at an average elevation of 435 feet above mean sea level, about 180 feet above the Broad River floodplain. The site is adjacent to Monticello Reservoir which was created by placing a series of dams across Frees Creek, a tributary of the Broad River. The reservoir provides water requirements for the Virgil C. Summer Nuclear Station, Unit 1 and a pumped storage facility. Plant grade is approximately 10 feet above the maximum operating level of Monticello Reservoir, which is at elevation 425 feet above mean sea level. A berm at elevation 438 feet above mean sea level is located along the north boundary of the site adjacent to Monticello Reservoir and, coupled with dams, forms the impoundment for the service water pond located to the east of the main plant structures.

Subsurface Conditions

To establish engineering properties of the residual soil and rock beneath seismic Category I structures, 108 borings were drilled at the site and the vicinity near the site to depths ranging from 12 feet to 235 feet below the original ground surface. Geophysical studies performed at the site included a seismic refraction survey, a surface wave survey, and micromotion measurements. In addition, detailed geologic studies of the exposed bedrock were performed including the comprehensive investigation of rock shear zones by trenching and construction excavation. See Sections 2.5.1 and 2.5.2 of this Safety Evaluation Report for details of the geologic studies.

Before grading, the upper five to 10 feet of natural soil principally consisted of stiff clayey soils containing variable quantities of sand. Some surface alluvium consisting of sand and/or silty soils was located in the area of the service water pond. Saprolite which is defined as rock that has weathered in place to form medium-dense to dense silty sand and/or sandy silt and which exhibits a slight to low plasticity because of weak cementation is found below the surface zone. After construction, soils present below the finished grade are in-situ saprolite and backfill, except in the valley area between the reactor building and service water pond, where fill overlies the upper zone of in-situ clayey soils. The overburden soils are underlain by bedrock consisting of metamorphic gneisses and schists of the Charlotte Belt with Paleozoic igneous intrusive zones. With the depth, the bedrock grades from highly weathered to moderately weathered to fresh rock containing some random thin zones of partially decomposed rock. Moderately weathered and/or fresh rock were encountered in borings for principal structures of the facility at depths (below the original ground surface) of from 65 to 115 feet (elevation 290 feet to 410 feet). Because of differential weathering, the elevation of fresh rock and the character of weathered rock changes appreciably over short distances (horizontal and vertical).

Observations of water levels in exploratory borings indicated that the groundwater table at and around the site occurs at depths ranging from approximately 20 to 90 feet (elevation 350 feet to 420 feet) below the original ground surface, generally in jointed bedrock. Sometime after Monticello Reservoir is impounded with a normal high water elevation of 425 feet and a normal drawdown elevation of 420.5 feet, it is estimated that the groundwater at the principal structures of the facility will rise to a maximum elevation of 420 feet. Groundwater did not constitute a major problem during construction of the facility. In those few areas where groundwater entered the excavations in sufficient quantities to require dewatering, it was controlled by a series of French drains and sumps.

Backfill Materials

Five types of backfill materials were used at the site beneath seismic Category I structures, i.e., fill concrete, river sand, and Zone I, II, and III materials.

Fill concrete was placed directly on rock beneath the base of the foundation mat of the reactor, control, and auxiliary buildings. The concrete was designed to obtain a 28-day strength of 1500 pounds per square inch.

Zone I and II materials were used as primary backfill beneath the diesel generator, fuel handling, and intermediate buildings which are supported on caissons. Zone I material is a reddish clayey soil classified as a CH or CL material in accordance with the Unified Soil Classification System. Zone II material is a multicolored saprolite ranging from an MH to an SM soil in accordance with the Unified Soil Classification System. Zone I and II materials are free of organic material and stones having a maximum dimension of over six inches. and were compacted in horizontal layers not exceeding eight inches (loose) in thickness to a dry density of at least 90 percent of the maximum dry density as determined by ASTM D-1557-70, Method C. The moisture content was held to within four percent above and two percent below the optimum moisture content.

Zone III material was used predominantly for support of the condensate storage tank and the diesel generator fuel oil storage tanks. This backfill consisted of a well-graded, durable crushed rock placed in 12- to 15-inch thick lifts and compacted to at least 85 percent of relative density as determined by the Department of the Army Standard EM-1110-2-1906.

The river sand was used for backfilling portions of excavations for the service water discharge pipes. The material was placed in layers not exceeding 12 inches and compacted to at least 75 percent of relative density as determined by the Department of the Army Standard EM-1110-2-1906.

Mat Foundations on Rock

After the site was cleared, grubbed, stripped of topsoil and organic material, and graded to elevation 435 feet, excavations were made for the foundation mats of the reactor, control, and auxiliary buildings. These excavations extended into rock (a maximum of approximately 100 feet below finished grade). Percussion rock drills were used to evaluate zones of highly weathered rock. The mats are founded on, as a minimum, moderately weathered rock with a compressional wave velocity of 8,000 to 10,000 feet per second. After the foundation rock was cleaned by air and/or water jetting, inspected, and approved, the excavations were backfilled with fill concrete.

The rock-bearing capacity was evaluated relative to the Rock Property Indicator Number (a convenient summary of weathering and jointing features). The rock property indicators and their bearing capacities based on the results of unconfined and triaxial compression tests are presented in Table 2-3 of this Safety Evaluation Report.

The design allowable rock-bearing capacity provides for safety factors of 30 for the Number 1 rock and 20 for the Number 2 and 3 rock. The total pressures on the bearing rock surface caused by the reactor, control, and auxiliary mat foundations do not exceed 25,000 pounds per square foot.

Elastic theory analysis indicated that the settlement of the reactor, control, and auxiliary buildings would be approximately 1/4 inch for loadings of up to 25,000 pounds per square foot. Stress-strain data indicated that the settlement will occur instantaneously as each increment of load is applied and, therefore, post-construction settlements of mat foundations on rock will be practically zero.

Caisson Foundation

The seismic Category I structures supported on caissons embedded in rock are the diesel generator, fuel handling, and intermediate buildings. Zone I and Zone II materials are used as primary backfill beneath these structures.

The caissons, which are 36 and 48 inches in diameter, are designed to be supported by end bearing and/or shaft resistance in the underlying rock. The depth of the rock sockets (minimum of one foot) are in accordance with the allowable end bearing and shaft resistance values relative to the compressional load, uplift load, and lateral resistance requirements of the individual caissons. Probe holes having a minimum depth of at least two times the caisson diameter were drilled beneath the bottom of each socket to investigate the competency of the bearing rock.

The static and dynamic allowable end bearing and shaft resistance for the various rock conditions are presented in Table 2-4 of this Safety Evaluation Report. The caissons are designed in accordance with the allowable end bearing and shaft resistance values presented in Table 2-4 of this Safety Evaluation Report with the exception that the bearing values for the Number 2 rock were also used for the Number 1 rock.

Caissons were estimated to settle 1/4 inch or less due to elastic compression and were expected to experience the settlement immediately as the load is applied.

Foundations on Soil

The seismic Category I structures and components supported on soils are the condensate storage tank, diesel generator fuel oil storage tanks, electrical duct bank, service water intake pipes, service water discharge pipes, service water intake structure, service water pumphouse, and service water discharge structure.

TABLE 2-3

FOUNDATION ROCK BEARING CAPACITIES

Rock Property Indicator Number	Description	Ultimate Bearing Capacity, thousand pounds per square foot	Design Allowable Bearing Capacity, thousand pounds per square foot
1	Massive fresh rock, some slightly jointed.	6,000	200
2	Moderately weathered rock, slightly jointed; and slightly weathered rock, moderately jointed.	2,000	100
3	Moderately weathered rock, highly jointed.	800	40

TABLE 2-4

CAISSON END BEARING AND SHAFT RESISTANCE

Rock Property Indicator	Ultimate End Bearing Capacity Thousand Pounds Per Square Foot	Allowable Static Loadings: End Bearing Thousand Pounds Per Square Foot	Allowable Static Loadings: Shaft Resistance Thousand Pounds Per Square Foot	Allowable Dynamic Loadings: End Bearing Capacity Thousand Pounds Per Square Foot	Allowable Dynamic Loadings: Shaft Resistance Thousand Pounds Per Square Foot
1	6,000	200	10	600	20
2	2,000	100	10	300	20
3	800	25	5	75	10

Each diesel generator fuel oil storage tank is supported on compacted Zone III material which is shaped to uniformly support the circular bottom of the tank. The Zone III material extends to approximately elevation 404 feet, the depth to which the natural in situ soils were removed. The service water pipes and electrical duct bank are supported below finished plant grade in compacted Zone I, II, and III material. The diesel generator fuel oil storage tanks, the electrical duct bank, and the service water pipes weigh less than an equal volume of compacted backfill. For these facilities, the safety factor against a soil bearing failure is extremely high.

The in-situ soils at the condensate storage tank site were excavated to elevation 409 feet and the excavation was backfilled with Zone III material to foundation grade, elevation 430 feet. The ultimate bearing capacity of the natural soils at elevation 409 feet was calculated to be 65,000 pounds per square foot. The factor of safety against bearing failure at this level is 45 for normal operating conditions and 20 for dynamic conditions. Total settlement of the condensate tank was estimated to be less than 1/2 inch.

The base slab of the service water discharge structure is supported on decomposed rock at elevation 408 feet. The ultimate bearing capacity for the structure was calculated to be greater than 90,000 pounds per square foot, resulting in a minimum factor of safety in excess of 15.

Table 2-5 presents a summary of foundation design information for seismic Category I structures.

Service Water Pumphouse and Intake Structure

The service water pumphouse foundation mat is supported on compacted fill at an elevation of 386 feet within the west embankment of the service water pond. The ultimate bearing capacity of the west embankment fill, after filling the service water pond, was computed to be 40,000 pounds per square foot resulting in a minimum factor of safety of six against a bearing capacity failure.

Based upon the results of original subsurface investigation, laboratory testing on block samples of the compacted fill and consolidation tests on samples of saprolite from areas other than beneath these structures, the applicant predicted that the pumphouse would experience about four inches of settlement and the pumphouse end of the intake structure about two inches over the life of the facility. A settlement monitoring program was established prior to construction of these structures to measure the actual settlement at various locations on the structures during construction.

On August 15, 1977, a settlement exceeding the estimate was measured at the settlement points. On August 22, 1977, the intake structure was inspected and nothing unusual was observed. On August 29, 1977, the intake structure was inspected again and several minor cracks were found in the walls and roof. These cracks widened at the roof and closed near the bottom of the walls. In most instances, they penetrated the full thickness of the walls and roof slab. The largest crack had a maximum width of 1/8 inch. Some of the cracks are nearly vertical and some are inclined to the vertical, with a maximum inclination of 25 degrees.

TABLE 2-5

FOUNDATION DESIGN INFORMATION FOR SEISMIC CATEGORY I STRUCTURES

Structure	Approximate Foundation Elevation (feet)	Foundation Embedment	Foundation Type	Foundation Materials
Reactor Building	396 to 408	39	mat	Fill concrete to rock
Control Building	407 to 411	28	mat	Fill concrete on rock
Auxiliary Building				
North	384 to 388	51	mat	Fill concrete on rock
South	370 to 374	65	mat	Fill concrete on rock
Diesel Generator Building	Cap elevation 394 to 421	41 and 14	caissons	Rock
Fuel Handling Building	Cap elevation 409 to 430	26 and 5	caissons	Rock
Intermediate Builindg	Cap elevation 394 to 409	39 and 26	caissons	Rock
Condensate Storage Tank	430	5	mat	Zone III
Diesel Generator Storage Tanks	419	16	-	Zone III
Service Water Pumphouse	386	49	mat	Zone I and II
Service Water Intake Structure	367	-	mat	Zone I and II
Service Water Discharge Structure	408	15	mat	Decomposed rock

On discovery of the cracks, a survey program was initiated by the applicant to monitor the cracks for changes of width and length and to record additional cracking which might be caused by further settlement during completion of construction of the embankment and pumphouse. The resulting data from this program indicates that as the settlement increased from August to December, the number of cracks increased and occurred over a longer length of the intake structure. Also, some of the existing cracks widened.

Soon after the unexpected large settlement, construction of the pumphouse and surrounding fill was temporarily halted while the cause was investigated. This investigation consisted of two additional test borings, settlement monitoring instrumentation of Borros points and settlement plates, and a revised settlement analysis based on the new data. Based on the results of this investigation and the available monitoring data, the applicant concluded that the pumphouse would experience a total settlement about 12 to 14 inces and the intake structure about 10 to 12 inches.

In order to accelerate the settlement, the applicant preloaded the soils under the pumphouse by placing an extra five feet of fill around the structure and filling the pump chamber in the pumphouse with 9.5 feet of water.

The preload accelerated the settlement to a point where it leveled off in December 1977 at from 11.5 inches of total settlement at the east corner to 13.5 inches of total settlement at the west corner.

After removal of the preload fill, pressure grouting of the cracks in the intake structure was begun on December 15, 1977 and completed on January 18, 1978. The service water pond was filled about five weeks after the completion of grouting in the intake structure. Since the settlement monitoring points in the intake structure would be inundated, three survey masts were affixed to its roof. The settlement data indicated that after an initial downward movement of approximately 0.25 inch due to filling the pond, the pumphouse rebounded about 0.5 to 0.6 inch from March through August, 1978, since which time movement has essentially ceased. The intake structure rebounded about 0.6 to 0.8 inch through November 1978, and since then movement has essentially ceased.

We are reviewing the information provided by the applicant to determine whether the soils beneath the pumphouse and intake structure would change its compressibility when saturated, i.e., whether additional settlement would be occurring upon saturation.

Two 30-inch-diameter service water pump discharge lines, one 36-inch-diameter circulating water bypass line, and an electrical duct bank are now connected to the pumphouse. These services were connected to the pumphouse as late as possible to allow maximum settlement or rebound to take place. They were stopped short of the pumphouse a minimum distance of 50 feet until connection was imminent.

The 36-inch bypass pipe line to the circulating water intake structure was connected in February 1978, prior to filling the pond. This pipe line has Dresser couplings close to the pumphouse. These couplings allow a radial movement of 3/8-inch and two degrees deflection. The 30-inch service water pipe lines were connected to the pumphouse at the beginning of April 1978, during the time that rebound occurred due to filling the pond. Some misalignments of piping were discovered which required re-excavation in order to connect the pipes to the pumphouse. The significance and the cause of this misalignment are being investigated and evaluated by the applicant and the staff.

The electrical duct bank was connected to the pumphouse on June 22, 1978. The joint provided for the duct bank at the pumphouse wall is designed to allow for a differential displacement of at least one inch in any direction all around. The applicant indicates that these connections and couplings will accommodate the differential movements anticipated from soil rebound and seismic events. A downward movement of 0.84 inch of the duct bank relative to the pumphouse was reported in October 1979. The significance and cause of the movement are also being investigated and evaluated.

Although significant movements of the pumphouse and intake structure have essentially ceased, the applicant will continue to monitor settlement and piezometric levels at the structure. Readings will be taken every six months throughout the life of the facility. Visual inspection of the pipelines connected to the pumphouse will also be made every six months.

Liquefaction Potential

The principal structures of the facility are supported on mats or caissons founded on the underlying rock and are therefore not susceptible to liquefaction. The applicant's evaluation of the potential for liquefaction under seismic loading of the saprolite layer beneath the diesel generator fuel oil storage tanks, condensate tank, electrical duct banks, and service water lines is based on the results of standard penetration tests. The maximum horizontal acceleration generated by the postulated safe shoutdown earthquake have been taken to be 0.15 times the gravitational acceleration in rock and 0.25 times the gravitational acceleration in soil. The applicant's analysis indicates that liquefaction of the saprolite will not occur under a peak surface acceleration of 0.25 times the gravitational acceleration.

The staff's review indicates that there is not enough information to establish an empirical correlation between the liquefaction potential and the standard penetration resistance of saprolite. The applicant's test results of undisturbed and recomputed saprolite under cyclic loading conditions show that the dynamic shear strengths of the saprolite are substantically greater than the calculated shear stresses for a peak ground surface accelration of 0.25 times the gravitational acceleration. The soft soils encountered at borings 4-1, 4-2, and 4-5 were found unsuitable for foundations and were replaced with compacted fills. These fills are not susceptiable to liquefaction. Based on the high shear strength of the saprolite and the conservative nature of the analysis on seismically-induced shear stresses, we conclude that no liquefaction problems exist in the founding materials.

Conclusion

Based on the results of the applicant's investigations, laboratory and field tests, analyses, and criteria implemented for design and construction, we and our consultants conclude that the site and facility foundations except those related to the service water pumphouse and intake structure meet the requirements of 10 CFR Part 100 and are therefore acceptable. We are continuing our evaluation of the service water pumphouse and intake structure including the discharge pipelines and electric duct bank. The acceptability of these structures and components to perform their safety function will be reported in a supplement to this Safety Evaluation Report.

2.5.5 Stability of Slopes

The three dams and the west embankment, constructed to impound the service water pond, are the only natural or man-made slopes the failure of which could prevent safe shutdown of or pose a hazard to the facility. The stability of these slopes is discussed in Section 2.5.6 of this Safety Evaluation Report.

2.5.6 Embankments and Dams

The service water pond is formed by impounding a segment of a tributary to Frees Creek with a north dam, east dam, south dam, and west embankment. The three dams and the west embankment are seismic Category I homogeneous earth structures. Criterion 45 of the General Design Criteria requires that the cooling water system be designed to permit appropriate periodic inspection of important components to assure the integrity of the system. To assure the continued integrity of the seismic Category I earth structures, the applicant will comply with Regulatory Guide 1.127, "Inspection of Water Control Structures Associated with Nuclear Power Plants."

Foundation and Abutment Treatment

The foundation preparation for the north dam included the removal of alluvium in the valley bottom, removal of all soft or loose surficial materials from the entire embankment foundation and abutment area, control of springs and seeps, and installation of a grout curtain and core trench along the dam centerline. In the valley bottom, the maximum depth of excavation of unsuitable materials was 21 feet. Flows from springs in the excavation were controlled by installing sumps for the larger flows and by dry packing with cement for the smaller flows. To reduce under-seepage a single-line grout curtain over 1300 feet long was installed along the centerline of the north dam and an auxiliary line was grouted on either side of the centerline to obtain a triple- line curtain where the highest permeabilities were encountered.

When preparing the foundation for the south dam, the colluvial soils were found to be excessively wet and to contain large amounts of organic inclusions. Consequently, all colluvial deposits were removed. This additional excavation resulted in up to 22 feet of colluvial materials being excavated from the south dam base. With the exception of one sump located in the toe drain area, all seeps were controlled by either dry packing cement or were of such small flow rates (less than 0.05 gallon per minute) that fill was placed directly over the area without adverse effect upon the compacted materials.

The foundation of the east dam was excavated from one to four feet into firm to stiff residual soil. No groundwater was encountered in the preparation of the east dam foundation.

The foundation preparation for the west embankment included the removal of the surficial soft or loose soils and organic materials, grading of gullies and installation of a French drain and sump system for a portion of the embankment. A peripheral French drain and sump system was constructed to reduce the moisture content of foundation soils wetted by flow from several seeps and springs.

Embankment Geometry and Materials

Each of the four earth structures is designed as a homogeneous embankment with riprap slope protection. The north and south dams contain an internal horizontal drainage blanket and a rock toe to control seepage in the event of sudden drawdown of Monticello Reservoir. The south dam also includes a relatively impervious upstream blanket for seepage control. The reservoir faces of the dams are inclined at 3.5 to one and the service water pond faces of the dams and west embankment are inclined at three to one. Other pertinent geometric aspects of the dams and embankment are given in Table 2-6 of this Safety Evaluation Report.

Select fill materials for the dams and embankment consist of residual soil and saprolite excavated from nearby borrowed sources. The liquid limit and plasticity index of fill materials did not exceed 70 percent and 25 percent respectively. The soil was placed in horizontal lifts not exceeding eight inches in loose thickness and was compacted to a minimum dry density of 90 percent of the modified maximum dry density as determined by ASTM Standard D-1557. The allowable compaction moisture content ranged from one percent below to six percent above the optimum moisture content as determined by ASTM Standard D-1557.

Settlement

Analyses of potential settlement of the north and south dams were performed, based on as-built dimensions and soil parameters to investigate the potential for the loss of freeboard due to post-construction consolidation and the potential for cracking within the embankment due to differential settlements.

The maximum anticipated post-construction settlement, a sum of immediate deformation and one-dimensional consolidation, was estimated to be 7.0 inches for the north dam and 4.6 inches for the south dam.

To assess the cracking potential, the results of a theoretical determination of tensile strains were compared to field observations of other embankments of similar materials. The maximum computed tensile strains along the crests of the north and south dam were compared to strains at cracking reported in the literature. The compression indicates a margin of safety greater than 1.15.

Slope Stability

The static stability of the service water pond dams was investigated using the circular arc method of analysis as performed by the ICES LEASE-I computer program. The design static conditions that were analyzed for the dams were end of construction, submerged condition, sudden drawdown, and steady seepage in both directions. The results of the analyses demonstrated that all of the embankments have an adequate factor of safety against shear failure under the design conditions.

TABLE 2-6

SUMMARY OF EMBANKMENT GEOMETRY

	North Dam	South Dam	East Dam	West Dam
Crest elevation (feet)	438	438	438	435
Crest width (feet)	30	30	40	50
Approximate crest length (feet)	1,500	765	1,150	1,900
Maximum height (feet)	129	98	28	96
Approximate volume (1,000 cubic yards)	785	273	44	1,169[1]

NOTE:

1. Includes non-safety class fill west of west embankment.

A detailed seismic evaluation of the earth dams was performed for conditions representative of the safe shutdown earthquake. The shear moduli and damping values representative of the foundation and embankment materials were obtained by means of field seismic wave velocity determinations and by laboratory cyclic torsion and cyclic shear tests. The factor of safety was calculated as the minimum ratio of shearing resistance to shear stress at any point within the embankment induced during the postulated earthquake. This conservative method of analysis resulted in a minimum factor of safety of 1.2 for the service water pond embankments under safe shutdown earthquake conditions.

However, it should be noted that the as-built conditions of the west embankment are different from those proposed in the original design. The significance of this design deviation on static and dynamic behavior of the west embankment and the effect on the service water pumphouse and intake structure are being investigated and evaluated. We will report the results of our evaluation in a supplement to this Safety Evaluation Report.

Seepage Test

A full-scale seepage test was performed to demonstrate the conservatism of the estimated seepage loss from the service water pond upon the postulated loss of Monticello Reservoir. The test was made during initial filling of Monticello Reservoir by preventing the simultaneous filling of the pond. Measured seepage into the service water pond under maximum differential head was approximately 20 percent of the calculated seepage in the reverse direction.

Instrumentation

Instrumentation was installed in the north and south dams in December 1977 to measure post-construction crest settlement and horizontal movement. The monuments are positioned at intervals of 100 feet along the dam crests. Measurements taken in December 1978 indicate that maximum post-construction settlement of the north dam was less than one inch and that the north dam is now experiencing rebound. Settlement of the south dam was negligible and maximum rebound has been 1.3 inches. Maximum net horizontal movements have been one inch and one-half inch for the north and south dams respectively.

A series of four piezometers are installed at each of three cross sections of the north dam. The piezometers located on the crest and on the service water pond side were used to monitor the transient phreatic water level within the dam during the reverse seepage testing. The piezometers located on the Monticello Reservoir side of the dam were installed primarily to monitor the phreatic surface near the toe of the north dam in the event that Monticello Reservoir is emptied.

Conclusion

Based on the results of the applicant's investigations, laboratory and field tests, analyses and criteria for design and construction, we and our consultants conclude that the man-made dams, except the west embankment, will function reliably and remain stable under safe shutdown earthquake conditions. At the west embankment, the as-built conditions are somewhat different from the design. The effect of this on the service water pumphouse and intake structure is being investigated and evaluated. We will report on the resolution of this matter in a supplement to this Safety Evaluation Report.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Conformance with General Design Criteria

In Section 3.0 of the Final Safety Analysis Report, the applicant presented an evaluation of the design bases against the NRC's General Design Criteria listed in Appendix A to 10 CFR Part 50. In a letter dated November 14, 1980 the applicant addressed compliance with 10 CFR Parts 20, 50, and 100 including the General Design Criteria. We evaluated the final design and the design criteria and conclude, subject to the applicant's adoption of the additional requirements imposed by us as discussed in this Safety Evaluation Report, that the facility has been designed to meet the requirements of the General Design Criteria.

Our review of structures, systems, and components relies extensively on the application of industry codes and standards that have been used as accepted industry practice. These codes and standards, as cited in this Safety Evaluation Report and attached bibliography, have been previously reviewed and found acceptable by us; and have been incorporated into our Standard Review Plan.

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

Criterion 2 of the General Design Criteria requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety function. These structures, systems, and components are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down 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 other than seismic Category I requirements. Included in this classification are those portions of seismic Category I 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 safe shutdown earthquake and remain functional have been identified in an acceptable manner in Tables 3.2-1 and 3.2-2 of the Final Safety Analysis Report.

The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for structures, systems, and components important to safety with the NRC's regulations as set forth in Criterion 2 of the General Design Criteria, and to Regulatory Guide 1.29, "Seismic Design Classification," Revision 2, NRC staff technical positions, and industry standards.

We conclude that structures, systems, and components important to the safety of the facility that must be designed to withstand the effects of a safe shutdown earthquake and remain functional are properly classified as seismic Category I items in conformance with the NRC's regulations, Regulatory Guide 1.29, and Criterion 2 of the General Design Criteria. Design of these items in accordance with seismic Category I requirements provides reasonable assurance that in the event of a safe shutdown earthquake, the facility will perform in a manner providing adequate safeguards to the health and safety of the public.

3.2.2 System Quality Group Classification

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

We have reviewed the applicant's classification systems for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves in fluid systems important to safety and the assignment by the applicant of quality groups or safety classes to those portions of systems required to perform safety functions.

The applicant has utilized the American Nuclear Society Safety Classes 1, 2a, 2b, 3 and non-nuclear safety as defined in the August 1970 draft of ANS N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." Safety Classes 1, 2a, 3 and non-nuclear safety correspond to the Quality Groups A, B, C, and D in Regulatory Guide 1.26, "Quality Group Classifications and Standards," Revision 3. In addition, this earlier version of the American Nuclear Society classification system has a Safety Class 2b which is based on those component codes within Quality Group C and the quality assurance requirements (administration - management and documentation) normally associated with components of Quality Group B. The applicant has applied the American Nuclear Society classification system to those fluid containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety where reliance is placed on these systems: (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and to maintain it in a safe shutdown condition, and (3) to contain radioactive material. These fluid system components are classified in conformance with Regulatory Guide 1.26 in Table 3.2-1 of the Final Safety Analysis Report. Piping and valves for these fluid systems are also classified in an acceptable manner on system piping and instrumentation diagrams in the Final Safety Analysis Report.

Fluid systems pressure-retaining components important to safety that are classified Quality Group A, B, or C are constructed to the following codes and standards:

NRC Quality Group	Applicant's Safety Class	Component Code ASME Section III, Division 1
A	1	Class 1
B	2a	Class 2
C	2b	Class 3
C	3	Class 3

Quality Group A components comply with Section 50.55a of 10 CFR Part 50. Quality Group B and C components comply with Subsection NNA-1140 of Section III of the ASME Code.

Quality Group D components are constructed to the following codes as appropriate: Division 1 of Section VIII of the ASME Boiler and Pressure Vessel Code, ANSI B31.1.0, API-620, API-650, AWWA D100, or manufacturer's standards.

The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves, in fluid systems important to safety, with the regulations as set forth in Criterion 1 of the General Design Criteria the requirements of the codes specified in Section 50.55a of 10 CFR Part 50, Regulatory Guide 1.26, and industry codes and standards.

We conclude that the pressure-retaining components of fluid systems important to safety that are designed, fabricated, erected, and tested to quality standards in conformance with the NRC's regulations, the applicable regulatory guides, and industry codes and standards are acceptable. Conformance with these requirements provide reasonable assurance that the facility will perform in a manner providing adequate safeguards to the health and safety of the public.

3.3 Wind and Tornado Loadings

3.3.1 Wind Loadings

All seismic Category I structures exposed to wind forces were designed to withstand the effects of the design wind. The design wind specified for this facility has a velocity of 100 miles per hour with a 400 year reocurrence. interval.

The procedures that were used to transform the wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gust factors are in accordance with "Wind Forces on Structures," transaction of the American Society of Civil Engineering, Paper No. 3269, Volume 126, Part 2.

We conclude that the procedures that were utilized to determine the loadings on seismic Category I structures induced by the design wind specified for the facility are acceptable. These procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that in the event of design basis winds, the structural integrity of the facility's seismic Category I structures will not be impaired. As a consequence, the seismic Category I systems and components located within these structures will be adequately protected and will perform their intended safety functions, if needed. Conformance with these procedures is an acceptable basis for satisfying, in part, the requirements of Criterion 2 of the General Design Criteria.

3.3.2 Tornado Loadings

All seismic Category I structures exposed to tornado forces and needed for the safe shutdown of the facility were designed to resist a tornado with 290 miles per hour tangential wind velocity and a 70 miles per hour translational wind velocity. The simultaneous atmospheric pressure drop was assumed to be three pounds per square inch in 1.5 seconds. Tornado missiles are also considered in the design as discussed in Section 3.5 of this Safety Evaluation Report.

The procedures that were used to transform the tornado wind velocity into pressure loadings are similar to those used for the design wind loadings as discussed in Section 3.3.1 of this Safety Evaluation Report. The effects of tornado missiles were determined using procedures discussed in Section 3.5 of this Safety Evaluation Report. The total effect of the design tornado on seismic Category I structures is determined by appropriate combinations of the individual effects of the tornado wind pressure, pressure drop and tornado associated missiles. Structures are arranged on the site and protected in such a manner that collapse of structures not designed for the tornado will not affect other safety-related structures.

The procedures utilized to determine the loadings on structures induced by the design basis tornado specified for the facility are acceptable since the procedures provide a conservative basis for engineering design to assure that the structures withstand such environmental forces.

The use of these procedures provides reasonable assurance that in the event of a design basis tornado, the structural integrity of the facility's structures that have to be designed for tornadoes will not be impaired and that safety-related systems and components located within these structures will be adequately protected and may be expected to perform their necessary safety functions as required. Conformance with these procedures is an acceptable basis for satisfying, in part, the requirements of Criterion 2 of the General Design Criteria.

3.4 Water Level (Flood) Design

3.4.1 Flood Protection

Our review included the applicant's design to protect safety-related systems, structures, and components from flood damage and to maintain the capability for a safe facility shutdown during a design basis flood.

The final grade for the facility is located at a minimum elevation of 435.0 feet. The probable maximum flood has been calculated at elevation 436.6 feet assuming the maximum impoundment pool level for Monticello Reservoir. The site is protected from the probable maximum flood by a three-foot-high dike (berm) to elevation 438.0 feet along the shoreline of Monticello Reservoir.

The maximum flood level for the service water pond, an impoundment created by dams within Monticello Reservoir, has been calculated at elevation 433.6 feet. This pond serves as the ultimate heat sink. The site is protected from this flood level by a sloped embankment to elevation 435.0 along the edge of the service water pond.

Since the dike and embankment are 1.4 feet above their maximum corresponding flood levels, the site, including the safety-related components, is adequately protected from the probable maximum flood.

The portions of seismic Category I structures located below finished grade are protected on their outside surfaces by a continuous water-proof membrane. In addition, the auxiliary building mat is protected by a water-proof membrane on the bottom surface.

Access to structures will be located above grade. Below grade penetrations through exterior walls for conduit and piping are provided with water-proof membrane covers to prevent any water leakage. In the event that in-leakage should occur, water will be carried by sloped floors to sumps and pumped away from these sumps.

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

3.4.2 Analysis Procedures

Seismic Category I structures are designed for bouyancy. No siesmic Category I structures become unstable with respect to uplift or overturning due to load combinations. The plant site is protected against potential floods up to elevation 438.0 feet. Therefore, dynamic effects of flooding are not applicable and were not considered. Seismic Category I structures will be adequately protected against expected water level thus satisfying, in part, the requirements of Criterion 2 of the General Design Criteria.

3.5 Missile Protection

In accordance with Criteria 2 and 4 of the General Design Criteria, the facility's seismic Category I structures, systems, and components will be shielded from, or designed for, various postulated missiles. Missiles considered in the design of structures include tornado-generated missiles and various missiles that may result from equipment failure both inside and outside the containment.

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

3.5.1 Missile Selection and Description

Internally Generated Missiles (Inside Containment)

Our review of possible effects of internally generated missiles inside containment included structures, systems, and components whose failure could prevent safe shutdown of the facility or result in significant uncontrolled release of radioactivity. The scope of our review in this area for the facility included general arrangement drawings, piping and instrumentation diagrams, and descriptive information for structures, systems, and components essential to the safe operation and shutdown of the facility.

The applicant analyzed the potential missiles within containment and identified the following missiles:

1. Resistance temperature detectors
2. Safety valve bonnets
3. Diaphragm-operated control valve drive shaft and diaphragm operator
4. Control rod drive mechanism housing plug
5. Control rod drive shaft
6. Instrument wells
7. Pressurizer heaters

The applicant also identified the systems, components, and structures requiring protection from these missiles and the design provisions that afforded the required protection. This included enclosing the systems or components in individual missile-proof compartments, physically separating redundant systems or components of the system, or providing special localized protective barriers.

The applicant did not consider the reactor coolant pump and motor component to be a credible source of missiles. The applicant's basis is the Westinghouse analysis presented in WCAP-8163, "Reactor Coolant Pump Integrity in LOCA," which concludes that the integrity of the flywheel during a loss of coolant accident will be maintained and that other pump and motor components, although they may fail, will not be a source of missiles.

We have not completed our review of the topical report and are pursuing this matter under a generic evaluation program. In conjunction with our review, the Electric Power Research Institute and Westinghouse have performed scaled pump tests with single- and two-phase flow intended to verify vendor analytical techniques and predictions on reactor coolant pump overspeed. The Electric Power Research Institute has contracted Combustion Engineering, CREARE, and the Massachusetts Institute of Technology to perform experimental and analytical work on two-phase flow reactor coolant pump performance. The pump tests were performed on a 1/5-scale test loop at Combustion Engineering, and a 1/20-scale test loop at CREARE. The Electric Power Research Institute has completed Phase I of the study. Westinghouse, together with Framatone, and the French Atomic Energy Commission, are also conducting a research program on pump testing and modeling using a 1/3-scale model of a Westinghouse reactor coolant pump. The test and analysis efforts are expected to be completed during 1979. We are following the development and performance of this program. If the results of these generic investigations indicate that additional protective measures are warranted to prevent excessive pump overspeed or to limit potential consequences to safety-related equipment, we will determine what modifications, if any, are necessary to assure that an acceptable level of safety is maintained, and the applicant will be required to comply with any changes we find necessary.

Section 3.5.1.2 of the Standard Review Plan was used as a basis for demonstrating conformance to Criterion 4 of the General Design Criteria. The missiles identified by the applicant and the proposed protection of the containment and engineered safety features from these internally generated missiles have been reviewed. We conclude that the applicant has identified all the systems and components, whose failure could prevent safe shutdown of the facility or result in a significant uncontrolled release of radioactivity. The acceptability of the analytical procedures and criteria used for the structures and barriers that protect the containment structure and liner, essential systems, and safety-related components from these missiles is addressed in Section 3.5.2 of this Safety Evaluation Report.

Internally Generated Missiles (Outside Containment)

Protection against postulated missiles associated with facility operation, such as missiles generated by rotating or pressurized equipment, is provided by any, or a combination of, compartmentalization, barriers, separation, and equipment design. The primary means of providing protection to safety-related equipment is through the use of physical arrangement. Safety-related systems are physically separated from nonsafety-related systems and the redundant components of safety-related systems are physically separated such that a potential missile could not damage both trains of the safety-related system.

The applicant provided an analysis of the effects of potential internal missiles in safety- related areas outside the containment. The postulated missile is an 18-inch check valve bonnet from the feedwater system. We have reviewed the applicant's analysis and agree that the postulated missile is conservative and representative of typical missiles in safety-related areas of the facility. We further agree that the applicant has shown to our satisfaction that these potential missiles will not adversely affect safety-related systems or components.

We have reviewed the adequacy of the applicant's design necessary to maintain the capability for a safe shutdown in the event of any internally generated missile outside containment. We have concluded that the design is in conformance with Criterion 4 of the General Design Criteria as it relates to structures housing essential systems and to the capability of the systems to withstand the effects of internally generated missiles, and Regulatory Guide 1.13 as it relates to protection from internal missiles, and is therefore acceptable.

Turbine Missiles

Criterion 4 of the General Design Criteria requires that all structures, systems, and components important to safety at a nuclear power plant be appropriately protected against dynamic effects including the effects of missiles that may result from equipment failures and from events and conditions outside the nuclear power plant.

The turbine-generator at the Virgil C. Summer Nuclear Station, Unit 1 has a non-peninsular orientation relative to the containment. The turbine is an 1800 revolutions per minute, tandem compound, four-flow two-stage reheat steam turbine with 43-inch last stage buckets, manufactured by the General Electric Company. The turbine-generator is located on the operating floor of the turbine building at elevation 463 feet. The turbine is flanked by the moisture separators and minor equipment intervening between the turbine and the turbine building walls. The turbine building walls from elevation 463 feet to the roof consists of steel super-structure covered by metal paneling. The sense of rotation of the turbine blades is towards the containment and intervening buildings for the part of the turbine below the turbine centerline and away from the site for the part of the turbine above the turbine centerline. The sense of rotation restricts the exposure of structures, systems and components to potential damage by low trajectory missiles to elevations of about 500 feet or greater.

The only structures vulnerable to low trajectory turbine missiles at the facility are in the upper regions (above elevation 500 feet) of the control building, auxiliary building and the containment.

All equipment necessary for safe shutdown or for mitigating the radiological consequences of an accident as specified in Regulatory Guide 1.115 are located in seismic Category I structures with the exception of the refueling water storage tank. Most of these structures have walls, roofs and barriers of steel reinforced concrete of 3000 pounds per square inch at least two feet thick with the exception of the fuel handling building.

In the event of a turbine overspeed, any turbine missile that exits the casing at an angle less than 60 degrees (measured down from the turbine horizontal centerline) must pass through a minimum of about 10 feet of steel reinforced concrete of 3000 pounds per square inch strength (turbine pedestal). These missiles would present no hazard to equipment necessary for a safe shutdown or for mitigating the radiological consequences of an accident. Any turbine missile that exits the casing at an angle between 60 degrees and 20 degrees could penetrate structures at elevations above 500 feet. For the purpose of our analyses, we have conservatively assumed that all missiles penetrating a safety structure pose unacceptable damage to the equipment inside the structure. The overall probability that turbine missiles would strike safety related structures housing vital equipment within this strike zone is conservatively calculated to be 5×10^{-7} per year. Any turbine missile that exits the casing at an angle between -30 degrees and 0 degrees will pass over the containment and is considered a high trajectory missile. The probability that high trajectory missiles would do damage to safety related equipment (as specified by Regulatory Guide 1.115) was conservatively calculated to be about 4×10^{-7} per year. Our analyses included the use of the overall turbine failure rate of 10^{-4} per year and assumed a damage probability on strike of 1.0 for those areas where horizontal or vertical protection is less than six feet of steel reinforced concrete. This assumption takes no credit for stopping of the missile by less concrete thickness and does not credit the possibility that if any area is struck, the critical equipment within may not be damaged.

The overall probability that a turbine missile could prevent a safe shutdown or damage equipment needed to prevent accidents which could lead to consequences approaching a significant fraction of the 10 CFR Part 100 guidelines is conservatively estimated to be less than 10^{-6} per year. In our review of the Virgil C. Summer Nuclear Station, Unit 1 for turbine missile protection has determined that the overall probability that turbine missiles could damage we have facility and lead to consequences in excess of 10 CFR Part 100 exposure guidelines is acceptably low. The essential systems are considered to be adequately protected against potential turbine missile damage.

Tornado Missiles

Criterion 4 of the General Design Criteria requires that all structures, systems, and components essential to the safety of the facility be protected from the effects of externally generated missiles. All safety-related structures with the exception of the fuel handling building are designed against penetration by tornado-generated missiles. All safety-related systems and components, with the exception of the condensate storage facility, are located within structures designed to afford protection from tornado missiles.

The fuel handling and roof deck and metal siding of the building superstructure are not designed to withstand tornadoes. Therefore, the structure can be penetrated by tornado-generated missiles. The new fuel racks and spent fuel pool and associated equipment are located beneath the fuel handling building's superstructure and roof deck. The applicant has provided an analysis of the effects of missile penetration of the fuel handling building. We have performed an evaluation of postulated missiles penetrating the building and the potential damage to the spent fuel and associated safety-related equipment which may result. Based on our evaluation of the consequences of tornado missiles, we conclude that no adverse effects to safety will result as a consequence of tornado missile penetration of the fuel handling bulding.

The condensate storage facility is discussed in Section 9.2.4 of this Safety Evaluation Report. Its safety function is to provide a source of emergency feedwater following a postulated accident. Its loss as a result of a tornado missile does not affect safety since the service water system which is protected from tornado missiles serves as an adequate backup supply of emergency feedwater. We therefore conclude that no adverse effects to safety will result as a consequence of loss of the condensate storage facility by tornado-generated missiles.

Based on the above, we conclude that the design of the facility is in accordance with the requirements of Criterion 4 of the General Design Criteria with regard to the protection of safety-related structures, systems, and components from externally generated missiles and is therefore acceptable.

3.5.2 Barrier Design Procedures

The seismic Category I structures, systems and components are shielded from, or designed for, various postulated missiles. Missiles considered in the design of structures include tornado-generated missiles and various containment internal missiles, such as those associated with a loss-of-coolant accident.

Information has been provided indicating that the procedures that were used in the design of the structures, shields and barriers to resist the effects of missiles are adequate. The requirements for concrete barriers for internally generated missiles are based on the modified National Research Defense Committee formula discussed in the paper entitled "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," Nuclear and

Systems Sciences Group, Holmes and Narver, Inc., September 1975, by R. P. Kennedy. For steel barriers, the BRL formula is used. The BRL formula is discussed in the paper entitled "Missile Generation and Protection in Light-Water-Cooled Power Reactor Plants," Oak Ridge National Laboratory, ORNL-NSIC-22, September 1968, by R. G. Gwaltney.

The adequacy of barriers against postulated tornado missiles is based on the test performed at Sandia Laboratories and Calspan Corporation. The test are summarized in the following reports: Stephenson, A. E., Tornado Vulnerability - Nuclear Production Facilities, Sandia Laboratories, April 1975; Stephenson, A. E., Addendum to Tornado Vulnerability - Nuclear Production Facilities, Sandia Laboratories, June 1975; Electric Power Research Institute, Full-Scale Tornado-Missile Impact Tests, EPRI NP-148, April 1976; and Calspan Corporation, Missile Impact Testing of Reinforced Concrete Panels, January 1975.

Effective loads due to impact of these missiles are derived by idealizing the target as an equivalent single-degree-of-freedom structure. The equivalent loads of missile impact, whether the missile is environmentally generated or accidentally generated within the facility, are combined with other applicable loads as is discussed in Section 3.8 of this Safety Evaluation Report.

The procedures that were utilized to determine the effects and loadings on seismic Category I structures and missile shields and barriers induced by design basis missiles selected for the facility are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures or barriers will adequately withstand the effects of such forces.

The use of these procedures provides reasonable assurance that in the event of design basis missiles striking seismic Category I structures or other missiles shields and barriers, the structural integrity of the structures, shields, and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected by these structures are, therefore, adequately protected against the effects of missiles and will perform their intended safety function, if needed. Conformance with these procedures is an acceptable basis for satisfying, in part the requirements of Criteria 2 and 4 of the General Design Criteria.

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

3.6.1 Inside Containment

The applicant presented in the Final Safety Analysis Report the criteria used to postulate pipe breaks in high and moderate energy lines both inside and outside containment. Based on our review and evaluation of these criteria conclude that they are consistent with the criteria of Regulatory Guide 46 and Section 3.6.2 of the Standard Review Plan.

The applicant has referenced Westinghouse Topical Report WCAP-8082-P-A, to describe the analytical methods for determining reaction loads on reactor coolant system piping and components due to postulated pipe breaks in the reactor coolant system. Topical Report BN-TOP-2, Revision 2, May 1974, is also referenced to describe analytical methods for dynamic effects of postulated pipe breaks on piping and components in the balance of plant. We have reviewed both WCAP 8082-P-A and BN-TOP-2, Revision 2, and found them to be acceptable.

The proposed design of piping restraints and measures to deal with jet impingement effects upon the reactor coolant pressure boundary and other safety-related systems provide adequate protection for the containment structure, reactor coolant pressure boundary elements, and other systems important to safety.

The provisions for protection against dynamic effects associated with pipe ruptures of the reactor coolant pressure boundary inside containment and the resulting discharge fluid provide adequate assurance that design basis loss-of-coolant accidents will not be aggravated by sequential failures of safety-related piping, and emergency core cooling system performance will not be degraded by these dynamic effects.

The proposed piping arrangement and applicable design considerations for high and moderate energy fluid systems inside and outside of containment will provide adequate assurance that the 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. The design will be of a nature to mitigate the consequence of a pipe break so that the facility can be safely shut down and maintained in a safe shutdown condition in the event of a postulated failure of a pipe carrying a high or moderate energy fluid inside or outside of containment.

3.6.2 Outside Containment

Our guidelines for protection against postulated piping failures in high-energy fluid systems outside containment are given in Section 3.6.1 of the Standard Review Plan and Branch Technical Positions MEB 3-1 and ASB 3-1. The facility design accommodates the effects of postulated pipe breaks and cracks in high energy fluid piping systems outside containment with respect to pipe whip, jet impingement and resulting reactive forces, and environmental effects. The means used to protect safety-related systems and components include physical separation, closure in suitably designed structures or components, pipe whip restraints, and equipment shields.

The protection provided against pipe failure outside containment is in conformance with the guidelines contained in Section 3.6.1 of the Standard Review Plan and Branch Technical Positions MEB 3-1 and ASB 3-1. The applicant analyzed high energy piping systems for the effects of pipe whip, jet impingement, and environmental effects on safety-related systems and structures. For moderate energy systems, protection of safety-related systems from the jet and environmental effects due to critical cracks is incorporated into the facility design.

The facility has the ability to sustain a high energy pipe break coincident with a single active failure in essential systems and retain the capability for safe shutdown. For postulated pipe failures, the resulting environmental effects do not preclude the habitability of the control room, the accessibility of other areas that have to be manned during and following an accident, and the loss of function of electric power supplies, controls, and instrumentation needed to complete a safety action.

Based on our review, we find that the applicant has adequately designed and protected areas and systems required for safe shutdown following postulated events, including the combination of pipe failure and single active failure. The facility design meets the criteria set forth in Section 3.6.1 of the Standard Review Plan and Branch Technical Positions MEB 3-1 and ASB 3-1 as regards the protection of safety-related systems and components from a postulated high energy line break and as regards the protection of safety-related systems and components from a postulated moderate energy line failure. We conclude the facility design for the protection of safety-related equipment against dynamic effects associated with the postulated rupture of piping outside containment is acceptable.

3.7 Seismic Design

3.7.1 Seismic Input

The seismic design response spectra for the operating basis earthquake and safe shutdown earthquake applied in the design of seismic Category I structures, systems, and components complies with the recommendations of Regulatory Guide 1.60, "Design Response Spectra for Nuclear Power Plants." The critical damping values used in the seismic analysis of seismic Category I structures, systems and components are in conformance with Regulatory Guide 1.61, "Damping Values for Seismic Analysis of Nuclear Power Plants."

The synthetic time history used for the design of seismic Category I structures, systems and components is adjusted in amplitude and frequency to obtain response spectra that envelope the response spectra specified for the site.

Conformance with Regulatory Guides 1.60 and 1.61 provides reasonable assurance that for an earthquake whose intensity is 0.15 for the operating basis earthquake, and 0.25 for the safe shutdown earthquake, the seismic inputs to seismic Category I structures, systems, and components are adequately defined to assure a conservative basis for the design of such structures, systems and components to withstand the consequent seismic loadings.

3.7.2 Seismic System and Subsystem Analysis

The scope of our review of the seismic system and subsystem analysis for the facility included the seismic analysis methods for all seismic Category I structures, systems and components. It included review of procedures for modeling, seismic soil-structure interaction, development of floor response spectra, inclusion of torsional effects, seismic analysis of seismic Category I dams, evaluation of seismic Category I structure overturning, and determination of composite damping. Our review has included design criteria and procedures for evaluation of interaction of non-seismic Category I structures and piping with seismic Category I structures and piping and effects of parameter variations on floor response spectra. Our review has also included criteria and seismic analysis procedures for reactor internals and seismic Category I buried piping outside the containment.

The system and subsystem analyses were performed by the applicant on an elastic bases. Response spectrum multi-degree of freedom and time history methods formed the bases for the analyses of all major seismic Category I structures, systems and components. When the modal response spectrum method was used, governing response parameters were combined by the square root of the sum of the squares rule. However, the absolute sum of the modal responses was used for modes with closely spaced frequencies. The square root of the sum of the squares of the maximum co-directional responses was used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. Floor spectra inputs used for design and test verifications of structures, systems, and components were generated from the time history method, taking into account variation of parameters by peaking widening. A vertical seismic system dynamic analysis was employed for all structures, systems and components where analyses showed significant structural amplication in the vertical direction. Torsional effects and stability against overturning were considered.

The lumped mass approach was used to evaluate soil- structure interaction and structure-to-structure interaction effects upon seismic responses.

For the analysis of seismic Category I dams, a finite element approach was used which took into consideration the time history of the forces, the behavior and deformation of the dam due to the earthquake, and applicable stress-strain relations.

We conclude that the seismic system and subsystem analysis procedures and criteria utilized by the applicant provide an acceptable basis for the seismic design.

There have been two other issues raised during plant construction for which we have not completed our review.

The first issue concerns cracks that may develop in the service water intake structure during an earthquake. This structure has experienced cracking due to settlement and has been repaired by grouting as described in Section 2.5.4 of this Safety Evaluation Report. Cracks in the structure have the possibility of degrading quality of service water by letting soil pass into the intake structure. The intake structure provides a passage for the service water to the plant. The second issue relates to seismic activities resulting from construction of Monticello Reservoir near the plant. The resolution of these matters will be reported in a supplement to this Safety Evaluation Report.

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 comply with Regulatory Guide 1.12. Supporting instrumentation is being installed on seismic Category I structures, systems and components in order to provide data for the verification of the seismic responses determined analytically.

The installation of the specified instrumentation in the reactor building and other seismic Category I structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical models used for the design of the facility are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with Regulatory Guide 1.12 and is acceptable.

3.8 Design of Seismic Category I Structures

3.8.1 Concrete Containment

The reactor coolant system is enclosed in a prestressed concrete containment as described in Section 3.8.1 of the Final Safety Analysis Report. The containment structure is designed in accordance with applicable subsections of Section III of the ASME Boiler and Pressure Vessel Code, and ACI 318 to resist various combinations of dead loads, live loads, environmental loads including those due to wind, tornadoes, operating basis earthquake, safe shutdown earthquake and loads generated by the design basis accident including pressure, temperature and associated pipe rupture effects.

The static and linear analyses used for the containment shell and base are well-established methods. Therefore, the analyses are acceptable to the staff.

Materials, construction methods, quality assurance and quality control measures are covered in the Final Safety Analysis Report. The are all based on well-accepted industry codes and standards such as those established by the American Society of Mechanical Engineers, the American Society for Testing and Materials and the American Concrete Institute. Detailed descriptions of the applied codes and standards are given in Section 3.8.1.2 of the Final Safety Analysis Report.

Prior to operation of the facility, the containment will be subjected to an acceptance test in accordance with Regulatory Guide 1.18 during which the internal pressure will be 1.15 times the containment design pressure.

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

The use of these criteria as defined by applicable codes, standards, guides, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occuring within the containment, the structure will withstand the specified design conditions without impairment of its structural integrity or safety function. Conformance with these criteria constitutes an acceptable basis for satisfying, in part, the requirements of Criteria 2, 4, 16, and 50 of the General Design Criteria.

3.8.2 Concrete and Structural Steel Internal Structures

The containment interior structures consist of walls, compartments and floors. The major code used in the design of concrete internal structures was ACI 318-71, "Building Code Requirements for Reinforced Concrete." For steel internal structures the AISC Specification, "Specification for the Design, Fabrication and Erection fo Structural Steel for Building," was used. For equipment supports, Subsection MF of Section III of the ASME Code was used.

The containment concrete and steel internal structures were designed to resist various combinations of dead and live loads, accident induced loads, including pressure and jet loads, and seismic loads. The load combinations used cover those cases likely to occur and include all loads which may act simultaneously. The design and analysis procedures that were used for the internal structures are the same as those used on previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-71 Code and in the AISC Specification for concrete and steel structures, respectively.

The containment internal structures were designed and proportioned to remain within limits established by the NRC staff under the various load combinations. These limits are, in general, based on the ACI 318-71 Code and on the AISC Specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction, their fabrication, construction and installation, are in accordance with the ACI 318-71 Code and AISC Specification for concrete and steel structures, respectively.

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

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Conformance with these criteria constitutes an acceptable basis for satisfying in part the requirements of Criteria 2 and 4 of the General Design Criteria.

3.8.3 Other Seismic Category I Structures

Those seismic Category I structures other than the containment and its interior structures are of structural steel and concrete construction. The structural components consist of slabs, walls, beams and columns. The major code used in the design of concrete seismic Category I structures was the ACI 318-71, "Building Code Requirements for Reinforced Concrete." For steel seismic

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

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

The design and analyses procedures that were used for these seismic Category I structures are the same as those approved on previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-71 Code and in the AISC Specification for concrete and steel structures, respectively.

The various seismic Category I structures are designed and proportioned to remain within limits established by the NRC staff under the various load combinations. These limits are, in general, based on the ACI 318-71 Code and on the AISC Specification for concrete and steel structures, respectively.

The materials of construction, their fabrication, construction and installation are in accordance with the ACI 318-71 Code and the AISC Specification for concrete and steel structures, respectively.

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

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

Conformance with these criteria, codes, specifications, and standards constitutes acceptable basis for satisfying, in part, the requirements of Criteria 2 and 4 of the General Design Criteria.

3.8.4 Foundations

Foundations of seismic Category I structures are described in Section 3.8.5 of the Final Safety Analysis Report. Primarily, these foundations are of reinforced concrete construction of the mat type. The major code used in the design of these concrete mat foundations is ACI 318-71. These concrete foundations have been designed to resist various combination of dead loads; live loads; environmental loads including winds, tornadoes, operating basis earthquake, safe shutdown earthquake and loads generated by postulated ruptures of high energy pipes.

The design and analysis procedures that were used for these seismic Category I foundations are the same as those approved on previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-71 Code. The various seismic Category I foundations were designed and proportioned to remain within limits established by the NRC staff under the various load combinations. These limits are, in general, based on the ACI 318-71 Code modified as appropriate for load combinations that are considered extreme. The materials of construction, their fabrication, construction and installation, will be in accordance with the ACI 318-71 Code.

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

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques, and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated events, seismic Category I foundations will withstand the specified design conditions without impairment to structural integrity and stability or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying in part the requirements of Criteria 2 and 4 of the General Design Criteria.

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

The criteria and methods of analysis the applicant has described for the design of all seismic Category I ASME Code Class 1, 2, 3 and CS components, component supports, reactor internals, and other non-Code items are in conformance with Section 3.9.1 of the Standard Review Plan. These criteria are acceptable to us and satisfy the applicable portions of Criteria 14 and 15 of the General Design Criteria. The use of these criteria for defining the applicable transients, computer codes used for analyses, analytical and experimental methods provides assurance that the stresses, strains, and displacements are within acceptable limits and are adequate for the design of these items.

In addition to our review of the applicant's Final Safety Analysis Report, we contracted with Battelle Pacific Northwest Laboratories to perform an independent confirmatory stress analysis of the facility's "C" feedwater line. The purpose of this analysis was to verify that the calculated stresses in the as-built piping were less than the applicable ASME Code stress allowables. This exercise also served as a random check of the applicant's ability to model its piping systems and use its computer programs.

The "C" feedwater line is an ASME Code Class 2 line. We analyzed this line for the loads due to pressure, dead weight, thermal expansion, and the safe shutdown earthquake in accordance with the rules of the 1971 Edition of the ASME Code, paragraph NC-3652. We conclude that the design of this line complies with the applicable ASME Code stress allowables. Additionally, we found reasonable agreement between our calculations and those of the applicant.

3.9.2 Dynamic Testing and Analysis

Preoperational Vibration and Dynamic Effects Piping Tests

The preoperational vibration test program which will be conducted during startup and initial operation on all safety-related nuclear steam supply system and balance-of-plant piping systems, restraints, components, and component supports classified as ASME Class 1, 2, and 3 and non-ASME portions of the main steam and feedwater piping systems is an acceptable program and is consistent with Section 3.9.2 of the Standard Review Plan. The tests 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 other operating modes associated with the design basis operational transients. The planned tests will develop loads similar to those experienced during reactor operation. Compliance with this test program constitutes an acceptable basis for fulfilling, in part, the requirements of Criterion 15 of the General Design Criteria.

Seismic Qualification of Safety-Related Mechanical Equipment

The qualification testing and analysis program described in the applicant's Final Safety Analysis Report for seismic Category I mechanical equipment, including their supports, have been further evaluated by our seismic qualification review team [illegible] part of the facility visit described in Section 3.10 of this Safety Evaluation [illegible] When this evaluation has been completed, the applicant's program will p[illegible] adequate assurance that such equipment will function properly under the loa[illegible] vibratory forces imposed by the safe shutdown earthquake and under the cond[illegible] of post-earthquake operation. This program will be consistent with Section [illegible] of the Standard Review Plan and will constitute an acceptable basis for satisfying, in part, the requirements of Criterion 2 of the General Design Criteria.

Preoperational Flow-Induced Vibration Testing of Reactor Internals

The designated prototype for the facility's reactor internals is the H. B. Robinson No. 2 reactor. The H. B. Robinson No. 2 reactor utilizes a thermal shield configuration, whereas this facility has neutron pads. It has been demonstrated by model test results which, in turn, have been verified by measurements at Indian Point No. 2, that three-loop reactors with neutron pads experience lower vibration levels than three-loop reactors having thermal shields. Based on the information presented in the Final Safety Analysis Report, we conclude that the H. B. Robinson No. 2 reactor is an acceptable prototype for the facility's reactor internals configuration.

The preoperational vibration program planned for the reactor internals provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of tests, predictive analysis, and post-test inspection provide adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced vibrations of reactor operation without loss of structural integrity. The integrity of the reactor internals in service is essential to assure the proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies to permit safe operation and shutdown of the facility. The preoperational vibration tests to be conducted on the facility's reactor internals conform with the provisions of Regulatory Guide 1.20 for non-prototype reactors. This program constitutes an acceptable basis for demonstrating design adequacy of the reactor internals and satisfies the applicable requirements of Criteria 1 and 4 of the General Design Criteria.

Analysis Methods Under Loss-of-Coolant Accident Loadings

The dynamic system analysis confirms the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic response loads of postulated loss-of-coolant accident and the safe shutdown earthquake. The analysis demonstrates that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction, and that the resulting deflections or displacements of any structural element of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The methods used for component analysis have been found to be compatible with those used for the systems analysis and both are acceptable. Results of the dynamic analysis verify structural integrity of the reactor internals under postulated loss-of-coolant accident conditions combined with the safe shutdown earthquake and provides added assurance that the facility will withstand a spectrum of lesser pipe breaks and seismic loading events. The dynamic system analysis is consistent with Section 3.9.2 of the Standard Review Plan and constitutes an acceptable basis for satisfying the applicable requirements of Criteria 2 and 4 of the General Design Criteria.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

Loading Combinations and Stress Limits

The specified design basis combinations of loadings as applied to safety-related ASME Code Class 1, 2, and 3 pressure-retaining components and their supports in systems designed to meet seismic Category I standards are such as to provide assurance that in the event of an earthquake affecting the site, or an upset, emergency, or faulted plant transient occurring during normal plant operation, the resulting combined stresses imposed on systems, components, and their supports will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity.

With respect to the method of combining dynamic responses to loss-of-coolant accident and safe shutdown earthquake loads, the NRC staff position as outlined in NUREG 0484, "Methodology for Combining Dynamic Responses," is that the square root of the sum of the squares method is acceptable for the reactor coolant pressure boundary systems, components, and supports. In addition, the NRC staff has accepted the square root of the sum of the squares methods of combining responses resulting from the loss-of-coolant accident and safe shutdown earthquake for all other ASME Class 1, 2, and 3 systems, components, and supports in the facility. In response to questions from the staff, the applicant has committed to the above position. Therefore, the NRC staff has concluded that the applicant's method of combining the responses to these two events is acceptable.

Pump and Valve Operability Assurance

The component operability assurance program for ASME Code Class 1, 2, and 3 active valves and pumps provides adequate assurance of the capability of such active components (a) to withstand the imposed loads associated with normal, upset, emergency, and faulted plant and component operating conditions without loss of structural integrity, and (b) to perform necessary "active" functions (e.g., valve closure or opening, pump operation) under accident conditions and conditions expected when plant shutdown is required. The specified component operability assurance test program is consistent with Section 3.9.3 of the Standard Review Plan and constitutes an acceptable basis for satisfying the applicable portions of Criteria 1, 2, and 4 of the General Design Criteria and is acceptable to the NRC staff.

Design of Pressure Relief Valve Mounting

The criteria used in the design of the mountings for ASME Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable 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 mountings for the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function. The criteria used for the design of the mountings for ASME Class 1, 2, and 3 overpressure relief devices constitute an acceptable basis for meeting the applicable requirements of Criteria 1, 2, and 4 of the General Design Criteria and are consistent with those specified in Regulatory Guide 1.67.

Asymmetric Blowdown Loads on the Reactor Coolant System

The applicant has performed a dynamic structural analysis to evaluate the effects of asymmetric blowdown loads on the reactor coolant system. These loads result from the postulated pipe breaks discussed in Section 3.6.2 of this Safety Evaluation Report. In the dynamic analysis, the pipe break thrust force, asymmetric subcompartment pressurization forces and asymmetric reactor internals hydraulic forces were applied as simultaneous time-history forcing functions. The resultant component and support reactions from these forces were combined with the appropriate normal operating and seismic reactions to arrive at maximum support loads. The dynamic load response methodology utilized by the applicant for combining responses due to loss-of-coolant accident and safe shutdown earthquake is acceptable for reactor coolant pressure boundary component supports.

As a part of NRC Task Action Plan A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant System," the NRC staff has performed an independent dynamic structural analysis on a similar three-loop Westinghouse reactor coolant system (North Anna Units 1 & 2). As a result of this analysis, the NRC staff concluded that the methodology and computer programs used to analyze the effects of asymmetric blowdown loads on the facility's reactor coolant system are acceptable. Therefore, the NRC staff agrees with the applicant's conclusions on the structural adequacy of the reactor coolant system and its supports.

3.9.4 Control Rod Drive Systems

The design criteria and the testing program conducted for verification of the mechanical operability and life cycle capabilities of the reactivity control system described in the applicant's Final Safety Analysis Report conforms with the guidelines outlined in Section 3.9.4 of the Standard Review Plan and is acceptable to the NRC staff. The use of these criteria provides reasonable assurance that the system will function reliably when required and is an acceptable basis for satisfying the mechanical reliability stipulations of Criterion 27 of the General Design Criteria.

3.9.5 Reactor Pressure Vessel Internals

The applicant has conducted a dynamic analysis of the reactor internals due to horizontal and vertical excitation under faulted condition loads to demonstrate structural integrity of the reactor internals components as discussed in Section 3.9.2 of this Safety Evaluation Report. In addition, the applicant's method of combining responses to loads is acceptable for reactor coolant pressure boundary components. The applicant has provided reasonable assurance that in the event of an earthquake or of a system transient during normal operation, the resulting deflections and associated stresses imposed on the reactor internals will not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events which have been postulated to occur during service lifetime without loss of structural integrity or impairment of function. The design procedures and criteria used by the applicant in the design of the reactor internals is consistent with Section 3.9.5 of the Standard Review Plan Section and constitutes an acceptable basis for satisfying the applicable requirements of Criteria 1, 2 and 4 of the General Design Criteria.

3.9.6 Inservice Testing of Pumps and Valves

To assure that all ASME Code Class 1, 2, and 3 pumps and valves will be in a state of operational readiness to perform necessary safety functions throughout the life of the facility, the applicant will conduct a test program which includes baseline preservice testing and periodic inservice testing. The program will provide the functional testing of the components in the operating state.

There are several safety systems connected to the reactor coolant pressure boundary that have a design pressure lower than the rated reactor coolant system pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below reactor coolant system pressure. In order to protect these systems from reactor coolant system pressure, two or more isolation valves are placed in series to form the interface between the high pressure and the low-pressure systems. The leak-tight integrity of these valves must be assured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an inter-system loss-of-coolant accident. Periodic leak testing of pressure isolation valves shall be performed after all disturbances to the valve are complete. The pressure isolation valves to be tested are listed in the Technical Specifications.

The applicant has agreed to categorize their pressure isolation valves for the safety injection, residual heat removal, and boron injection systems, as Category A or AC. These categorizations meet our requirements and we find them acceptable. Pressure isolation valves are required to be Category A or AC and to meet the appropriate valve leak rate test requirements of IWV-3420 of Section XI of the ASME Code except as discussed below. The allowable leakage rate shall not exceed 1.0 gallon per minute for each valve as stated in the Technical Specifications.

The applicant has committed to test all pressure isolation valves to the 1.0 gallon-per-minute leak rate criteria.

The applicant will leak test the residual heat removal suction and low head safety injection to the cold legs pressure isolation valves (two check valves or two motor-operated valves for each) once per refueling but not after seat disturbances due to flow. As an alternative, so as to reduce the probability of an intersystem loss-of-coolant accident from occurring in the low head safety injection to the cold legs, the applicant has proposed to leak test a third check valve in each line (located inside the containment). We find this acceptable provided the applicant leak tests these valves once each refueling as described above.

The applicant has also proposed to test the residual heat removal system pressure isolation valves once per refueling as described above. The staff finds this acceptable for the following reasons: (1) full closure of these valves is verified in the control room by direct monitoring position indicators, (2) inadvertent opening of these valves is prevented through interlocks which require the plant to be below residual heat removal system operating pressure prior to opening, and (3) gross leakages due to valve failure would be detected by increasing levels in the pressurizer relief tank. Therefore, full closure of these valves is assured after opening, inadvertent opening is prevented and gross reactor coolant system leakages can be readily detected.

Limiting conditions for operation will be added to the Technical Specifications which will require corrective action i.e., shutdown or system isolation when the leakage limits are not met. Also surveillance requirements, which will state the acceptable leak rate testing frequency, will be provided in the Technical Specifications.

We conclude that the applicant's commitments to periodic leak testing of the pressure isolation valves between the reactor coolant system and low pressure systems will provide reasonable assurance that the design pressure of the low pressure systems will not be exceeded, and thus reduce the probability of an occurrence of an inter-system loss-of-coolant accident and satisfies in part Criterion 55 of the General Design Criteria.

The applicant has stated that the inservice test program for all ASME Code Class 1, 2, and 3 pumps and valves will be submitted 30 days prior to fuel loading. This commitment is consistent with Section 3.9.6 of the Standard Review Plan and constitutes an acceptable basis for satisfying the applicable portions of Criteria 37, 40, 43, and 46 of the General Design Criteria.

3.10 Seismic Qualification of Seismic Category I Instrumentation and Electric Equipment

Mechanical and electrical (includes instrumentation, control, and electrical) equipment and components required to perform a safety function are designed to meet seismic Category I design criteria. Seismic requirements established by the seismic system analysis have been incorporated into equipment specifications to assure that the purchased or designed equipment meets seismic requirements equal to or in excess of the requirements for seismic Category I equipment and components, either by appropriate analysis, by qualification testing, or a combination of analysis and testing.

The applicant has implemented a seismic qualification program for seismic Category I mechanical and electrical equipment, and the associated supports for that equipment. The purpose of this program is to provide assurance that such equipment can be expected to function properly, and that structural integrity of the equipment and its supports will not be impaired during the excitation and vibratory forces imposed by the safe shutdown earthquake and under the conditions of post-accident operation. The applicant's qualification program was implemented while Sections 3.9.2 and 3.10 of the Standard Review Plan were being published and therefore was directed toward full compliance with these sections of the Standard Review Plan. Conformance with these criteria satisfies the applicable portions of Criterion 2 of the General Design Criteria. Section 3.10 of the Standard Review Plan references Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants," and IEEE Standard 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The principal change from earlier criteria is to require consideration of equipment multi-mode response and biaxial coupling effects.

Our seismic qualification review team performed a review at the facility on October 14-17, 1980 to determine the extent to which the qualification of the equipment, as installed in the facility, meets current licensing criteria as described in Sections 3.9.2 and 3.10 of the Standard Review Plan. During this review, we evaluated a representative sample of 22 pieces of seismic Category I mechanical and electrical equipment, both in the nuclear steam supply system and the balance of plant. Among the eighteen pieces of balance-of-plant equipment selected, a review of the qualification of the reactor building cooling unit damper actuator and the radiation monitoring control panel had not been completed by the applicant's rchitect engineer, and therefore final qualification reports were not available for review. The qualification documents for the main steam isolation valve, although approved by the architect-engineer were not available for review during our visit. The documentation was provided to us at the conclusion of our visit. The complete documentation for the 480 volt substations was reviewed briefly during our visit and will be reviewed further. In addition, the hydrogen analyzer panels had not been delivered to the facility and complete information was also not available during our visit. Of the four pieces of nuclear steam supply system equipment selected, only the qualification documents for the post-accident monitoring indicators were not available for review during our visit.

In addition to the six outstanding qualification reports identified above, our seismic qualification review team, at the conclusion of the visit requested the applicant to provide the test and analysis reports for three additional pieces of equipment ecompassing both the balance of plant and nuclear steam supply system, to be included in a follow-up confirmatory review. The equipment selected includes the diesel generator and associated equipment (electrical and air starting controls), accumulator tanks, and electrical containment penetrations and miscellaneous connectors.

Our review of the available balance-of-plant equipment qualification when compared with the current criteria of the Standard Review Plan Sections 3.9.2 and 3.10, identified the needs to clarify the details of the qualification for some pieces of equipment. For example, (1) the design of the supports for the battery charger need to be clarified since they were bolted to the test table, but are welded to the floor in the plant, (2) on the charging pump, some small pipes are loosely supported, and clarification of the safety significance is needed, and (3) in all the safety related valves reviewed, the justification of the acceleration levels used for qualification need to be documented and verified with the as-built piping analysis results. The details of our review and the concerns identified for the qualification of both the nuclear steam supply system and balance of plant equipment are described in the report of our October 14-17 trip to the plant.

In order to complete our review we have requested the applicant to provide the following information:

1. Identify all equipment still to be qualified and provide documentation to demonstrate the completion of the qualification program. Provide seismic qualification review team "Qualification Summary of Equipment" forms for this equipment and update the forms provided for the site visit.

2. Review and revise, as necessary, the tables in Chapter 3 of the Final Safety Analysis Report updated information for all safety-related systems and components.

3. Provide a copy of the revised seismic qualification review team tables which include a list of equipment and the summary of the qualification program.

4. For all safety-related valves describe the design procedure used to demonstrate that accelerations used in the valve qualification meet or exceed the accelerations obtained in the final as-built piping analysis. Provide specific information for the valves reviewed by the seismic qualification review team.

5. Provide qualification reports for the four pieces of equipment not available during the visit and the three additional pieces of equipment selected by the staff at the conclusion of the visit.

6. Provide confirmation that Westinghouse's generic response spectra for equipment qualification envelope the corresponding plant specific required response spectra.

7. Clarify details as discussed in our trip report concerning the qualification of the component cooling water pump and motor, turbine appurtenances for the turbine driven emergency feedwater pump, charging pump, residual heat removal system pumps, battery chargers, control valves, and pressure and differential pressure transmitters.

Based on the results of our review to date, we concude that the equipment qualification program has been defined for the seismic Category I mechanical and electrical equipment which will provide adequate assurance that such equipment will function properly during and after the exitation from vibratory forces imposed by the safe shutdown earthquake. We are continuing our review and will report our conclusions, including our evaluation of the additional information requested of the applicant as discussed above, in a supplement to this Safety Evaluation Report.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

We have reviewed the applicant's estimate of the chemical and radiological environment to which engineered safety feature equipment will be qualified during a postulated design basis accident. The chemical environment inside the containment structure will be dominated by water from the borated water storage tank, the sodium hydroxide storage tank (both water sources for the containment spray), the reactor primary coolant system and the emergency core cooling system. The boric acid in these water sources is neutralized by the sodium hydroxide to a final pH of approximately nine. Based on this final pH and the characteristics of the other chemicals mentioned above, we conclude that the chemical environment inside the containment after a postulated design basis accident is not hostile to engineered safety feature equipment.

The radiological environment inside the containment is attributed to the source term resulting from the 10 CFR Part 100 design basis accident source term. Specifically, this is equivalent to a release of 50 percent of the halogens in the core, 100 percent of the noble gases in the core and one percent of the solid fission product inventory in the core, as stated in Regulatory Guide 1.7. Using these assumptions the applicant has calculated a gamma dose of 4×10^7 rads for the first ten days following a loss of coolant accident and 10^8 rads for 90 days. Since the applicant's calculated doses are based on the source term assumptions of Regulatory Guides 1.4 and 1.7 and the results agree with our independent estimates in connection with ongoing generic Task Action Plan A-24 on environmental qualification (which indicates integrated doses on the order of 10^8 rads could result), we conclude the applicant's calculated doses are acceptable.

We have published guidance to be used for the environmental qualification of safety-related electrical equipment, NUREG-0588, "Interim Staff Position on

Environmental Qualification of Safety-Related Electrical Equipment." We requested the applicant to reassess the qualification documentation for all safety-related electrical equipment in accordance with the guidance provided in NUREG-0588. The purpose of this request was to determine the degree of conformance of the applicant's environmental qualification program to the program as outlined in NUREG-0588. In response to this request, the applicant provided an environmental qualification submittal dated September 24, 1980. This submittal identified several items of electrical equipment which were inadequately qualified for the expected service environment. The applicant agreed to submit additional information concerning the open items identified, and this information was provided in January 1981.

On the basis of the information reviewed we cannot finalize our conclusions regarding equipment qualification for the facility. However, we will continue the equipment qualification review for this facility. The review will include an audit of the utilities qualification documentation and site visit after we have received and reviewed a completed NUREG-0588 submittal from the applicant.

Our review, audit/site visit, and safety evaluation relating to equipment qualification for the facility will be completed prior to full power operation.

4 REACTOR

4.1 General

The nuclear steam supply system design for the Virgil C. Summer Nuclear Station, Unit 1 is similar to that reviewed and approved for North Anna, Units 1 and 2 (Docket Nos. 50-338 and 50-339). A comparison of the principal thermal-hydraulic parameters is presented in Table 4-1 of this Safety Evaluation Report.

4.2 Fuel Design

4.2.1 Description

The fuel assemblies proposed for the facility will consist of 264 fuel rods, 24 guide thimbles, and one instrumentation thimble arranged in a 17x17 array. The instrumentation thimble will be located at the center of the assemblies and will facilitate the insertion of neutron detectors. The guide thimbles will provide channels for inserting various reactivity controls. The fuel rods will contain uranium dioxide ceramic pellets hermetically clad in Zircaloy-4 tubes. The fuel assembly structure is held together by Zircaloy thimble tubes and the stainless steel fuel assembly nozzles at the top and bottom. Alignment and transverse spacings will be maintained by eight spacer grids separated uniformly along the vertical axis of the fuel assembly.

All fuel rods will be internally prepressurized with helium during final welding to reduce fuel cladding compressive stresses during service. The level of prepressurization is designed to preclude flattening of the cladding. The specific level of prepressurization will be dependent upon the planned fuel burnup and will be determined prior to establishing technical specifications.

The fuel assembly design (17x17 array) is identical to the assemblies operating in Trojan, Farley Unit 1, Beaver Valley Unit 1, Salem Units 1 and 2, D. C. Cook Unit 2, and North Anna Units 1 and 2. This design is only a slight modification of the previously used Westinghouse 15x15 fuel assembly. Those mechanical aspects which differ from the previous 15x15 designs are presented in Table 4-2 of this Safety Evaluation Report. The differences are essentially geometric and will result in a lower linear power density and other increased safety margins for the 17x17 type fuel assembly.

The evaluation of the Westinghouse fuel mechanical design is based upon mechanical tests, in-reactor operating experience, and engineering analyses. Additionally, the in-reactor performance of the fuel design will be subjected to the continuing surveillance programs of Westinghouse and individual utilities. These programs provide confirmatory and current design performance information.

TABLE 4-1

THERMAL-HYDRAULIC DESIGN COMPARISON

Thermal-Hydraulic Design Parameter	Virgil C. Summer Nuclear Station, Unit 1	North Anna, Units 1 & 2
Core power, thermal megawatts	2775.0	2775.0
Minimum steady-state pressurizer pressure, pounds per square inch, absolute	2220.0	2220.0
Reactor coolant system flow 10^6 pounds per hour	107.5	105.2
Coolant inlet temperature, degrees Fahrenheit	554.8	546.8
Enthalpy rise factor	1.55	1.55
Departure from nucleate boiling ratio correlation	W-3 R-grid	W-3 R-grid
Minimum departure from nucleate boiling ratio at nominal conditions	2.01 typical cell 1.69 thimble cell	2.15 typical cell 1.77 thimble cell
Minimum departure from nucleate boiling ratio for design transients	$\geq$1.30	$\geq$1.30
Average heat flux, British thermal units per hour-foot squared	189,800	189,000
Heat transfer surface area, square feet	48,600	48,600
Average linear heat rate, kilowatts per foot	5.44	5.44
Peak linear heat rate for normal operation, kilowatts per foot	12.6	13.6

TABLE 4-2

FUEL MECHANICAL DESIGN COMPARISON

Design Parameter	Virgil C. Summer Nuclear Station, Unit 1	Trojan	Typical Westinghouse Operating Fuel
Fuel			
Rod array	17x17	17x17	15x15
Number of fuel rods	264	264	204
Fuel column length, inches	144	144	144
Number of spacer grids	8	8	7
Number of guide thimbles	24	24	20
Inter-rod pitch, inches	0.496	0.496	0.563
Average thermal output (four loop), kilowatts per foot	5.44	5.44	7.0
Fuel Pellets			
Density (theoretical, percent)	95	95	94
Fuel weight/unit length (per rod, not assembly), pounds per foot	0.364	0.364	0.364
Fuel Cladding			
Outside radius, inches	0.187	0.187	0.211
Thickness, inches	0.0225	0.0225	0.0243
Radius/thickness ratio	8.31	8.31	8.68

TABLE 4-3

RANGE OF DESIGN PARAMETER EXPERIENCE

Parameter	Range of Power Reactor Experience
Fuel rod array	14x14, 15x15, and 17x17
Rod per assembly	179 to 264
Guide thimbles per assembly	16 to 24
Assembly envelope, inches	7.76 to 8.43
Inter-rod pitch, inches	0.563 to 0.463
Plenum length, inches	3.27 to 6.69
Prepressurization, pounds per square inch, absolute	14.7 to over 400
Diametral gap, inches	0.0065 to 0.0075
Spacer grids/assembly	7 to 9
Fuel column height, inches	120 to 144

4.2.2 Thermal Performance

In our evaluation of the thermal performance of the reactor fuel, we assume that densification of the uranium oxide fuel pellets may occur during irradiation in light water reactors.

The initial density of the fuel pellets and the size, shape, and distribution of pores within the fuel pellets influence the densification phenomenon.

Briefly stated, in-reactor densification (shrinkage) of oxide fuel pellets (a) may reduce gap conductance, and hence increase fuel temperatures, because of a decrease in pellet diameter; (b) increases the linear heat generation rate because of the decrease in pellet length; and (c) may result in gaps in the fuel column as a result of pellet length decreases. These gaps produce local power spikes and the potential for cladding creep collapse.

The engineering methods to be used by Westinghouse to analyze the densification effects on fuel thermal performance have been previously submitted to the staff in WCAP-8219 and approved for use in licensing. The methods include testing, mechanical analyses, thermal and hydraulic analyses, and accident analyses. The results of our review are reported in an NRC staff report, "Technical Report on the Densification of Westinghouse PWR Fuel," and additional information on densification methods can be found in NUREG-0085.

The improved Westinghouse fuel thermal performance code as described in WCAP-8720 was used for the safety analysis. This code contains a revision of an earlier fission gas release model and revised models for helium solubility, fuel swelling, and fuel densification.

The new Westinghouse code was approved with four restrictions as described in our safety evaluation of February 9, 1979, NRC staff letter from J. Stolz to T. Anderson, Westinghouse. Three of those restrictions deal with numerical limits and have been complied with. The fourth restriction relates to the use of the PAD-3.3 code for the analysis of fission gas release from uranium dioxide for power increasing conditions during normal operation. This restriction applies to the safety analysis of this facility. However, Westinghouse has stated that this restriction does not adversely affect the results of the safety analyses performed for the plant. Although we believe that this is essentially correct for the planned operation of this facility, Westinghouse has prepared and submitted a detailed evaluation of this restriction in WCAP-8720.

At this time, we have not completed our review of the Westinghouse evaluation of this restriction. However, our review has progressed to the point where the following conclusions can be made:

1. The Westinghouse evaluation of our restriction on the use of the PAD-3.3 code supports their earlier statement that the restriction does not adversely affect the results of the safety analyses performed for the facility.

2. We continue to believe that this result is essentially correct and anticipate some additional information from Westinghouse to confirm this conclusion.

3. Because the restriction pertains to the release of fission gases from the fuel, any change in our conclusions would not have significant impact at low burnup, when the fission gas inventory in the fuel is low.

At this time, we can therefore state that for first cycle operation at full power, the restriction for PAD-3.3 is not significant and the analyses as presently docketed are acceptable. We anticipate completion of our review of the Westinghouse evaluation prior to operation at extended burnup.

For the safety analysis, revised internal fuel rod pressure criteria, as described in an approved Westinghouse topical report WCAP-8963-A, were used. Briefly stated, these criteria allow the fuel rod internal pressure to exceed the external system pressure. The approved criteria are as follows: (a) the internal pressure is limited such that the fuel-to-cladding gap does not increase during steady-state operation and (b) extensive departure from nucleate boiling propagation does not occur to postulated transients and accidents. Based on the analyses already submitted in support of this facility, we know that these rod pressure criteria will be satisfied for fuel burnups up to the peak target burnup.

Westinghouse topical report WCAP-8377 which describes the details of a revised cladding flattening model, which, for a given fuel region, predicts initial flattening time and the flattened rod frequency for pressurized rods containing relatively stable fuel, was revised by the staff. This revised analysis was based on the results of examinations of irradiated fuel rods via television, and the results indicated that the original flattening model in WCAP-7982 significantly underpredicted the time and frequency of collapse. The COLLAP computer code is used to perform these calculations. The revised model was accepted for use in safety analysis related to licensing subject to provisions specified in our safety evaluation report, which required that no alterations to the specified curves used as input to the model be made. We have verified that the model has been applied in the approved manner, therefore cladding collapse calculations have been performed acceptably.

4.2.3 Mechanical Performance

Although limited operating experience exists on 17x17 fuel assemblies, substantially all of the in-reactor operating experience with Westinghouse fuel rods and assemblies is applicable to the facility fuel design since the 17x17 fuel assembly is only a slight mechanical extrapolation from the 15x15 fuel assembly. The current use of similar fuel rods and assemblies has yielded operating experience that provides confidence in the acceptable performance of the fuel assembly design. The range in design parameters for which in-reactor experience is specifically applicable has been tabulated in Table 4-3 of this Safety Evaluation Report. The assemblies referred to in Table 4-2 of this Safety Evaluation Report have been irradiated for up to six years and have peak exposures of 3 gigawatt days per metric ton, totaling more than 70 million megawatt hours of power generation.

Verification tests on the 17x17 (12-foot core) assemblies have been completed and reported in Westinghouse topical reports WCAP-8279 and WCAP-8288. We have reviewed these two topical reports and have approved WCAP-8288 for use in the safety analysis. Our review of WCAP-8279 has progressed to the point that we can forecast a favorable evaluation.

The consideration of fuel rod bowing in the 17x17 design was initially analyzed by Westinghouse in the report WCAP-8346. Subsequently, Westinghouse reassessed its analysis in light of new information and documented its findings in WCAP-8692. Neither of these reports has been approved for use in licensing applications. We issued an interim safety evaluation report on Westinghouse fuel rod bowing in April 1976, and in February 1977, we issued a revised interim evaluation report. In the February 1977 report, we accepted the burnup-dependent approach used by Westinghouse with modifications to account for extensions to the 17x17 design and with an increase in rod bowing from as-measured values (cold dimensions) to those in-reactor (hot dimensions). While revised generic methods of analysis have been submitted by Westinghouse in WCAP-8692, they have not been reviewed; however, the interim method is conservative and acceptable for use for this facility. The departure from nucleate boiling analyses was performed using the interim method of February 1977, therefore fuel rod bowing in the facility is acceptably considered. Seismic effects and vertical loads from postulated double-ended hot and cold leg breaks during the loss-of-coolant accident were analyzed in topical report WCAP-8288. We found the methodology acceptable for 17x17 assemblies with either seven or eight spacer grids. Westinghouse subsequently postulated a new asymmetric (horizontal) hydraulic load caused by a postulated pipe break within the biological shield. Westinghouse has performed a preliminary analysis that indicated that the fuel assemblies will be able to accommondate this load. In a letter from C. Eicheldinger to D. Vassallo, NRC staff, dated March 1, 1976, Westinghouse stated that although the experiments and calculational techniques supplied in WCAP-8288 may be applicable in assessing the adequacy of the fuel assembly to withstand these loads, it would be expected that they would be reviewed on a plant-by-plant basis. The applicant has performed an analysis for the most limiting main coolant pipe break and states that the maximum grid impact force for the combined loss-of-coolant accident and safe shutdown earthquake is approximately 38 percent of the minimum grid strength. Therefore, the response for the fuel assemblies for seismic and loss-of-coolant accident loads has been analyzed with acceptable methodology and the results show that the assemblies will accommodate these loads in an acceptable manner.

Limitations on power rate changes will affect pellet-cladding interaction, which is being reviewed as a generic item. The Westinghouse 17x17 fuel rod design used in the facility incorporates features that reduce, compared with the 15x15 design, cladding strain due to pellet-cladding interaction. These features include (a) pellet chamfering, (b) rod prepressurization, (c) lower linear heat rating, and (d) smaller cladding diameter-to-thickness ratio. Based on the available experimental and commercial reactor data, these design features should result in a reduction or delay of pellet-cladding interaction failures to later in the fuel design life. Although the failure thresholds are probably lower at high burnup than at low burnup, the fuel duty is also less severe. While pellet-cladding interaction is being studied generically to determine if licensing criteria should be revised, current criteria are satisfied for the fuel design. Should licensing criteria related to pellet-cladding interaction change in the future, the effects of such a change would be reviewed for all plants including this facility.

We have reviewed the safety aspects of waterlogging fuel rod failures. A recent NRC survey (NUREG-0303) of available information included (a) results of tests in the capsule driver core at SPERT and the Japanese test reactor NSRR, and (b) observations of waterlogging failures in test and commercial reactors. It was concluded that (a) operating restrictions to reduce pellet-cladding interactions also reduce the potential for waterlogging failures during transients, (b) tests to simulate accident conditions produced the worst waterlogging failures, and (c) there is no apparent threat from waterlogging failures to the overall coolability of the core or to safe reactor shutdown. We thus agree that the evaluation of waterlogging failures as presented in the Final Safety Analysis Report is correct.

Fuel assembly fretting and wear test results from 17x17 fuel assemblies were reported in the Westinghouse hydraulic flow test report, WCAP-8279. These tests with a seven-grid assembly indicated that fuel rod wear under both normal and transient operating conditions was within the Westinghouse predicted values and that, even for fuel rods with deliberately damaged grid cells, the wear was within acceptable limits. Westinghouse has since submitted the tests of an eight-grid 17x17 fuel assembly loop which simulated actual in-reactor conditions, showed that no anomalous vibrations were observed or could be induced; and, therefore, no modification to the 17x17 fuel assembly design was required.

The Westinghouse flow test report WCAP-8279 also presented results for fretting wear at contact points between the control rods and thimble tubes. Contact is usually observed in two locations; (1) at the top nozzle for fully withdrawn control rods, and (2) in the dashpot transition section for inserted rods. In both regions, the observed wear was significant but was stated to be within the design limits. Because of excessive guide tube wear experienced in a non-Westinghouse pressurized water reactor fuel design, this wear phenomenon is being reviewed carefully for all pressurized water reactor plants.

In response to the NRC staff's attempt to assess the susceptibility and impact of guide thimble tube wear in Westinghouse plants, two meetings were held with Westinghouse, and information was submitted on their experience and understanding of the issue. This information consisted of guide thimble tube wear measurements taken on irradiated fuel assemblies from Point Beach, Units 1 and 2 (two-loop plants using 14x14 fuel assemblies). Also described was a mechanistic wear model (developed from the Point Beach data) and the impact of the model's wear predictions on the safety analyses of plant designs. Westinghouse believes that their fuel designs will experience less wear than that reported in some other nuclear steam supply system designs because the Westinghouse designs use thinner, more flexible control rods that have relatively more lateral support in the guide tube assembly of the upper core structure. Such construction provides the housing and guide path for the rod cluster control assemblies above the core and thus restricts control rod vibration due to lateral exit flow. Also, Westinghouse believes that their wear model conservatively predicts guide thimble tube wear and that even with the worst anticipated wear conditions (both in the degree of wear and the location of wear) their guide thimble tubes will be able to fulfill their design functions.

The NRC staff concluded that the Westinghouse analysis probably accounts for all the major variables that control this wear process. Because of the complexities and uncertainties in (a) determining contact forces, (b) surface-to-surface wear rates, (c) forcing functions, and (d) extrapolations of these variables to the new 17x17 fuel assembly design, the staff required several near-term operating license applicants to submit to a surveillance program. For acceptability, the minimum objective of such program was to demonstrate that there is no occurrence of hole formation in rodded guide thimble tubes.

To satisfy this request for confirmation of the Westinghouse analytical predictions, a cooperative owners group was established which is now sponsoring a program to obtain post-irradiation examination data from the Salem Unit 1 facility. This post-irradiation examination program will examine all guide thimble tubes in six rodded fuel assemblies having either one or two cycles of burnup. It is our expectation that the program will confirm the Westinghouse predictions. On the basis of the data and analyses mentioned above and the confirmation surveillance program that will be performed, we conclude that this issue is resolved for this facility.

An additional fretting problem has arisen in some fuel rods that are adjacent to baffle plate joints on the periphery of the core. The baffle plates are not always tightly joined, and pressure differences across the baffle sometimes result in cross-flow impingement on nearby fuel rods. In several instances this baffle jetting has resulted in gross failures of one or two isolated fuel rods.

To eliminate baffle-jetting problems, the applicant has modified the lower internals by (a) adding edge bolts at center injection points, and (b) peening all joints as necessary to close the gaps. The applicant, in a letter dated July 1, 1980, has also agreed to examine all fuel rods residing near such locations at the first refueling outage. Should damage be observed at that time, corrective action would be taken.

Recent experiences in Westinghouse plants have indicated an actual or potential operating problem with some of the core hardware items. Specifically, these items are spacer grids and control rodlet fingers. Spacer grid damage to a significant number of grids occurred at a single plant during a refueling operation. Westinghouse and the NRC staff have issued notices with recommended revisions to operating procedures that should eliminate this problem.

A small number (eight) of control rodlet fingers in a single reactor core failed from stress corrosion cracking and this allowed single control rodlets to be inserted into fuel assemblies. The probable cause for stress corrosion cracking has been identified as a tapping lubricant, and this lubricant has been eliminated from the manufacturing process. The most significant effect of dropped rodlets is in the core physics area. Local regions of power depression will occur and power tilts may result. Again, Westinghouse and the NRC staff have issued memoranda on this subject with recommendations for increased attention to the hot zero power flux maps. No significant safety problem from dropped rodlets is anticipated.

In summary, actual or potential problem areas for two core hardware items have been discussed. However, no design changes have been required for the items concerned. We conclude that these problems have been satisfactorily addressed for this facility.

The NRC staff has been generically evaluating three materials models that are used in emergency core cooling system evaluations. These models predict cladding rupture temperature, cladding burst strain, and fuel assembly flow blockage. We have (a) discussed our evaluation with vendors and other industry representatives, (b) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," and (c) required licensees to confirm that their operating reactors would continue to be in conformance with Section 50.46 of 10 CFR Part 50 if the NUREG-0630 models were substituted for the present materials models in their emergency core cooling system evaluations and certain other compensatory model changes were allowed.

Until we complete our generic review and implement new acceptance criteria for cladding modes, we have been requiring that plant emergency core cooling system analyses be accompanied by supplemental calculations performed with the materials models of NUREG-0630. For these supplemental calculations, we have been accepting other compensatory model changes that may not yet be approved by the NRC, but are consistent with the changes allowed for the confirmatory operating reactor calculations mentioned above.

By letter dated October 29, 1980, the applicant provided a supplemental calculation. This calculation also accounted for a non-conservativsm recently identified by Westinghouse in their February 1978 emergency core cooling system evaluation model. As described in their letter dated November 16, 1979 Westinghouse had discovered that loss-of-coolant accident analyses of actual plant heatup rates were at relatively slow temperature-ramp rates; whereas, the 1978 emergency core cooling system evaluation model was, in part, based on cladding burst tests that were conducted at relatively fast temperature-ramp rates. The applicant's submittal assessed the combined impact of this calcuational error and the NUREG-0630 models to be worth 855 degrees Fahrenheit peak cladding temperature above that previously calculated. Subsequently Westinghouse calculated that a reduction in total peaking factor, of 0.0318 would offset the 855 degrees Fahrenheit increase in peak cladding temperature. However, Westinghouse also identified a margin in total peaking factor available through the use of thermohydraulic models already approved for some applications. This margin was worth 0.15 in total peaking factor. Thus no total peaking factor reduction is required for the facility and we conclude that the applicant has satisfied our concerns related to the swelling and rupture issue.

4.2.4 Surveillance

Performance of the fuel is indirectly monitored by measurement of the activity of the primary coolant for compliance with Technical Specification limits. Westinghouse has proposed a fuel surveillance program for several plants that will use the 17x17 fuel assemblies. A summary of this program is given in the fuel rod bowing report, WCAP-8692. This program includes lead assemblies in Surry Units 1 and 2 and the initial core loadings for Trojan, Beaver Valley Unit 1, Farley Unit 1, and Salem Unit 1.

Surry Units 1 and 2 each have two lead burnup 17x17 fuel assemblies. One of the lead assemblies in each unit has removable rods. These assemblies were carefully measured prior to insertion and will be examined between cycles for dimensional changes, fretting corrosion near the spacer grids, fuel rod bowing, axial gamma distribution, cladding defects, and surface deposits. Inspections after two cycles in Surry Units 1 and 2 have revealed no anomalies. Surry Unit 2 assemblies have completed three full cycles with an estimated burnup of 28,000 megawatt days per ton.

The other four reactors included in the surveillance program will each have an initial core loading of 17x17 fuel assemblies (Trojan, Beaver Valley Unit 1, Farley Unit 1, and Salem Unit 1). Each core will include a removable-rod assembly except for Beaver Valley Unit 1. Only two of the four, however, will be examined as part of the 17x17 fuel assembly surveillance program, and these will be selected on the basis of the first two to actually reload fuel. The surveillance program includes visual examination (100 percent scanning) of the initially loaded (first core) fuel assemblies to be removed during the first three refueling outages. If any anomalies are detected, further examination will be performed using the removable fuel rod assemblies.

The first visual examination has been completed at the Trojan facility, and the results show the fuel assemblies with burnups up to 17,800 megawatt days per metric ton to be in excellent condition. Preliminary results from the second inspection at Trojan revealed baffle-jetting failures in two fuel rods that had resided near an inside corner. Corrective action was taken at Trojan, and additional surveillance for baffle-jetting failures will be performed at Trojan and in the Virgil C. Summer Nuclear Station, Unit 1.

4.2.5 Conclusions

On the basis of the safety analysis, confirmatory data from both in-reactor and out-of-reactor tests, and satisfactory experience with this fuel type in other operating reactors and anticipated receipt of acceptable supplemental calculations on the emegency core cooling system, we conclude that the fuel for this facility will perform its function adequately and that all applicable requirements have been met. All applicable requirements related to the reactor fuel are described in Section 4.2, "Fuel System Design," of the Standard Review Plan. The applicable Regulations and Regulatory Guides are: Section 50.46 of 10 CFR 50; Appendix A to 10 CFR Part 50 (Criterion 10 of the General Design Criteria); Appendix K to 10 CFR Part 50; Regulatory Guide 1.3; Regulatory Guide 1.4; Regulatory Guide 1.25; Regulatory Guide 1.77; Regulatory Guide 1.126. Some of these requirements are satisfied in Section 15 rather than in Section 4.2 of the Final Safety Analysis Report.

4.3 Nuclear Design

The reactor is a pressurized water reactor containing 157 fuel assemblies of the Westinghouse 17x17 type. It has a core heat output of 2775 thermal megwatts and is essentially identical in design to the North Anna, Units 1 and 2 reactors. We have reviewed the nuclear design of the reactor for the facility. Our review was based on information contained in the Final Safety Analysis Report, amendments thereto, and the referenced topical reports. Our review was conducted within the guidelines provided by Section 4.3 of the Standard Review Plan.

4.3.1 Design Bases

We have reviewed the design bases and functional requirements used in the nuclear design of the fuel and reactivity control systems of the facility. The basic requirement for the core and control system is that the consequences of each event be appropriate to the category for that event. To meet this requirement, several specific design bases are presented. These include:

1. Specification of acceptable fuel design limits.

2. Specification of a negative prompt feedback coefficient.

3. Requirement that power oscillations be inherently damped or that the control system be capable of detecting and suppressing them.

4. Requirement for a control and monitoring system which automatically initiates a rapid negative insertion to prevent fuel design limits from being exceeded during normal operation and anticipated transients.

5. Requirement that the design of the control system be such that no single malfunction or operator error lead to a violation of fuel design limits.

6. Requirement that shutdown be assured even when the single rod cluster control assembly (control rod) of highest worth is assumed to be stuck out of the core.

7. Requirement for a chemical shim system capable of controlling power changes in normal operation and of bringing the reactor to cold shutdown.

8. Requirement that the control system, when combined with the engineered safety features, be capable of controlling reactivity changes during accident conditions.

9. Requirement that reactivity insertion rates and amounts be controlled so that only limited damage occurs to the pressure boundary and the core remains in a coolable geometry following a reactivity insertion accident.

Based on our review, we conclude that the nuclear design bases presented in the Final Safety Analysis Report are in conformance with Criteria 10, 11, 12, 13, 20, 25, 26, 27, and 28 of the General Design Criteria and are therefore acceptable.

4.3.2 Design Description

The Final Safety Analysis Report contains the description of the first cycle fuel loading which consists of three different enrichments and has a first cycle length of approximately one year. The enrichment distribution, burnable poison distribution, soluble poison concentration and higher isotope (actinide) content as a function of core exposure are presented. Values presented for the delayed neutron fraction and prompt neutron lifetime at beginning and end of cycle are consistent with those normally used and are acceptable.

Power Distribution

The design bases affecting power distribution are:

1. The peaking factor in the core will not be greater than 2.32 during normal operation at full power in order to meet the initial conditions assumed in the loss-of-coolant accident analysis.

2. Under normal conditions (including maximum overpower) the peak fuel power will not produce fuel centerline melting.

3. The core will not operate during normal operation or anticipated operational occurrences, with a power distribution that will cause the departure from nucleate boiling ratio to fall below 1.3 (W-3 correlation with modified spacer grid effect).

The applicant has described the manner in which the core will be operated and power distributions monitored so as to assure that these limits are met. The core will be operated in the constant axial offset control mode which has been shown to result in peaking factors less than 2.32 for both constant power and load following operation. A recently discovered error* in the loss-of-coolant accident analysis may lead to a requirement for operation with a peaking factor less than 2.32. In this event, operation at full power may be performed with the axial power distribution monitoring system. This mode of operation has been required in several operating Westinghouse-designed reactors and is acceptable. The requirement for this mode of operation will be inserted into the Technical Specifications, if required. Another option is the performance of a plant-specific analysis to support operation with a lower power peaking factor using excore monitoring.

Two types of instrumentation systems are provided to monitor core power distribution measurements - excore detectors which monitor core power, axial offset and azimuthal tilt, and incore detectors which permit detailed power distributions to be measured. These systems are used in operating reactors supplied by Westinghouse and we find their use acceptable for this facility.

Reactivity Coefficients

The reactivity coefficients are expressions of the effect on core reactivity of changes in such core conditions as power, fuel and moderator temperature, moderator density, and boron concentration. These coefficients vary with fuel burnup and power level. The applicant has presented values of the coefficients in the Final Safety Analysis Report and has evaluated the uncertainties of these values. We have reviewed the calculated values of reactivity coefficients and have concluded that they adequately represent the full range of expected values. We have reviewed the reactivity coefficients used in the transient and accident analyses and conclude that they conservatively bound the expected values, including uncertainties. Further, moderator and power Doppler coefficients along with boron worth are measured as part of the startup physics testing to assure that actual values are within those used in these analyses.

Control

To allow for changes in reactivity due to reactor heatup, load following, and fuel burnup with consequent fission product buildup, a significant amount of excess reactivity is built into the core. This excess reactivity is controlled by a combination of full-length control rods and soluble boron. Soluble boron is used to control reactivity changes due to:

*The error in the Zirconium-water reaction calculation discovered early in 1978.

1. Moderator density and temperature changes from ambient to operating temperatures.

2. Equilibrium xenon and samarium buildup.

3. Fuel depletion and fission product buildup - that portion not controlled by lumped burnable poison.

4. Transient xenon resulting from load following.

Control rods are used to control reactivity changes due to:

1. Moderator reactivity changes from hot zero to full power.

2. Fuel temperature changes (Doppler reactivity changes).

Burnable poison rods placed in some fuel assemblies are used for radial flux shaping and to control part of the reactivity change due to fuel depletion and fission product buildup.

The applicant has provided data to show that adequate control exists to satisfy the above requirements with enough additional control rod worth to provide a hot shutdown effective multiplication factor less than the design basis value of 0.982 during initial and equilibrium fuel cycles with the most reactive control rod stuck out of the core. In addition, the chemical and volume control system will be capable of shutting down the reactor by adding soluble boron and maintaining it shut down in the cold, xenon-free condition at any time in core life. These two systems satisfy the requirements of Criterion 26 of the General Design Criteria.

Comparisons have been made between calculated and measured control rod bank worth in operating reactors and in critical experiments. These comparisons lead to the conclusion that bank worths may be calculated to within approximately ten percent. In addition, bank worth measurements are performed as part of the startup test program to assure that conservative values have been used in safety analyses.

Based on these comparisons, we conclude that the applicant has made suitably conservative assessments of reactivity control requirements and that adequate control rod worths have been provided to assure shutdown capability.

Provision is made in the design for the use of part-length control rods. However, the applicant has informed us that the use of part-length rods is not presently contemplated. All analyses have been performed without part-length rods, and therefore the use of part-length rods will be prohibited.

Control Rod Patterns and Reactivity Worths

The full-length control rods are divided into two categories - shutdown rods and regulating rods. The shutdown rods are always completely out of the core when the reactor is at operating conditions. Core power changes are made with regulating rods which are nearly out of the core when it is operating at full power. Regulating rod insertion will be controlled by power-dependent insertion limits which will be established to assure that:

1. There is sufficient negative reactivity available to permit rapid shutdown of the reactor with adequate margin.

2. The worth of a control rod that might be ejected is not greater than that which has been shown to have acceptable consequences in the safety analyses.

We have reviewed the calculated rod worths and the uncertainties in these worths and conclude that rapid shutdown capability exists at all times in core life assuming the most reactive control rod assembly is stuck out of the core.

Stability

The stability of the core to xenon induced spatial oscillations is discussed in the Final Safety Analysis Report. The overall negative reactivity (power) coefficient provides assurance that the reactor will be stable against total power oscillation. It is also concluded that sustained radial or azimuthal xenon oscillations are not possible. This conclusion is based on measurements on an operating reactor of the same dimensions which showed stability against these oscillations.

This core is predicted to be unstable with respect to axial xenon oscillations after about 12,000 megawatt days per ton of exposure. The applicant has shown that axial xenon oscillations may be controlled by the regulating rods to prevent reaching any fuel damage limits.

Criticality of Fuel Assemblies

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer and storage facilities. The applicant has presented information on calculational techniques and assumptions used to assure that criticality is avoided. We have reviewed this information and the criteria which will be employed and find them to be acceptable.

Vessel Irradiation

Values are presented for the neutron flux in various energy ranges at mid-height of the pressure vessel inner boundary. Core flux shapes calculated by standard design methods are input to a transport theory calculation which results in a value of 2.9×10^{10} neutrons per square centimeter per second having energy greater than 10^6 electron-volts at the reactor vessel boundary. This results in a fluence of 2.9×10^{19} neutrons per square centimeter for a 40-year reactor vessel life with an 80 percent use factor. The methods used for these calculations are state of the art, and we conclude that acceptable analytical procedures have been used to calculate the reactor vessel fluence.

4.3.3 Analytical Methods

The applicant has described the computer programs and calculational techniques used to obtain the nuclear characteristics of the reactor design. The calculations consist of three distinct types, which are performed in sequence: determination of effective fuel temperatures, generation of macroscopic few-group parameters, and space-dependent few-group diffusion calculations. The computer programs used (e.g., LASER, TWINKLE, LEOPARD, TURTLE and PANDA) have been applied as part of the applications for most earlier Westinghouse-designed nuclear plant facilities and the predicted results have been compared with measured characteristics obtained during many startup tests for first cycle and reload cores. These results have validated the ability of these methods to predict experimental results. We, therefore, conclude that these methods are acceptable for use in calculating the nuclear characteristics of the core.

4.3.4 Summary

The applicant has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of the analyses to predict reactivity and physics characteristics of the reactor.

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity balances for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to make the reactor subcritical with an effective multiplication factor no greater than 0.982 in the hot condition at any time during the cycle with the most reactive control rod stuck in the fully withdrawn position.

On the basis of our review, we conclude that the applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control system to assure safe shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes available. We also conclude that nuclear design bases, features, and limits have been established in conformance with the requirements of Criteria 10, 11, 12, 13, 20, 25, 26, 27, and 28 of the General Design Criteria.

4.4 Thermal-Hydraulic Design

4.4.1 Thermal-Hydraulic Criteria and Design Bases

The safety criteria for the reactor core design as stated in Section 4.4.1 of the Final Safety Analysis Report are as follows:

1. "Fuel damage (defined as penetration of the fission product barrier; i.e., the fuel rod clad) is not expected during normal operation and operational transients (Condition I) or any transient conditions arising from faults of moderate frequency (Condition II). It is not possible, however, to preclude a very small number of rods damaged. These will be within the capability of the plant cleanup system and are consistent with the plant design bases.

2. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged (see above definition) although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

3. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events."

These safety criteria are implemented through the thermal-hydraulic design bases for departure from nucleate boiling ratio, fuel temperature, and hydrodynamic stability.

The margin to departure from nucleate boiling at any point in the core is expressed in terms of the departure from nucleate boiling ratio which is defined as the ratio of the heat flux required to produce departure from nucleate boiling at the calculated local coolant conditions to the actual local heat flux.

The thermal-hydraulic design basis for departure from nucleate boiling ratio in Section 4.4.1.1 of the Final Safety Analysis Report is as follows:

Departure from nucleate boiling will not occur on at least 95 percent of the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition I and II events) at a 95 percent confidence level. Historically, this has been conservatively met by limiting the minimum departure from nucleate boiling ratio to 1.30 and for this application a minimum departure from nucleate boiling ratio of 1.30 will continue to be used.

The fuel temperature design basis in Section 4.4.1.2 of the Final Safety Analysis Report is as follows:

During modes of operation associated with Condition I and Condition II events, the maximum fuel temperature shall be less than the melting temperature of uranium dioxide. The uranium dioxide melting temperature for at least 95 percent of the peak kilowatts per foot fuel rods will not be exceeded at the 95 percent confidence level. The melting temperature of uraium dioxide is taken as 580 degrees Fahrenheit unirradiated and decreasing 58 Fahrenheit degrees per 10,000 megawatt days per metric ton.

The hydrodynamic stability design basis in Section 4.4.1.4 of the Final Safety Analysis Report is as follows:

Modes of operation associated with Condition I and II events shall not lead to hydrodynamic instability.

The safety criteria and the thermal-hydraulic design bases are based upon the classification of events specified in the American National Standards Institute document ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." This classification recognizes the transient resulting from loss of power to all three reactor coolant pumps to be a Condition III event. The applicant does not consider this transient to be subject to the design bases stated above. We consider this transient to be an anticipated operational occurrence, as defined by Appendix A to 10 CFR Part 50, and require that it meet the safety criteria which have been specified for faults of moderate frequency, or Condition II events. We reviewed the analysis provided in Section 15 of the Final Safety Analysis Report and concluded that it meets the Condition II safety criteria.

The thermal-hydraulic design basis for departure from nucleate boiling has been reviewed and found to be one of the acceptable methods listed in Section 4.4. of the Standard Review Plan. The applicant has proposed to implement the thermal hydraulic design basis for departure from nucleate boiling through the use of the Westinghouse W-3, R-grid correlation. This correlation has been reviewed and approved by the NRC staff as described in "Topical Report Evaluation - WCAP-8536, "Critical Heat Flux Testing of 17 x 17 Fuel Assembly Geometry with 22-Inch Grid Spacing," December 10, 1976.

An important parameter that influences the departure from nucleate boiling ratio calculations is rod-to-rod bowing within fuel assemblies. Only limited experimental data on the extent of bowing in the 17 x 17 fuel design are available. However, an acceptable method based on data obtained with the 15 x 15 fuel design is available at this time. The applicant has provided a commitment in response to our request number 221.11 to comply with our interim position on fuel rod bowing "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," February 16, 1977.

In steady-state, two-phase flow in parallel channels, the potential for hydrodynamic stability always exists. For years, Westinghouse has used the HYDNA code to predict the inception of hydrodynamic instability for its reactors. The HYDNA code assumes that the core consists of parallel closed channels. Westinghouse performed experiments intended to demonstrate that flow in parallel open channels is more stable than in parallel closed channels. Westinghouse's experimental data were provided in Topical Report WCAP-7240, "An Experimental Investigation of the Effect of Open Channel Flow on Thermal Hydrodynamic Instabilities." This report did not describe the HYDNA code or details of its use in reactor calculations. We reviewed the topical report and concluded that, while the experimental data are useful as background information, they alone are not sufficient to support a conclusion that the HYDNA code conservatively predicts the onset of flow instability in the core. We also concluded that the experiments described by Westinghouse in support of the hydrodynamic design are not sufficient to justify that the design basis is satisfied.

The applicant has submitted additional information in support of the thermal hydrualic stability of the reactor in response to our request number 221.12. The discussion describes alternate analyses, not related to the HYDNA code for predicting the onset of hydraulic instability. The applicant has concluded that a power margin greater than twice the rated power exists to the predicted inception of instabilities.

We are presently reviewing the submitted material as part of a generic study of the hydrodynamic stability characteristics of light water reactors under normal operation, anticipated transients and accident conditions under Task Action Plan B-19, "Thermal Hydraulic Stability." The results of this study will be applied to our review and acceptance of stability analyses and analytical methods now in use by the reactor vendors. In the interim, we conclude that past operating experience, stability tests and the inherent thermal hydraulic characteristics of light water reactors provide a basis for accepting the stability evaluation for normal operation and anticipated transient events for this facility. Any required actions resulting from our study will be applied to this facility.

The design basis on fuel temperature is implemented through the reactor protection system overpower trip setpoints. These setpoints will be selected to assure that the calculated fuel centerline temperature does not exceed 4700 degrees Fahrenheit.

We conclude that the thermal-hydraulic criteria and design bases for this facility are acceptable.

4.4.2 Thermal-Hydraulic Analytical Models

For the design of this facility as well as other Westinghouse-designed reactors which we have recently reviewed, the THINC computer code has been used to calculate core thermal-hydraulic performance characteristics. The THINC code considers cross-flow between adjacent assemblies in the core and cross-flow and thermal diffusion between adjacent subchannels in the assemblies.

The THINC code is described in Westinghouse Topical Reports WCAP-7956, "THINC-IV - An Improved Program for Thermal and Hydraulic Analysis of Rod Bundle Cores," and WCAP-8054, "Application of the THINC-IV Program to PWR Design," We have completed our review of these reports and conclude that the THINC-IV code is acceptable as described in "Staff Evaluation of WCAP-7956, WCAP-8054, WCAP-8507, and WCAP-8762," J. Stolz to C. Eicheldinger, April 19, 1978).

Crud deposition in the core and an associated change in core pressure drop and flow have been observed on some pressurized water reactors. We reviewed the input assumptions used in the facility design and questioned the treatment of possible crud buildup in the core. The applicant replied that: 1) operating experience from several Westinghouse reactors indicates very low levels of crud buildup on the core; 2) some margin for uniform crud buildup is included in the clad surface roughness factor used in their analysis; and 3) significant changes in core pressure drop and flow would be observed during periodic core flow measurement. We have reviewed this information and the list of instrumentation to detect significant changes in core flow and concluded that it adequately addresses our concerns relative to crud deposition in the core. The Technical Specifications will provide appropriate considerations for detection and actions relevant to significant crud deposition.

We also questioned the effect of a radial pressure gradient at the core exit on the thermal-hydraulic design. We first raised this matter as a result of our review of the Westinghouse 1/7 scale hydraulic tests which showed a radial pressure gradient in the upper plenum. The analyses assume a uniform core outlet pressure distribution. In response to our question on the radial pressure gradient, the applicant referenced a sensitivity study with the THINC-IV code for a reactor with a 193 assembly core and has presented an argument that the effect would be even smaller for the design of its facility, which has a 157 assembly core. We have reviewed this information and concluded it is acceptable.

4.4.3 Thermal-Hydraulic Design Comparison

The thermal-hydraulic design parameters for the reactor are listed in Table 4-1 of this Safety Evaluation Report. A comparison of these parameters with those of the Koshkonong design was given in the Final Safety Analysis Report.

The design parameters of the two plants are identical except for the allowable linear heat generation rate which is lower for Virgil C. Summer Nuclear Station, Unit 1 as a result of more stringent limits resulting from the loss of coolant accident analysis.

We have compared the Virgil C. Summer Nuclear Station, Unit 1 design parameters with those of North Anna Units 1 and 2 since we consider it more appropriate to compare it to a design which has been approved for an operating license. This comparison is provided in Table 4-1. We have reviewed the differences in flow and inlet temperature and have found that these differences are consistent with the difference in minimum departure from nulceate boiling ratio. The comparisons were done using sensitivity factors supplied by Westinghouse and previously accepted.

4.5 Reactor Materials

4.5.1 Reactor Vessel Internals Materials

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

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

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

The controls imposed upon components constructed of austenitic stainless steel, as used in the reactor internals, satisfy the recommendations of the NRC staff interim position MTEB 5-1 on Regulatory Guide 1.31, "Control of Ferrite Content of Stainless Steel Weld Metal," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Material selection, fabrication practices, examination procedures, and protection procedures performed in accordance with these recommendations provide reasonable assurance that the austenitic stainless steel used for reactor internals will be in a metallurgical condition which precludes susceptibility to stress corrosion cracking during service. The use of materials proven to be satisfactory by actual service experience and conformance with the recommendations of these regulatory guides constitutes an acceptable basis for meeting in part the requirements of Criteria 1 and 14 of the General Design Criteria.

4.5.2 Control Rod System Structural Materials

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

The controls imposed upon the austenitic stainless steel of the system satisfy the recommendations of the NRC staff interim position MTEB 5-1 on Regulatory Guide 1.31, "Control of Stainless Steel Welding," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices performed in accordance with these recommendations provide added assurance that stress corrosion cracking will not occur during the design life of the components.

The compatibility of all control rod system materials that are in contact with the reactor coolant satisfies the criteria for Articles NB-2160 and NB-3120 of Section III of the ASME Code. Both martensitic and precipitation- hardening stainless steels have been given tempering or aging treatments in accordance with NRC staff positions. Cleaning and cleanliness control are in accordance with ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

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

4.6 Functional Design of Reactivity Control Systems

The functional designs of the reactivity control systems for the facility have been reviewed to confirm that the systems have the capability to shut down the reactor with appropriate margin during normal, abnormal, and accident conditions. The reactivity control systems reviewed included the control rod drive system and the chemical and volume control system (see also Section 9.3.4 of this Safety Evaluation Report for additional discussion). The scope of our review included layout drawings and descriptive information for the systems and for the supporting systems that are essential for operation of the systems.

The chemical and volume control system regulates the concentration and makeup of the boric acid solution in the reactor coolant system used to control reactivity. A portion of the chemical and volume control system (the centrifugal charging pumps, the boron injection tank), injects a high concentration boron solution into the reactor coolant system to assure facility shutdown in the event of accidents. The boric acid concentration in the reactor coolant system is controlled by the boron thermal regeneration system and by the reactor makeup subsystem of the chemical and volume control system.

The concentration of boron in the reactor coolant system is changed manually for the following operating conditions:

1. Startup-boron concentration decreased.

2. Load follow-boron concentration increased or decreased to compensate for xenon transients following load change.

3. Fuel burnup-concentration decreased to compensate for burnup.

4. Cold shutdown-boron concentration increased to prevent return to power.

The rod cluster control assemblies are the main shutdown mechanism in the event of most transients. The rods contain a silver-indium-cadmium alloy. In the event of an accident, the rod cluster control assemblies are inserted automatically. Concentrated boric acid solution is injected by the emergency core cooling system in the event of a loss-of-coolant accident or steam line break, thereby satisfying the requirements of Criterion 20 of the General Design Criteria.

The chemical and volume control system can maintain the reactivity of the reactor within required bounds by means of the automatic makeup system to replace minor leakage without significantly changing the boron concentration in the reactor coolant system. Dilution of the reactor coolant system boron concentration required for the reactivity losses occurring as a result of fuel and burnable poison depletion is accomplished by operator action.

The applicant has stated that a single failure will not result in loss of the protection system nor will a loss of redundancy occur as a result of removal of a channel or components service. The control rod drive mechanisms utilized in the facility are essentially identical to those supplied on previously reviewed Westinghouse plants. A functional test program has been conducted on a full scale prototype assembly under simulated conditions of reactor temperature, pressure and flow for 1000 hours which included 3,000,000 steps and 600 trips without failure. All the control rod drives for the facility are production tested prior to shipment to confirm their ability to meet the design specification operational requirements. In addition, preoperational trip time tests will be performed to verify that the control rods will insert within the time requirements identified in the Technical Specifications. This satisfies the recommendations of Regulatory Guide 1.68 with regard to the control rod drive system.

Checks of rod movement also will be made on every full-scale rod cluster control assembly periodically during the reactor operation. Rod cluster control assembly drop tests will be performed at each refueling shutdown to demonstrate the ability of the rod clusters to meet required drop times. The foregoing periodic testing, reliability, and redundancy conforms to the requirements of Criterion 21 of the General Design Criteria.

The vulnerability of the trip system to common mode failures has been analyzed in Topical Reports WCAP 7306, 7706 and 7486. As a result of these studies Westinghouse has concluded that the high reliability and functional diversity of the Westinghouse reactor protective system makes complete failure to trip on demand during an anticipated transient not credible. We have completed our review of these reports and published our evaluation in a report titled, "Status Report on Westinghouse Analyses of Anticipated Transients Without Scram" dated December 9, 1975.

After review of these reports and subsequent studies on trip systems, we have concluded that regardless of the high reliability of the current trip systems, protection from anticipated transients without scram events must be provided.

A further discussion regarding this subject matter is provided in Section 15.3.5 of this Safety Evaluation Report.

Failure of electrical power to a rod cluster control assembly will cause insertion of that assembly as will shearing of the connection between the rod cluster control assembly and control rod drive mechanism. Single failure of a rod cluster control assembly is considered in transient and accident analyses which includes the most reactive rod cluster control assembly stuck outside the core. Analysis of accidental withdrawal of a rod cluster control assembly is found to have acceptable results. This conforms to the requirements of Criteria 23 and 25 of the General Design Criteria.

The applicant has stated that control rod drive mechanisms and latches are designed with sufficient clearances for thermal expansion so that a loss of forced air cooling for an indefinite period will not interfere with a tripped mechanism. The forced air cooling is primarily provided to limit degradation of the control rod drive mechanism coils over prolonged operation and assure that the commercial design life is met. A failure of one or more coils does not impair trip since the reactor is tripped by de-energizing the coils which allows the rods to drop. We find this design feature of the control rod drive mechanism to be acceptable.

Soluble poison concentration is used to control normal operating reactivity changes. If necessary, rod cluster control assembly movement can also be used to accommodate such changes but is used mainly to control anticipated operational occurrences even with a single malfunction, such as a stuck rod. In either case, fuel design limits were not exceeded. The soluble poison control is capable of maintaining the core subcritical under conditions of cold shutdown, which conforms to the requirements of Criterion 26 of the General Design Criteria.

The reactivity control systems, including the addition of concentrated boric acid solution by the emergency core cooling system, are capable of controlling all anticipated operational changes, transients, and accidents, including the full spectrum of loss-of-coolant accidents. All accidents are calculated with the assumption that the most reactive rod cluster control assembly is stuck and cannot be inserted, which complies with the requirements of Criterion 27 of the General Design Criteria.

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited so as to preclude rupture of the coolant pressure boundary or disruption of the core internals to a degree which would impair core cooling capability due to any postulated reactivity accident (rod injection, steam line break, etc.), which complies with the requirements of Criterion 28 of the General Design Criteria.

In summary, the basis for our acceptance is conformance of the applicant's designs, design criteria, and design bases for the reactivity control systems and their supporting systems to the NRC's regulations as set forth in the General Design Criteria. We conclude that the designs of the reactivity control systems conform to all applicable regulations and are acceptable.

5 REACTOR COOLANT SYSTEM

5.1 Summary Description

The reactor coolant system consists of three similar heat transport loops connected to the reactor pressure vessel. Each loop contains a reactor coolant pump, steam generator, and associated piping. In addition, the system includes a pressurizer, a pressurizer relief tank, interconnecting piping, and instrumentation necessary for operational control. All of these components are located within the containment building.

During operation, the reactor coolant system transfers the heat generated in the core to the steam generators where steam is produced to drive the turbine-generator. Borated demineralized water is circulated in the reactor coolant system at a flow rate and temperature consistent with achieving the required reactor core thermal-hydraulic performance. The coolant also acts as a neutron moderator and reflector, and as a solvent for the neutron absorbing boric acid used for chemical shim control.

The reactor coolant system pressure boundary provides a second barrier against the release of radioactivity generated within the reactor and is designed to assure a high degree of integrity throughout the life of the facility.

The reactor coolant system pressure changes during normal operation are controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters and water spray. Spring-loaded safety valves and power-operated relief valves are mounted on the pressurizer and discharge to the pressurizer relief tank where steam is condensed and cooled by mixing with water.

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary Components

Components of the reactor coolant pressure boundary as defined by the rules of Section 50.55a of 10 CFR Part 50 have been properly identified and classified as ASME Section III, Class 1 components in Table 5.2-1 of the Final Safety Analysis Report. Those components within the reactor coolant pressure boundary are constructed in accordance with the requirements of the applicable codes and addenda as specified by the rules of Section 50.55a of 10 CFR Part 50.

We conclude that construction of the components of the reactor coolant pressure boundary in conformance with the ASME Code and the NRC's regulations is expected to result in a component quality commensurate with the importance of the safety function of the reactor coolant pressure boundary, and is acceptable.

The ASME Code Cases whose requirements have been applied in the construction of pressure- retaining ASME Section III, Class 1 components within the reactor coolant pressure boundary (Quality Group Classification A) are acceptable to the NRC staff except for Code Case 1528-1 which was used in the manufacture of the steam generators for the facility. This revision of Code Case 1528-1 is not acceptable for general use in the construction of ASME Section III, Class 1 components.

In order to demonstrate the adequacy of the forging material used in the manufacture of the steam generators, the applicant has conducted a test program and provided additional data which we find to be acceptable.

We conclude that compliance with the requirements of these code cases, in conformance with the NRC's regulations, is expected to result in a component quality level that is commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

5.2.2 Overpressurization Protection

The pressure relief system prevents overpressurization of the reactor coolant pressure boundary under the most severe transients and limits the reactor pressure during anticipated operational occurrences. Overpressure protection for the reactor coolant pressure boundary is accomplished by utilizing three spring-loaded safety valves and three power-operated relief valves located on the pressurizer. The safety valves have a bellows arrangement that compensates for backpressure. The steam release from these valves discharges to the pressurizer quench tank through a common header from the pressurizer. The reactor coolant system safety valves, in conjunction with the steam generator safety valves, and the reactor protection system, protect the reactor coolant system against overpressure, limited to 110 percent of the design pressure of 2485 pounds per square inch gauge, following a complete loss of steam flow to the turbine. The relief valves, which have a setpoint pressure of 2335 pounds per square inch gauge, are designed to limit system pressure to a value below the safety valve setpoints to prevent excessive safety valve opening. The pressurizer spray system is designed to maintain the reactor coolant system pressure below the relief valve setpoints during a step reduction in power level of up to 10 percent. The relief valves also limit the pressurizer pressure to a value below the high pressure reactor trip setpoint of 2385 pounds per square inch gauge for all design transients up to and including the design percentage step load decrease with steam dump; however, credit is taken only for safety valves in analyzing anticipated operational occurrences and accidents.

The safety valves and the power operated relief valves are not designed for two-phase or subcooled liquid relief. However, the applicant in order to satisfy the requirement of TMI item II.D.1 (See Section 22.2 of this Safety Evaluation Report has committed to an Electric Power Research Institute program to test the valves and confirm their capability for two-phase or subcooled liquid relief for all conditions under which this relief would be expected, including low pressure. The applicant will be required to submit the test results and confirm their applicability to the Virgil C. Summer Nuclear Station, Unit 1. We will report the results of our evaluation in a supplement to this Safety Evaluation Report.

Westinghouse Topical Report WCAP-7769, Revision 1, was referenced as the basis for the design requirements of the overpressure protection system for the Virgil C. Summer Nuclear Station, Unit 1. In WCAP-7769, Revision 1, the overpressure analyses were performed in two major parts. The first case considered a complete loss of steam flow and assumed main feedwater flow maintained with no credit taken for reactor trip. This case was performed strictly as a conservative method of sizing the pressurizer safety valves based on maximum surge rate. The second case involved taking credit for reactor trip and a complete loss of steam flow with a simultaneous loss of all feedwater. This analysis was performed to verify the adequacy of the sizing method. The assumptions used in the overpressure analysis for the sizing and verification of performance adequacy of the pressurizer safety valves included taking no credit for operation of reactor coolant system power-operated relief valves, steam line power relief valves, steam dump system, reactor coolant system pressurizer level control system, and pressurizer spray.

WCAP-7769, Revision 1, shows that for the analyzed complete loss of steam flow transient with a simultaneous loss of all feedwater, and credit taken for reactor trip on reactor coolant temperature differential (the second safety-grade trip signal), the peak pressurizer safety valve flow capacity would be 86 percent of rated. This analyzed event is consistent with Section 5.2.2 of the Standard Review Plan which requires the use of the high pressure trip signal or the second safety grade trip signal, whichever is later, for relief valve sizing. Although, the margin for overpressure predicted in WCAP-7769, Revision 1, is acceptable, our review of this report has not been completed. The analyses in WCAP-7769, Revision 1, were performed for a four-loop, 3423 thermal megawatts plant compared to the Virgil C. Summer Nuclear Station, Unit 1 which is a three-loop plant. The use of WCAP-7769, Revision 1, is justified since the ratio of available pressurizer safety valve capacity to peak surge rate into the pressurizer during the sizing transient is greater for a three-loop plant than for the four-loop plant.

The analyses in WCAP-7769, Revision 1, are consistent with the Final Safety Analysis Report in terms of the initial condition assumed for power, i.e., 102 percent of the licensed power level. The Virgil C. Summer Nuclear Station, Unit 1 has a peak surge rate ratio of 1.107 (43.3 cubic feet per second relief rate compared to a 39.1 cubic feet per second pressurizer surge rate used in WCAP-7769, Revision 1. The staff finds that WCAP-7769, Revision 1, provides a conservative calculation for the sizing of the relief valves for overpressurization protection.

Incidents of reactor vessel overpressurization in pressurized water reactors have been reported during startup and shutdown in which the limitations of Appendix G to 10 CFR Part 50 have been exceeded. The applicant recognized this concern on overpressurization when the reactor coolant system is water solid and has provided an automatic reactor coolant system pressure control to maintain pressures within allowable limits during low temperature operation. This feature is provided by incorporating an independent actuation logic to each of the two nitrogen-operated pressurizer power-operated relief valves. The system logic will continuously monitor reactor coolant system temperature and pressure conditions whenever plant operation is at a temperature below referenced nil ductility temperature and will actuate a signal to open the power-operated relief valves when required to prevent pressure-temperature conditions from exceeding allowable limits. This system will be testable and the electrical power supply to the power-operated relief valves' control circuits will be independent of offsite power.

Two power-operated relief valves are supplied with an independent, seismically designed supply of nitrogen which is sized to asure that no operators action is required to terminate an abnormal transient in 10 minutes. The applicant has stated that the basis for this 10 minute limit is that there are sufficient indications available inside the control room for the operator to identify and terminate the cause of the transient in 10 minutes and the Appendix G to 10 CFR Part 50 limits will be maintained for upset conditions during low temperature operation assuming water-solid operation. We find this reactor coolant pressure boundary overpressurization protection system design acceptable. However, we will require the applicant to comply with any changes recommended by the final revision of WCAP-7769.

5.2.3 Materials

Material Specifications and Compatibility with Reactor Coolant

The materials used for construction of components of the reactor coolant pressure boundary, including the reactor vessel and its appurtenances, have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code. Special requirements of the applicant with regard to control of residual elements in ferritic materials have been identified and are considered acceptable.

The reactor coolant pressure boundary materials of construction that will be exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion of all materials except carbon and low alloy steel will be negligible. For these materials, conservative corrosion allowances have been provided for all exposed surfaces of carbon and low alloy steel in accordance with the requirements of Section III of the ASME Code.

The external non-metallic insulation used on austenitic stainless steel components conforms with the requirements of Regulatory Guide 1.36, "Non-metallic Thermal Insulation for Austenitic Stainless Steels."

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

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

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

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

The controls imposed on welding preheat temperatures and weld cladding satisfy the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," and Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steels." These recommendations provide reasonable assurance that cracking of components made from low alloy steels will not occur during fabrication and will minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment.

The welding procedures used for ferritic steels in limited access areas satisfy the recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility." The ultrasonic method for examination of ferritic steel tubular products meets the requirements of the ASME Code. The fabrication practices and examination procedures performed in accordance with these recommendations provide reasonable assurance that welds in the reactor coolant pressure boundary will be satisfactory in locations of restricted accessibility and that unacceptable defects in components of the reactor coolant pressure boundary will be detected regardless of shape, size, or orientation.

Conformance with the ASME Code and the regulatory guides mentioned constitutes an acceptable basis for meeting the requirements of Criteria 1 and 14 of the General Design Criteria.

Fabrication and Processing of Austenitic Stainless Steel

Within the reactor coolant pressure boundary, no components of austenitic stainless steel have a yield strength exceeding 90,000 pounds per square inch, in accordance with the NRC staff position.

The controls imposed upon components constructed of austenitic stainless steel used in the reactor coolant pressure boundary and for the reactor vessel and its appurtenances satisfy the recommendations of NRC staff interim position MTEB 5-1 on Regulatory Guide 1.31, "Control of Stainless Steel Welding," Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants," and Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility."

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

Fracture Toughness of Class 2 Components

We have reviewed the requirements for fracture toughness testing and properties that the applicant met to provide assurance that the pressure-retaining ferritic materials of Code Class 2 components will have adequate toughness. The ferritic materials are specified to meet the toughness requirements of the ASME Code.

The fracture toughness tests and properties required by the ASME Code provide reasonable assurance that safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for the pressure-retaining ferritic materials of ASME Code Class 2 components.

5.2.4 Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

Criterion 32 of the General Design Criteria requires, in part, that components which are part of the reactor coolant pressure boundary be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity.

To assure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be periodically inspected at the facility. The design of the ASME Code Class 1 and 2 components of the reactor coolant pressure boundary in the facility incorporates provisions for access for inservice inspection in accordance with Section XI of the ASME Code. Methods have been developed to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel.

Section 50.55a(g) of 10 CFR Part 50 defines the detailed requirements for the preservice and inservice inspection programs for light water cooled nuclear power facility components. Based upon a construction permit date of March 21, 1973, this section of the Code of Federal Regulations requires that a preservice inspection program be developed and implemented using at least the edition and addenda of Section XI of the ASME Code in effect six months prior to the date of issuance of the construction permit. Also, the initial inservice inspection program must comply with the requirements of the latest edition and addenda of the ASME Code in effect 12 months prior to the date of issuance of the operating license. The applicant has made a commitment to meet the preservice and inservice inspection requirements of Section 50.55a(g) of 10 CFR Part 50. The preservice inspection program is currently under review by the NRC staff and is based upon the 1974 Edition of Section XI of the ASME Code through the Summer 1975 Addenda. Our evaluation of the preservice inspection program will be included in a supplement to this Safety Evaluation Report. The inservice program will be evaluated after the applicable ASME Code edition and addenda have been determined and before the initial inservice inspection.

The conduct of periodic inspections and hydrostatic testing of pressure-retaining components of the reactor coolant pressure boundary in accordance with the requirements of Section XI of the ASME Code and 10 CFR Part 50 will provide reasonable assurance that evidence of structural degradation or loss of leaktight integrity occurring during service will be detected in time to permit corrective action before the safety functions of a component are compromised. Compliance with the inservice inspections required by the ASME Code and 10 CFR Part 50 constitutes an acceptable basis for satisfying the inspection requirements of Criterion 32 of the General Design Criteria.

5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary

We have reviewed the reactor coolant pressure boundary leakage detection systems proposed by the applicant as described in the Final Safety Analysis Report. The areas of our review included the proposed leakage detection methods for continuous monitoring of both identified and unidentified leakage rates, intersystem leakage, leakage detection system sensitivity and response times, seismic capability of systems, indicators and alarms, and testability.

Identified Leakage

Identified leakage is collected in the pressurizer relief tank and in the reactor coolant drain tank. This includes all anticipated reactor coolant pressure boundary leakage excluding that to other systems. Included in this identified leakage is fluid from the following areas: (1) pressurizer safety relief valves, (2) reactor vessel head gasket, (3) reactor coolant pump seals, (4) excess letdown heat exchanger drain, and (5) flange seals and valve steam leakoff. All leakages, with the exception of (1), above, are collected in the reactor coolant drain tank. Any leakage from the pressurizer safety valves will collect in the pressurizer relief tank. Changing levels in the pressurizer relief tank and reactor coolant drain tank will be used to measure the identified leakage rate from these sources. Leakage between the double O-ring in the reactor vessel main flange will be detected by a temperature detector in the leakoff line. Collection of identified leakage in closed containers and measurement of its flow rate independent of the unidentified leakage satisfy the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection."

Unidentified Leakage

Unidentified leakage from the reactor coolant pressure boundary inside containment is detected by three primary methods: 1) the containment atmospheric particulate radioactivity monitoring system, 2) the reactor building cooler condensate drain flow monitoring system, and 3) the reactor building sump monitoring system. Reactor building temperature and pressure monitors, low pressurizer level indication or unexplained discrepancies between net letdown and makeup flow are also available to provide less sensitive indications of reactor coolant leakage.

The leakage detection sump is a 30 gallon capacity sump located within a larger reactor building drainage sump. Level indicators in the sump, which sound an alarm in the control room are positioned such that the time to identify a one-gallon per minute leak is 15 minutes from the time the first alarm is sounded. The common drain headers from each of the reactor building condensate coolers have a flow switch set to actuate an alarm in the control room if a flow rate exceeding 0.5 gallon per minute occurs. This system has been designed to detect a one gallon per minute leak in approximately 15 minutes. Both the sump and the cooler monitoring systems meet the recommendations of Regulatory Guide 1.45 with regard to the capability to detect a leak rate of one gallon per minute in less than one hour and are therefore acceptable.

The airborne particulate radiation monitor system has been designed to detect a one gallon per minute leak in one hour based on the presence of a normal background level in the reactor building resulting from very small reactor coolant pressure boundary leaks. This capability is consistent with previously accepted plants and is acceptable to the NRC staff.

In accordance with the recommendations of Regulatory Guide 1.45 and Criterion 30 of the General Design Criteria, the applicant has stated that unidentified leakage detection systems are capable of performing their function following seismic events that do not require shutdown of the facility, and the airborne particulate radioactivity monitoring system can withstand a safe shutdown earthquake. The acceptability of the seismic qualification of seismic Category I instrumentation and electrical equipment is addressed in Section 3.10 of this Safety Evaluation Report.

In summary, based on the information provided by the applicant, the types of unidentified leakage detection systems, their sensitivities and their seismic qualifications are in compliance with the recommendations of Regulatory Guide 1.45 and are acceptable.

Intersystem Leakage

The applicant has identified the possible intersystem leakage paths which included: the safety injection system, accumulators, residual heat removal system, secondary system (steam generators) and the component cooling water system. The detection methods used for determination of intersystem leakage from each of the leakage paths were identified. These methods include various combinations of pressure, temperature, flow and level sensors, lifting of relief valves, radiation monitors, sampling and inventory balances. The intersystem leakage detection capabilities provided for the facility are in agreement with the recommendations of Regulatory Guide 1.45. Additional intersystem leakage detection capability will be available by means of periodic testing of the emergency core cooling system components as discussed in Section 6.3 of this Safety Evaluation Report.

Indicators and alarms to monitor all three types of leakage have been provided in the control room. In response to our request, the applicant has provided a tabulation which lists all the control room, indicators and alarms and their associated leakage detection instrumentation for all three types of leakage monitoring features.

The applicant has stated that the leakage detection systems are designed to permit testing and instrument calibration and that the containment radiation monitoring systems have a radioactive source ("check source") built into the system to permit test and calibration during operation.

The leakage detection systems provided to detect leakage from components of the reactor coolant pressure boundary furnish reasonable assurance that structural degradation, which may develop in pressure-retaining components of the reactor coolant pressure boundary and result in coolant leakage during service, will be detected on a timely basis. The leakage detection systems assure that corrective actions can be made before such degradation could become sufficiently severe to jeopardize the safety of the system, or before the leakage could increase to a level beyond the capability of makeup systems to replenish the coolant loss. We have concluded that the systems are in compliance with the recommendations of Regulatory Guide 1.45 and satisfy the requirements of Criterion 30 of the General Design Criteria. We find the leakage detection systems to be acceptable for issuance of an operating license.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

Criterion 31 of the General Design Criteria requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to assure that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. Criterion 32 of the General Design Criteria, requires, in part, that the reactor coolant pressure boundary be designed to permit an appropriate material surveillance program for the reactor coolant pressure boundary.

We have reviewed the materials selection, toughness requirements, and extent of materials testing conducted by the applicant to provide assurance that the ferritic materials used for pressure-retaining components of the reactor coolant pressure boundary possess adequate toughness under operating, maintenance, testing and anticipated transient conditions. The ferritic materials were specified to meet the toughness requirements of Section III of the 1971 edition of the ASME Boiler and Pressure Vessel Code.

The guidelines specified for the fracture toughness requirements for the ferritic materials of the reactor coolant pressure boundary are defined in Appendix G and Appendix H to 10 CFR Part 50. The ferritic pressure boundary material of the facility was qualified by impact testing in accordance with Section III of the 1971 Edition of the ASME Code and evaluated in accordance with Appendix G to Section III of the 1971 Edition of the ASME Code and the 1972 Summer Addenda of the ASME Code. The edition and addenda of the ASME Code used by the applicant to qualify and evaluate the reactor coolant pressure boundary material meet the requirements of Section 50.55a of 10 CFR Part 50.

Compliance with Appendix G to 10 CFR Part 50

We have evaluated the applicant's Final Safety Analysis Report to determine the degree of compliance with the fracture toughness requirements of Appendix G to 10 CFR Part 50. Our evaluation indicates that the applicant has complied with Appendix G to 10 CFR Part 50, except for Paragraphs I and IV.A.3, which will remain open items and Paragraph III.B.4 for which the applicant has provided sufficient information to justify an exemption. Our evaluation of each of these areas follows:

1. Paragraph I states that the adequacy of the fracture toughness of ferritic materials used in the reactor coolant pressure boundary, having a minimum specified yield strength greater than 50,000 pounds per square inch, be demonstrated to the NRC on an individual case basis. Table 5.2-8 of the Final Safety Analysis Report indicates that SA-508 Class 2a and SA-533 Grade A Class 2 steels have been used in the reactor coolant pressure boundary. These steels have a specified minimum yield strength of 65,000 and 70,000 pounds per square inch, respectively.

As part of the demonstration of the adequacy of these materials, Westinghouse Topical Report WCAP-9292 has been submitted to the NRC staff for review. The purpose of this topical report is to demonstrate the adequacy of the subject materials to be described by the reference stress intensity factor curve of Appendix G to Section III of the ASME Code.

We have conducted our review of WCAP-9292 and the additional information provided by Westinghouse in the September 18, 1980 letter from T. Anderson to J. Miller of the staff and have found the report acceptable. Therefore, the applicant's use of WCAP 9292 satisfies the generic fracture toughness requirements for SA 508 Class 2a and SA 533 Class 2 material. To demonstrate compliance with the plant specific fracture toughness requirements of Appendix G to 10 CFR Part 50, the applicant has identified the pressurizer to be the only component in which SA 533 Class 2 material was used, and has also submitted results from both dropweight and Charpy V-notch tests. However, the applicant has not supplied any impact energy data for the high strength ferritic welds in the pressurizer. Before compliance with Paragraph I of Appendix G to 10 CFR Part 50 can be determined, the applicant must identify all high strength ferritic welds and submit the necessary fracture toughness test results for these welds. We will report the resolution of this matter in a supplement to this Safety Evaluation Report.

2. Paragraph III.B.4 requires that the testing personnel shall be qualified by training and experience and should be competent to perform the tests in accordance with written procedures. For Virgil C. Summer Nuclear Station, Unit 1 component testing, no written procedures were in existence as required by the later regulation; however, the applicant has supplied sufficient information to demonstrate that the intent of Paragraph III.B.4 has been met. The applicant has stated that individuals who conducted the testing were qualified by education, training, and years of experience and were certified by qualified supervisory personnel. Because these tests are relatively routine in nature, are continually being performed in the laboratory, and were conducted by qualified, experience personnel, we conclude that it is unlikely that the tests were conducted improperly. Consequently, we conclude that an exemption for not performing the tests in accordance with written procedures is justified.

3. Paragraph IV.A.2.a requires that an initial reference temperature, RT_{NDT}, be defined and used as a basis for providing adequate margins of safety for reactor operation. Values of RT_{NDT} are based on the fracture toughness properties of the ferritic pressure-retaining materials of the reactor coolant pressure boundary. RT_{NDT} is defined by the ASME Code as the higher of either a) the nil ductility temperature, as defined by the dropweight test, or b) a temperature of 60 degrees Fahrenheit less than the temperature at which the material exhibits 50-foot pounds energy and 35 mils lateral expansion, whichever is higher, by the Charpy impact tests. The Charpy impact tests must be conducted using specimens oriented in the transverse direction.

Impact testing of the ferritic material was conducted to meet the requirements of the 1971 ASME Code, Section III, through 1971 Winter Addenda. According to our evaluation of impact test results presented in the Final

Safety Analysis Report, the applicant has not submitted any impact energy data for the ferritic pressure-retaining materials (including base, weld, and heat-affected zone material) of the steam generator, and consequently we cannot verify that adequate margins of safety exist for reactor operation. The applicant has stated this information will be submitted in an amendment dated November 17, 1980. However, until such information is received and evaluated, compliance with Paragraph IV.A.2.a, Appendix G, will remain an open item. We will report the resolution of this matter in a supplement to this Safety Evaluation Report.

4. Paragraph IV.A.3 requires that ferritic materials for bolting and other fasteners within the reactor coolant pressure boundary meet the fracture toughness requirements of Paragraph NB-2333 of the ASME Code. In order to demonstrate compliance with this requirement, the applicant must submit the data identified to be in the quality assurance data packages at the Virgil C. Summer Nuclear Station, Unit 1. We cannot complete our evaluation of compliance with Paragraph IV.A.3 until the applicant supplies this data. We will report the resolution of this matter in a supplement to this Safety Evaluation Report.

Compliance with Appendix H to 10 CFR Part 50

The toughness properties of the reactor vessel beltline materials must be monitored throughout the service life of the facility by a materials surveillance program that meets the requirements of ASTM Standard E-185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactors," and Appendix H to 10 CFR Part 50. We have evaluated the information in the applicant's Final Safety Analysis Report for degree of compliance with these requirements and conclude that the applicant has met all of the requirements of Appendix H to 10 CFR Part 50.

The materials surveillance program at the facility will be used to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region, resulting from exposure to neutron irradiation and the thermal environment. Under the applicant's surveillance program, fracture toughness data will be obtained from material specimens that are representative of the limiting base, weld, and heat-affected zone materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture toughness throughout its service life.

Conclusions for Compliance with Appendices G and H to 10 CFR Part 50

Based on our evaluation of compliance with Appendices G and H to 10 CFR Part 50, we conclude that the applicant has met all the material surveillance program requirements of Appendix H, but has not met all the fracture toughness requirements of Appendix G. The areas of noncompliance include Paragraph III.B.4 for which the applicant has supplied sufficient information to justify an exemption, and Paragraphs I, IV.A.2.a, and IV.A.3, which will remain open items until the applicant submits the necessary data and analyses.

Appendix G, "Protection Against Non-Ductile Failure," to Section III of the ASME Boiler and Pressure Vessel Code, will be used, together with the fracture toughness test results required by Appendices G and H to 10 CFR Part 50, to calculate the reactor coolant pressure boundary pressure-temperature limitations for the Virgil C. Summer Nuclear Station, Unit 1.

The fracture toughness tests required by the ASME Code and by Appendix G to 10 CFR Part 50 will provide reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating fracture can be established safe operating procedures, and use of the results of the fracture toughness tests performed in accordance with the ASME Code and NRC regulations, will provide adequate safety margins during operating, testing, maintenance, and anticipated transient conditions. Compliance with these Code provisions and NRC regulations constitutes an acceptable basis for satisfying the fracture toughness requirements of Criterion 31 of the General Design Criteria.

The materials surveillance program, required by Appendix H to 10 CFR Part 50, will provide information on material properties and the effects of irradiation on material properties so that changes in the fracture toughness of the material in the Virgil C. Summer Nuclear Station, Unit 1 reactor vessel beltline caused by exposure to neutron radiation can be properly assessed, and adequate safety margins against the possibility of vessel failure can be provided.

Compliance with ASTM E-185-73 and Appendix H to 10 CFR Part 50 assures that the surveillance program constitutes an acceptable basis for monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and satisifies the materials surveillance requirements of Criteria 31 and 32 of the General Design Criteria.

5.3.2 Pressure-Temperature Limits

Appendix G and Appendix H to 10 CFR Part 50, describe the conditions that require pressure-temperature limits and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. Appendix G to 10 CFR Part 50 requires additional safety margins whenever the reactor core is critical, except for low-power physics tests.

The pressure-temperature limits imposed on the reactor coolant pressure boundary during operation and tests are reviewed to assure that they provide adequate safety margins against non-ductile behavior or rapidly propagating failure of ferritic components, as required by Criterion 31 of the General Design Criteria for:

1. Preservice hydrostatic tests,
2. Inservice leak and hydrostatic tests,
3. Heatup and cooldown operations, and
4. Core operation.

The applicant has proposed the use of an alternative method of calculating the shift in the reference temperature, as required by Appendices G and H to 10 CFR Part 50. This method estimates the shift in the reference temperature for the first 10 effective full power years as conservatively as using the methods in Regulatory Guide 1.99, "Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," Revision 1, and is acceptable. Subsequent to operation, predictions of radiation damage can be based on the actual measured shifts in reference temperature that are from the results of the surveillance program at the facility.

We have evaluated the information supplied by the applicant to demonstrate compliance with Appendices G and H to 10 CFR Part 50 and conclude that the proposed pressure-temperature limits in Figures 3.4-2 and 3.4-3 of the Technical Specifications have been constructed using acceptable procedures. However, the pressure-temperature limits cannot be accepted at this time because the applicant has not established an initial reference temperature, RT_{NDT}, for every ferritic material within the reactor coolant pressure boundary, and therefore, a basis for the pressure-temperature limits cannot yet be determined. Until the applicant submits the necessary impact energy data for the ferritic materials of the steam generator, pressurizer, and bolting, the proposed pressure-temperature limit curves will be considered unacceptable.

Contingent upon approval of pressure-temperature limits and subsequent to operation, predictions of radiation damage can be based on the actual measured shifts in the reference temperature as determined by the materials surveillance program. The data obtained will be compared to the pressure-temperature limits to be imposed on the reactor coolant system for all operating and testing conditions to assure adequate safety margins against non-ductile or rapidly propagating failure are in conformance with established criteria, codes and standards acceptable to the NRC staff. The use of operating limits based on these criteria, as defined by applicable regulations, codes, and standards, provides reasonable assurance that non-ductile or rapidly propagating failure will not occur, and constitutes an accepable basis for satisfying the applicable requirements of Criterion 31 of the General Design Criteria.

5.3.3 Reactor Vessel Integrity

We have reviewed the Final Safety Analysis Report sections related to reactor vessel integrity. Although most areas are reviewed separately in accordance with other review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted.

We have reviewed the information in each area to assure that it is complete and that no inconsistencies exist that would reduce the certainty of vessel integrity. The areas reviewed are:

1. Design (Safety Evaluation Report Section 5.3.1)
2. Materials of Construction (Safety Evaluation Report Section 5.3.1)
3. Fabrication Methods (Safety Evaluation Report Section 5.3.1)
4. Operating Conditions (Safety Evaluation Report Section 5.3.2)

We have reviewed the above factors contributing to the structural integrity of the reactor vessel and conclude that the applicant has complied with Appendices G and H to 10 CFR Part 50, except for Paragraph IV.A.3 of Appendix G, which will which will remain an open item until the applicant supplies sufficient data and analyses, and Paragraph III.B.4 of Appendix G, for which the applicant has provided sufficient information to justify an exemption.

Paragraph III.B.4 of Appendix G, requires the applicant to conduct impact testing according to specific written procedures. Although the tests were not conducted to formal written procedures for impact tests, the applicant has supplied sufficient information to demonstrate that the tests were conducted correctly, and therefore, we have concluded that an exemption to Paragraph III.B.4 of Appendix G, is justified.

Paragraph IV.A.3 of Appendix G, requires that the impact energy properties of the fasterners meet the levels required by Section NB-2333 of the ASME Code. The applicant has not supplied the necessary data to determine compliance with Paragraph IV.A.3.

Until the applicant has supplied the information necessary to complete our evaluation of Virgil C. Summer Nuclear Station, Unit 1 compliance with Appendices G and H to 10 CFR Part 50, we cannot complete our evaluation of the structural integrity of the reactor vessel.

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps

The reactor coolant pumps are vertical, single-stage, centrifugal, shaft seal pumps having a bottom suction and horizontal discharge. These pumps have been sized to provide adequate core cooling flow to maintain a departure from nucleate boiling ratio within the design bases discussed in Section 4.4 of this Safety Evaluation Report. The mechanical design loop flow is 100,700 gallons per minute.

Sufficient pump rotational inertia has been provided by a flywheel in conjunction with the impeller motor assembly to provide flow during coastdown which is adequate to maintain a departure from nucleate boiling greater than the minimum acceptable value of 1.30 in the event of loss of power to the reactor coolant pumps. A ratchet device is also provided to prevent reverse rotation of an idle reactor coolant pump.

The seals and bearings of the reactor coolant pumps and motors are continuously cooled by the component cooling water system and the chemical and volume control system. A thermal barrier heat exchanger, located above the pump impeller, which is cooled by the component cooling water system limits heat transfer between the hot reactor coolant system water and the seal injection water. High pressure seal injection water from the chemical and volume control system is introduced through the thermal barrier wall. A portion of this water flows up around the bearing through the seals; the remainder flows down through the thermal barrier where it acts as a buffer to prevent system water from entering the radial bearing and seal section of the unit.

Criterion 4 of the General Design Criteria requires, in part, that structures, systems, and components of nuclear power plants important to safety be protected against the effects of misiles that result from equipment failures. At our request the applicant, in Amendment 15 and 22 of the Final Safety Report, addressed the consequences of the loss of component cooling water to the reactor coolant pumps. The applicant stated that should a loss of component cooling water to the reactor coolant pumps occur, the chemical and volume control system continues to provide seal injection flow to the reactor coolant pumps; the seal injection flow is sufficient to prevent damage to the seals with a loss of reactor coolant pumps thermal barrier cooling. Consequently, the reactor coolant pumps can continue to run following a loss of thermal barrier cooling. However, the loss of component cooling water to the reactor coolant pumps motor bearing oil coolers will result in an increase in oil temperature and a corresponding rise in motor bearing metal temperature. Two reactor coolant pump motors of the same type as the summer reactor coolant pumps have been tested with interrupted component cooling water flow to the motor bearing oil coolers. The bearing metal temperature reached 185 degrees Fahrenheit in the first 10 minutes, and it was demonstrated that the reactor coolant pumps would continue to function.

Two safety-related transmitters are provided to redundantly monitor component cooling water flow to the upper and lower reactor coolant pump motor bearings. Two additional safety-related transmitters are provided to redundantly monitor component coolant water flow to the reactor coolant pumps thermal barriers. These transmitters provide flow indication and actuate low flow alarms in the control room.

To assure that reactor coolant pumps would not be run beyond the range of the 10 minutes test data, and to protect the pumps themselves, operating procedures are provided for a loss of component cooling water and seal injection to the reactor coolant pumps. Include in these operating procedures is the provision to trip the reactor and the reactor coolant pumps if component cooling water flow, as indicated by the instrumentation discussed above, is lost to the reactor coolant pumps and cannot be restored within 10 minutes.

We have reviewed the above provisions and conclude that in the event of a loss of component cooling water flow to the reactor coolant pumps the procedures for identifying the cause and tripping the pumps would be executed in 10 minutes, thereby preventing unacceptable damage to the reactor coolant pumps due to a loss of component cooling water to the reactor coolant pumps.

Because flywheels have large masses and rotate at speeds of approximately 1200 revolutions per minute during normal reactor operation, a loss of flywheel integrity could result in high energy missiles and excessive vibration of the reactor coolant pump assembly. The safety consequences could be significant because of possible damage to the reactor coolant system, the containment, or the engineered safety features.

Adequate margins of safety and protection against the potential for damage from flywheel missiles can be achieved by the use of suitable materials, adequate design, and inservice inspection. The flywheels have been fabricated from SA-533 Grade B Class 1 steel, produced by a process that minimizes flaws and improves fracture toughness, and have been cut, machined, finished, and inspected in accordance with Section III of the ASME Boiler and Pressure Vessel Code and Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1.

The integrity of the reactor coolant pump flywheel has been provided by designing it to 125 percent of the normal synchronous speed of the motor (approximately 1500 revolutions per minute). The minimum speed for ductile failure is estimated to be much higher than 125 percent of operating speed for flywheels of the design used at the facility. The lowest design operating temperature is specified to be at least 110 degrees Fahrenheit. The applicant has stated that the reference temperature will be no higher than 10 degrees Fahrenheit based on dropweight tests (two for each flywheel) exhibiting no-break performance at 20 degrees Fahrenheit. Consequently, operation will occur at the nil ductility transition temperature plus 100 degrees Fahrenheit. To show further compliance with Regulatory Guide 1.14, the applicant has submitted impact energy data for each flywheel to demonstrate that the upper shelf energy in the weak direction is at least 50 foot-pounds.

Based on our evaluation we conclude that the reactor coolant pump flywheels in the facility possess a margin of safety against flywheel missiles equivalent to that recommended in Regulatory Guide 1.14. Compliance with Regulatory Guide 1.14 provides a basis acceptable to the staff for satisfying the requirements of Criterion 4 of the General Design Criteria.

5.4.2 Steam Generators

The three steam generators will be vertical shell and U-tube evaporators with integral moisture separators. The steam generator design to be used for the facility is the Westinghouse Model D3 series. The primary reactor coolant will enter the steam generator lower hemispherical head and flow through the U-tubes giving up heat to generate steam on the shell side of the unit. The U-tube and tube-sheet boundary will be designed to withstand full reactor coolant side design pressure and temperature with atmospheric pressure on the secondary side so as to prevent the activity generated within the primary system from passing over to the secondary system. Since the steam generators must provide a heat sink for the primary reactor coolant system during certain shutdown conditions, they are at a higher elevation than the core to assure natural circulation flow for decay heat removal.

Feedwater flows through an integral flow restrictor into a preheater section and is heated almost to saturation temperature before entering the boiler section. Subsequently, the water-steam mixture flows upward through the tube bundle and into the steam drum section. A set of centrifugal moisture-separators, located above the tube bundle, removes most of the entrained water from the steam. The remaining steam will then pass through steam dryers to raise the steam quality before leaving the steam generator.

Integral safety grade flow restrictors with a 1.4 square foot area are located in the steam line nozzles of the steam generators. Each restrictor is designed to limit the blowdown rate from the steam generators in the event of a main steam line rupture.

The steam generators have carbon steel support plates with drilled flow holes. The materials used in Class 1 components of the steam generators were selected and fabricated according to codes, standards, and specifications acceptable to the NRC staff. The steam generator pressure-retaining parts are designed and manufactured to meet Section III of the ASME Code. The pressure boundary materials comply with fracture toughness requirements of Article NB-2300 of Section III of the ASME Code.

The primary side of the steam generator is designed to ASME Code, Class 1 requirements, as required by the NRC staff. The secondary side pressure boundary parts of the steam generator are also designed, manufactured, and tested in accordance with the requirements of the ASME Code.

The steam generators are of the preheat design. The water entering the preheater will be of feedwater quality (low chloride and copper concentrations) and the impurities contained therein will not be subject to concentration due to the absence of recirculation. This and the thermal and hydraulic characteristics minimize the potential for chemical hideout in the preheater. Since sludge may still settle out at the tubesheet elevation outside the preheaters, blowdown pipes are provided which have been designed to achieve maximum removal of the sludge from the steam generators.

The onsite cleaning and cleaniness controls during fabrication conform to the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Po r Plants." Conformance with applicable codes, standards, and regulatory guides constitutes an acceptable basis for meeting, in part, the requirements of Criteria 14, 15, and 31 of the General Design Criteria.

Inspection Ports

Recent operating experience with some Westinghouse plants has revealed problem areas associated with steam generator tube deformation in the form of a reduction in tube diameter (i.e., a phenomenon known as tube denting) and tube support plates information. Tube denting is a related phenomena resulting from corrosion product buildup in the crevices formed between the tubes and the tube support plates or tubesheet due to secondary side contamination and corrosion of the carbon steel support plates. Denting was first discovered during the inspection of the Surry Unit 2 steam generator in April, 1975. Since then, Westinghouse was conducted a comprehensive research program is order to determine the cause and extent of the problem and establish corrective actions to be implemented in all Westinghouse plants. Tube support plate deformation is a secondary effect concurrent with tube denting.

For these forms of steam generator degradation which have occurred, eddy current testing and tube gauging alone are not sufficient to assess and monitor tube and support plate degradation. In order to perform an adequate assessment and monitoring of these areas, we require that inspection ports be installed. These ports should be installed just above the upper support plate, and between the tubesheet and the lower support plate, with both ports in line with the tube line.

Under the as low as reasonably achievable concept, we are requesting that all possible steam generator modifications be made prior to the start of operation. Based upon experience we have determined that these ports can be installed in the steam generators after the start of operations at a personnel exposure of 7.5 man-rem. The NRC staff has determined that this exposure is not significant enough to justify the delay of startup of the plant to permit the installation of inspection ports. However, since the probability of secondary side contamination will increase as the operating time increases, we require that these ports be installed prior to startup after the first refueling.

Row 1 Steam Generator Tubes

Operating experience has shown that in the region of the small bend radius of the Row 1 tubes in the steam generators of Westinghouse design leads to early onset of cracking. It is our position that, unless information is developed to demonstrate that potential cracking in the U-bend region of row 1 tubes can be avoided, we will, in the interest of as low as reasonably achievable industrial dose exposures, require the plugging of all row 1 tubes prior to issuance of the full power license. At the present time, Westinghouse has committed (letter from T. M. Anderson to R. H. Vollmer, May 12, 1980) to a program to determine the particular susceptibility of row 1 tubes to cracking. The program involves removing numerous tubes from the Trojan plant and subjecting them to nondestructive and destructive testing to identify the cause of the cracking and to develop a field inspection method capable of detecting potential leaking tubes. The results of this evaluation are expected to be available in October 1980; thus a sound engineering decision on the need to plug row 1 tubes can be made prior to the issuance of the full power license. We shall review the program results and decide at that time on the necessity to plug the row 1 tubes.

Although the possibility of tube and tube support plate degradation exists, we have concluded that, with the additional measures mentioned above and discussed further below, operation of the steam generators will not constitute an undue risk to the health and safety of the public for the following reasons:

1. Primary to secondary leakage rate limits and associated surveillance requirements will be established to provide assurance that the occurrence of tube cracking during operation will be detected and appropriate corrective action, such as tube plugging, will be taken such that any individual crack present will not become unstable under normal operating, transient, or accident conditions.

2. Augmented inservice inspection requirements and preventative tube plugging criteria will be established to provide assurance that the great majority of degraded tubes will be identified and removed from service before leakage develops.

Criterion 32 of the General Design Criteria requires, in part, the components which are part of the reactor coolant pressure boundary or other components important to safety be designed to permit periodic inspection and testing of critical areas of structural and leaktight integrity.

The components in the steam generator are classified as ASME Boiler and Pressure Vessel Code Class 1, depending on their location in either the primary or secondary coolant systems, respectively. The steam generators are designed to permit inservice inspection of the Class 1 components, including individual tubes. The design aspects that provide access for inspection and the proposed inspection program should follow the recommendations of Regulatory

Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," Revision 1, and must comply with the requirements of Section XI of the ASME Code, with respect to the inspection methods to be used, provisions of a baseline inspection, selection and sampling of tubes, inspection intervals, and actions to be taken in the event that defects are identified.

We have reviewed the steam generator inspection program presented in Section 3/4 4-5 of the Technical Specifications for the facility and conclude that this inspection program meets the requirements, except for the following:

1. The Technical Specifications do not contain the details of the required preservice inspection; and

2. Technical Specification Sections 4.4.5.2.b, 4.4.5.2.b.3, 4.4.5.2.c, and 4.4.5.3.b should be rewritten to convey the same meaning that is found in corresponding sections of NUREG-0452, Revision 2, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors."

Conformance with Regulatory Guide 1.83 and Section XI of the ASME Code will constitute an acceptable basis for meeting, in part, the requirements of Criterion 32 of the General Design Criteria.

5.4.3 Residual Heat Removal System

The residual heat removal system includes the piping, valves, pumps, heat exchangers, instrumentation, and controls used to remove core decay heat and provide long-term cooling following the initial phase of reactor cooldown. The scope of our review of the residual heat removal system for the facility included piping and instrumentation diagrams, plant arrangement drawings, and design performance specifications for essential components. The review has included an assessment of the applicant's design criteria and design bases for compliance of the design to the General Design Criteria, Regulatory Guides and the corresponding section of the Standard Review Plan.

The residual heat removal system consists of two parallel flow trains each consisting of a heat exchanger, pump, and the associated piping, valves, and instrumentation necessary for operational control. The inlet lines to the residual heat removal system are connected to the hot legs of two of the reactor coolant system loops and the return lines are connected to each of the cold legs of the reactor coolant system loops. The residual heat removal system lines are isolated from the reactor coolant system by two motor-operated valves in series located inside the containment. Each discharge line is isolated from the reactor coolant system by two check valves located inside the containment and by a normally open motor-operated valve located outside the containment. Thus, the facility design incorporates two independent and redundant barriers whenever the reactor coolant system pressure will be above the residual heat removal system design pressure. The vulnerability of these barriers to single electrical failures is addressed in Section 7.6.2 of this Safety Evaluation Report.

During power operation, electric power is locked out to the four inlet lines valves to assure that these valves will stay the in closed position under post-loss-of-coolant accident flooding conditions. When the service of the residual heat removal system is required, power supply to these valves can be restored by manual action.

In the Virgil C. Summer Nuclear Station Unit 1 residual heat removal system design, one of the two isolation valves in the suction lines of each of the two trains is powered from the same source and sensor. Failure of this electrical source or pressure sensor would prevent operating of both residual heat removal trains (see Section 7.6.2 of this Safety Evaluation Report). In response to our requests, the applicant has stated that for a single failure in the electrical power supply, an alternative power supply can be temporarily connected to the valves through a manual action at the motor control center, and for a single failure in the pressure sensor, the valve associated with the failed sensor can be operated from the control room. The staff finds this acceptable.

When the reactor coolant system is open to the atmosphere and the steam generators are not available for decay heat removal (the facility cannot be put in hot standby), electrical power will be locked out to the suction isolation valve in each suction line which is powered from an emergency power source different from that of its respective pump. With this arrangement, the automatic initiation of at least one out of the two residual heat removal system trains will not be prevented by a single failure in the electric power supply or the pressure sensor.

When the residual heat removal system is in operation, heat removal is controlled by regulating primary coolant flow through the residual heat removal system heat exchanger bypass valves. The potential for exceeding the allowable cooldown rate of the residual heat removal system and the reactor coolant system during the shutdown cooling mode, assuming loss of the non-safety-grade instrument air system which controls the residual heat removal heat exchangers outlet and bypass valves, has been evaluated. The applicant has stated that the maximum cooldown rate, assuming failure of these heat exchanger bypass valves and no operator action, would not exceed 200 degrees Fahrenheit per hour, would be limited to a short time duration, and would not exceed 100 degrees Fahrenheit in the first hour. Westinghouse has performed an analysis to show that even at a constant cooldown rate of 200 degrees Fahrenheit over a reactor coolant temperature range from 350 degrees Fahrenheit to 250 degrees Fahrenheit, the resulting stresses are acceptable. We have reviewed the information provided and determined that these cooldown rates at the temperatures and pressures expressed in the residual heat removal system cooling mode, and at short time intervals, will not result in unacceptable stresses to the reactor vessel.

Overpressure protection for each residual heat removal system line is provided by use of pressure relief valves. The pressure relief valves in the discharge lines from the residual heat removal system to the reactor coolant system are capable of relieving 20 gallons per minute of possible back-leakage through the check valves that provide pressure boundary separation between the residual heat removal system and the reactor coolant system. Periodic testing of the check valves, in accordance with the requirements of Article IWV-2000 of Section XI of the ASME Code, will be performed to verify that they do not leak. Each inlet line to the residual heat removal system is equipped with a pressure relief valve designed to relieve the combined flow of all charging pumps at the relief valve set pressure of 450 pounds per square inch, gauge. The applicant has performed analyses to confirm that one relief valve has adequate capacity to prevent the residual heat removal system maximum pressure from exceeding the ASME Code limit.

Recent plant operating experience has identified a potential problem related to the loss of shutdown cooling during certain reactor coolant maintenance operations, when the reactor coolant system was partially drained, that has resulted in air binding of the residual heat removal pumps and subsequent loss of shutdown cooling. In response to our concerns the applicant has provided operating procedures which include instructions for draining the reactor coolant system for vessel head removal prior to refueling operations. The instructions include the installation of clear hose between one of the reactor coolant system loop drains and the pressurizer relief line vent to provide level indication after the pressurizer level indication is off scale. The procedure also states that after the reactor coolant system level has reached four to 12 inches below the reactor vessel flange, draining operations are secured. These actions assure that air will not be introduced into the residual heat removal system via the reactor coolant system during refueling operations. The staff finds this approach acceptable to prevent air binding of the residual heat removal system pumps.

Branch Technical Position RSB 5-1 which is attached to Section 5.4.7 of the Standard Review Plan requires that the capability be provided for transferring heat from the reactor to the environment from normal operating conditions to cold shutdown using only safety-grade systems, with only offsite or onsite power available, and assuming the most limiting single failure. There are four processes that are involved in taking the plant from hot standby to cold shutdown conditions. These are: (1) removal of residual heat and stored energy; (2) circulation of the reactor coolant; (3) makeup and boration of the reactor coolant to the cold shutdown boron concentration; and (4) depressurization. With loss of offsite power the reactor coolant pumps, main condenser and the main feedwater pumps are unavailable. Heat removal and coolant circulation under natural circulation conditions is then controlled by use of the steam generator atmospheric dump valves and the emergency feedwater system.

The three air-operated atmospheric dump valves at the facility (one per steam generator) are safety-grade, seismic Category I valves. They are provided with handwheels and can be operated locally to permit plant cooldown. Since this is a control function, we require the applicant to perform the necessary tests which confirm the feasibility of this type of manual action. In case of a mechanical failure that prevents the opening of one of the dump valves, limited maintenance to correct for this failure could be taken by closing the isolation valve upstream of the affected atmospheric dump valve. However, to assure cooldown capability, the applicant is required to confirm the capability of plant cooldown with only two steam generators in case one of the atmospheric dump valves fails to open.

The water supply to the emergency feedwater system is provided initially from the seismic Category I condensate storage tank which has a reserve of 150,000 gallons dedicated to the emergency feedwater system. This supply is backed up by the seismic Category I service water system. The emergency feedwater system is designed such that no single failure prevents delivery of the minimum feedwater flow to at least two steam generators. The emergency feedwater system can take suction from the service water system for an indefinite period of time.

During a normal plant cooldown from hot standby conditions, the chemical and volume control system letdown line from the reactor coolant system would be used during both the initial boration to the required boron shutdown concentration and while the reactor coolant system inventory is controlled during the cooldown. Loss of the non-seismic air supply results in loss of letdown due to the air-operated valves failing closed in the letdown line. Under these conditions, boration without letdown could still be accomplished using safety-grade equipment. Borated water (four weight percent boric acid) could be supplied to the suction of the centrifugal charging pumps from one of the two boric acid tanks using one of the boric acid transfer pumps. The capacity of one boric acid tank is sufficient to provide boration to the required shutdown concentration. Makeup above that provided by the boric acid tanks is obtained from the refueling water storage tank (2000 parts per million boric acid). Two motor-operated valves, each powered from different emergency power trains and connected in parallel, will transfer the suction of the charging pumps to the refueling water storage tank. Makeup from the refueling water storage tank can be monitored using Class 1E instrumentation in the control room.

The applicant stated that boration and depressurization could be accomplished without letdown in two steps. First the operators integrate the cooldown and boration functions taking advantage of the reactor coolant system inventory contraction resulting from the cooldown. Finally, the operators use auxiliary spray from the chemical and volume control system to depressurize the plant to residual heat removal system initiation conditions. The applicant indicated that the available volume in the pressurizer steam space is greater than that needed to achieve cold shutdown concentration in the reactor coolant system without taking credit for letdown and without taking full credit for contraction of the primary coolant in cooldown. In addition, the available volume for borated water injection without letdown which results from the contraction of the primary coolant is much larger than that required to cool and hence depressurize the pressurizer to 425 pounds per square inch gauge by injection of borated water through the pressurizer spray. This pressure must be reached to permit shutdown cooling with the residual heat removal system.

Under natural circulation conditions the normal coolant supply to the pressurizer spray from the cold legs of two coolant loops is lost due to loss of forced circulation. In this case, the pressurizer spray needed for depressurization can be supplied by auxiliary spray flow from the centrifugal charging pumps through a line branching off from the charging line of the chemical and volume control system. This supply could be lost by a single failure involving either closing of a single valve in the supply line or opening of one of several valves in lines connected to the supply line. If manual actions to correct for such failures were not successful, a backup method of depressurization, after boration and reactor coolant system cooldown to 450 degrees Fahrenheit, would consist of discharging reactor coolant from the pressurizer to the pressurizer relief tank via the pressurizer power-operated relief valves. The power-operated relief valves are designed to meet seismic Category I requirements but their operators are not. The applicant will be required to upgrade these valves if they fail the Electric Power Research Institute testing program on safety valves and power-operated relief valves discussed in Section 22 of this Safety Evaluation Report. Each power-operated relief valve has a motor-operated isolation valve which is designed to meet seismic Category I requirements and is powered by emergency buses. Manual action may be taken in the event of a power-operated relief valve failure to position or repair the valve, if necessary. The applicant stated that another alternative method of depressurization after boration and reactor coolant system cooldown to 450 degrees Fahrenheit, would be to allow the pressurizer to cool via ambient heat losses as the reactor coolant system is maintained at 350 degrees Fahrenheit via natural circulation.

The residual heat removal system design has parallel lines from the hot legs to the suction of the residual heat removal system pumps each having two motor-operated isolation valves in series powered from different emergency power trains. In the event of a single failure (either electrical or mechanical), access to the residual heat removal system could be accomplished when required by manual operator action. The residual heat removal system performance may be monitored by control room flow indication which alarms in the control room on low residual heat removal system flow. RSB 5-1 requires that a natural circulation test with supporting analysis be conducted to demonstrate the ability to cool down and depressurize the plant and to demonstrate that boron mixing is sufficient under such circumstances. Comparison with performance of previously tested plants of similar design may be substituted for these tests, if justified. The applicant plans to reference tests to be conducted at Diablo Canyon which might affect boron mixing under natural circulation.

The applicant's comparisons of system and upper head region characteristics for the Virgil C. Summer Nuclear Station, Unit 1 and Diablo Canyon suggest that the results of the Diablo Canyon test and supporting analysis should satisfy the RSB 5-1 requirements. However, the staff plans to defer reaching a conclusion on this matter until the Diablo Canyon results have been reviewed. Moreover, it has recently come to our attention that voiding in the vessel can occur if cooldown rates under natural circulation are too high. In establishing the applicability of the Diablo Canyon natural circulation test to this facility, the applicant should account for this phenomenon and assure that the potential for its occurrence is properly reflected in the testing. If the Diablo Canyon tests are not completed or do not provide satisfactory results, the applicant has committed to submit such test results applicable to the Virgil C. Summer Station, Unit 1 prior to startup following the first refueling.

The applicant has committed to providing a summary of procedures for cooldown by natural circulation. We require that the applicant commit to provide these specific procedures when the test and analyses are completed.

This testing is not necessary for first cycle operation of the facility. The major purpose of the natural circulation test is to obtain information on the time needed to take the plant from hot standby to the cut-in point of the residual heat removal system under conditions such as extended loss of offsite power when the reactor coolant system pumps are not available. It would be preferable to run this test after the first reload when the decay heat is relatively large. This would result in more meaningful test data and testing under conditions more representative of those occurring over the 40-year plant life.

A remaining concern that is still under discussion with the applicant is the power lockout to the four inlet residual heat removal system valves which provide suction from hot legs 1 and 3.

In order to preclude low pressure injection system valve misalignment during normal operation, single failure considerations require that power sources to the valve operators be administratively locked out by opening the motor operator circuit breakers. At present, the circuit breakers for the motor operators are located in multiple motor control centers, a considerable distance outside of, and away from, the control room. RSB 5-1 requires bringing the plant from normal operation to cold shutdown without the operators having to the leave the control room, except to correct for failures. For Class 2 plants, in which only partial implementation of RSB 5-1 is required, the staff position has been to require complete compliance with this item, with exceptions considered on a case-by-case basis.

Based on our review of the motor control center locations with respect to the control room, and our understanding of the operator action necessary to supply power to the valve motor controllers for normal residual heat removal system realignment, we do not conclude the applicant to be in compliance with RSB 5-1. We informed the applicant that the design must be modified so that alignment to the shutdown cooling system can be accomplished from the control room. We require that the applicant comply with RSB 5-1 by either installing a switch in the control room that would lock or unlock the power to these residual heat removal system suction valves or provide an acceptable alternative to the staff. The applicant will be required to make the necessary design modification by the end of the first refueling outage.

Subject to the conditions discussed above, we find that the capability to achieve cold shutdown for the Virgil C. Summer Nuclear Station, Unit 1 satisf[illegible] the requirements of RSB 5-1 and is acceptable.

5.4.4 Loose Parts Monitor

The applicant has provided a description of the loose parts monitoring system to be provided for the facility. The design will include redundant sensors at each natural collection region. The sensors will be mounted with threaded or clamped adapters on the various reactor coolant system components. Reasonable assurance has been provided that the sensor mounting will be capable of withstanding seismic events up to and including the operating basis earthquake. The system will be capable of detecting a loose part with an impact energy of 0.2 foot-pound at a distance within three feet of a sensor. Operator training in the operation and calibration of the system will be provided by the vendor. The applicant has stated that data records will be used and maintained based on established administrative procedures. We have reviewed the information provided by the applicant and have found that the loose parts monitoring system for the facility is consistent with the guidelines established by the NRC staff and is acceptable.

5.4.5 Reactor Coolant Piping

The reactor coolant system piping includes all sections of piping interconnecting the reactor vessel, steam generator and reactor coolant pump for all three primary loops. Also included is the piping for all systems directly connected to the reactor coolant system up to the system isolation valve (where applicable) such as letdown, charging, accumulator, residual heat removal, safety injection, pressurizer, safety injection and instrumentation piping. Conformance of this piping to the General Design Criteria is identified in Section 3 of this Safety Evaluation Report.

6 ENGINEERED SAFETY FEATURES

6.1 Design Considerations

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 facility. The reactor containment systems and the emergency core cooling systems are described in this section of the Safety Evaluation Report. Certain of these systems, or parts of these systems, will have functions for normal facility operations as well as serving engineered safety features.

We have reviewed the proposed systems and components designated as engineered safety features. These systems and components are designed to be capable of assuring safe shutdown of the facility under the adverse conditions of the various postulated design basis accidents described in Section 15 of this Safety Evaluation Report. Therefore, these systems are designed to seismic Category I requirements and must function in the event of complete loss of offsite power.

Components and systems are provided with sufficient redundancy so that a single failure of any component or system will not result in the loss of the capability to achieve and maintain safe shutdown of the facility. The instrumentation systems and emergency power systems for the engineered safety features are designed to the same seismic and redundancy requirements as the systems they serve. These systems are described in Sections 7 and 8 respectively of this Safety Evaluation Report.

6.1.1 Engineered Safety Features Materials

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

The controls on the pH of the reactor containment sprays and the emergency core cooling water following a postulated loss-of-coolant accident are adequate to assure freedom from stress corrosion cracking of the austenitic stainless steel components and welds of the containment spray and emergency core cooling systems throughout the duration of the postulated accident to the completion of cleanup. The controls on the use and fabrication of the austenitic stainless steel of the systems satisfy the requirements of the NRC staff interim position MTEB 5-1 on Regulatory Guide 1.31, "Control of Stainless Steel Welding" and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices performed in accordance with these requirements provide added assurance that stress corrosion cracking will not occur during the postulated accident time interval. The controls placed on concentrations of leachable impurities in non-metallic thermal insulation used on austenitic stainless steel components of the engineered safety features are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

The control of the pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, is in accordance with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and provides assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution resulting from corrosion of containment metal or cause serious deterioriation of the materials in containment. The protective coating systems have been qualified by test acceptable to the staff. This qualification provides reasonable assurance that the coating systems will not degrade the operation of the engineered safety features by delaminating, flaking or peeling. The applicant states that the engineered safety features were designed and constructed in accordance with the applicable edition of Section III of the ASME Code and the addenda of the 1974 edition of Section III of the ASME Code.

Conformance with the ASME code and regulatory guides and with the staff positions mentioned above, constitute an acceptable basis for meeting in part the requirements of Criteria 16, 34, 35, 38, 41 and 44 of the General Design Criteria.

Conformance with the codes and regulatory guides mentioned above, and with the NRC staff's position on the allowable maximum yield strength of cold worked stainless steel, and the minimum level of the pH of containment sprays and emergency core cooling water constitutes an acceptable basis for meeting the applicable requirements of Criteria 35, 38, and 41, of the General Design Criteria.

6.1.2 Organic Materials Inside Containment

We have evaluated the applicant's proposed coating systems to be used inside the containment to determine their suitability under design basis accident conditions and to determine any potential adverse interaction with the engineered safety features equipment.

The applicant has proposed to select protective coatings which satisfy ANSI N101.4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities." The applicant has also committed additional quality assurance requirements in a letter C. Eicheldinger, Westinghouse to C. Heltemes, NRC, dated February 1, 1977, which we have reviewed and found acceptable. Accordingly, we conclude that the protective coatings selected by the applicant will meet the recommendations of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," with the exception of a small quantity (0.18 cubic foot) of unqualified paints. The quantity of unqualified paints is sufficiently small that potential decomposition products from this source will not pose a safety problem for the facility.

The applicant has further indicated that the organic materials inside containment will consist principally of paints, coatings, and insulation and that there will be no significant quantity of organic materials exposed directly to the containment atmosphere.

In summary, the coating systems chosen by the applicant have been qualified under conditions which take into account the postulated design-basis accident conditions. No adverse interactions (under design-basis accident conditions) between the decomposition products and the engineered safety features have been identified. We conclude, therefore, that the proposed nature and quantity of coating materials is acceptable.

6.1.3 Post-Accident Chemistry

The post-accident chemical environment inside the containment will be dominated by containment spray water recirculated from the containment sumps. The water in these sumps will come from the refueling water storage tank, the reactor coolant system, and the emergency core cooling system. Chemicals expected to be added to this water are boric acid (2000 parts per million) added through the emergency core cooling system, and sodium hydroxide (pH of nine) from the containment spray.

Since the sump water is controlled to a pH level of approximately nine, this will reduce the probability of chloride stress corrosion cracking leading to equipment failure or loss of containment integrity. The method and procedures for controlling the pH of solutions expected to be recirculated in containment following design basis accidents have been found adequate. The proposed controls provide assurance that the pH will be maintained at a level which minimizes the possibility of stress corrosion cracking of mechanical systems and components.

6.2 Containment Systems

The containment systems for the Virgil C. Summer Nuclear Station, Unit 1 include the containment structure, containment heat removal system, containment isolation system, containment combustible gas control systems, and provisions for containment leakage rate testing.

6.2.1 Containment Functional Design

The containment structure will be a cylindrical, carbon steel lined, prestressed, reinforced concrete structure with a net free volume of 1,840,000 cubic feet. The containment structure will house the nuclear steam supply system, which includes the reactor vessel, reactor coolant piping, reactor coolant pumps, pressurizer, and steam generators, as well as certain components of the facility's engineered safety feature systems. The containment structure is designed to withstand internal pressurization resulting from postulated high energy pipe breaks inside containment and external pressurization due to inadvertent actuation of the containment heat removal systems. The containment structure is designed for an internal pressurization resulting from postulated high energy pipe breaks inside containment and external pressurization due to inadvertent actuation of containment heat removal systems. The containment structure is designed for an internal pressure of 57 pounds per square inch, gauge, and a temperature of 283 degrees Fahrenheit. The containment structure is designed for an external differential pressure of 3.4 pounds per square inch, gauge.

Containment Analysis

The data for mass and energy release to the containment following a primary system rupture were calculated using the Westinghouse SATAN-V code for the blowdown period, the WREFLOOD code for the reflooding period and the FROTH code to calculate post-reflood steam boil-off from the core and steam generators. These methods are designed to conservatively maximum steam flow to the containment pressure. The method and assumptions are documented in the Westinghouse Topical Report WCAP-8312A, and were approved by the NRC staff in our topical report evaluation dated March 12, 1975.

The blowdown rates from postulated primary system ruptures within containment subcompartments were calculated using the SATAN-V code. This code uses the modified Zaloudek correlation to calculate flow when the break fluid is saturated. Stagnation conditions at the break are approximated by removing the momentum flux option from the SATAN-V code. This method is also documented in the Westinghouse Topical Report WCAP-8312A, which was approved by the NRC staff on March 12, 1975.

The applicant has analyzed the containment pressure response to various postulated pipe break accidents in the manner described below. Mass and energy release rate data were input to the NRC staff's CONTEMPT-LT/22 and CONTEMPT-LT/26 computer codes which were used by the applicant to perform the containment pressure response analysis.

The applicant analyzed a spectrum of reactor coolant system pipe breaks, considering various single failures, to identify the containment design basis loss-of-coolant accident. The containment design basis loss-of-coolant accident was identified as the postulated double-ended rupture at the pump suction of the reactor coolant system, which resulted in a peak calculated pressure of 44.7 pounds per square inch, gauge. For the containment peak pressure analysis, the most severe single active failure was determined to be the loss of one diesel generator train, which results in the loss of one containment spray train, one containment emergency fan cooler train and one safety injection system train.

We have also analyzed the containment pressure response to a postulated double-ended rupture at the pump suction of the reactor coolant system using the CONTEMPT-LT/26 computer code. Our confirmatory analysis was based on the mass and energy release, containment structure heat sink, and containment heat removal system performance data provided by the applicant. Conservative condensing heat transfer coefficients to the structures inside the containment were used. Our confirmatory analysis resulted in a peak calculated pressure of 43.1 pounds per square inch, gauge, which confirms the acceptablity of the containment design pressure of 57 pounds per square inch, gauge.

The applicant analyzed a spectrum of main steam line break accidents, considering various single active failures, to determine both the containment peak pressure and the temperature profile for use in the qualification of safety-related equipment.

Following a postulated main steam line break inside the containment, steam will initially be discharged from each of the steam generators. Flow from the steam generators in the unbroken loops will be terminated following the main steam line isolation signal. Flow from the steam generator in the broken loop will continue until the fluid is discharged.

The mass and energy available to flow into the containment is the mass of fluid initially in the steam generators and the additional water added by the feedwater system. For the Virgil C. Summer Nuclear Station, Unit 1 steam generators, the initial water mass is greatest during hot standby conditions since feedwater flow is proportional to the power level.

For the long-term mass and energy release calculations used to verify the containment design pressure, the applicant has performed a bounding calculation designed to maximize the total mass and energy release. In the bounding calculation, the steam generator water mass is set at the hot standby value, however, feedwater is assumed to flow at the full power value. A double ended break is assumed and, following isolation of the main steam isolation valves and the redundant valves in the feedwater lines, the entire inventory of the ruptured steam generator is added to the containment as steam. The primary system is assumed to provide an infinite heat source for this process. The flow rate to the containment is maximized by use of the Moody critical flow correlation. We conclude that this method is conservative for verification of the containment design pressure.

To assess the maximum temperature that might occur within the containment for the purposes of instrument qualification analysis, the applicant has calculated mass and energy releases for a spectrum of steam line breaks at various power levels using the methods and assumptions described in Westinghouse Topical Report WCAP-8860. These data are calculated using the MARVEL code which describes both the primary and secondary systems. Entrained liquid is calculated to exit the steam generator for large break sizes and entrained liquid acts to reduce the amount of superheat within the containment. No liquid entrainment is calculated for smaller break sizes and these breaks release the greatest amount of energy to the containment.

The amount of liquid entrainment is calculated using the TRANFLØcode described in the Westinghouse Topical Report WCAP-8859. In WCAP-8859, the results of field tests are presented for operating steam generators which indicate that significant liquid entrainment will occur during postulated large steam line breaks. Both the MARVEL and TRANFLØ codes are under review by the NRC staff. Our review to date indicates that there is reasonable assurance that the mass and energy release rates will not be appreciably altered upon completion of our review of these codes.

The applicant identified the worst case main steam line break, with respect to containment pressure, to be a double-ended guillotine rupture. The applicant has also performed a single active failure analysis and determined the limiting single failure to be the failure of an emergency feedwater flow-control valve to close.

During normal operation the emergency feedwater system is idle and the emergency feedwater flow-control valves are open. In the event of a secondary side pipe rupture, the emergency feedwater system automatically injects feedwater into the steam generators. In addition, the system automatically terminates flow to the affected steam generator by closing the flow-control valves. Operator action is required to verify that flow to the affected steam generator has been terminated. For this analysis, the applicant has conservatively assumed that one of the two flow-control valves for the affected steam generator fails to close, and the operator does not terminate flow until 30 minutes following onset of the accident. Peak containment pressure is reached approximately 142 seconds after the pipe break.

For this worst case main steam line break, the applicant calculated a peak containment pressure of 47.1 pounds per square inch, gauge.

We have performed a confirmatory analysis of the containment pressure response to the worst-case main steam line break identified by the applicant, and have calculated a peak pressure of 48.5 pounds per square inch gauge. Our results confirm the acceptability of the containment design pressure.

The applicant has identified the worst-case main steam line break, with respect to containment temperature, to be a 0.681 square foot split rupture at 70 percent power. For this postulated break, the failure of a diesel generator to start was assumed. The applicant calculated a peak containment temperature of 324 degrees Fahrenheit.

In determining the worst-case single active ilure for the peak temperature, the applicant analyzed a spectrum of break sizes and power levels. The highest containment temperatures resulted from small split ruptures that had pure steam blowdown (no water entrainment) and maximized the actuation time of the containment spray system. The split ruptures represent the largest breaks which will neither generate a steam line isolation signal nor produce entrainment. Failure of the emergency feedwater flow control valve and the main steam isolation valve were not considered for the small split ruptures because the containment spray system would be actuated before these valves would receive isolation signals.

We have performed a confirmatory analysis, and our results confirm the acceptability of the applicant's temperature profile.

The applicant has determined the maximum external differential pressure on the containment structure due to inadvertent operation of the containment spray system. For this analysis the applicant assumed the containment is cooled to a temperature of 40 degrees Fahrenheit (minimum refueling water storage tank temperature), resulting in an external differential pressure of 3.4 pounds per square inch. The applicant's analysis conservatively neglects natural convection heat transfer from the containment heat sinks to the containment atmosphere. In addition, operator action is not relied on to terminate the sprays. Therefore, we conclude that the applicant's analysis is conservative, and that the specified containment external design pressure of 3.4 pounds per square inch is acceptable.

Containment Subcompartment Analysis

The applicant has analyzed the pressure response of subcompartments inside the containment due to postulated high energy line breaks occurring in the reactor cavity, steam generator, and pressurized compartments.

We have reviewed the postulated pipe break sizes and locations and found them to be acceptable. The mass and energy release data were then used with the RELAP 4/MOD 5 computer code to perform the subcompartment analysis for the reactor cavity and steam generator compartments. The applicant has used the FLASH-2 computer program to predict the transient pressure behavior of the pressurizer compartment.

The applicant has performed nodalization sensitivity studies for the reactor cavity and steam generator compartments. These studies showed the nodal volume averaged pressures changed only insignificantly as the nodalization schemes were varied. Therefore, we conclude that the nodalization of the compartments is acceptable.

We have performed confirmatory analyses of the applicant's 33 node, 150 square-inch cold leg break in the reactor cavity; 25 node, hot leg double-ended rupture in the steam generator compartment; 20 node, cold leg double-ended rupture in the steam generator compartment, and both the pressurizer surge and spray line breaks in the pressurizer compartment. Using the applicant's input data and the COMPARE computer program, we were able to verify the applicant's results. We conclude that the noding models and the calculated pressure transients for both the reactor cavity and pressurizer compartments are acceptable for use in the subcompartment structural analysis.

Similar to the applicant's results, we observed that the calculated differential pressures in the break nodes of the steam generator compartment would slightly exceed the specified design pressure. However, the applicant has pointed out that since the calculated pressure acting on the structural walls vary spatially, the integrated load on the structural walls is less than the design load. A discussion of the acceptability of the steam generator compartment structural wall design is provided in Section 3.8.2 of this Safety Evaluation Report.

The applicant has provided the transient loads and moments acting on the reactor vessel, steam generators, and reactor coolant pumps for the component supports design evaluation. Sufficient justification has also been provided for the nodalization of the subcompartments for the component supports design evaluation.

We have reviewed the applicant's analyses and conclude that an acceptable model has been developed for use in the design of the component supports. Upon resolution of the NRC staff's Generic Task A-2, "Asymmetric LOCA Loads," we will further review the subcompartment analysis. We will require the applicant to comply with any analytical or design requirements resulting from the resolution of Generic Task A-2.

ECCS Containment Pressure Evaluation

Appendix K to 10 CFR Part 50 requires that the effect of the operation of all installed containment pressure reducing systems and processes be included in the emergency core cooling system evaluation. For this evaluation, it is conservative to minimize the containment pressure since this will increase the resistance to steam flow in the reactor coolant loops and reduce the reflood rate in the core. Following a loss-of-coolant accident, the pressure in the containment building will be increased by the addition of steam and water from the primary reactor system to the containment atmosphere. After initial blowdown, heat transfer from the core, primary metal structures, and steam generators to the emergency core cooling system water will produce additional steam. This steam together with any emergency core cooling system water spilled from the primary system will flow through the postulated break and into the containment. This energy will be released to the containment during both the blowdown and later, the emergency core cooling system operational phases, i.e., reflood and post-reflood.

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

The emergency core cooling system containment backpressure calculations were performed with the Westinghouse emergency core cooling system evaluation model of October 1975. We have reviewed this model and concluded that it is acceptable for the evaluation of the containment backpresssure, subject to the review of the facility-dependent input parameters used in the analysis. We have reviewed the facility parameters used for the analysis of the containment pressure for emergency cooling system evaluation and find them to be suitably conservative. We, therefore, conclude that the containment pressure analysis for emergency core cooling system evaluation is acceptable and meets the requirements of Appendix K to 10 CFR Part 50.

Conclusions

We have evaluated the containment system functional design for conformance with the General Design Criteria and, in particular, Criteria 16 and 50. We conclude that the containment internal and external design pressure are acceptable and that the subcompartment analysis is adequate for the determination of loads on subcompartment structural walls and component supports.

6.2.2 Containment Heat Removal Systems

The containment heat removal systems for the facility consist of the containment spray systems and the containment emergency fan cooler system.

The containment heat removal systems return the containment pressure to a low value following a break in either the primary or secondary system piping inside the containment. Heat is transferred from the containment atmosphere to the spray water and the containment emergency fan cooler system, respectively. In addition, spray water drawn from the containment engineered safety feature sump is cooled via the residual heat removal heat exchangers in the recirculation mode of safety injection system operation.

The containment spray system consists of two redundant and independent trains. The containment spray system serves as an engineered safety feature and will not be used for normal operation. The system will be safety grade (Quality Group B and seismic Category I) and all active components will be located outside of the containment building.

The containment spray system is automatically initiated by a containment spray actuation signal that is initiated by the combination of any two containment high pressure signals (with a setpoint of 11.4 pounds per square inch gauge) and a safety injection signal. The containment spray actuation signal which may also be initiated manually in the control room, starts the containment spray pumps and opens the spray control valves to the containment. The spray water is discharged into the containment upper region through spray nozzles arranged on headers. The containment spray pumps initially take suction from both the refueling water storage tank and the sodium hydroxide storage tank. When a predetermined low level is reached in the refueling water storage tank, low level alarms are actuated and the operator stops the spray pumps and closes the valves in the suction lines. The containment spray pump suction is then automatically switched from the refueling water storage tank to the containment emergency sump.

The facility has two physically separated recirculation sumps. An outer trash rack acts to prevent large pieces of debris from entering the sumps. Two inner, fine screens surround the intake to each sump; the screens are sized to prevent the entrance of particles which may degrade systems served by the pump. Furthermore, the sumps are designed to minimize vortex formation.

We have reviewed the design of the containment emergency sumps to determine the extent of compliance with the provisions of Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems." Since the construction permit for this facility was issued prior to the publication of Regulatory Guide 1.82, full compliance with the Regulatory Guide is not required. However, we have found that the sump design does satisfy most of the provisions of the guide and conclude that the design is acceptable.

Sufficient net positive suction head will be available to the spray pumps for the recirculation mode of operation. The applicant's evaluation of the available net positive suction head is consistent with the guidelines of Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." The results of the applicant's evaluation show that the available net positive suction head for the containment spray pumps in trains A and B are 27.2 feet and 28.1 feet respectively. The required net positive suction head for a spray pump is 20 feet.

The reactor building cooling system consists of four separate fan cooler units inside the containment. The reactor building cooling system is used during normal plant operation and as an engineered safety feature system in the event of an accident. The reactor building cooling system is separated into two trains with two fan cooler units on each train. The two trains are supplied from separate cooling water trains and power sources.

In the event of an accident, the reactor building cooling system will automatically switch from a normal to an emergency operating mode. This involves changing motor speeds from high to low, changing the source of cooling water from the industrial cooling water system to the service water system, and closing the reactor building high efficiency particulate air filter bypass damper.

The automatic switchover is initiated upon receipt of either a safety injection or a loss of offsite power signal, or can be manually initiated from the control room.

Based on our review of the containment heat removal system, we conclude that the system design is in accordance with the requirements of Criteria 38, 39, and 40 of the General Design Criteria and is, therefore, acceptable.

6.2.3 Containment Isolation System

The containment isolation system is designed to automatically isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of closed systems and isolation valves, is provided to assure that no single active failure will result in the loss of containment integrity. The containment isolation provisions are of safety grade design (ASME Boiler and Pressure Vessel Code, Section III, Class 2, and seismic Category I) and are protected from missiles.

Criteria 54 through 57 of the General Design Criteria explicitly state the isolation requirements for piping systems that penetrate the containment. However, Criteria 55 and 56 also allow for deviations from the explicit isolation requirements if the isolation provisions can be found acceptable on some other defined basis. In this regard, the isolation provisions for lines have been found acceptable for the reasons given below:

1. Containment Emergency Sump Recirculation

Criterion 56 of the General Design Criteria requires each line that connects directly to the containment atmosphere and penetrates primary reactor containment to have two containment isolation valves, one inside containment and one outside containment. The containment isolation valves must be locked closed or capable of automatic isolation.

The containment sump suction lines are part of the emergency core cooling system and the containment heat removal system, and must be opened following a loss-of-coolant accident to satisfy their post-accident functional requirement, which is to permit long-term cooling of the reactor core and the containment atmosphere. As a result, automatic isolation capability of these lines is not desirable and remote manual isolation capability is provided for the isolation valves. Also, the containment sump suction lines each have only a single containment isolation valve. The valve, located outside containment, is in a concentric, leaktight guardpipe which acts as an extension of the containment. Since the emergency core cooling system reliability is greater with only a single isolation valve in the line, we find this isolation arrangement acceptable for the containment sump suction lines. The emergency core cooling system, which is a closed engineered safety feature grade system outside containment, serves as the second containment isolation barrier. We have concluded that the isolation provisions for these lines represent an acceptable defined basis which differs from the requirements of Criterion 56 of the General Design Criteria regarding the number of isolation valves and the actuation provisions for them.

2. Safety Injection Lines

Criterion 55 of the General Design Criteria requires each line that is part of the reactor coolant boundary and penetrates primary reactor containment to have two containment isolation valves, one inside containment and one outside containment. The containment isolation valves must be either locked closed or capable of automatic isolation.

The containment isolation provisions for certain emergency core cooling system safety injection lines consists of a check valve inside containment and a remote manual valve outside containment. A remote manual isolation valve is provided in lieu of an automatic isolation valve because the lines which are part of the emergency core cooling system, have a post-accident safety function. We have concluded that the isolation provisions for these lines represent an acceptable defined basis which differs from the requirements of Criterion 55 of the General Design Criteria regarding the actuation provisions for the valves outside containment.

3. Residual Heat Removal System Return Lines From Hot Legs to Residual Heat Removal System Pumps)

The system isolation provisions for the residual heat removal system return lines consist of two normally closed, motor-operated gate valves in series inside containment. The valves are interlocked to prevent them from being inadvertently opened. Since the residual heat removal system return lines have no post-accident safety function, they remain isolated following a loss-of-coolant accident. Also, the lines connect to the closed, emergency core cooling system outside containment. In view of the above system design considerations, we have concluded that the normally closed, system isolation valve closest to the containment and the closed, engineered safety feature system outside containment represent an acceptable defined basis which differs from the requirements of Criteria 55 of the General Design Criteria regarding the number of containment isolation valves.

4. Reactor Coolant Pump Seal Injection Lines

The containment isolation provisions for the reactor coolant pump seal injection lines consist of a check valve inside containment and a remote manual isolation valve outside containment. A remote manual valve is preferred over an automatic isolation valve since closure of this valve due to a spurious containment isolation actuation signal could result in reactor coolant pump damage.

The reactor coolant pump seal injection system is connected to the emergency core cooling system which is a closed, engineered safety features system, and is of the same design quality. In view of this, we have concluded that the remote manual actuation of the isolation valve outside containment represents an acceptable defined basis which differs from the requirements of Criterion 55 of the General Design Criteria regarding the actuation provisions.

Our review of the containment isolation system includes verification that there is diversity of parameters sensed for the initiation of containment isolation, as called for by Section 6.2.4 of the Standard Review Plan. The containment isolation system design meets this requirement. The parameters sensed for the initiation of containment isolation include high containment pressure, and the various parameters sensed for safety injection system actuation.

Our review of the containment isolation system has also included the containment purge system which will be used to reduce airborne radioactivity in the containment and allow personnel entry. The facility has both a six-inch and 36-inch purge system.

With respect to the 36-inch purge system, the applicant has not addressed the analyses called for in Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations." Therefore, we will include in the Technical Specifications a requirement that this system be locked closed during all operating modes requiring containment integrity; i.e., startup, normal operation, hot standby, and hot shutdown.

The six-inch purge system has been designed to operate during operating modes requiring containment integrity. The applicant has performed an analysis of the consequences of a loss-of-coolant accident occurring while purging the containment, using the guidelines of Branch Technical Position CSB 6-4. The analysis was done to determine the offsite radiological consequences from the release of containment atmosphere to the environs and to determine the effect on the emergency core cooling system effectiveness. The analysis was done assuming only the six-inch purge system was operating.

The applicant has included the incremental dose resulting from the release through the purge lines in the design basis accident loss-of-coolant accident analysis, and has concluded that the total dose is well within the limits of 10 CFR Part 100. In addition, the applicant's analysis shows that the amount of steam which escapes through the purge isolation valves prior to closure has an insignificant effect on the containment backpressure used in the emergency core cooling system evaluation. See Section 6.2.1 of the Safety Evaluation Report for a discussion of the emergency core cooling system backpressure calculation.

The six-inch purge isolation valves will close within five seconds after receipt of an isolation signal. Diverse parameters, including safety injection, containment high pressure, and high radiation signals, are sensed to initiate automatic isolation of the purge system isolation valves. These valves are also capable of being manually closed by the operator from the control room.

We have determined that the six-inch purge system isolation valves are active valves and must be capable of closing during the design basis loss-of-coolant accident. The valve operability is addressed in Section 3.9.3 of this Safety Evaluation Report.

Based on our review, we conclude that the containment isolation system design conforms to Criteria 54, 55, 56, and 57 of the General Design Criteria, Regulatory Guide 1.141, "Containment Isolation Provisions for Fluid Systems," and the provisions of Section 6.2.4 of the Standard Review Plan. Therefore, we conclude that the system is acceptable.

6.2.4 Combustible Gas Control System

Following a loss-of-coolant accident, hydrogen may accumulate inside the containment as a result of (1) a chemical reaction between the fuel rod cladding and steam resulting from vaporization of emergency core cooling water, (2) corrosion of construction materials by the spray solution, and (3) radiolytic decomposition of the cooling water in the reactor core and the containment sumps.

In order to limit the hydrogen accumulation in the containment, the applicant has provided a combustible gas control system. The combustible gas control system consists of a hydrogen monitoring subsystem that measures the containment atmosphere, hydrogen concentration, and a hydrogen recombiner subsystem that provides the means of reducing the containment hydrogen concentration.

The hydrogen recombiner subsystem is designed to meet the quality assurance, redundancy, energy source, and instrumentation requirements of an engineered safety feature, and is described in the Westinghouse Topical Report, WCAP 7820, "Electrical Hydrogen Recombiner for PWR Containments," Supplements 1, 2, 3, 4, and 6. We have previously reviewed this topical report and found it acceptable for reference.

The two recombiners located inside containment are seismic Category I design, powered from separate Class 1E electric buses and are designed to function in the post-accident containment environment.

The facility has two hydrogen analyzers to monitor the hydrogen concentration within the containment.

In Amendment 13 of the Final Safety Analysis Report, the applicant provided the following reasons to justify the adequacy of a single hydrogen concentration monitor:

1. Redundant manual sampling capability exists.

2. The hydrogen analyzer can be calibrated against a known sample, onsite, by facility personnel.

3. Sufficient time exists following an accident to obtain an additional hydrogen concentration monitor from offsite sources in the unlikely event of failure of the existing hydrogen concentration monitor and manual sampling capability. A replacement analyzer is available from other locations within two hours.

4. A hydrogen grab sample can be obtained and analyzed on-site by facility personnel.

Although there is only one hydrogen concentration monitor on-site, there are other means available to monitor hydrogen concentrations inside the containment such as manual sampling and grab sampling. We agree with the applicant that one hydrogen concentration monitor with manual sampling and grab sampling as a backup meets the guidelines of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

Based on our review, we conclude that the hydrogen monitoring system is acceptable.

Natural convection currents within the containment following an accident will mix the containment atmosphere sufficiently to preclude high concentrations of combustible gases from occurring locally. However, mixing of the containment atmosphere will also be enhanced by the containment spray system and the containment emergency fan cooler system, both of which are designed to engineered safety feature system criteria. This will assure that samples drawn by the hydrogen monitoring subsystem are representative of the containment atmosphere.

The applicant has performed an analysis of the post-accident production and accumulation of hydrogen within the containment that is consistent with the provisions of Regulatory Guide 1.7 and Branch Technical Position CSB 6-2. The applicant's analysis was performed assuming the operation of one recombiner, actuated one day after the loss-of-coolant accident. The analysis showed that the hydrogen concentration would remain below the lower flammability limit of four volume percent.

In the Final Safety Analysis Report, sufficient information did not exist to determine that the applicant properly considered the zinc- and aluminum-based coatings as a source of hydrogen in the hydrogen accumulation analysis. We required additional information from the applicant to determine if zinc- and aluminum-based coatings were considered as hydrogen sources.

Using the additional information provided by the applicant in Amendment 13 to the Final Safety Analysis Report, we performed confirmatory analysis of the hydrogen accumulation and verified that the applicant properly considered the zinc- and aluminum-based coatings as sources of hydrogen.

Based on our review, we conclude that the hydrogen accumulation analysis conforms to Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," Revision 2 and the provisions of Section 6.2.5 of the Standard Review Plan is, therefore, acceptable.

We have reviewed the proposed combustible gas control system with regard to the design and performance requirements of Criteria 41, 42, and 43 of the General Design Criteria and section 50.44 of 10 CFR Part 50, "Standards for Combustible Gas Control Systems in Light Water-Cooled Power Reactors," and Regulatory Guide 1.7, Revision 2. We conclude that the design of the combustible gas control system meets our criteria and is acceptable.

6.2.5 Containment Leakage Testing Program

We have reviewed the applicant's containment leak testing program, as presented in Section 6.2 of the Final Safety Analysis Report, for compliance with the containment leakage testing requirements specified in Appendix J to 10 CFR Part 50. [illegible] compliance will provide adequate assurance that the containment leak-tight integrity can be verified throughout its service lifetime and that the leakage rates will be periodically checked during service on a timely basis to maintain the leakage within specified limits. Maintaining containment leakage within limits provides reasonable assurance that, in the event of any radioactivity release within the containment, the loss of the containment atmosphere through leak paths will not be in excess of the limits specified for the site.

The applicant has provided a detailed discussion of the containment integrated leak rate (Type A) test procedure and acceptance criteria. All systems penetrating containment are identified as being either vented and drained to the containment atmosphere so that the accident differential pressure will exist across the containment isolation valves, or the systems are identified as remaining fluid filled for the Type A test. Justification was provided for each system that was not vented and drained to the containment atmopshere for the Type A test.

The applicant has listed all the containment penetrations and has itemized all the local leak testing that will be performed. Schematic drawings of each piping system penetrating containment have been submitted showing the isolation valve arrangements. The location of test connections and vents for each isolation valve subject to local (Type C) leak testing is such that the test pressure will be applied in the same direction as the pressure existing when the valve performs its safety function.

With the exception of the secondary system penetrations, all containment penetrations will be subject to local Type B (electrical penetrations, personnel air locks, and flanged penetrations) or Type C leak tests.

If primary to secondary steam generator tube leakage is postulated to occur, containment atmosphere leakage could pass through the steam generator and the secondary system isolation valves would become containment atmosphere leak paths. However, post loss-of-coolant accident procedures call for covering the steam generator tube bundles with feedwater. By the time the steam generator depressurizes to the containment pressure, the head of feedwater will prevent atmospheric leakage from occurring across the tube bundles. We concur with the applicant that the secondary system containment isolation valves will not become potential containment leak paths and, therefore, local Type C leak tests should not be required for these valves.

Closed systems outside containment (e.g., the emergency core cooling system and the containment spray system) will become extensions of the containment boundary following a loss-of-coolant accident. We have requested that the applicant provide the capability for leak testing of the closed systems located outside containment and to include the leakage in the off-site dose calculation. The applicant has committed to include, in the plant Technical Specifications, requirements for periodic hydrostatic testing of the emergency core cooling system and containment spray system located outside containment.

Section III.D.2 of Appendix J to 10 CFR 50 requires airlocks to be leak tested at six-month intervals, and after each opening during the intervals. Section III.B.2 of Appendix J requires all penetrations to be leak tested at the calculated peak containment internal pressure, Pa, corresponding to the design basis accident.

Based on plant operating experience, requiring an airlock to be leak tested after each opening is an impractical requirement when frequent airlock usage is necessary over a short period of time. Furthermore, the airlock design for the facility incorporates dual seals on the airlock doors with the capability to pressurize the volume between the seals. Therefore, the applicant proposes to leak test the airlock door seals within three days after opening an airlock. This will permit door seal integrity to be demonstrated without pressurizing the entire airlock. This is an acceptable test method for tests other than the six-month test. Testing of the door seals is more practical and still provides the desired confidence that the leak tightness of the airlock is within acceptable limits.

The airlock door seal tests will be performed at a pressure less than the calculated peak containment internal pressure related to the design basis accident. The acceptance criteria for the door seal tests is that seal leakage be less than, or equal to 0.01 times the maximum allowable leakage rate at the calculated peak containment internal pressure related to the design bases accident when the volume between the seals is pressurized to eight pounds per square inch gauge for at least 30 seconds. The lower test pressure of eight pounds per square inch, gauge is sufficient to verify that door seal integrity is being maintained and that the door seals are free of dirt and foreign objects. The test pressure is recommended by the air lock manufacturer, and testing at the lower pressure is expected to extend the seal life. We, conclude that the use of a test pressure of eight pounds per square inch gauge for the door tests is acceptable, although it is lower than the test pressure called for by Appendix J to 10 CFR Part 50.

Additional NRC staff effort on containment leak testing that will lead to a revision of Appendix J to 10 CFR Part 50 is being done in conjunction with the Office of Standards Development. The outcome of this task will be applicable to all plants depending on their licensing status and design.

We conclude that the design of the reactor building and associated systems will permit leakage rate testing in compliance with the proposed "Reactor Containment Leakage Testing for Water-Cooled Power Reactors," 50.54 (o), Appendix J, published in the Federal Register on August 27, 1971.

6.3 Emergency Core Cooling System

6.3.1 Design Bases

The emergency core cooling system is designed to cool the reactor as well as to provide additional shutdown capability for the following postulated accidents:

1. Pipe breaks in the reactor coolant system which cause a discharge greater than that which can be made up by the normal makeup system, up to and including the double-ended break of the largest pipe in the reactor coolant system.

2. Rupture of a control rod drive mechanism causing a rod cluster control assembly ejection.

3. Pipe breaks in the secondary system including the double-ended break of the largest steam system pipe.

4. A steam generator tube rupture.

The Virgil C. Summer Nuclear Station, Unit 1 has a core output of 2775 megawatts thermal. The emergency core cooling systems' design bases are to prevent fuel and cladding damage that would interfere with adequate emergency core cooling and to mitigate the amount of clad-water reaction for any size break up to and including a double-ended rupture of the largest primary coolant line. The applicant has stated that the level of performance for core cooling is met even with minimum engineered safeguards available, such as the failure of a diesel generator to start or component failure of the electrical supply bus. The Virgil C. Summer Nuclear Station, Unit 1 emergency core cooling system is designed to withstand the safe shutdown earthquake and to have the required number, diversity, reliability, and redundancy of components such that no single active component failure during the short-term or no single active or passive failure during the long-term of an accident results in inadequate cooling of the reactor core.

The boric acid injection portion of the emergency core cooling system is designed to control the reactivity insertion accompanying the rapid cooldown following any single steam line rupture or spurious relief valve lifting. Control of the reactivity insertion is accomplished by injection of high concentration boric acid solution into the reactor coolant system. The range of steam line ruptures protected against is up to and including the double-ended circumferential rupture of the largest pipe in the steam system.

6.3.2 System Design

The emergency core cooling system design for the facility consists of two independent subsystem trains. Each series aligned train consists of a high head safety injection/centrifugal charging pump, a low head residual heat removal pump, and associated valves in the flow path providing sufficient borated water flow capacity needed to satisfy the design basis. Power sources for actuation of the emergency core cooling system components will be supplied from separate emergency buses and separate diesel generators in the event of loss of offsite power. The pumps in each train for the emergency core cooling system injection mode of operation will initially take suction from the refueling water storage tank and deliver flow to the reactor through three cold leg connections. For the recirculation mode of operation, the low head residual heat removal pump in each train will automatically switch over to take suction from the containment sump upon receiving a lo-lo refueling water storage tank level alarm in the control room and the suction of the high head safety injection pumps will be aligned to the discharge of the low head pumps. Following a postulated loss-of-coolant accident, the emergency core cooling system will operate initially in the active high head injection mode and passive accumulator mode, then in the active low head injection mode, and finally in the recirculation mode.

The accumulators represent a passive system since the only mechanical operation which is required for injection is from two swing disc check valves in series, which occurs when the primary coolant system pressure decreases to 600 pounds per square inch. Each of the three accumulators is connected to one of the reactor coolant system cold legs. These accumulators each have a total volume of 1450 cubic feet with a nominal volume of 450 cubic feet containing nitrogen gas. A normally open motor-operated gate valve is located in the lines between the accumulator and cold leg piping. Administrative procedures identified in the Technical Specifications will require the power to these normally open accumulator isolation valve motor operators to be disconnected by removal of the breaker from the circuit during reactor power operation. This procedure provides additional assurance that these valves will be open during power operation when the accumulators are needed.

The high pressure injection mode, upon actuation of a safety injection signal, consists of operation of two high head safety injection/centrifugal charging pumps (rated at 150 gallons per minute each at a design head of 5800 feet) which provide high pressure injection of boric acid solution (the boron injection tank maintaining a nominal boron concentration of 21,000 parts per million) into the reactor coolant system. These pumps are automatically aligned to take suction from the refueling water storage tank with a minimum boron concentration of 2000 parts per million.

The boric acid injection portion of the emergency core cooling system consists of the boron injection tank, boron injection surge tank, boron injection recirculation loop, charging pumps, and the associated valves. The boron injection tank contains 900 gallons of 21,000 parts per million boric acid solution and is connected to the reactor coolant system by means of a loop from the refueling water storage tank, through the high head safety injection/centrifugal charging pumps, to the boron injection tank inlet. The boron injection tank outlet is connected through a common manifold pipe to pipes connected to each of the three reactor coolant cold legs. The boron injection surge tank contains 75 gallons of the same concentration of boric acid as the boron injection tank and is used to provide surge capacity for the boron injection tank recirculation loop. During normal operation the boron acid solution is continuously recirculated by the two recirculation pumps in a closed loop consisting of the boron injection tank and boron injection surge tank to maintain mixing and prevent stratification. The safety injection signal automatically stops the recirculation pumps and closes the valves in the recirculation lines. Redundant tank heaters, line heat tracing cables and low temperature alarms are provided to assure that the boron will remain in solution.

Low pressure injection is provided by two residual heat removal low head safety injection pumps, rated at 3750 gallons per minute each at a design head of 240 feet. These pumps take their suction from the refueling water storage tank. Upon actuation of the low level set point from the refueling water storage tank, automatic switchover from the injection to the recirculation mode occurs. In the recirculation mode, the long-term cooling requirements are achieved by recirculating the spilled reactor coolant from the rupture pipe (collected in the sump) back through the cold legs to the reactor vessel. This automatic action opens the two sump suction valves in each train to align the residual heat removal pumps, which continue to operate during the transfer, to take suction from the containment sump. The two charging pumps continue to take suction from the refueling water storage tank, following the above automatic action, until manual operator action is taken to align the suction of the charging pumps to the discharge of the residual heat removal pumps.

To reduce the potential for boron precipitation subsequent to a loss-of-coolant accident, after switchover to the recirculation mode, the applicant has stated that the emergency core cooling system will be manually realigned alternately between the hot leg and cold leg recirculation mode.

6.3.3 Design Evaluation

As discussed above, the emergency core cooling system includes the piping, valves, pumps, heat exchangers, instrumentation, and controls used to transfer heat from the core following a loss-of-coolant accident. The scope of review of the emergency core cooling system for the facility included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analyses, and design specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the emergency core cooling system and the manner in which the design conforms to these criteria and bases. Specifically, we evaluated the system's ability to withstand a single active failure during the short term or a single active or passive failure during the long term following a postulated loss-of-coolant accident.

Manually controlled electrically operated valves in the emergency core cooling system which will have power lock out have been identified. These valves are:

1. Accumulator isolation valves: 8808A,B, and C

2. Low head safety injection/residual heat removal pumps discharge to hot leg valve: 8889

3. High head safety injection/centrifugal charging pump discharge to hot leg valves: 8884 and 8886

We find the lock out of the valves identified and the proposed method for locking out to be acceptable.

The electrical review of the emergency core cooling system with regard to single failure of the instrumentation and controls is provided in Section 7.3 of this Safety Evaluation Report, and of the electrical power supplies in Section 8.3 of this Safety Evaluation Report.

The applicant has also identified the manual (handwheel) valves critical to the operation of the emergency core cooling system, if inadvertently left in the wrong position. The applicant has stated that these valves all have locking provisions, are administratively controlled, and that a system-level indication has been provided on the main board in the control room for bypass or inoperable status of these valves for compliance with Regulatory Guide 1.47. During our review we expressed concern regarding the potential consequences of an inadvertent closure of the single manual refueling water storage tank isolation valve (6700 SF). The applicant, in response to our request, has confirmed in Amendment 14 that in addition to the lock-open feature provided for this valve, position indication is provided in the main control room. We find this acceptable.

The potential for valve motors and associated valve control/actuation to become submerged within the containment following a loss-of-coolant accident has been reviewed. In the calculation of the maximum predicted water level inside containment, the applicant conservatively accounted for the total amount of water available to flood the bottom of the reactor building, the volumes of all significant structures, equipment, and sumps, and the effective height of the reactor building basement floor. Based on these calculations, the applicant has identified the safety-grade valves which have some portion of the electrical circuits associated with the valve operators below the maximum predicted water level. These valves were identified as not being required to function after being flooded. We therefore conclude that their submergence would not inhibit effective emergency core cooling system operation.

We have recently been notified of a potential design deficiency regarding double sealing gate valves which are used in the emergency core cooling systems of some pressurized water reactor plants. The concern is that when fluids, trapped in the interal body cavity of the valve, are heated due to the increased temperatures of adjacent piping systems or of the environment, substantial pressure increases may result in these cavities that could rupture the valve.

In response to our concern, the applicant stated that the only gate valves of the double-disk design used in the facility are the three main feedwater containment isolation valves. These valves, however, have incorporated in this design a trapped fluid release feature between the parallel disks to prevent overpressurization of the internal body cavity. We find this acceptable.

With regard to long-term recirculation cooling, the applicant was requested to evaluate the emergency core cooling system performance by applying a single failure analysis; which includes considering limited passive as well as active failures in the emergency core cooling system (assuming no prior failure during the short-term phase). As part of the passive failure evaluation, the applicant addressed the concern of post-loss-of-coolant accident water leakage from emergency core cooling system components such as a failed pump seal or valve stem packing which could degrade more than one subsystem. The maximum leakage from a passive failure has been determined by the applicant to be less than 50 gallons per minute for the case of a sudden pump shaft seal failure. Valve packing leaks have been considered to be less severe than the pump seal failure. The detection of leakage from emergency core cooling system components is accomplished by monitoring the sump levels in the engineered safety features areas of the auxiliary building. The leakage collected in the sumps in this building is detected by instruments sensitive enough to initiate (by alarm) operator action to isolate a 50 gallons per minute leak. The leak detection system has the capability to identify the faulted emergency core cooling system train and allow isolation of the leak prior to adversely affecting other systems by flooding. The system design is based on not assuming any operator action for 30 minutes prior to isolation of a leak. The level sensors, power supplies, and alarms meet the criteria in IEEE Standard-279 with the exception that the detection system need not meet the single failure requirements. We find this acceptable.

Manual switchover to hot leg recirculation for long-term cooling following a loss-of-coolant accident is employed as the means to prevent excessive boron precipitation. The applicant has determined that this switchover between hot leg and cold leg recirculation should occur every 24 hours, after initiation of hot leg recirculation, which occurs 14 hours after the loss-of-coolant accident. We have determined that the method to be used by the applicant for preventing excessive boron precipitation is acceptable and have verified the adequacy of the times identified by the applicant for initial and subsequent switchovers between hot and cold leg recirculation.

We have reviewed the loss-of-coolant accident emergency procedures which may permit early manual reset of the safety injection signal during the injection phase. The applicant has stated that the loss-of-coolant emergency operating procedure provides steps to assure that no operator action will be taken to reset a safety injection signal resulting from a loss-of-coolant accident following a safety injection signal unless the following conditions are met:

1. Reactor coolant system pressure is greater than 2000 pounds per square inch absolute and increasing.

2. Pressurizer level is greater than 50 percent.

3. Steam generator level indication is in the narrow range or the emergency feedwater flow is at least 285 gallons per minute.

4. Reactor coolant system must be subcooled by at least 50 Fahrenheit degrees.

This procedure is consistent with the NRC staff position outlined in NUREG-0138.

In the event of a loss of offsite power during the injection mode following safety injection signal reset the operator had to manually reset the residual heat removal pumps. The staff found this position to be unacceptable, and per our request, the applicant changed the residual heat removal pumps to be automatically sequenced onto the diesel following a manual reset of the safety injection signal that is followed by an offsite power failure. We now find this acceptable.

The applicant's design bases for the refueling water storage tank included allowances for the following:

1. injection in the event of a loss-of-coolant accident (386,950 gallons)
2. instrumentation error (25,070 gallons)
3. working allowance (6,850 gallons)
4. transfer allowance (43,240 gallons)
5. single failure allowance (17,425 gallons)
6. unusable volume (50,760 gallons)

The minimum injection requirement for the loss-of-coolant accident case is 350,000 gallons. The refueling water storage tank is designed to be kept at the "normal full" level during normal power operation. The volume of water in the refueling water storage tank between the "normal full" level and the "lo-lo" level is 386,950 gallons. The volume of water available in the refueling water storage tank between the "lo-lo" level with instrument error and the minimum submergence level for the outlet nozzle is 43,240 gallons. This volume is the transfer allowance and includes single failure allowance. The unusable volume in the refueling water storage tank, from the tank bottom to the minimum submergency level is 50,760 gallons. At the time of the "lo-lo" level in the refueling water storage tank, the residual heat removal system suction valves from the reactor building recirculation sumps automatically open. To assure that the emergency core cooling system pumps have a suction source, the operator must complete the following actions following switchover from injection to recirculation mode and prior to loss of remaining refueling capacity. These will be accomplished within the approximate times indicated.

<u>Action 1</u>: Stop both reactor building spray pumps (30 seconds)

<u>Action 2</u>: Verify that the residual heat removal system suction valves from the reactor building recirculation sumps are open and close the refueling water storage tank to residual heat removal system suction valves (one minute)

<u>Action 3</u>: Close valves in the 10 inch crossover leg between the residual heat removal system discharge lines (30 seconds)

<u>Action 4</u>: Open the residual heat removal system to charing pump suction valves (30 seconds)

The staff finds the required operator actions and the time needed to perform these actions satisfactorily. The staff also finds the capacity of the refueling water storage tank to be sufficient.

The instrumentation needed to monitor and control the emergency core cooling system equipment following a loss-of-coolant accident has been reviewed. The applicant has provided safety grade flow indicators for high pressure injection flow monitoring and control grade flow indicators for low pressure injection flow monitoring. This emergency core cooling system flow monitoring system design will be evaluated against the recommendations in Regulatory Guide 1.97 (not yet issued) and our evaluation will be reported in a supplement to this Safety Evaluation Report.

6.3.4 Performance Evaluation

The applicant has submitted an evaluation of the emergency core cooling system performance pursuant to the requirements of Section 50.46 of 10 CFR Part 50 of the NRC's regulations. The results of small and large break loss-of-coolant accident analyses have been provided in Section 15.3.1 and 15.4.1, respectively, of the Final Safety Analysis Report. The analyses submitted by the applicant for both the large and small breaks did not contain the correction for the error in the metal/water reaction heat generation rate that has been identified by Westinghouse.

On March 23, 1978, Westinghouse informed the NRC that a programming error was uncovered in the metal/water reactor correlation used in the LOCTA computer program. The LOCTA code is part of the Westinghouse licensing model used for evaluating emergency core cooling system performance in the event of loss of coolant accidents. On August 23, 1978, the staff approved a revised version of the LOCTA code in which the metal/water reactor coding error was corrected and thus conformed to applicable regulations. The applicant was requested to resubmit analyses for the limiting small- and large-break loss-of-coolant accidents using the corrected version of the LOCTA code. The staff has informally received new analyses performed with the corrected, approved LOCTA code. These new analyses demonstrate that the facility does conform to present licensing requirements. The staff requires that the revised analyses be incorporated in the Final Safety Analysis Report prior to issuance of a license for full power operation.

The information currently provided by the applicant in the Final Safety Analysis Report for the small and large break is described below.

The large break loss-of-coolant accident analysis was limited to a spectrum of three double-ended guillotine breaks with discharge coefficients of 0.4, 0.6, and 1.0. To supplement the analysis, the applicant has referenced the Westinghouse Topical Report WCAP-8853 which covers other break sizes, types, and locations, and demonstrates that the double-ended cold leg breaks in the reactor coolant pump discharge piping are the limiting breaks for this type plant. In this generic study and in the applicant's analyses the upper head temperature of the reactor vessel was assumed to correspond to the hot leg temperature.

The analyses submitted by the applicant identified the worst break as the double-ended cold leg guillotine break with a discharge coefficient (Moody multiplier) of 0.6. The calculated peak clad temperature was 2067 degrees Fahrenheit which is within the acceptable limit of 2200 degrees Fahrenheit as specified in Section 50.46(b) of 10 CFR Part 50. In addition, the maximum local metal/water reaction of 6.79 percent and total core-wide metal/water reaction of less than 0.3 percent were well below the allowable limits of 17 percent and one percent, respectively.

The analyses were performed based on an assumed total peaking factor of 2.32, 102 percent of the engineered safety features nuclear steam supply system power of 2900 megawatts thermal, and 102 percent of a peak linear power density of 12.63 kilowatts per foot. Analyses were performed using the combination of emergency core cooling subsystems available assuming the most severe single failure to conform to the requirements of Appendix K to 10 CFR Part 50.

The applicant performed the small break loss-of-coolant analyses with an approved version of the Westinghouse emergency core cooling system evaluation model presented in Westinghouse Topical Report WCAP-8971. The postulated small break size (less than 1.0 square foot) loss-of-coolant accident analysis included a three-break spectrum. The break sizes evaluated were three-inch, four-inch, and six-inch breaks. Table 15.3-2 of the Final Safety Analysis Report did not conclusively demonstrate that the limiting small-break loss-of-coolant accident was evaluated. The predicted peak clad temperature for the six-, four-, and three-inch cold leg breaks increased with decreasing break area. The staff requested the applicant to provide confirmatory analyses that the three-inch cold leg break was indeed limiting. During the reevaluation of the small-break loss-of-coolant accident, an input error to the WFLASH computer code was uncovered. The applicant, therefore, evaluated the small-break spectrum with the input error corrected, which resulted in a decrease in peak clad temperature of 100 degrees Fahrenheit for the three-inch cold leg break. The revised analyses included a two-inch break, which confirmed that the three-inch cold leg break was indeed limiting. The staff therefore concludes that the facility complies with the requirements of Section 50.46 of and Appendix K to 10 CFR Part 50 regarding emergency core cooling system performance. Prior to issuance of a full power operating license the applicant will be required to revise Table 15.3-2 and the corresponding figures of the Final Safety Analysis Report for the small- break loss-of-coolant spectrum.

In response to our question on manual actions following a small break loss-of-coolant accident, the applicant has stated the operator is instructed to control the pressurizer and steam generator levels. This requires control of the safety injection and feedwater flows. However, credit for operator action is taken after satisfying the four conditions specified in Section 6.3.3 of this Safety Evaln Report. These flows have to be terminated to mitigate the consequences of this accident on reactor vessel degradation. Current licensing criteria for material fracture toughness and initial safety margins are under generic review in Task Action Plan A-11. Appendix C of this Safety Evaluation Report discusses the basis for continued reactor operation for the Virgil C. Summer Nuclear Station, Unit 1.

6.3.5 Tests and Inspections

The applicant has performed flow tests on the containment recirculation sump to meet the requirements of Regulatory Guides 1.68 and 1.79 to demonstrate the performance of the recirculation sump. However, in-plant sump tests did not accurately replicate expected post-loss-of-coolant accident conditions, and this did not fully confirm acceptable sump performance under emergency core cooling system recirculation conditions. Specifically, the plant tests only took suction from a single line, when there are two lines in each of two sumps. This resulted in approach flow velocities which were lower in the test than would be expected during actual loss-of-coolant accident conditions.

Additionally, various flow approach directions were not investigated to determine if undesirable rotation could be induced in the sump area, which could lead to vortex formation.

Finally, sump screen blockage due to debris entrainment was not considered, with the correspondingly higher screen velocities which also could aggravate vortex formation.

The applicant plans to conduct a scale model sump test to investigate these areas and to demonstrate that recirculation sump performance will be acceptable in the expected post loss-of-coolant accident environment. The applicant has committed to make any modifications to the recirculation sump which are identified as necessary by the model test program. We will require that the results of the model test be submitted for our review. The resolution of this matter will be reported in a supplement to this Safety Evaluation Report.

The applicant has committed to perform routine periodic testing of the emergency core cooling system components and all necessary support systems with the plant at power. Valves that are required to operate after a loss-of-coolant accident will be operated through a complete cycle, and pumps will be operated individually on their miniflow lines. Test lines will be provided to perform periodic tests on emergency core cooling sytem check valve operability.

The staff has requested additional information on the applicant's leak testing program to verify the integrity of pressure isolation check valves. The applicant's response has been evaluated and is reported in Section 3.9.6 of this Safety Evaluation Report.

6.3.6 Conclusions

We have reviewed the drawings, component descriptions, design criteria, performance analyses, and testing of the emergency core cooling system. Based on this review, we have determined that the emergency core cooling system will meet the acceptance criteria and conform to the Commission's requirements as set forth in General Design Criteria, regulatory guides, and staff technical positions provided that matters discussed in Section 6.3.5 of this Safety Evaluation Report are resolved.

6.4 Habitability Systems

6.4.1 Radiological Dose Protection

The applicant proposes to meet Criterion 19 of the General Design Criteria by use of shielding and by installing a filtration system to remove radioactive particulates and gaseous iodines after an accident. Normal ventilation of the control room is by means of dual air inlets located approximately 60 feet apart on top of the control building that supply 20,000 cubic feet per minute of filtered, cooled or heated air. After an accident, redundant isolation series dampers close automatically in response to the accident signal (safety injection) or the high gaseous radioactivity signal for inlet air. This places the control room ventilation system in a recirculation mode, with 20,000 cubic feet per minute being circulated through redundant particulate and carbon filtration components. The isolation dampers may also be manually positioned for purging the control room of smoke or noxious gases.

Based on information presented in the FSAR through Amendment 22, the control room will be operated in the emergency mode using a filtered pressurization and recirculation system. In evaluating the potential dose to control room operators, we have assumed an infiltration rate of outside contaminated air. We have evaluated the proposed design (through Amendment 22) to determine the potential radiological consequences to the control room operators following a design basis loss of coolant accident using the criteria presented in Section 6.4 of the Standard Review Plan. The assumptions and parameters used in our analysis can be found in Table 15-1 of this Safety Evaluation Report and our estimates of the potential control room operator doses can be found in Table 15-2 of this Safety Evaluation Report.

In addition, the applicant has incorporated a requirement into the plant Technical Specifications to periodically test the ability of the proposed system to pressurize the control room under the conditions given in Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release" Rev. 1, to assure the assumed leak-tightness of the control room used in our dose calculations.

Based upon our evaluation, the calculated radiological consequences are within the acceptance criteria contained in Section 6.4 of the Standard Review Plan and we conclude that the design of the control room emergency ventilation system, as described through Amendment 22 to the Final Safety Analysis Report, is acceptable for the purpose of preventing significant radiological exposure to operating personnel in the control room. We conclude that the design and proposed operation of the control room habitability systems as described in the Final Safety Analysis Report through Amendment 22 of the Final Safety Analysis Report meet the requirements of Criterion 19 of the General Design Criteria.

6.4.2 Toxic Gas Protection

The toxic gas hazards with respect to the control room were evaluated using the procedures described in Regulatory Guides 1.78 and 1.95. The applicant plans to store chlorine on-site in a chlorine shed adjacent to the water treatment building that is greater than 100 meters from the closest control room intake. No chlorine container will have an inventory of greater than 150 pounds. If a tank should leak, a detector in the chlorine shed will annunciate both locally and in the control room, allowing sufficient time for the operator to isolate the control room from the outside atmosphere. We conclude that the control room is adequately protected against any chlorine hazard from the on-site chlorine tanks.

With respect to off-site toxic gas hazards, there is no major highway traffic within five miles of the site. We have evaluated the possible effect of a failure of one rail car of methanol one mile from the control room intakes and have determined that this release does not pose a hazard to the control room operator. We have also determined that the hazard from a failure of one rail car of chlorine or ammonia is less than the hazard from a failure of one rail car of methanol. Accordingly, we conclude that there are no toxic gases off-site that are postulated to pose a hazard to the control room operator.

6.5 Fission Product Removal and Control System

6.5.1 Engineered Safety Features Atmosphere Cleanup Systems

The engineered safety features atmosphere cleanup systems for the Virgil C. Summer Nuclear Station, Unit 1 consist of process equipment and instrumentation to control the releases of radioactive materials in gaseous effluents (radioactive iodine and particulate matter) following a design basis accident. In the facility, there are three filtration systems designed for this purpose: the reactor building cooling system, the control room emergency filter system, and the fuel handling building charcoal exhaust system.

Reactor Building Cooling System

The reactor building cooling system is a redundant system consisting of two trains. Each train has a design capacity of 120,000 cubic feet per minute; the engineered safety features filter system components consist of demisters and high efficiency particulate air filters. Charcoal adsorbers are not included, since the reactor building spray system serves to remove radioiodine in the event of an accident.

The equipment and components are designed to Quality Group C and seismic Category I and are located in a seismic Category I structure.

We have determined that the engineered safety features filter system of the reactor building cooling system is designed in accordance with the guidelines of Regulatory Guide 1.52 and is capable of controlling the releases of radioactive materials in gaseous effluents in accordance with applicable regulations following a postulated design basis accident. We therefore find the design of the system acceptable.

Control Room Emergency Filter System

The function of the control room emergency filter system is to process potentially radioactive air in the control room after a design basis accident. This system permits operating personnel to remain in the control room following a design basis accident. The control room emergency filter system is a redundant recirculation system consisting of two trains which can be operated without supply air; each train has a recirculating design capacity of 20,000 cubic feet per minute of air. Each train contains, in series, a prefilter, high efficiency particulate air filter, carbon adsorber, high efficiency particulate air filter and fan.

The equipment and components are designed to Quality Group C and seismic Category I and are located in a seismic Category I structure.

We conclude that the in-place and laboratory testing efficiencies for iodine removal by the 2 inch charcoal adsorber should be at least equal to the efficiencies given in Regulatory Guide 1.52 for maintaining a suitable control room environment following a design basis accident. The applicant will therefore be required to amend the Technical Specifications for iodine removal accordingly for our acceptability of the system.

Fuel Handling Building Charcoal Exhaust System

The function of the fuel handling building charcoal exhaust system is to process potentially radioactive air exhausted from the fuel handling areas. The system consists of two redundant 100 percent capacity exhaust fans which draw air through three 50 percent capacity high efficiency particulate air/ charcoal plenums. Any two of the three plenums can be serviced by any one of the two exhaust fans.

The equipment and components are designed to Quality Group C and seismic Category I and are located in a seismic Category I structure.

With the exception of adsorber efficiencies, we have determined that the fuel handling building charcoal exhaust system complies with Regulatory Guide 1.52 requirements. The applicant will be required to amend the Technical Specifications on adsorber efficiencies for iodine removal as determined by in-place and laboratory testing in accordance with Regulatory Guide 1.52 for our acceptability of the system.

6.5.2 Containment Spray as a Fission Product Cleanup System

The reactor building spray system is designed with two independent and redundant trains, either or both of which would be acceptably efficient for fission product removal from the containment atmosphere in the event of a design basis release.

This system employs sodium hydroxide solution as a means of assuring that the boric acid refueling water supply is sufficiently alkaline to reduce elemental iodine in the containment atmosphere.

The spray system is designed to activate automatically, even in the event of a single active component failure. Upon exhaustion of the refueling water supply, the system will automatically switch over into the recirculation mode of operation, drawing water from the containment sump, and will continue to operate in the recircuation mode for a period of two hours. During this recirculation mode, the sump water will continue to draw down any sodium hydroxide remaining in the chemical addition tank to keep the sump pH greater than 8.5.

The system, as designed, mixes the refueling water supply with the sodium hydroxide solution by gravity flow rather than by eduction or positive displacement pumps. As a result, the composition of the spray solution is dependent upon the containment pressure and the relative liquid levels in the refueling water and sodium hydroxide storage tanks, which are, in turn, affected by potential failures of this and other systems also drawing upon the refueling water supply.

The applicant has analyzed several combinations of scenarios for emptying the refueling water storage tank, and has concluded that the spray system can be regulated within acceptable bounds of spray solution composition by choice of suitable orifices within the system piping. The applicant has also provided the results of preoperational tests of the drawdown capability of the spray system, as built, using water instead of a sodium hydroxide solution.

We have also estimated the injection pH under a number of extremes for spray system failures and continued operation. Based upon our review of the spray system design, and our accident analysis given in Section 15.4.1 of this Safety Evaluation Report, we conclude that the injection pH will provide an acceptable spray removal coefficient for scavenging elemental radioiodine released to the containment atmosphere following a postulated accident. We also conclude that the proposed post-operational testing and surveillance, and proposed limiting conditions for the spray system, in accordance with the Technical Specifications, provide adequate assurance that the iodine scrubbing function of the containment spray system will meet or exceed the effectiveness assumed in the accident analysis.

6.6 Inservice Inspection of Class 2 and 3 Components

Criterion 36, Criterion 39, Criterion 42, and Criterion 45, of the General Design Criteria, require, in part, that the emergency core cooling, containment heat removal, and containment atmosphere cleanup systems be designed to permit appropriate periodic inspection of important component parts to assure integrity and capability of the system.

The applicant has made a commitment to meet the preservice inspection requirements for ASME Code Class 2 components and the inservice inspection requirements for ASME Code Class 2 and 3 components as detailed in Section 50.55a(g) of 10 CFR Part 50. The preservice inspection program is currently under review by the NRC staff and is based upon the 1974 Edition, Summer 1975 Addenda, of Section XI of the ASME Code. Our evaluation of the preservice inspection program will be provided in a supplement to this Safety Evaluation Report. The inservice inspection program will be evaluated after the applicable ASME Code Edition and addenda have been determined and before the initial inservice inspection in accordance with Section 50.55a(g) of 10 CFR Part 50.

Compliance with the inservice inspection required by the ASME Code and 10 CFR Part 50 contitute an acceptable basis for satisfying the applicable requirements of Criteria 36, 39, 42, and 45 of the General Design Criteria.

6.7 Fracture Prevention of Containment Pressure Boundary

In our review we assessed the ferritic materials in the Virgil C. Summer Nuclear Station, Unit 1 containment system that constitute the containment pressure boundary to determine if the fracture toughness of the materials is in compliance with the requirements of Criterion 51 of the General Design Criteria.

Criterion 51 of the General Design Criteria requires that under operating, maintenance, testing, and postulated accident conditions, the ferritic materials of the containment pressure boundary behave in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

The reactor building for the facility is a reinforced concrete structure with a thin steel liner on the inside surface which serves as a leak-tight membrane. The ferritic materials of the containment pressure boundary which were considered in our assessment are those which have been used in the fabrication of the equipment hatch, personnel air lock, penetrations and fluid system components, including the valves required to isolate the system. These components are the parts of the containment system which are not backed by concrete and must sustain loads.

The containment pressure boundary is comprised of ASME Code Class 1, Class 2, and Class MC components. In late 1979, we reviewed the fracture toughness requirements of the ferritic materials of Class MC, Class 1 and Class 2 components which typically constitute the containment pressure boundary. Based on this review, we determined that the fracture toughness requirements contained in AMSE Code editions and addenda typical of those used in the design of the containment pressure boundary may not assure compliance with Criterion 51 of the General Design Criteria for all areas of the containment pressure boundary. We initiated a program to review fracture toughness requirements for containment pressure boundary materials for the purpose of defining those fracture toughness criteria that most appropriately address the requirements of Criterion 51. Prior to completion of this study, we have elected to apply in our licensing reviews of Class 1, Class 2 and Class MC ferritic containment pressure boundary materials the criteria for Class 2 components identified in the Summer 1977 Addenda of Section III of the ASME Code. We chose the criteria of the Summer 1977 Addenda of Section III of the ASME Code to provide a uniform review, consisted with the safety function of the containment pressure boundary materials. Since the fracture toughness criteria that have been applied in construction typically differ in Code classification and Code editions and addenda.

We considered in our review components of the containment system which are load bearing and provide a pressure boundary in the performance of the containment function under operating, maintenance, testing and postulated accident conditions as addressed in Criterion 51. These components are the equipment hatch, personnel airlocks, penetrations and elements of the main steam and main feedwater systems.

In some cases, materials were not fracture toughness tested; typically these materials are those in the main steam and main feedwater systems. Generally, those materials which were not fracture toughness tested were not tested because the ASME Code edition and addenda in effect at the time the components were ordered did not require that they be tested. Our assessment was based on the metallurgical characterization of these materials and fracture toughness data presented in NUREG-0577, "Potential for Low Pressure Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," USNRC, October 1979 and ASME Code Section III, Summer 1977 Addenda, Subsection NC.

The metallurgical characterization of these materials, with respect to their fracture toughness was developed from a review of how these materials were fabricated and the thermal history they experienced during fabrication. The metallurgical characterization of these materials, when correlated with the data presented in NUREG-0577 and the Summer 1977 Addenda of Section III of the ASME Code provided the technical basis for our finding for the materials which were not fracture toughness tested.

Based on our review of the available fracture toughness data and material fabrication histories, and the use of correlations between metallurgical characteristics and material fracture toughness, we conclude that the ferritic components in the containment pressure boundary meet the fracture toughness requirements that are specified for Class 2 components by the 1977 Addenda of Section III of the ASME Code. Compliance with the ASME Code requirements assures that the reactor containment pressure boundary will behave in a non-brittle manner, that the probability of rapidly propagating fracture will be minimized and that the requirements of Criterion 51 of the General Design Criteria are satisfied.

7 INSTRUMENTATION AND CONTROLS

7.1 General

The instrumentation and control systems for the Virgil C. Summer Nuclear Station, Unit 1 were reviewed using "The Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, NUREG 75-094 and the Standard Review Plan, NUREG 75-087.

7.1.1 Acceptance Criteria

Acceptance criteria and requirements used to evaluate the instrumentation and control systems for this facility are listed in Table 7-1 of this Safety Evaluation Report. These criteria and requirements include the General Design Criteria, various Institute of Electrical and Electronic Engineers (IEEE) Standards, Regulatory Guides for power reactors and staff positions listed in Table 7-A of the Standard Review Plan.

7.1.2 Identification of Safety Related Systems

Safety related systems included in our review of the facility design included:

1. The reactor trip system discussed in Section 7.2 of this Safety Evaluation Report.

2. The engineered safety features systems discussed in Section 7.3 of this Safety Evaluation Report.

3. The safe shutdown systems discussed in Section 7.4 of this Safety Evaluation Report.

4. The safety related display instrumentation discussed in Section 7.5 of this Safety Evaluation Report.

5. All other systems required for safety discussed in Section 7.6 of this Safety Evaluation Report.

7.1.3 Separation of Electric Equipment and Systems

We requested the applicant to perform a detailed field audit to confirm conformance to the separation criteria for the installed electrical equipment and systems. This audit was to include a sample not less than five percent of the installed equipment and cables, and emphasize but not necessarily be limited to:

1. Separation of redundant Class 1E equipment (e.g., motors, valve operators, instruments).

2. Separation of redundant Class 1E cables.

3 Separation of associated circuits.

TABLE 7-1

ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS

Criteria	Title	Reactor Trip System		Engineered Safety Features				Systems for Safe Shutdown		Post Accident Monitoring System	Other Systems Required For Safety				Reactor Control, Rod Control Pressurizer Pressure & Water Level Control, Steam Generator Water Level Control, Steam Dump Control and Incore Instrumentation
		Turbine Trip	All Other Trips	Emergency Core Cooling System	Emergency Boration System	Engineered Safety Features Actuation System	Containment Systems	Residual Heat Removal System	Emergency Feedwater System		Emergency Switchover	Accumulator Isolation Valves	Resi Isolation Valves	Instrumentation & Control Power Supply	
10 CFR Part 50	Contents of Application Technical Information	X	X	X	X	X	X	X	X	X	X	X	X	X	X
	Technical Specifications	X	X	X	X	X	X	X	X	X	X	X	X	X	X
	Codes & Standards	X	X	X	X	X	X	X	X	X	X	X	X	X	X
General Design Criteria App. A to 10 CFR Part 50	(See Standard Review Plan Table 7-1 for Specific General Design Criteria and Table	X	X	X	X	X	X	X	X	X	X	X	X	X	X

TABLE 7-1

ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS

Criteria	Title	Reactor Trip System		Engineered Safety Features				Systems for Safe Shutdown		Post Accident Monitoring System	Other Systems Required For Safety				Reactor Control, Rod Control Pressurizer Pressure & Water Level Control, Steam Generator Water Level Control, Steam Dump Control and Incore Instrumentation
		Turbine Trip	All Other Trips	Emergency Core Cooling System	Emergency Boration System	Engineered Safety Features Actuation System	Containment Systems	Residual Heat Removal System	Emergency Feedwater System		Emergency Switchover	Accumulator Isolation Valves	Residual Isolation Valves	Instrumentation & Control Power Supply	
IEEE STANDARDS															
IEEE STD 279-1971 (ANSI N42.7 1972)		X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE STD 308-1971				X	X	X	X	X	X	X	X	X	X	X	
IEEE STD 317-1972		X	X	X	X	X	X	X		X	X	X	X	X	X
IEEE STD 323-1974		X	X	X	X	X	X	X	X	X	X	X	X	X	
IEEE STD 336-1971		X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE STD 338-1971		X	X	X	X	X	X	X	X	X	X	X	X	X	
IEEE STD 344-1975			X	X	X	X	X	X	X	X	X	X	X	X	

TABLE 7-1

ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS

Criteria	Title	Reactor Trip System		Engineered Safety Features				Systems for Safe Shutdown		Post Accident Monitoring System	Other Systems Required For Safety				Reactor Control, Rod Control Pressurizer Pressure & Water Level Control, Steam Generator Water Level Control, Steam Dump Control and Incore Instrumentation
		Turbine Trip	All Other Trips	Emergency Core Cooling System	Emergency Boration System	Engineered Safety Features Actuation System	Containment Systems	Residual Heat Removal System	Emergency Feedwater System		Emergency Switchover	Accumulator Isolation Valves	Resi- Isolation Valves	Instrumenta- & Control Power Supply	
IEEE STD 379-1972		X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE STD 384-1974		X	X	X	X	X	X	X	X	X	X	X	X	X	X
REGULATORY GUIDES															
RG 1.6				X	X	X	X	X	X	X	X	X	X	X	
RG 1.11			X	X	X	X	X	X		X	X	X	X		
RG 1.22		X	X	X	X	X	X	X	X	X	X	X	X	X	
RG 1.29			X	X	X	X	X	X	X	X	X	X	X	X	
RG 1.30		X	X	X	X	X	X	X	X	X	X	X	X	X	X
RG 1.32				X	X	X	X	X	X	X	X	X	X	X	
RG 1.47		X	X	X	X	X	X	X	X	X	X	X	X	X	X
RG 1.53		X	X	X	X	X	X	X	X	X	X	X	X	X	
RG 1.62			X	X	X	X	X	X	X		X			X	
RG 1.63			X	X	X	X	X	X		X	X	X	X	X	X

TABLE 7-1

ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS

Criteria	Title	Reactor Trip System		Engineered Safety Features				Systems for Safe Shutdown		Post Accident Monitoring System	Other Systems Required For Safety				Reactor Control, Rod Control Pressurizer Pressure & Water Level Control, Steam Generator Water Level Control, Steam Dump Control and Incore Instrumentation
		Turbine Trip	All Other Trips	Emergency Core Cooling System	Emergency Boration System	Engineered Safety Features Actuation System	Containment Systems	Residual Heat Removal System	Emergency Feedwater System		Emergency Switchover	Accumulator Isolation Valves	Resi Isolation Valves	Instrumenta- & Control Power Supply	
RG 1.68		X	X							X				X	
RG 1.70		X	X	X	X	X	X	X	X	X	X	X	X	X	X
RG 1.75		X	X	X	X	X	X	X	X	X	X	X	X	X	X
RG 1.89			X	X	X	X	X	X	X	X	X	X	X	X	
BRANCH TECHNICAL POSITIONS (BTP)															
BTP EICSB 1		X	X	X	X	X	X	X	X					X	
BTP EICSB 3								X					X		
BTP EICSB 4				X						X					
BTP EICSB 5		X	X												
BTP EICSB 9		X	X	X	X	X	X	X	X	X	X	X	X		
BTP EICSB 10			X	X	X	X	X	X	X	X	X	X	X		
BTP EICSB 12			X				X								

TABLE 7-1

ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS

Criteria	Title	Reactor Trip System		Engineered Safety Features				Systems for Safe Shutdown		Post Accident Monitoring System	Other Systems Required For Safety				Reactor Control, Rod Control Pressurizer Pressure & Water Level Control, Steam Generator Water Level Control, Steam Dump Control and Incore Instrumentation
		Turbine Trip	All Other Trips	Emergency Cor Cooling System	Emergency Boration System	Engineered Safety Features Actuation System	Containment Systems	Residual Heat Removal System	Emergency Feedwater System		Emergency Switchover	Accumulator Isolation Valves	Resi Isolation Valves	Instrumenta- & Control Power Supply	
BTP EICSB 13						X			X						
BTP EICSB 14			X												X
BTP EICSB 15			X												
BTP EICSB 18				X	X	X	X	X	X		X	X	X		
BTP EICSB 19						X	X				X	X			
BTP EICSB 20				X		X				X	X				
BTP EICSB 21		X	X	X	X	X	X	X	X	X	X	X	X	X	
BTP EICSB 22		X	X	X	X	X	X	X	X	X	X	X	X	X	
BTP EICSB 23										X					
BTP EISCS 24		X	X	X	X	X	X		X		X	X	X		
BTP EICSB 25				X						X	X				
BTP EICSB 26		X													
BTP EICSB 27				X	X		X	X	X		X	X	X		

4. identification (marking) of redundant Class 1E equipment and cables.

5. Identification (marking) of associated circuits to a level indicative of the Class 1E system with which they are associated.

6. Separation of redundant wiring, indicators and controls at panels and control boards.

In response to our request, the applicant has performed a detailed fie d audit. The results of audit were submitted to NRC by a letter dated January 6, 1981. The audit included a five percent sample of plant-wide installed Class 1E and associated field routed circuits and control boards and panels internal wiring. The selection method of equipment or circuits was by means of random sampling by computer algorithm identified in circuit summary printouts. The selected samples are fairly representative. The criteria identified in Sections 7.1.2.2, 7.1.2.3, 8.3.1.4, 8.3.1.5, Appendix 3A, and Table 8.3-4 of the Final Safety Analysis Report and Regulatory Guide 1.75 were used as audit criteria.

For each audited item, an audit report was completed to either confirm conformance with the criteria or documenting the discrepancy. The applicant has justified each discrepancy and committed to take corrective actions. Based on our review of the information presented by the applicant, we conclude that the applicant's field audit has met the guidelines of the audit requirements and is acceptable.

7.1.4 IE Bulletins 79-27 and 80-06

Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation (IE Bulletin 79-27)

On November 30, 1979, the Office of Inspection and Enforcement issued IE Bulletin 79-27 "Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation" to all power reactor facilities with an operating license and to those nearing full power licensing. This bulletin outlined actions to be taken to address control system malfunctions and significant loss of information to the control room operator as a potential consequence of the loss of Class 1E and non-Class 1E buses supplying power to these plant systems. Further, IE Information Notice 80-10, issued on March 7, 1980, provided information relating to the Crystal River, Unit 3 event of February 26, 1980 in which a significant loss of information to the operator resulted from a loss of power to a portion of the plant instrumentation system.

The applicant conducted a review using the guidelines of IE Bulletin 79-27 and concluded that no deficiencies existed based upon the capability to achieve shutdown conditions using plant procedures. We find this acceptable.

Potential Design Deficiencies in Bypass, Override, and Reset Circuits of Engineered Safety Features (IE Bulletin 80-06)

On March 13, 1980, the Office of Inspection and Enforcement issued Bulletin 80-06, "Engineered Safety Feature (ESF) Reset Control", to address the concern that the use of reset pushbuttons alone could permit certain engineered safety feature components to revert to their normal state following safety system actuation. On May 14, 1980, we requested South Carolina Electric & Gas Company to provide additional information related to this subject. Both IE Bulletin 80-06 and our May 14, 1980 letter require that, upon reset of an engineered safety feature actuation signal; all associated safety-related equipment should remain in its emergency mode. A design review at the schematic diagram level was requested and IE Bulletin 80-06 requires confirmatory testing of all engineered safety feature reset actions.

In response to these concerns for operating plants, the applicant provided responses stating that it reviewed the design and determined that it was in conformance with the applicable guidelines except for four cases as noted below.

The first case is the battery room supply dampers. The trains A and B battery rooms are normally supplied air from both the trains A and B ventilation systems. On a safety actuation signal, dampers are closed such that each room is supplied air only from its associated train ventilation system. On a reset of the safety actuation signal, the dampers would reopen to realign the ventilation system to the normal operating mode. We find that this action does not result in a safety concern and, therefore, the present design is acceptable.

The second is the chemical volume and control system letdown orifice isolation valves. Upon a reset of a safety actuation signal, these orifice isolation valves will remain closed as designed; however, the Train A solenoid on each valve does not have a valve limit switch contact "seal-in" interlock to assure that reopening does not occur. The applicant committed to add a "seal-in" interlock to assure that reopening does not occur. We find this to be acceptable.

The third case is the feedwater bypass control valves. Upon a reset of a safety actuation signal, these valves will go on modulation; however, reset of the signal will not reopen the feedwater isolation valves downstream from these bypass control valves. Therefore the isolation function will be maintained. We find this to be acceptable.

The fourth case is the component cooling water booster pumps. Upon reset of a safety actuation signal, these non-safety related pumps may restart against a closed discharge line and consequently affect pump reliability. The applicant committed to modify the design to include a discharge valve limit switch interlock to preclude a restart of a pump against a closed discharge valve. We find this to be acceptable.

During the site visit we found that all engineered safety feature circuits had not been reviewed due to a misinterpretation of the scope of the review. We requested that the applicant review all engineered safety features control circuits with respect to deficiencies in bypass, override and reset of engineered safety feature actions. We further requested that a test be conducted to confirm the conclusion of this review as was required for operating plants by IE Bulletin 80-06.

In response to our request, the applicant committed, by a letter dated January 9, 1981, that it would review all engineered safety features control circuits with respect to deficiencies in bypass, override and reset of engineered safety features action. Any deviations from these criteria will be justified or corrected. The applicant also committed to perform testing to conform the conclusions of the review. We find this acceptable and will address this matter further review in a supplement to this Safety Evaluation Report.

7.2 Reactor Trip System

The reactor trip system is designed to provide timely protection against the onset and consequences of transients and other conditions that threaten the integrity of the fuel barrier and the nuclear system process barrier.

The reactor trip system includes sensors and analog circuitry arranged in two to four channels for monitoring various plant parameters. Signals from these analog channels are sent to redundant logic trains, composed of digital circuitry, which causes the reactor trip breakers to open, resulting in a reactor trip.

Each redundant train is capable of opening a separate and independent reactor trip breaker. Opening either trip breaker will trip the reactor and all control rods will fall into the core by gravity.

The reactor trip system is similar to the system used in the James M. Farley nuclear plant design. No significant differences have been identified by the applicant. We approved the James M. Farley reactor trip system in our review of the operating license application for that facility on Docket Nos. 50-348 and 50-364. See NUREG 75/034 dated May 2, 1975.

7.2.1 Reactor Trip System Component Systems

The component systems which make up the reactor trip system are:

1. The process instrumentation system.

2. The nuclear instrumentation system.

3. The solid state logic protection system.

These component systems have been reviewed and approved by the NRC staff in previous applications, e.g., for the James M. Farley facility. These reviews emphasized that (1) the equipment be seismically and environmentally qualified and (2) the implementation of the system requirements that assure the independence and redundancy of the protection system.

The seismic and environmental qualification of the reactor protection system equipment is discussed in Sections 3.10 and 3.11 of this Safety Evaluation Report. Preservation of independence and redundancy is achieved in the protection system racks by using isolation amplifiers to separate individual channel inputs from both the control system outputs and the protection system outputs.

Westinghouse performed tests to demonstrate that credible faults or electrical interference in cables associated with the reactor trip system would not degrade the system performance requirements for the nuclear instrumentation system and the solid state protection system. These tests were made in support of the Diablo Canyon license application. Similar tests were made by Westinghouse on the 7300 Series process control system. The tests which demonstrate acceptance of the 7300 Series process control system were presented in Westinghouse topical report WCAP 8892A. We have reviewed and approved the procedures for and results of these tests. All cables external to the reactor trip system racks are installed to satisfy the requirements specified in the Westinghouse test reports.

For the Virgil C. Summer Nuclear Station, Unit 1 the cables leaving the protection system racks will not be exposed to potentials higher than those used in performance of the Westinghouse tests. We conclude that electrical interference in cables associated with the reactor trip system at the facility will not degrade the reactor trip system performance and is acceptable.

7.2.2 Reactor Trip System Trips

The following trips are provided as inputs to the reactor trip system:

1. Nuclear Overpower Trips

   - Power range, high neutron flux trip
   - Intermediate range, high neutron flux trip
   - Source range, high neutron flux trip
   - Power range, high positive neutron flux rate trip
   - Power range, high negative neutron flux rate trip

2. Core Thermal Overpower Trips

   - Over temperature delta temperature trip
   - Overpower delta temperature trip

3. Reactor Coolant System Pressurizer Pressure and Water Level Trips

   - Low pressure trip
   - High pressure trip
   - High water level trip

4. Reactor Coolant System Low Flow Trips

   - Low flow trip
   - Reactor coolant pump undervoltage trip
   - Reactor coolant pump underfrequency trip

5. Steam Generator Trips

   - Low feedwater flow trip
   - Low-low water level trip

6. Turbine Trip (Anticipatory)

7. Safety Injection System Actuation Trip

8. Manual Trip

7.2.3 Trip Setpoints and Margins

The setpoints for the various functions in the reactor trip system are determined based on the accident analyses requirements. As such, during any anticipated operation and occurrence, the reactor trip system limits the following parameters to:

1. Minimum departure from nucleate boiling ratio-1.30

2. Maximum system pressure-2750 pounds per square inch (absolute)

3. Fuel rod maximum linear power-18.0 kilowatts per foot

The reactor trip system bistable setpoints are established considering the following:

1. Safety limit setpoint - value assumed in the accident analysis

2. Limiting setpoint - Technical Specification value

3. Normal setpoint - value set into the equipment and obtained by subtracting allowance for instrument drift, calibration uncertainty, transmitter error and base starting margin from the limiting setpoint

The detailed trip setpoint review is being performed as part of our review the plant Technical Specifications and will be completed prior to issuance of the operating license. The applicant was requested to provide an evaluation and/or an analysis of the effect of post-accident environmental conditions on the setpoint for the reactor trip system instrumentation (Technical Specification Table 2.2-1), and the engineered safety feature actuation system instrumentation (Technical Specification Table 3.3-4). The margins from the normal setpoint to the limiting setpoint should include the instrument drift, calibration uncertainty, transmitter error and base starting margin.

A generic letter on the staff concerns of level measurement errors due to environmental temperature effects on level instrument reference legs was sent to the applicant. We will report our findings in a supplement to this Safety Evaluation Report following the review the applicant's response.

7.2.4 Anticipated Transients Without Scram

The status of our evaluation of anticipated transients without scram is presented in Section 15 of this Safety Evaluation Report.

7.2.5 Conclusion

Based on our review of the description and analyses presented in the Final Safety Analysis Report, except the trip setpoints and margins as stated in Section 7.2.3, we conclude that the reactor trip system satisfies the IEEE Standard 279-1971 and the acceptance criteria referenced in Secti 7.1.1 of this Safety Evaluation Report and is acceptable.

7.3 Engineered Safety Features Systems

Section 7.3 of the Final Safety Analysis Report describes the portion of the protection system used to initiate and control operation of the engineered safety features systems and their auxiliary supporting systems. The descriptive information, functional control diagrams, piping and instrument diagrams, electrical schematics and physical arrangement drawings as presented in the Final Safety Analysis Report were reviewed. The objective of our review was to determine that the engineered safety feature systems satisfy applicable design criteria referenced in Section 7.1 of this Safety Evaluation Report and will perform as intended during all operating conditions and accident conditions for which its function is required.

7.3.1 Engineered Safety Features Actuation System

The engineered safety features actuation system receives inputs from selected facility parameters and initiates operation of the necessary safety-related equipment to control the facility within acceptable limits.

The applicant stated that this system is similar to the one used in James M. Farley facility and no significant differences were identified.

The engineered safety features actuation system includes:

1. Inputs from the process instrumentation and control system.

2. Inputs from the solid state protection system.

3. The engineered safety features test cabinets.

4. The engineered safety features loading sequence control panels.

5. Manual actuation circuits.

Major safety functions initiated by the engineered safety features actuation system are:

1. Trip reactor when a trip has not been generated by the reactor trip system.

2. Open the cold leg injection isolation valves so that borated water may be injected into the reactor coolant system by the charging pumps.

3. Operate the charging pumps and associated valves required to inject emergency makeup water into the reactor coolant system cold legs, following a loss of coolant accident.

4. Initiate Phase A containment isolation to prevent fission product release.

5. Isolate the main steamline.

6. Isolate the main feedwater line.

7. Start the emergency diesel generators.

8. Isolate the control room air intake ducts following a loss of coolant accident.

9. Initiate reactor building spray.

10. Start the engineered safety features emergency power loading sequencers.

The engineered safety features actuation systems include the instrumentation and controls used to detect a condition requiring operation of an engineered safety features system, to initiate action of the engineered safety features system, and to control its operation. The scope of our review of the engineered safety features actuation system for the facility included schematic diagrams and descriptive information, analysis of the manner in which the design of the engineered safety features actuation system and the auxiliary supporting system conform to the applicable General Design Criteria and the Regulatory Guides, on-line testing capabilities and the overall testing programs. We conclude that the engineered safety features actuation system satisfies the IEEE Standard 279-1971 and the acceptance criteria referenced in Section 7.1.1 of this Safety Evaluation Report and is acceptable.

7.3.2 Containment Systems

Safety-related instrumentation required to monitor containment parameters during normal and accident conditions are discussed in Section 7.5 of this Safety Evaluation Report. Our evaluation of the instrumentation and controls to initiate operation of the safety-related containment systems follows:

Reactor Building Heat Removal Systems

The reactor building heat removal system is composed of the reactor building spray system and the reactor building cooling system. These systems are designed to remove reactor building heat following an accident.

The reactor building spray system also removes iodine that may be released from the core following an accident. The reactor building cooling system is used to remove reactor building heat during normal operation.

Reactor Building Spray System

Ther reactor building spray system is composed of two independent and redundant subsystems which are normally in standby condition. Following a high energy line break inside containment, this system is started automatically by two-out-of-four logic on high-high containment pressure which is part of the engineered safety features actuation system.

Spray actuation also can be manually initiated from the control room. During the construction permit review we were concerned that the provisions for manual actuation of containment spray would not meet the single failure criteria. The modified design provides redundant manual actuation controls. Actuation is effected only if both controls are operated simultaneously. This design precludes the inadvertent initiation of the spray system. We conclude that the design is acceptable.

Reactor Building Cooling System

The reactor building cooling system is composed of four units and each one can be operated manually at high or low fan speed. Two units are energized from power train A and two from power train B. Control room selector switches are provided to determine which unit in each train will respond to an engineered safety feature loading sequence signal. The units are tripped on the receipt of a safety injection signal or a loss of offsite power signal. One unit in each train is then automatically started, at low speed, by the engineered safety feature actuation system and the engineered safety feature loading sequencer for its associated emergency diesel generator.

Containment Isolation System

There are two phases of containment isolation provided in the Virgil C. Summer Nuclear Station, Unit 1 design. Phase A containment isolation is initiated by safety injection or the following parameters:

1. Containment high pressure

2. High differential pressure between steam lines

3. Pressurizer low pressure

4. Low steam line pressure

5. Manual

Phase B containment isolation is initiated by two-out-of-four logic on a high-high containment pressure signal. This signal also initiates containment spray.

The definition of essential and non-essential systems has been given; the essential systems are those which are required immediately to mitigate the consequences of an accident. All other systems are considered non-essential. All non-essential systems are automatically isolated by the phase A containment isolation signals, except the power operated containment isolation valves associated with penetrations for component cooling water to the reactor coolant pump are automatically closed by the containment isolation phase B signal.

The power-operated valves associated with penetrations for containment ventilation are automatically closed by a containment phase A isolation signal, or high radiation in the reactor building purge exhaust vent (A train valves), or containment atmosphere high radiation (B train valves) signal. Each isolation valve is controlled by two solenoid valves in series. Both solenoid valves have to be energized to open the isolation valve. The containment isolation signal trips one solenoid, and the radiation signal trips the other solenoid. Resetting the isolation signal will not result in the automatic reopening of containment isolation valves.

Combustible Gas Control

Redundant electric hydrogen recombiners and an alternate reactor building purge system are provided to remove hydrogen following an accident. Each recombiner is manually operated from a separate power panel and control panel which is supplied from a separate engineered safety feature bus. The alternate reactor building purge system serves as backup to the hydrogen recombiners. This system is used to remove hydrogen only in the unlikely event of failure of both hydrogen recombiners. A hydrogen concentration analyzer operates continuously to confirm proper operation of the systems. A backup hydrogen concentration analyzer is provided which can be placed into operation if needed.

We have reviewed the instrumentation and control systems to initiate operation for the reactor building heat removal system, the containment isolation system, and the combustible gas control system and conclude that they satisfy IEEE Standard 279-1971 and the acceptance criteria referenced in Section 7.1.1 of this Safety Evaluation Report and are acceptable.

7.3.3 Emergency Core Cooling System

Accumulator Isolation Valves

Each accumulator is provided with motor-operated isolation valves. During shutdown, these valves are closed to prevent loss of accumulator water inventory. During startup and power operation, these valves are required to be fully open so that accumulators will be available in the event of a loss of coolant accident. Power is removed from the valves during both of these modes of operation to prevent inadvertent operation of the isolation valves. Redundant valve position sensors are provided, one from a cam-operated switch within the motor operator, the other from a stem-mounted limit switch. One set of valve position indication is located on main control panel near the control switch. The other set of valve position indication is located on the engineered safety feature monitor panel. An audible alarm will warn the operator when any accumulator isolation valve is closed. The engineered safety feature monitor panel is powered from a Class 1E source which is independent from the source used to power the valve motor operator and the valve control circuit.

This design satisfies our requirements specified in Branch Technical Positions ICSB 4 and 18 and is therefore acceptable.

Boron Injection System

Major components in the boron injection system are (1) the boron injection tank, (2) the boron injection surge tank, and (3) the boron injection recirculation pumps. During operation, this system is operated in the recirculation mode. In the event of an accident, the recirculation pumps are tripped and the charging pumps are automatically started and deliver flow, through the boron injection tank, into the reactor coolant system. A safety injection signal, originating in the engineered safety features actuation system, starts the charging pumps, opens the charging pump discharge valves and trips the boron injection recirculation pumps.

Redundant boron injection tank heaters, heat tracing for the boron injection system piping and heater controls are provided to maintain the boron solution at a temperature above 135 degrees Fahrenheit. This prevents boron from precipitating out of solution. Based on our review, we conclude that the boron injection system satisfies the criteria referenced in Section 7.1.1 of this Safety Evaluation Report and is acceptable.

Residual Heat Removal System

Residual Heat Removal System Overpressurization

The Virgil C. Summer Nuclear Station, Unit 1 design includes two motor operated valves connected in series in each residual heat removal system suction line.

These valves are provided to prevent overpressurization of the residual heat removal system in the event that a single failure caused one of the valves to open.

Branch Technical Position ICSB 3 requires that independent and diverse interlocks be provided for these valves to preclude their being opened under conditions which could overpressurize the residual heat removal system. The initial design satisfied the independent interlock requirement but diversity was not provided. The applicant modified the design to include two different pressure transmitters from different manufacturers. We find that this satisfies Branch Technical Position ICSB 3 and is therefore acceptable.

Residual Heat Removal System Availability

Criterion 34 of the General Design Criteria requires that the system provided to remove residual heat be designed to perform its safety function with only onsite power or with only offsite power available. Criterion 34 also requires that the residual heat removal system safety function be accomplished assuming a single failure. However, we found that the failure of a power supply could eliminate the capability to open one of the two suction valves in both residual heat removal system trains and result in loss of the entire residual heat removal system.

In response to our concern, the applicant stated that the facility design allows the facility to be maintained in hot shutdown status for a long period of time. While failure of either power train could prevent initiation of residual heat removal system cooling in the normal manner from the control room, the affected valve could be opened with its handwheel. In case the containment became inaccessible, an alternate power source could be wired to provide the power to the valve. These manual actions could be performed within 36 hours as stated in the staff's interim position BTP RSB 5-1. We conclude that this portion of the design satisfies BTP RSB 5-1 and therefore is acceptable.

7.3.4 Single Failure of Engineered Safety Features System

The applicant has identified eight motor-operated valves in the Virgil C. Summer Nuclear Station, Unit 1 emergency core cooling system which must not change position during certain phases of a postulated loss of coolant accident. To assure that these valves do not change position when the facility is operating, power is removed from these valves.

Three valves are used to isolate the accumulators during shutdown. See also Section 7.3.3 of this Safety Evaluation Report. During normal reactor operation, these valves are required, by the Technical Specifications, to be fully open with power removed from the valve operators. Power is removed by removing the circuit breakers.

Removal and restoration of electrical power for high head to hot leg injection valves (8884, 8885), low head to cold leg cross tie valves (8888A, 8888B), and low head to hot leg cross tie valve (8889) will be accomplished through two power contactors. Controls for these contactors are located on the main control board in the main control room. With this arrangement, power will be removed from the valve operators during reactor operation, but power can be restored quickly by the operator if and when required. These valves are under Technical Specification surveillance requirements.

We have reviewed this design and find that it satisfies the requirements of Branch Technical Position ICSB 18 and is therefore acceptable.

7.3.5 Failure Modes and Effects Analysis

The applicant has referred to the Westinghouse Topical Report WCAP-8584, "Failure Mode and Effects Analysis (FMEA) of the Engineered Safety Features Actuation System," as the supporting document on failure mode and effects analysis for the engineered safety features equipment within the Westinghouse scope of supply. We have reviewed WCAP-8584, and find the methodology and the general conclusions to be acceptable. However, in Appendices B and C of WCAP-8584, Westinghouse specifies the interface requirements for electrical circuit and instrument impulse lines separation involving other plant systems included in the balance of plant. The conformance to these requirements was not addressed in the Final Safety Analysis Report. We therefore requested that the applicant identify the difference between the Virgil C. Summer Nuclear Station, Unit 1 design and the typical design described in WCAP-8584. In response to our request, the applicant stated in a letter dated December 16, 1980, that there are no deviations with respect to the Virgil C. Summer Nuclear Station, Unit 1 design and the requirements of WCAP-8584. Appendix B of WCAP-8584 deals with independence of redundant safety related systems. In the design, the physical separation criteria for redundant safety-related system sensores, sensing lines, wireways, cables, and components on control boards/racks meet the recommendations contained in Regulatory Guide 1.75 "Physical Independence of Electric Systems". We find this to be acceptable.

Appendix C of WCAP-8584 specifies the interface requirements for isolation, power supply, turbine-driven emergency feedwater pumps startup on loss of voltage to emergency electric power buses, and environmental considerations. In the design, the control signals and other non-protective functions are derived from individual protective channels through isolation amplifiers. The isolation amplifiers are designed such that a short circuit, open circuit, or the application of credible fault voltages from within the cabinets on the isolated output portion of the circuit will not affect the input (protective) side of the circuit. The signal obtained through the isolation amplifiers are not returned to the protective system. This design meets the requirements of Criterion 24 of the General Design Criteria and paragraph 4.7 of IEEE Standard 279-1971.

The instrumentation and control power supply for the facility has been addressed in Section 8.3.1 of this Safety Evaluation Report and found to be acceptable. The turbine-driven emergency feedwater pump startup has been addressed in Section 7.4.1 of this Safety Evaluation Report and found to be acceptable. The environmental considerations will be addressed in Section 3.11 of a supplement to this Safety Evaluation Report.

Based on our review of the information presented by the applicant, we conclude that the interface requirement for the failure mode and effects analysis of the engineered safety features actuation system have been notified and therefore this aspect of the design is acceptable.

7.3.6 Main Steamline Isolation

The main steamline isolation valves for the facility are designed to close five second after receipt of a closure signal. Automatic closure signals originate from (1) high containment pressure, (2) low steamline pressure, and (3) high steamline flow coincident with low average primary coolant temperature. Capability for manual closure is also provided. These closure signals are redundant such that initiation from either train A or train B will cause isolation of the main steamlines.

Issue 1 listed in NUREG-0138, "Staff Discussion of Fifteen Technical Issues," identifies a concern that a main steamline break accident in coincidence with the failure of a main steamline isolation valve in another line could permit the uncontrolled blowdown of more than one steam generator. Our position, as stated in NUREG-0138, is that for this purpose, nonsafety grade components can be used to back up the single failure of the safety grade components used in the secondary system if these components receive a signal to close.

We requested the applicant to perform a study to verify that all valves used to back up the main steamline isolation valve receive isolation signals following a steamline break accident. The study should also verify that the total steam flow in all uncontrolled paths, downstream of each main steam line isolation valve, will not cause an unsafe condition.

In Amendment 19 of the Final Safety Analysis Report, the applicant provided the detailed steam generator blowdown path analysis, which identifies the various flow paths that branch off between the main steam line isolation valves and the turbine stop valves. Most of those flow paths are normally closed. The main steam lines to the steam-driven feedwater pumps will be isolated by a feedwater siolation signal. The study indicates that total steam flow in all uncontrolled paths downstream of each main steam line isolation valve will not cause an unsafe condition. We conclude that this is acceptable.

7.3.7 Feedwater Isolation

In the event of a feedwater line break either inside or outside the reactor building, the engineered safety features actuation system signal will initiate the feedwater containment isolation valves. These valves close in five seconds upon receipt of an engineered safety features actuation system signal. Any of the following engineered safety features actuation system channel A feedwater isolation signals initiates valve closure:

1. Safety injection signal
2. Steam generator high level
3. Low T_{avg} coincident with reactor trip
4. Low steam generator level
5. Coincidence of low steam generator pressure, low feedwater temperature and low feedwater flow.

Engineered safety features actuation system channel B feedwater isolation signals trips the feedwater pumps.

The feedwater isolation signal also closes the feedwater isolation bypass valves, the feedwater control valves and the control bypass valves as a backup function. Feedwater isolation can be accomplished by manual action from the control room.

We conclude that the electrical design for feedwater isolation satisfies the applicable criteria listed in Section 7.1.1 of this Safety Evaluation Report and is acceptable.

7.4 Systems Required for Safe Shutdown

The staff has issued Branch Technical Position RSB 5-1 which requires that the capability be provided from normal operating conditions to achieve cold shutdown using only safety-grade systems. This issue is also addressed in Sections 5.4.3 and 7.3.3 of this Safety Evaluation Report.

The following systems have been identified in the Final Safety Analysis Report as having the capability for achieving and maintaining a safe shutdown condition:

1. Boration system

2. Residual heat removal system

3. Emergency feedwater system

The applicant states that the safe shutdown systems are similar to the Joseph M. Farley, Unit 1 systems. We have identified no significant differences.

As stated in Section 7.3 of this Safety Evaluation Report, we conclude that the boration and residual heat removal systems are acceptable. The emergency feedwater system is discussed in Section 7.4.1 which follows.

7.4.1 Emergency Feedwater System

The emergency feedwater system is a part of the engineered safety features. This system consists of two motor-driven pumps and one turbine-driven pump. The motor-driven pumps start automatically on low water level in any steam generator, a trip of all three steam generator main feedwater pumps, a safety injection signal, or undervoltage on engineered safety features buses. Operation of the turbine-driven emergency feedwater pump is automatically initiated by the opening of the steam supply valves to the turbine drive on either low water level in any steam generator or undervoltage on both engineered safety features buses. The automatic initiation signals and circuits for the emergency feedwater system comply with the single failure criterion of IEEE Standard 279. Both the turbine- and motor-driven emergency feedwater pumps are tested monthly by manual initiation from the control room. Channel functional test for the emergency feedwater system automatic initiation circuitry for steam generator low-low level, reactor coolant pump bus undervoltage, and safety injection are performed monthly. The emergency feedwater pumps are demonstrated to be operable at least once every 18 months by verifying that each pump starts automatically upon receipt of an emergency feedwater actuation signal which simulates emergency operation of the system.

The automatic initiation signals and associated circuitry used to actuate the emergency feedwater system are part of the engineered safety features actuation system and are powered from the emergency buses. The redundant actuating channels which provide signals to the pumps and valves are physically separated and electrically independent. The alternating current powered motor-driven pumps and valves in the emergency feedwater system are powered from the emergency buses and are included in the automatic sequencing of loads onto these buses. The turbine-driven pump and associated valves are independent from alternating current power sources.

There are four pressure transmitters provided in the emergency feedwater pump suction header from the condensate storage tank. Two-out-of-four logic on low pump suction header pressure automatically open the cross connect valves to the service water system which automatically provides service water for the emergency feedwater system when the condensate storage tank approaches empty.

No single failure within the manual or automatic initiation systems for the emergency feedwater system will prevent initiation of the system by manual or automatic means. The emergency feedwater system can be manually operated from the control room or from the evacuation panels remote from the control room.

The environmental qualification of all safety-related systems, including the emergency feedwater system, is being reviewed for conformance to NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." Our evaluation will be addressed in Section 3.11 of a supplement to this Safety Evaluation Report.

Two emergency feedwater system flow transmitters are provided for each steam generator loop. One flow instrument provides alarm and flow indication in the control room and the evacuation panel. The other flow transmitter provides signals for computer input. Both transmitters provide signals for interlocks which automatically closes the emergency feedwater system flow control valves to a faulted steam generator. A single failure in the flow control circuits will not terminate all emergency feedwater supply to steam generator. The emergency feedwater system flow channels are powered from the Class 1E buses. Testing of the emergency feedwater system flow channels will be performed at 18-month intervals. The system is tested to verify that each pump starts automatically upon receipt of each emergency feedwater actuation signal.

An additional indication of emergency feedwater system performance is provided by one wide-range level channel per steam generator. These wide-range level are environmentally and seismically qualified and tested at 18-month intervals in accordance with the Technical Specifications.

Based on our review of the emergency feedwater system automatic initiation system, we find that the initiation signals, logic, and associated circuitry comply with the long-term safety-grade requirements of Section 2.1.7.2 of NUREG-0578, and therefore, is acceptable.

A further in-depth review of the emergency feedwater system was conducted during our site visit to the facility. During this review we noted a number of concerns related to the control and protection aspects of the design for the auxiliary feedwater system. Subsequently the applicant provided a committment to implement modifications to resolve these concerns. A summary of these concerns and the the corrective actions to be taken are discussed below.

A control valve is provided in each of the two separate feedlines to each steam generator. The control valves permit manual control of emergency feedwater flow to maintain the desired steam generator level for safe shutdown. They also permit manual isolation of a steam generator on feedwater/steam line breaks to protect the containment from overpressurization and to assure an adequate supply of emergency feedwater for the remaining steam generators for safe shutdown.

A single channel system is provided to automatically close the control valves on high flow to a steam generator to permit isolation of a steam generator for feedwater/steam line breaks. The automatic isolation signal for the control valves actuates a solenoid valve that connects an accumulator air supply to the valve to close the control valve. This is provided to assure the availability of an air supply to close the valve, since the normal air supply is from the non-safety grade instrument air system. The electrical power for the two control valves for each steam generator are provided from separate power sources. Thus, from a single failure viewpoint, at least one of the two control valves would be automatically closed on a high flow condition. The alternative means for terminating flow to a faulted steam generator from the control room is to shut off either the turbine-driven or motor-operated emergency feedwater system pumps. The design of the control system does not include the capability to manually close the control valves independent of the availability of the non-safety grade instrument air system. The system will be modified to permit the control valves to be manually closed from the control room using the accumulator air source provide to effect valve closure by the automatic isolation system. In addition the accumulator system design will be modified to permit periodic testing to confirm its availability, and surveillance requirements will be included in the Technical Specifications.

The automatic system for isolating the steam generator on high emergency feedwater flow does not include provisions to indicate that a steam generator has been isolated on a high flow condition. Further, the features which would permit a reset of the high flow trip conditions and reopening of the control valves is provided by a control switch that is used to reset automatic initiation of the system to permit control of emergency feedwater flow. The system will be modified to provide direct indication of an automatic isolation of a steam generator on high flow. This change will be implemented by modifying existing high flow alarms or by separate indicating lights included with modifications for manual closure of the valves. The set point for the high flow alarm will be set at a value not to exceed the isolation set point if not otherwise modified as indicated. Also the reset capability for the automatic isolation will be modified to be independent of the reset for the control valve on automatic initiation of the emergency feedwater system.

As previously noted, tripping of the turbine-driven emergency feedwater pump provides an alternate means to terminate flow to a faulted steam generator for a failure of the control valve to provide isolation. The system design is such that a common mode failure exists which could prevent closure of a control valve and tripping the turbine-driven emergency feedwater pump. In addition this action is dependent upon the availability of the non-safety grade instrument air system to effect closure of the steam admission valve for the the turbine-driven pump. The system will be modified such that a single failure will not prevent the capability from the control room to isolate emergency feedwater flow to a faulted steam generator. An air accumulator will be provided for the steam admission valve to assure the availability of an air supply to effect valve closure. Test features and surveillance requirements in the Technical Specifications will be provided for the air accumulator.

The atmospheric relief valves provide means to control steam generator pressure, without reliance on the safety valves to effect shutdown with the emergency feedwater system. The protection system includes interlocks to close these valves for over-cooling events. These features have been implemented such that the loss of electrical power to either channel of the protective function causes all relief valves to fail closed and preclude their operability from the control room for a plant shutdown. This system will be modified such that a loss of electrical power will not block the relief operability. This improves the sytem tolerance to potential failures which could preclude the capability to control these valves from the control room.

We find that the applicant's commitment to modify the emergency feedwater system design is adequate to address our concerns. We will review the details of these modifications and provide our evaluation of these changes in a supplement to this Safety Evaluation Report.

7.4.2 Remote Shutdown Capability

To meet the remote shutdown requirements of the Criterion 19 of the General Design Criteria and to satisfy the alternate shutdown capability for the fire protection program, the Virgil C. Summer Nuclear Station, Unit 1 provides two control room evacuation panels which include indicators and controls for monitoring several facility parameters and controlling plant equipment necessary to bring the facility to a hot shutdown condition. Two panels are provided to satisfy the requirements for channel separation; however, they are not redundant and both are used to effect hot shutdown. The indicators include:

1. Wide range water level indication for each steam generator.
2. Pressure indication for each steam generator.
3. Pressurizer water level indication.
4. Pressurizer pressure indication.
5. Reactor building temperature indication.
6. Volume control tank level indication.
7. Charging pressure and flow indication.
8. Emergency boration flow indication.
9. Condensate storage tank level indication.
10. Letdown flow indication.

The controls include:

1. Pressurizer heater controls.
2. Charging flow control.
3. Controls for the emergency boration valve
4. Controls for the turbine-driven emergency feedwater system pump flow control valves.
5. Control for the motor-driven emergency feedwater system pumps flow control valves.
6. Controls for the letdown line isolation valves.

7. Controls for the letdown orifice isolation valves.
8. Controls for the steam supply valve to the emergency feedwater pump turbine.
9. Speed control for the emergency feedwater pump turbine.
10. Controls for the service water pumps.
11. Controls for boric acid transfer pump B.
12. Controls for pressurizer power-operated relief valves.

Transfer switches are provided on the control room evacuation panel to permit transferring control of the equipment from the main control room. These transfer switches and associated circuits are designed with the capability to prevent the loss of equipment controls due to failures that may occur in the control room or the cable spreading room.

During the fire protection review, a concern was raised that fire damage in the cable spreading room area could negate some of the capability to be performed at the control room evacuation panels. This matter will be addressed in Section 9.5.1 of a supplement to this Safety Evaluation Report.

The control system for the emergency feedwater flow control valves includes a feature to fully open the valves on automatic initiation of the system. Low steam generator level is one of the signals which initiates this action. This action must be reset before control can be regained to regulate flow. Since the capability to control flow is provided at the remote shutdown panels, modifications will be made to permit a reset of these automatic initiation systems from this location. The applicant will implement this change prior to fuel loading.

7.4.3 Conclusion

Based on our design review, we conclude that systems required to achieve and maintain a safe shutdown are acceptable and satisfy the criteria referenced in Section 7.1.1 of this Safety Evaluation Report. We have also determined that the Virgil C. Summer Nuclear Station, Unit 1 can be safely shut down both from within and outside the main control room. This satisfies the requirements of Criterion 19 of the General Design Criteria and is therefore acceptable.

7.5 Safety-Related Display Instrumentation

7.5.1 Engineered Safety Feature and Reactor Protection System Status Monitoring System

The following instrumentation readouts are provided in the main control room available to the operator during condition II, III, and IV events. There are a minimum of two channels for each parameter with one channel recorded. Two channels are on separate power supplies.

1. Reactor coolant temperature
2. Pressurizer water level
3. Reactor coolant system wide range pressure
4. Reactor building pressure
5. Steam line pressure
6. Steam generator water level (narrow and wide range)
7. Refueling water storage tank level

8. Boric acid tank level
9. Reactor building sump temperature
10. Reactor building spray pump discharge flow
11. Reactor building temperature
12. Reactor building sump level
13. Reactor building water level
14. Condensate storage tank level
15. Emergency feedwater flow
16. Reactor building cooling unit service water flow
17. Service water temperature into and out of reactor building cooling unit
18. Sodium hydroxide storage tank level

The above information is required to maintain the plant in a hot shutdown condition or to proceed to cold shutdown within the limits of the Technical Specifications.

The applicant will perform a detailed control room design review to identify and correct any design deficiencies. This review will include an assessment of control room layout, the adequacy of the information provided, the arrangement and identification of important control and instrumentation displays, the usefulness of the audio and visual alarm systems, the information recording and recall capability, lighting, and other considerations of human factors that have an impact on operator effectiveness.

We will review the applicant's assessments and any corrective actions implemented to assure that the human factor's concerns are addressed. Our evaluation will be provided in a supplement to this Safety Evaluation Report as discussed in Section 22 of this Safety Evaluation Report.

7.5.2 Post-Accident Monitoring

Revision 2 to Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," was issued in December 1980. The operating license will be conditioned to require that the applicant by June 1983 will comply with Regulatory Guide 1.97, Revision 2 or provide justification for any alternatives.

7.5.3 Bypassed and Inoperable Status Indication for Safety-Related Systems

The bypassed and inoperable status indication system provides the control room operator with a continuous system level indication of a bypassed or inoperable condition for the systems comprising the engineered safety features. The system considers the actual status of individual components including systems-level bypasses and the operator-entered inputs for components removed from service.

The bypassed and inoperable status indication component inputs include the following:

1. Emergency core cooling valves open/shut
   Emergency core cooling pumps operable
   Emergency core cooling process (level,pressure) high/low etc.

2. Emergency feedwater valves open/shut
Emergency feedwater pumps operable
Emergency feedwater process high/low etc.

3. Containment spray values open/shut
Containment spray pumps operable
Containment spray process high/low

4. Containment isolation valves open/shut

5. Auxiliary power system breakers open/closed/out
Auxiliary power system generators operable
Auxiliary power system voltage high/low

6. Containment ventilation valves open/shut
Containment ventilation motors operable

7. Containment hydrogen recombiners valves open/shut
Containment cooling recombiners motors operable

8. Component cooling water valves open/shut
Component cooling water pumps operable

9. Service water valves open/shut
Service water pump operable

The systems level bypass functions include the following systems.

1. Safety injection

    - Low pressurizer pressure
    - Low steamline pressure
    - Manual reset

2. Steamline isolation
3. Steam dump interlock
4. Steam generator blowdown isolation.

The interface between the operator and this system is provided by redundant cathode ray tube displays and keyboard consoles located in the control room. The primary display contains bypassed or inoperable status indication for each affected subsystem or either a system level or train level basis. Identification is provided as to whether the condition is due to the inoperable status of a component or unavailability of an auxiliary support such as cooling water or power supply. Other levels of display provide supporting information on individual components within each subsystem, or the operator-entered inputs. Whenever the status of a system becomes inoperable or bypassed, an audible alarm alerts the operator and a video display on a cathode ray tube indicates the affected system and subsystem.

Regulatory Guide 1.47 recommends that system level indication be provided in the control room to indicate when a safety-related system is bypassed or inoperable.

Automatic indication of the bypassed or inoperable status should be provided when the disabled system meets the following criteria:

1. The disabled system will affect the emergency core cooling system, the reactor building cooling or spray system, and/or any auxiliary supporting systems for these systems.

2. The disablement occurs more than once a year.

3. The disablement is expected to occur when the affected system is required to operate.

Manual initiation of indication should provide for all other conditions which could disable safety systems.

Based on our review, the facility design satisfies the recommendations of Regulatory Guide 1.47 and, therefore, we conclude that it is acceptable.

7.6 Other Systems Required for Safety

Other systems required for safety include: (1) the instrumentation and control power supply system discussed in Section 8.3.1 of this Safety Evaluation Report, (2) the residual heat removal isolation valves discussed in Section 7.3.3 of this Safety Evaluation Report, (3) the refueling interlocks discussed in Section 9.1 of this Safety Evaluation Report, (4) the accumulator valves discussed in Section 7.3.3 of this Safety Evaluation Report, (5) switchover from injection to recirculation discussed in Section 6.3.3 of this Safety Evaluation Report, and (6) the leak detection systems discussed in Section 5.2.5 of this Safety Evaluation Report.

These systems are similar to systems provided in the Joseph M. Farley, Unit 1 facility, with no significant differences identified by the staff.

Based on our review, we conclude that the other systems required for safety satisfy the criteria referenced in Section 7.1.1 of this Safety Evaluation Report and are acceptable.

7.7 Control Systems Not Required for Safety

The Final Safety Analysis Report identifies the following control systems not required for safety:

1. Reactor control system,
2. Rod control system,
3. Plant control interlocks,
4. Pressurizer pressure control,
5. Pressurizer water level control,
6. Steam generator water level control, and
7. Steam dump control.

As a result of the accident at TMI-2, the Commission identified a number of actions to be taken for all plants related to control systems not required for safety. In addition, a number of IE Bulletins have been issued which address problems that have been identified as a result of operating reactor experience.

We are continuing our review of generic concerns related to control systems and, as specific problems are identified, their resolution will be applied to the Virgil C. Summer Nuclear Station, Unit 1 as necessary. All currently identified concerns have been addressed in the review of control systems for the Virgil C. Summer Nuclear Station, Unit 1, and we find this area of the facility design acceptable.

8 ELECTRIC POWER SYSTEMS

8.1 General

The requirements in Criterion 17 and Criterion 18 of the General Design Criteria, Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," Regulatory Guide 1.75, "Physical Independence of Electric Systems," Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light Water Cooled Nuclear Power Plants," and IEEE Standard 308-1974 "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations" served as the primary bases for evaluating the adequacy of the emergency power systems for the facility. A complete list of acceptance criteria is included in Table 8.1 of the Standard Review Plan. Section 8 of the Standard Review Plan provided guidance for conducting our review.

The following subsections provide our evaluation of the design criteria and design description in the Final Safety Analysis Report. The conclusions in the following subsections are subject to acceptable implemention of design changes that may be required as a result of our electrical drawing review and fire protection review.

8.2 Offsite Power System

The offsite power system is the preferred source of power for the facility. This system includes the grid, transmission lines, transformers, switchyard components and associated control systems provided to supply electric power to safety-related equipment and other equipment. The electrical grid is the source of energy for the offsite power system. The safety function of the offsite power system (assuming that the onsite power systems are not available) is to provide sufficient capacity and capability to assure that the specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary will not be exceeded and to assure that core cooling, containment integrity and other vital functions will be maintained in the event of postulated accidents. The objectives of our review are to determine that the offsite power system (1) satisfies the applicable criteria set forth in Section 8.1 of this Safety Evaluation Report, and (2) can reliably perform its design functions during normal operation, anticipated operational occurrences, and accident conditions.

Six 230 kilovolt transmission lines connect the Virgil C. Summer Nuclear Station, Unit 1 switchyard to the South Carolina Electric & Gas Company transmission system. Two additional 230 kilovolt transmission lines interconnect the facility with the South Carolina Public Service Authority system. In addition, two 230 kilovolt transmission lines extend directly from the facility substation bus section 3 to the South Carolina Electric & Gas Company's Fairfield Pumped Storage

Facility. The 230 kilovolt system serves as only one of the two required offsite power sources, which feeds the emergency buses through a step down transformer.

One 115 kilovolt transmission line extends from South Carolina Electric & Gas Company's Parr Generating Complex and serves as an alternate source of offsite power for the engineered safety features. This line terminates in the substation in a rigid bus construction for the crossover of the 230 kilovolt bus has no connection to the 23 kilovolt bus. The two separate sources of offsite power have sufficient separation and isolation so that no single event such as transformer failure or transmission line tower failure can cause simultaneous disruption of both sources. Power from both the 115 kilovolt preferred source tranformer and the 230 kilovolt transformer is brought into the plant by physically separated and independently supported buses to the redundant 7.2 kilovolt Class 1E electric system.

The facility's 230 kilovolt substation is a single bus single breaker arrangement with the buses divided into three sections. Primary and backup relay protection has been provided for each 230 kilovolt line as well as for the 115 kilovolt backup source line. The breaker control power is obtained from 125 volt direct current battery located in the substation control room with a backup feed from the non-Class 1E battery.

During normal operating conditions the main generator supplies electrical power through isolated phase buses to the main step-up transformer, and the unit auxiliary transformer located adjacent to the isolated-phase but at a point between the generator circuit breaker and the low voltage connections to the main step-up transformer. During normal operation, station auxiliary power is provided from the main generator through the unit auxiliary transformer. During normal startup and shutdown, auxiliary power is supplied from the 230 kilovolt substation through the main transformer, with the unit generator isolated by the unit generator circuit breaker. Two emergency auxiliary transformers are connected to the 230 kilovolt substation bus. The two secondary windings on each bank are rated at 7200 volts. Three of the four windings are used as an emergency power source for the three 7200 volts non-Class 1E auxiliary system buses. The fourth winding is the preferred power source for either or both of the 7200 volt essential safety features buses. A second preferred power source for either or both of the 7200 volt essential safety features buses is provided by separate 115 kilovolt essential safety feature transformers fed from 115 kilovolt transmission lines as discussed above. A manual transfer will be required for switching bus power supply to the alternate source. This transfer can be accomplished from the control room. This configuration meets the requirements of Criterion 17 of the General Design Criteria and is acceptable.

The substation components are testable during reactor power operation. The breakers are inspected, maintained and tested on an individual basis while allowing the 230 kilovolt substation to remain energized. We find this capability to be in conformance with the requirements of Criterion 18 of the General Design Criteria and acceptab.

8.2.1 Grid Stability

The applicant has conducted grid stability studies on the portion of the network contiguous to and in the vicinity of the South Carolina Electric & Gas Company grid supplying the offsite power for Virgil C. Summer Nuclear Station, Unit 1.

The simulated contingencies included simultaneous loss of any system generator including Virgil C. Summer Nuclear Station, Unit 1 and the most critical transmission line associated with its loss. In addition, the system is stable for the most severe fault condition on any transmission line or substation bus. The results of these grid stability studies indicate that the grid which supplies the offsite power for the facility remains stable for the conditions noted above.

Based on our review of the applicant's results of the stability studies presented in the Final Safety Analysis Report, there is reasonable assurance that the ability of the South Carolina Electric & Gas Company grid to provide offsite power to the facility will not be impaired by the loss of the largest single supply to the grid. This satisfies our requirements set forth in Section 8.2 of the Standard Review Plan and is acceptable.

8.2.2 Sustained Degraded Grid Voltage Position and Offsite/Onsite Power System Interaction

As a result of the Millstone Unit 2 low grid voltage occurrence, the NRC staff developed additional requirements concerning (1) sustained degraded voltage conditions at the offsite and onsite emergency power systems, and (2) interaction of the offsite and onsite emergency power systems. We have compared the Virgil C. Summer Nuclear Station, Unit 1 design to our established position on offsite/onsite power system interaction, and have reached the following conclusions. Our position is in four parts and is addressed separately below.

Part 1 of the position requires a second level of undervoltage protection for low grid voltages. The undervoltage relays traditionally used to detect loss of voltage (offsite power) at the emergency buses have had setpoints around 70 to 75 percent of nominal bus voltage. This protection alone does not protect the facility loads from damaging low voltages which are maintained above this setpoint. Our requirement has, therefore, been to require an additional protective trip at approximately 90 percent of nominal bus voltage with a time delay to avoid spurious trips due to short duration transients such as those occurring when starting large motors.

The scheme providing undervoltage and degraded voltage protection for the Virgil C. Summer Nuclear Station, Unit 1 power distribution system consists of two sets of three undervoltage relays per safety-related bus. The first level undervoltage relays are set at 80 percent of bus voltage with no time delay (instantaneous). The second level undervoltage relays are set at 90 percent of the bus voltage with a time delay of 10 seconds. The actuation of the undervoltage relays of either level will trip the offsite power feeder breakers to the bus. A three-out-of-three coincidence logic per bus is provided to preclude spurious trips of the offsite power source. The second level of undervoltage protection system meets the requirements of IEEE Standard 279-1971 and is acceptable.

The Technical Specifications have been modified to include limiting conditions for operation and surveillance requirements, as well as trip setpoints for allowable values for the second level of voltage protection. We find this aspect of the design to be acceptable.

The undervoltage setpoint and allowable time duration of a degraded voltage must not result in failure of safety systems or components. It is our position that the applicant must demonstrate that the time delay chosen does not exceed the maximum time delay considered in the accident analyses in the Final Safety Analysis Report. The applicant has stated that the time delay is consistent with the safety analyses and has committed to supply the supporting information. We find this acceptable subject to documentation of the supporting information. We will report on our review of this information in a supplement to this Safety Evaluation Report.

Part 2 of our position requires that when the diesel generators are supplying power to the emergency buses, the load shedding feature be automatically bypassed during the load sequencing cycle and this load shedding feature be automatically reinstated when the load-sequencing cycle is complete. This is required so that the voltage drops encountered during load sequencing on the diesel generators will not interact with the load shedding feature and negate the loading sequence.

In the design of this facility, the load shedding feature, through the essential safety features loading sequencer is automatically disconnected when the buses are transferred to onsite power. When the buses are returned to the offsite power sources, the undervoltage tripping feature is automatically reinstated. We find this design to be consistent with our position and acceptable.

Part 3 of our position deals with incorporating onsite power tests and test frequencies into the Technical Specifications to assure continued adherence to this position throughout the facility lifetime. These provisions have been incorporated into the Technical Specifications proposed by the applicant and we find this acceptable.

Part 4 of our position requires that the tap settings on the plant transformers be optimized and verified at the preoperating testing stage by measurement. The applicant has stated that the voltage fluctuates on the preferred sources between 99 percent and 104 percent of the nominal value of 230 kilovolt and 115 kilovolt respectively for the facility. Taps on the stepdown transformers in the distribution network are selected to provide proper voltage level for safety-related buses. The initial taps are selected considering a full load on the buses. When the is at the startup stage, the selected taps are applied and the voltage on the buses is verified. If necessary, the taps will be adjusted to obtain the desired voltage range of plus or minus five percent of the nominal bus voltage.

We have reviewed the above information and conclude that this is inadequate and we will require the applicant to submit for our approval (a) the analytical method used for calculating voltage at all distribution levels to demonstrate that the transformer tap settings have been fully optimized for the facility, and (b) test plans and test results to demonstrate that the analytical method used for calculating these voltages at all distribution levels is valid. The applicant has committed to provide this information. We find this acceptable subject to our review of the applicant's submittal. We will report the results of our review in a supplement to this Safety Evaluation Report.

The design modifications, described in part 1 above, have been implemented by the applicant in accordance with the requirements of IEEE Standard 279-1971, IEEE Standard 308 and Criteria 17 and 18 of the General Design Criteria. We find this aspect of the design to be acceptable subject to documentation and our evaluation of the information to be provided by the applicant for parts 1 and 4 of our position as discussed above. We will report the results of our evaluation in a supplement to this Safety Evaluation Report.

8.2.3 Conclusion

The scope of our review included the descriptive information, functional logic diagrams, electrical single line diagrams, selected electrical schematics for the offsite power system, and the applicant's analyses of the adequacy of these criteria and bases.

On the basis of our review, we have concluded that the offsite power system satisfies our requirements set forth in Section 8.1 of this Safety Evaluation Report and are acceptable upon satisfactory completion of the items discussed in Section 8.2.2 of this Safety Evaluation Report.

8.3 Onsite Power Systems

8.3.1 Alternating Current Power System

The alternating current onsite emergency power system serves as a standby to the offsite power system. The safety function of the alternating current onsite emergency power system (assuming the offsite power system is not functioning) is to provide sufficient capacity and capability to assure that the structures, systems and components important to safety perform as intended. The objectives of our review are to determine that the alternating current onsite emergency power system has the required redundancy, meets the single failure criterion, is testable, and has the capacity, capability, and reliability to supply power to all required safety loads. In addition to verifying the above, our review will determine if the system is designed in accordance with the applicable criteria set forth in Section 8.1 of this Safety Evaluation Report.

The onsite power and distribution system is compared of an emergency portion qualified as a Class 1E system and a normal portion which is a non-Class 1E system. There is one auxiliary unit and two emergency auxiliary transformers which supply normal power to the onsite system. The unit auxiliary transformer has three low voltage windings and the emergency auxiliary transformers each have two low voltage windings. The unit auxiliary transformer supplies power to the reactor coolant pumps in a one pump per bus configuration. The primary sides of the two emergency auxiliary transformers are connected in parallel to the 230 kilovolt substation bus. The two secondary windings on each bank are rated at 7.2 kilovolts. Three of the four windings supply backup power to the reactor coolant pump buses. The fourth winding is the preferred power source for either or both 7.2 kilovolt essential safety feature buses.

The non-Class 1E alternating current system of the facility is normally supplied by the main generator through the unit auxiliary transformer, which is connected to the unit main generator isolated phase bus, is backfed from the 230 kilovolt bus through the main power transformer and unit auxiliary transformer with the unit generator isolated by the unit generator breaker. Upon tripping of the normal feeder breaker, the non-Class 1E electrical system is automatically transferred to the emergency auxiliary transformers which are the emergency power sources as indicated above. This automatic transfer is initiated when the normal feedwater breaker is tripped oy the main and unit auxiliary transformer lockout relay, generator differential protection relay, generator and main transformer backup and field failure and main transformer backup and field failure relaying and overall backup lockout relaying. There is no automatic transfer when a bus over-current condition exists.

The design also includes a generator breaker on the isolated phase bue between the main generator and the unit auxiliary transformer, to serve as means of isolating the main generator from the transmission network. This generator breaker is used only to provide startup and shutdown power to the non-Class 1E loads by isolating the generator from the system. This action eliminates the necessity for a transfer from the emergency auxiliary transformer to the normal unit auxiliary transformer.

The Class 1E portion of the onsite power system is comprised of two redundant and independent 7.2 kilovolt distribution systems each with their 480 volt load centers and motor control centers, 120 volt vital alternating current power system and the standby power supplies (diesel generator units). The normal source of power for the two independent Class 1E distribution networks are the essential safety features transformers fed from 115 kilovolt transmission line as discribed in Section 8.2 of this Safety Evaluation Report and a winding of the emergency auxiliary transformer as described above. These two sources of power also serve as an alternate source of power to each other, however, the transfer to the alternate offsite power source can only be initiated manually and this can be accomplished from the control room.

The onsite emergency power is supplied by two diesel generator units. Both units are automatically started by either a safety injection actuation signal or an emergency bus undervoltage signal. The units will be connected automatically to their respective emergency bus upon loss of offsite power and, under accident conditions, the safety loads will be automatically connected in a predetermined sequence to their respective diesel generator. There is one diesel generator per bus. Each diesel generator with its supporting auxiliaries is located in a separate seismic Category I structure and is rated at 4250 kilowatts for continuous operation, and 4676 kilowatts for short time operation. The total loads do not exceed the short time rating of the diesel generator and analysis has shown that during the loading sequence, the frequency and voltage are maintained above a level which would degrade the performance of any load below minimum requirements. This meets the requirements of Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies" and we find this to be acceptable, subject to successful pre-operational testing. We will request that our Office of Inspection and Enforcement provide followup at the facility on this item. We will report the results in a supplement to this Safety Evaluation Report.

Branch Technical Position ICSB 2 (PSB), of Appendix 8A of the Standard Review Plan, requires that new and previously untried diesel generator designs to be used in nuclear power plant service undergo a prototype reliability qualification testing program. The applicant has provided the results of the reliability qualification testing for our review. We have reviewed this information and concluded that it meets our position and is acceptable.

Branch Technical Position ICSB 17 (PSB) of Appendix 8A of the Standard Review Plan requires that the diesel generator protective trips be bypassed when the diesel generator is required for a design basis event. All protective trips are required during periodic testing. The allowed exceptions to the above requirement for bypassing are diesel overspeed and generator differential. Any other trips retained must utilize coincident logic in order to avoid spurious trips. The applicant has provided the two trips mentioned above plus low lube oil pressure. A matrix arrangement is provided for tripping the diesel generator for lube oil pressure. This matrix consists of four pressure relays set at 60, 55, 50, and 50 pounds per square inch. To cause a diesel generator trip due to low lube oil pressure, two of the low pressure switches must be activated and at least one of the two activated switches must be one of the two with the 50 pounds per square inch setpoints. This is in full conformance with our position and is acceptable.

We have reviewed the diesel generator alarms and status information provided for the control room operator. The control room annunciation consists of single input alarms and common alarms. The annunciator window engraving for the single input alarm identifies the specific nature of the problem. The window engraving for the common alarms is generalized and the applicant has presented in the Final Safety Analysis Report a list of conditions that render the diesel generator units incapable of responding to an automatic emergency start signal. We have reviewed this information and conclude that each condition which can render a diesel generator unit incapable of responding to an automatic emergency start signal is alarmed in the control room except for the engine running (greater than 335 revolutions per minute) condition. Although not alarmed, this condition is obvious from the metering and indication on the main control board, therefore, we find this to be acceptable.

There are six redundant and independent divisions of 120 volt Class 1E vital instrumentation and control power subsystems that provide power to the four channels (A, B, D, and E) of the reactor trip and engineered safety features actuation system and two channels of balance of plant vital loads which are associated with channels A and B. Each vital instrumentation and control alternating current power supply consists of one solid-state inverter/rectifier and one distribution panel. Normally, the distribution panel is supplied from the inverter/rectifier. This inverter/rectifier is fed from the Class 1E 480 volt bus. The Class 1E direct current power system constitutes the standby power source, three inverters connected to each of the two Class 1E station batteries. In the event of loss of 480 volt power, the power source for the vital bus inverter is the station battery. The change in power source, from normal to standby, occurs by actioneering without exceeding the stated inverter output voltage and frequency regulation. An alternate source power supply for the 120 volt vital buses is provided through 480/120 volt transformers from 480 volt Class 1E buses for each inverter for use when the inverters are out of service. These alternate supply circuit breakers are mechanically interlocked with the normal power supply circuit breakers so that only one circuit breaker can be closed at a time. Further, there are no provisions for either manually or automatically transferring loads or sources between the redundant subsystems. Inverter trouble alarms are annunciated in the control room. Based upon our review we have determined that the four vital alternating current subsystems are independent.

The Class 1E portion of the emergency onsite power and distribution system is designed to permit the following testing and inspections:

1. During equipment shutdown, periodic inspection and testing of wiring, insulation, connections, and relays to assess the continuity of the systems and the condition of components.

2. During normal operation, periodic testing of the operability and functional performance of standby onsite power supplies circuit breakers and associated control circuits, relays, and buses.

3. During shutdown, testing of the operability of the Class 1E system as a whole. Under conditions as close to design as practical, the full operational sequence that brings the system into operation, including operation of signals of the engineered safety features actuation system and the transfer of power between the offsite and the standby onsite power systems, will be tested.

We find that the above is in conformance with Criterion 18 of the General Design Criteria and is acceptable.

The applicant has applied the following design criteria to the Class 1E equipment. The criterion for motor size is that the motor develop sufficient horsepower to drive the mechanical load for maximum expected flow and pressure. Motors are sized to permit the driven equipment to develop its specified capacity without exceeding the temperature rise rating of the motor when operated at the duty cycle of the driven equipment. The motors are designed for across the line starting. Essential safety features motors rated at 6900 volts are capable of accelerating the driven equipment to rated speed at 70 percent of the motor nameplate voltage. Essential safety features motors rated 460 volts, however, are capable of accelerating the driven equipment to rated speed at 80 percent of rated voltage. We have informed the applicant that this higher starting voltage for 460 volt essential safety features motors is unacceptable unless it can be demonstrated that, at no time during sequencing of safety loads on diesel generators, will the voltage at the 460 volt level go below 80 percent of the rated voltage. The applicant will provide the analysis to demonstrate this. We find this acceptable subject to documentation of the applicant's analysis. We will report the results of our evaluation in a supplement to this Safety Evaluation Report.

We have reviewed the emergency onsite power system and have determined the following: There are no automatic transfers of loads or sources between redundant emergency buses, which is in accordance with Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems." The two divisions of the emergency power and distribution system are independent, meet the single failure criterion and have the required capability, capacity and stability as required by Criteria 17 and 18 of the General Design Criteria. The design is also in conformance with IEEE Standard

308-1974 as endorsed by Regulatory Guide 1.32. We, therefore, find the emergency onsite alternating current power system to be acceptable subject to documentation and evaluation of the applicant's forthcoming analysis as discussed above.

8.3.2 Direct Current Power System

The direct current power system provides, (1) the alternating current offsite and onsite emergency power systems with control power as required, (2) power to the six inverters of the Class 1E vital instrumentation and control alternating current power subsystem and (3) motive and control power to selected safety-related equipment. The objectives of our review are to determine that the direct current power system is designed in accordance with the applicable criteria set forth in Section 8.1 of this Safety Evaluation Report; and to establish that it has the required redundancy, capacity, capability, and reliability to supply power to all required safety loads.

The direct current power system is comprised of the Class 1E direct current power system and the non-Class 1E direct current power system. The non-Class 1E 125 volt direct current power system provides power for non-Class 1E loads, and is also a manually switched emergency backup direct current power source for the 230 kilovolt substation direct current system. The system consists of one 125 volt battery, two static battery chargers and a distribution panel. Based on our review we have determined that the non-Class 1E direct current power system is independent of the Class 1E system.

The Class 1E direct current power system for control and instrumentation consists of two 125 volt, lead calcium, 60 cell batteries, two 125 volt battery buses and three static battery chargers. Two of the three battery chargers are supplied from separate, redundant motor control centers. One of these three static battery chargers serves as a standby charger and is provided for use during maintenance of, and to back up, either of the normal power supply chargers. The Class 1E batteries, chargers and direct current power distribution panel, are located in the seismic Category I intermediate building. The battery rooms are ventilated by a system that is designed to preclude the possibility of hydrogen accumulation.

During normal operation, the 125 volt direct current (Class 1E and non-Class 1E) load is supplied from the battery chargers with the batteries floating on the system. Upon loss of station alternating current power, the entire load is powered from the batteries until the alternating current power is restored by the emergency diesel generator or the preferred power source. No provisions exist for either manually or automatically transferring loads or sources between the redundant direct current systems in accordance with Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems." Based upon our review, we find that the Class 1E direct current subsystems are independent.

Each Class 1E battery has sufficient capacity to independently supply the required safety loads for two hours. Each battery charger has enough capacity to recharge the battery from its designed minimum charge state to the fully charged state within 12 hours while simultaneously supplying the largest combined demand of various steady-state and transient loads irrespective of the status of the facility during which these demands occur. The battery chargers also have the capacity to perform their required function if their associated battery is disconnected for any reason. This in accordance with Regulatory Guide 1.32, "Criteria for Safety-Related Power Systems for Nuclear Power Plants," and is acceptable.

We have reviewed the provisions described in the Final Safety Analysis Report for testing the Class 1E direct current power system. Our review was conducted to determine the capability to perform surveillance tests that are included in the Technical Specifications and the testing capability required by Criterion 18 of the General Design Criteria. On the basis of our review, we conclude that the design as presented will be capable of meeting these requirements. We find this aspect of the design to be in accordance with Criterion 18 of the General Design Criteria and acceptable.

In conclusion, the Class 1E direct current power system has the required independence, redundancy, and capability to perform its safety functions while degraded by a single failure. This fulfills the requirements of Criterion 17 of the General Design Criteria and is acceptable.

8.3.3 Physical Identification and Independence of Redundant Safety-Related Electrical Systems

The applicant has provided in the Final Safety Analysis Report the criteria for physical identification and separation of electrical equipment to preserve the independence of redundant equipment. Physical identification of safety-related electrical systems is accomplished as follows. Each cable and raceway is color coded to indicate its separation group. Name plates of appropriate color background are provided for all equipment. This identification provides a means of distinguishing a cable, raceway and equipment associated with a particular separation group. We find the above identification criteria to be acceptable.

The applicant has stated that the separation of redundant safety circuits is in accordance with Regulatory Guide 1.75, "Physical Independence of Electric Systems." The redundant Class 1E circuits are run in separate and independent raceways. In general plant areas, not subject to hazards such as missiles, open ventilated cable trays for redundant circuits are separated by a minimum of three feet horizontally or five feet vertically. In the cable spreading room, open ventilated cable trays are separated by a minimum of one foot horizontally or three feet vertically. Totally enclosed raceways for redundant circuits are separated by a minimum of one inch. Where these separation criteria cannot be met, barriers are placed between the raceways. Where non-Class 1E circuits are connected to Class 1E equipment or are routed in the same raceways with Class 1E circuits they are designated as associated circuits. Circuits designated as associated circuits are routed with the designated separation channel throughout their length. Where non-Class 1E circuits are connected to Class 1E equipment, an isolation device is provided to protect the Class 1E equipment (see Section 8.4.4 of this Safety Evaluation Report for additional details). Where the separation between the raceways for non-Class 1E circuits and raceways for Class 1E circuits does not meet the criteria for raceways carrying redundant Class 1E circuits, case by case analysis has been performed. Testing and/or analyses are acceptable methods for demonstrating adequate separation in accordance with the requirements of IEEE Standard 384-1974, "Criteria for Independence of Electrical Circuits," and the recommendations of Regulatory Guide 1.75, "Physical Independence of Electrical Circuits."

We initially reviewed the detailed analyses of three typical cases presented by the applicant and concluded that it was inadequate to make a fair evaluation of the rest of the cases which are presented in a tabular form in the Final Safety Analysis Report. We informed the applicant that an analysis for each case must be presented for our review where the separation criteria are not met including cases where non-Class 1E trays run close and are even on the same hangers with Class 1E trays. The applicant has now provided the requested information. We have reviewed the applicant's analysis and conclude that it has been demonstrated that an acceptable separation between trays for non-Class 1E circuits and trays has been provided such that a fire in the non-Class 1E cable trays will not jeopardize the independence of the redundant Class 1E cable trays. We find this acceptable.

8.3.4 Fire Protection

Special requirements needed for the plant electrical systems to satisfy Appendix A to Branch Technical Position APCSB 9.5-1 "Fire Protection of Nuclear Power Plants," will be reviewed at a later date during the site visit and fire protection review of the facility. Additional recommendations may be proposed to further improve the capability of the electrical systems resulting from the site visit and completion of the fire protection review. We will report on this matter in a supplement to this Safety Evaluation Report.

8.3.5 Conclusions

The scope of our review included the descriptive information, functional logic diagrams, electrical single line diagrams, selected physical arrangement drawings and selected electrical schematics for the onsite emergency power systems and for those auxiliary systems that are vital to the proper operation of the onsite emergency power sytems and their connected emergency loads. Our review includes the applicant's design bases and their relation to the proposed design criteria for the onsite emergency power systems and for the vital supporting systems, and the applicant's analyses of the adequacy of those criteria and bases.

On the basis of our review, we have concluded that the alternating current onsite emergency power system and the direct current onsite power system satisfy our requirements as set forth in Section 8.1 of this Safety Evaluation Report and are acceptable subject to the acceptable resolution of the matters discussed in Sections 8.3.1 and 8.3.4 of this Safety Evaluation Report.

8.4 Other Electrical Features and Requirements for Safety

This section presents other electrical features and requirements applicable to the safety of the facility and which deal with distinct aspects of the design of the offsite power system and alternating current onsite power systems. The objective of our review is to determine that these electrical features and requirements are implemented in accordance with all applicable acceptance criteria set forth in Section 8.1 of this Safety Evaluation Report.

8.4.1 Containment Electrical Penetrations

In order to meet the requirements set forth in IEEE Standard 317-1972, "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," as augmented by the recommendations of Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," the containment electrical penetration assemblies for the facility are designed to withstand, without loss of mechanical integrity, the maximum available fault current for a period of time sufficiently long to allow back-up circuit protection to operate assuming a failure of the primary protective device. The applicant has applied the following design criteria to the containment electrical penetration circuits.

For the reactor coolant pump circuits fed from 7.2 kilovolt switchgear, the motor feeder protective relays are coordinated with, and backed up by, the bus protective relays.

For circuits fed from 480 volt switchgear, the motor feeder air circuit breakers are coordinated with, and backed up by, the overcurrent relays of the bus protective breaker.

For circuits fed from motor control centers, the normal overcurrent protective devices (breaker) are backed up by a thermal-magnetic current limiting circuit breaker added to each circuit.

Control rod drive power circuits are protected by two sets of fuses in series which are integral with the rod drive control system.

Power circuits supplied from alternating current and direct current panels are protected with thermal-megnetic circuit breakers which are backed up by fuses in series with the breakers.

Direct current circuits are supplied from the ungrounded direct current power systems and each circuit is protected with either two fuses or two thermal magnetic circuit breakers.

We have reviewed the above information and conclude that the applicant has provided a primary and backup circuit protection device for each circuit that passes through the penetration and we find this acceptable. However, the circuit overload protection employing breakers requiring control power does not meet the single failure criterion set forth in IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." This is because the control power used for tripping the primary and backup breakers are powered from the same battery. We informed the applicant that this was unacceptable and that we require the containment penetration circuit protection to meet the single failure criterion, and that we required the submittal of a modified design. The applicant has modified the design such that the control power used for tripping the primary and backup breaker is powered from separate batteries. We conclude that the modified design meets our position and is acceptable.

8.4.2 Thermal Overload Protection Bypass

Motor-operated valves with thermal overload protection devices for the valve motors are used in safety systems and their auxiliary supporting systems. Operating experience has shown that indiscriminate application of thermal overload protection devices to the motor associated with these valves could result in needless hindrance to successful completion of safety functions. Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," addresses this subject and recommends in position C.1 bypassing of thermal overload relays during accident conditions or in position C.2 properly selecting the setpoints for the thermal overloads in a manner that precludes spurious trips.

The facility's motor-operated valves activated by a safety injection signal in the event of a loss-of-coolant accident have their respective thermal overload protection devices bypassed by the same safety injection signal contact that initiates the valve operation. This is in conformance with position C.1 of Regulatory Guide 1.106 and is acceptable.

8.4.3 Power Lockout to Motor-Operated Valves

The applicant has provided in the Technical Specifications a list of valves that require power lockout in order to meet the single failure criterion in the fluid systems. Branch Technical Position ICSB 18 (PSB) of Appendix 8A to the Standard Review Plan requires that all such valves be listed in the Technical Specifications and that the position indication for these valves meet the single failure criterion. The power lockout for the facility will be accomplished as follows: Power is locked out to accumulator isolation valves 8808A, B and C by removal of the breakers from the circuits. For motor-operated valves 8884, 8888A and 8889, a power lockout contactor will be installed in series with their starters. When the contactor is open it will break power to starters of these valves. Before the valve position can be changed, the power lockout contractor must be closed from the main control board to provide power to the starters. Thus two manual operator actions will be required to close the valves.

In order to meet our requirement that redundant valve status indication be provided to the control room operator, the applicant has provided the following design. Redundant and separate valve position switches are provided. One position switch is a cam-operated switch within the motor operator. The second position switch is mounted on the valve stem and is actuated by mechanical motion of the valve stem. In the event that one position switch is inoperable, the second will be available. Each position switch actuates a separate indicating light in the main control room. These indicating lights are powered from redundant and diverse power supplies. One is the 480/120 volt alternating current control transformer in the motor control center and the other is the 120 volt direct current control power from the train opposite to the valve motive power train. We find that the listing of the valves, the methods of power lockout and the valve position indication are in accordance with our position and are acceptable.

8.4.4 Non-Safety Loads on Emergency Sources

Present regulatory practice for operating license applications allows the connection of non-safety loads in addition to the required safety loads to Class 1E (emergency) power sources if it can be shown that the connection of the non-safety loads will not degrade the emergency sources below an acceptable level. This is accomplished by disconnecting the non-safety loads automatically from the Class 1E sources upon detection of an emergency condition.

The facility design provides for the connection of both safety and non-safety loads to the Class 1E emergency buses of the alternating current and direct current onsite emergency power systems. The connection of these non-safety loads to the Class 1E buses does not exceed the continuous rating of the emergency power sources, i.e., diesel generators and batteries.

The design provides for the automatic disconnection of non-safety loads connected to the 7.2 kilovolt and 480 volt switchgear upon detection of an emergency condition. Reconnection of these non-safety loads to the emergency buses requires subsequent deliberate operator action.

The non-safety loads that are connected to motor control centers and the Class 1E batteries are not automatically disconnected upon detection of an emergency signal. This is because these non-safety loads are important for safe operation of the plant and the emergency power sources have been designed to handle these loads. For these loads the design includes two Class 1E protective devices in series for each non-Class 1E circuit.

Based on our evaluation of the information provided by the applicant we conclude that the design for loading the 7.2 kilovolt and 480 volt switchgear non-Class 1E loads onto the emergency buses is in accordance with Regulatory Guide 1.75 and, therefore, acceptable. We further conclude that the non-safety loads connected to the motor control centers and Class 1E batteries with two Class 1E protective devices in series provide diverse and redundant isolation capability which poses no threat to the Class 1E system and, therefore, we find this alternate design to be acceptable.

8.4.5 Use of a Load Sequencer with Offsite Power

Recently, in a meeting with the applicant discussing grid stability, we were informed that the facility design includes the use of a load sequencer for the connection of emergency safety features loads to the emergency buses when power is being supplied either from offsite or from the diesel generators. This information had not been included in Section 8 of the Final Safety Analysis Report. It is our understanding that the basis for this design feature is to assure sufficient voltage profiles on the safety buses when the grid is being operated at the low end of its design voltage range. We have informed the applicant that in order for us to accept the use of a single sequencer for both offsite and onsite power sources, we would require the following additional information:

1. A full description of this design feature in the Final Safety Analysis Report. This should include sequencer components, power supplies, test features and alarms.

2. A reliability study on the sequencer.

3. A detailed analysis to assure that there are no credible sneak circuits or common mode failures in the sequencer design that could render both onsite and offsite power sources unavailable.

The applicant has been requested to provide this information and we will report the resolution of this item in a supplement to this Safety Evaluation Report.

9 AUXILIARY SYSTEMS

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

We reviewed the auxiliary systems which are necessary for safe facility shutdown. These include the service water system, component cooling water system, ultimate heat sink, portions of the chemical and volume control system, and safety-related ventilation systems.

We reviewed the systems necessary to assure safe handling of fuel and adequate cooling of spent fuel which include the new and spent fuel storage facilities, portions of the spent fuel pool cooling and cleanup system, portions of the fuel handling system, and portions of the fuel handling building ventilation system.

We reviewed the equipment and floor drainage system the failure of which would not prevent safe shutdown, but, indirectly could be a potential source of radiological release to the environment.

We also reviewed certain auxiliary systems whose failure would neither prevent safe shutdown nor result in potential radioactive releases but could affect safety-related systems. These include pressurizer relief tank, "sterile" water system, demineralized water makeup system, reactor makeup water system, compressed air systems, and the non-safety-related ventilation systems. The acceptability of these systems was based on our determining that: (a) where the system interfaces or connects to a seismic Category I system or components, seismic Category I isolation valves will be provided to physically separate the non-essential portions from the essential system or components, and (b) the failure of non-seismic systems or portions of the systems will not preclude the operation of safety-related systems or components located in close proximity. We find the above listed systems meet our criteria and, therefore, find them acceptable.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

The new fuel storage racks provide dry storage for approximately one-third of the full core load of 157 fuel assemblies. The racks are designed to maintain the fuel assemblies in an array which is sufficient to maintain an effective multiplication factor of 0.95 or less in the event that the new fuel area were flooded with unborated water and will limit the effective multiplication factor to 0.98 with fuel of the highest anticipated enrichment in the event that optimum moderation occurs under dry, fogged, or flooded conditions. The rack design precludes the inadvertent placement of a fuel assembly in the rack closer than the prescribed spacing. The new fuel storage racks are bolted together and anchored to the new fuel storage facility floor. The new fuel racks and storage structure are designed to seismic Category I requirements. The new fuel storage racks are located in the fuel handling building.

We have reviewed the adequacy of the applicant's design for the new fuel storage facility necessary to maintain a subcritical array during normal, abnormal, and accident conditions. We have concluded that the design of the facility is in conformance with Criterion 62 of the General Design Criteria and the positions of Regulatory Guide 1.13, "Fuel Storage Facility Design Basis," and, therefore, is acceptable.

9.1.2 Spent Fuel Storage

Spent fuel will be stored under water in the spent fuel pool. In Amendment 3 to the Final Safety Analysis Report, the applicant proposed to increase the spent fuel storage pool capacity to 4-1/3 cores at an assembly center-to-center spacing of 14 inches. Space between storage positions is blocked to prevent insertion of a fuel assembly in any other than its prescribed location. With the above spacing, the effective multiplication factor will not exceed 0.95. The spent fuel storage racks are designed to withstand the impact of a dropped fuel assembly without unacceptable damage to the fuel. The spent fuel racks, the storage pool, and pool liner are designed to seismic Category I requirements. The spent fuel racks can withstand the maximum uplift forces exerted by the fuel handling machine. The spent fuel storage facility is located in the fuel handling building. The design of missile protection for the spent fuel pool is discussed in Sections 3.5.1 and 3.5.2 of this Safety Evaluation Report.

We have reviewed the adequacy of the applicant's design for the spent fuel storage facility necessary to maintain a subcritical array during all operating conditions. Our evaluation of fuel cask handling is provided in Section 9.1.4 of this Safety Evaluation Report. We conclude that the design for the spent fuel storage facilities is in conformance with the requirements of Criteria 61 and 62 of the General Design Criteria and the positions of Regulatory Guides 1.13 and 1.29, including the positions on seismic design and missile protection, and, therefore, is acceptable.

9.1.3 Spent Fuel Pool Cooling and Purification System

The spent fuel pool cooling and purification system is designed to maintain the water quality and clarity of the spent fuel pool water and to remove the decay heat generated by the spent fuel assemblies stored in the fuel pool.

The fuel pool cooling system is designed to Quality Group C and seismic Category I requirements. It consists of two 100-percent-capacity trains. Each train includes a fuel pool cooling pump and a fuel pool heat exchanger. The fuel pool cooling pumps are powered from the Class 1E electrical system. The safety-related component cooling water system provides cooling water to the fuel pool heat exchangers. In order that the pool cannot be inadvertently drained to uncover the stored fuel, piping entering and exiting the pool is located between the normal water level and design low water level and in addition is provided with anti-syphoning holes.

Each spent fuel cooling train is capable of maintaining the pool water temperature below 137 degrees Fahrenheit with a heat load based upon decay heat generation from one-third of a core that has been irradiated for 24,000 effective full-power hours and cooled for six days plus nine 1/3 cores cooled for more than one year. This "normal" heat load temperature is below our acceptance criterion of 140 degrees Fahrenheit. The fuel pool water temperature can be maintained below 150 degrees Fahrenheit with both of the fuel pool cooling trains in service when the pool contains one full core load placed in the pool six days after shutdown plus one-third of a core stored in the pool for 106 days plus nine 1/3 cores stored in the pool for more than one year. This "abnormal" heat load temperature is below our acceptance criterion. The spent fuel pool water temperature can also be maintained within acceptable limits with both fuel pool cooling trains in service in the maximum storage condition when a full core is placed in the pool six days after shutdown and the remaining ten 1/3 core storage spaces are filled with spent fuel that has undergone irradiation and storage in accordance with normally expected refueling cycles.

Normal makeup water to the pool is supplied from the nonsafety grade demineralized water storage tank. Emergency makeup water to the spent fuel pool is supplied from the seismic Category I refueling water storage tank or from the seismic Category I reactor makeup water storage tank. The spent fuel pool cooling system is located within the auxiliary building which is designed against the effects of missiles.

The fuel pool purification system is a nonsafety-related system and is designed to non-seismic requirements. The system pumps, piping, and valves are connected to the spent fuel pool cooling system, but the cross-connections have redundant valves for isolation. The failure of this system does not have an adverse effect on any safety-related equipment.

Based on our review, we conclude that the design of the spent fuel pool cooling and cleanup system is in conformance with Branch Technical Position ASB 9-2 with respect to decay heat loads, the guidelines of Regulatory Guides 1.13 and 1.29 including the positions on availability of assured makeup sources, the seismic design and missile protection, and the requirements of Criteria 61 and 62 of the General Design Criteria. We, therefore, find the fuel pool cooling and cleanup system acceptable.

9.1.4 Fuel Handling System

The fuel handling system, in conjunction with the fuel storage area, provides the means of transporting, handling, and storing of fuel. The fuel handling system consists of equipment necessary for the safe handling of the spent fuel cask and for safe disassembly, handling, and reassembly of the reactor vessel head and internals during refueling operations. The system also includes additional equipment designed to facilitate the periodic refueling of the reactor.

The 125-ton overhead traveling bridge crane in the fuel handling building is used for the handling of the spent fuel shipping cask. This crane is designed to seismic Category I requirements and is equipped with two independent and automatically operated brake systems.

The spent fuel cask loading pit is located approximately 15 feet from the pool, separated from the fuel pool by reinforced concrete walls and a transfer canal which can be isolated from the pool. Mechanical stops on the fuel handling building crane rails prevent the crane from moving over the spent fuel pool. There is no safety-related equipment along the path of travel of the cask; therefore, damage to the safety equipment or radioactive releases are precluded in the event of a cask drop accident.

Based on our review, we conclude that the fuel handling system can adequately perform its intended functions. Further, based on the design of the fuel handling building crane, we conclude that handling of spent fuel and the consequences of a cask drop will not impair safe shutdown capability nor result in unacceptable damage to the spent fuel storage facility and is therefore acceptable.

9.2 Water Systems

9.2.1 Service Water System

The service water system supplies cooling water to the plant from the service water pond, which is the ultimate heat sink discussed in Section 9.2.3 of this Safety Evaluation Report. The service water system provides cooling for the emergency diesel generators, component cooling heat exchangers, and heating ventilating and air conditioning mechanical water chiller condensers during all plant operating conditions. The system also cools the reactor building cooling units under abnormal conditions when the normal non-essential cooling supply is not available. The service water system is also a backup water source for the emergency feedwater system and component cooling water system makeup The system consists of two independent 100-percent-capacity loops each with a 100-percent-capacity service water pump, service water heat exchanger, and service water booster pump. A third 100-percent-capacity service water pump can be manually aligned to either service water loop. The service water booster pumps automatically start on a safety injection signal. This signal also simultaneously isolates the normal supply provided by the non-essential cooling water system and aligns the service water system to provide the safety cooling function normally performed by the non-essential system. One of the two redundant loops and its corresponding booster pump and any one of the three service water pumps is capable of providing the required cooling water flow after a postulated design basis accident, thus assuring adequate water supply in the event of a single failure of a system component.

Essential portions of the service water system are designed to Quality Group C, seismic Category I requirements, and are protected to withstand adverse environmental occurrences, such as tornadoes and floods. Each train is powered from a separate essential alternating current power bus. The service water pond, in conjunction with the service water system intake and discharge structures, serves as the ultimate heat sink for the service water system and is discussed in Section 9.2.3 of this Safety Evaluation Report.

The service water system operates during normal opertion; therefore, it does not require additional periodic tests and inspections of the system safety functions. The service water system booster pumps will operate only during emergency conditions. Their availability is assured by periodic tests and inspections as delineated in the facility Technical Specifications. In addition, the applicant has committed to an inservice inspection program and further tests as delineated in the facility Technical Specifications.

Based on our review, we conclude that the service water system design is in conformance with the requirements of Criterion 44 of the General Design Criteria regarding the ability to transfer heat from safety-related components to the ultimate heat sink and regarding the single-failure criterion. It is also in conformance with requirements of Criteria 45 and 46 of the General Design Criteria regarding the system design for periodic tests and inspections, including functional testing and confirmation of heat transfer capabilities. We, therefore, conclude that the system is acceptable.

9.2.2 Component Cooling Water System

The component cooling water system provides an intermediate closed cooling loop for removing heat from reactor auxiliary systems and transferring it to the service water system. It consists of two independent closed-loop flow paths for safety-related systems and a common supply to non-essential systems. Each loop consists of a 100-percent-capacity component cooling water pump, component cooling heat exchanger, and non-essential component cooling water booster pump.

One of the two redundant flow paths is required during a design basis accident to meet the minimum engineered safety feature requirements. A third-full capacity pump is provided and may be manually aligned to either of the independent loops should one of the other pumps fail. These provisions assure adequate water supply in the event of a single failure of a system component.

Essential portions of the system are designed to Quality Group C, seismic Category I requirements, and are protected to withstand adverse environmental occurrences, such as tornadoes and floods. Each essential train is powered from a separate essential alternating current power bus. Since non-essential portions of the system including the supply to the reactor coolant pumps are also designed to Quality Group C, siesmic Category I requirements, double isolation valves are not required, and manual isolation is acceptable.

The component cooling water system operates durng normal operation; therefore, it does not require additional periodic tests and inspections of the systems safety functions. In addition, the applicant has committed to an inservice inspection program and further tests as delineated in the facility Technical Specifications.

Based on our review, we conclude that the component cooling water system design is in conformance with the requirements of Criterion 44 of the General Design Criteria regarding the ability to transfer heat from safety-related components to the ultimate heat sink and the single-failure criterion. It is also in conformance with the requirements of Criteria 45 and 46 of the General Design Criteria regarding the system design for periodic tests and inspections, including functional testing and confirmation of heat transfer capabilities. We, therefore, conclude that the system is acceptable.

9.2.3 Ultimate Heat Sink

The ultimate heat sink provides cooling water to the service water system during all modes of plant operation including loss of offsite power or safe shutdown of the plant. The ultimate heat sink consists of a seismic Category I impoundment created by dams within the Monticello Reservoir, and the seismic Category I service water system intake structure, service water pump house, where the service water pumps are located, and service water system discharge structure. The service water pumps take suction from the ultimate heat sink of the intake structure. Our evaluation of the service water system is discussed in Section 9.2.1 of this Safety Evaluation Report.

At our request, the applicant demonstrated by analysis that the ultimate heat sink has the capability to provide adequate water inventory and provide sufficient heat dissipation to keep operating temperatures of system components within acceptable design ranges. Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," was used to establish the heat input due to fission produce decay and heavy element decay. In addition, the applicant has stated, and we agree, that the ultimate heat sink design meets the guidelines of Regulatory Guide 1.27 regarding the capability of the system to provide sufficient cooling for 30 days following accident conditions.

Based on our review, we conclude that the design of the ultimate heat sink meets the positions of Branch Technical Position ASB 9-2 and Regulatory Guide 1.27 and, therefore, is acceptable.

9.2.4 Condensate Storage Facility

The condensate storage facility consists of a 500,000-gallon seismic Category I storage tank located outdoors adjacent to the turbine building. Of the condensate storage capacity, 150,000 gallons is reserved for use by the emergency feedwater system.

This reserve capacity is maintained by having all non-safety-grade connections to the tank above the 150,000-gallon level. All connections below this level are seismic Category I including the emergency feedwater supply line. This tank serves as a reliable primary water source for the emergency feedwater system.

Redundant level indicators are provided at the tank and in the control room, and low level alarms are provided in the control room to assure that the 150,000-gallon reserve capacity is maintained. The non-safety-grade demineralized water storage tank provides makeup water to the seismic Category I condensate storage tank.

Based on our review, we conclude that the condensate storage facility is designed to meet its intended safety function and is therefore acceptable.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

The compressed air system provides both instrument air and service air for the plant. The main system consists of two identical 100-percent capacity oil lubricated air compressors which supply one air receiver. The instrument air and service air subsystems supply headers are taken separately from this common air receiver. Instrument air is filtered and dried before going to the intended services. The reactor building has an independent instrument air system. It consists of two identical 100-percent-capacity non-lubricated air compressors and receivers and supplies filtered and dried air to instruments and valves inside the reactor building. A backup supply of air to the reactor building is provided by a connection with the instrument air subsystem. On low air pressure in the reactor building instrument air system, the valve on the interconnecting line will automatically open to provide air from the instrument air system to the reactor building.

The compressed air system is classified as non-safety related except the portions that penetrate containment walls, including isolation valves which are designed to Quality Group B and seismic Category I requirements. All air-operated valves in safety-related plant systems are designed to fail to the safe position upon loss of instrument air. A seismic Category I air volume tank is provided for the emergency feedwater flow control valves to assure their function in the event of loss of instrument air.

The design of the compressed air system is in accordance with Regulatory Guides 1.26 and 1.29 with regard to Quality Group and seismic Category classification of the safety-related portions of the system and provides the continued supply of air to safety-related components during anticipated plant operating conditions. We, therefore, conclude that the system is acceptable.

9.3.2 Process Sampling System

The process sampling system is designed to provide representative samples of radioactive, as well as nonradioactive, fluid streams from systems throughout the plant for chemical and radiochemical analysis. The seismic design and quality group classification of sampling lines and components conform to the classification of the system to which each sampling line and component is connected, as described in Regulatory Guide 1.26 and 1.29. The process sampling system consists of piping, fittings, isolation and throttling valves, sample coolers, sample flasks, hooded sample sink, gaseous sample pump, and instrumentation. The process sampling system is designed to obtain representative samples from the systems and components listed in Table 9-1 of this Safety Evaluation Report.

Our review included the provisions proposed to sample all principal fluid process streams associated with plant operation and the applicant's proposed design. The review has included descriptive information for the process sampling system and the location of sampling points, as shown on piping and instrumentation diagrams.

The basis for acceptance in our review has been conformance of the applicant's design for the process sampling system to applicable regulatory guides, as well as to industry standards. Based upon our evaluation, we find the proposed system acceptable.

9.3.3 Equipment and Floor Drainage System

The equipment and floor drainage system accommodates drains from potentially radioactive sources and non-potentially radioactive sources through separate subsystems. The system is designed to prevent backflooding of safety-related areas by providing adequately sized drains and/or sumps. The radioactive sumps and drains systems collect potentially radioactive liquid waste from

TABLE 9-1

PROCESS SAMPLING SYSTEM SAMPLE POINTS

Points of Extraction	Number of Points Sampled
Residual heat removal loops	2
Pressurizer steam space	1
Pressurizer liquid space	1
Reactor coolant loop B (hot leg)	1
Reactor coolant loop C (hot leg)	1
Steam generator A, Secondary Water	2 (shell and blowdown)
Steam generator B, Secondary Water	2 (shell and blowdown)
Steam generator C, Secondary Water	2 (shell and blowdown)
Accumulators A, B and C	3 (one each accumulator)
Chemical and volume control system	
Downstream of letdown heat exchanger	1
Downstream of mixed bed demineralizer No. 1	1
Downstream of mixed bed demineralizer No. 2	1
Downstream of demineralizers and upstream of reactor coolant filter	1
Volume control tank, gas space	1
Reactor coolant drain tank	1
Thermal regeneration demineralizer outlet	1

equipment and floor drainage including waste resulting from piping or tank ruptures in the reactor building, auxiliary building, intermediate building, fuel handling building, penetration access area, and hot laboratories in the control building. These drains are discharged to the liquid radwaste system. The liquid radwaste system is discussed in Section 11 of this Safety Evaluation Report. Drains from non-potentially radioactive sources, such as the turbine building are discharged to the industrial and sanitary waste treatment system. Floor drain sumps in the engineered safety features equipment and piping areas are provided with alarms which will annunciate in the control room should flow into them exceed the expected flow rate. Engineered safety features equipment room sumps are equipped with automatic redundant full-capacity sump pumps. Additional flood protection is provided in the intermediate building to protect the emergency feedwater and component cooling water pumps by sump high level alarms which automatically close the main feedwater isolation valves and thus eliminate one of the potential causes of flooding in this area.

The equipment and floor drainage system is classified as non-safety related except for the reactor building sump discharge line containment penetration and isolation valves which are seismic Category I and Quality Group B. Adequate protection against flooding is provided by the measures discussed above to justify the non-safety classification of the equipment and floor drainage system.

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

9.3.4 Chemical and Volume Control System

The chemical and volume control system is designed to control and maintain reactor coolant inventory and to control the boron concentration in the reactor coolant through the process of makeup and letdown. The system purifies the primary coolant by demineralization.

An essential portion of the system consists of three centrifugal charging pumps. These pumps are used during normal operation and also for high pressure safety injection when the emergency core cooling system is required to function. This latter function is evaluated in Section 6 of this Safety Evaluation Report.

Boric acid at approximately four percent by weight is used for chemical reactivity control. The boric acid will be stored in two boric acid tanks. When dilution of the core is required, a preset amount is added to the system from the reactor makeup water tank. To avoid high water inventory in the primary system, coolant is bled off to the boron recycle system as necessary.

The boric acid solution is made up in a batching tank and transferred to one of two boric acid tanks. The combined boric acid tank capacity is sufficient to allow a cold shutdown from full-power operation immediately following refueling with the most reactive control rod not inserted. As a backup, borated water from the refueling water storage tank can be used. All portions of the chemical and volume control system that contain concentrated boric acid (four weight percent) are either located in heated rooms or in some cases are heat traced in order to maintain the solution temperature high enough to prevent precipitation.

Control of the coolant inventory and maintenance of proper water chemistry is achieved by a continuous feed and bleed process during which the feed rate will be automatically modulated by the pressurizer level. The letdown flow from the reactor coolant system is reduced in pressure, cooled in heat exchangers and processed through one of two mixed-bed demineralizers. If the inlet fluid temperature exceeds 104 degrees Fahrenheit, the flow bypasses the demineralizers to protect the resin bed. From the demineralizers, the flow is routed to the volume control tank where hydrogen is added to inhibit formation of oxygen in the coolant. From there the letdown flow is charged back into the reactor coolant system by the charging pumps. If the coolant inventory needs to be reduced, part or all of the letdown flow is routed to the boron recycle system.

Other chemicals that are added to the primary coolant via the chemical and volume control system are hydrazine to scavenge oxygen during startup and lithium hydroxide for pH control.

The boron thermal regeneration subsystem is designed to control the changes in reactor coolant boron concentration to compensate for xenon transients during load following operations without adding makeup for either boration or dilution. Storage and release of boron is controlled by the temperature of the fluid entering the thermal regeneration demineralizers.

The chemical and volume control system also supplies seal water injection flow for reactor coolant pump seal cooling and collects the controlled leakoff from the reactor coolant pump seals. In addition, it provides a means of filling, draining, and pressure testing of the reactor coolant system. The portions of the system required for safe shutdown of the reactor are designed to meet the seismic Category I requirements, the single failure criteria, and are powered from essential buses.

Based on our review of the chemical and volume control system and the requirements for system performance of necessary functions during normal, abnormal, and accident conditions, we conclude that the design of the chemical and volume control system and supporting systems is in conformance with the NRC's regulations as set forth in Criteria 2, 4, and 33 of the General Design Criteria and meets the guidelines of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Regulatory Guide 1.29, "Seismic Design Classification," and, therefore, is acceptable.

9.4 Heating, Ventilation, and Air Conditioning

9.4.1 Control Building Area Ventilation System

The safety-related portions of the control building area ventilation system consists of the control room system and relay room system. Each of the above systems is designed to seismic Category I requirements and is powered from the Class 1E essential emergency power supplies. Each system consists of two redundant 100-percent-capacity air handling units each with separate, independent, and redundant supply, return, relief, and outside air ducts and filtering subsystems. These provisions assure adequate air handling capability in the event of a single failure of a system component. The outside air intakes are not subject to tornado missiles.

The control room system is designed to maintain the control room within the environmental limits required for operation of plant controls and uninterrupted safe occupancy of required manned areas during all operational modes including the design basis accident conditions. The system is designed to maintain the control room under positive pressure.

If the radiation level in the control room rises above set limits, a high radiation signal automatically closes the outside air dampers, places the system in the recirculation mode, and starts the normal air handling unit and emergency filter fan which passes control room air through the emergency filters. The safety injection signal will also initiate this sequence of events. In addition, control room isolation and system operation in the emergency recirculation mode can be manually initiated in the control room.

The relay room system is designed to maintain the relay room environmental conditions during all modes of operation. Upon receipt of a high radiation or safety injection signal, the outside air dampers are automatically positioned for the recirculation mode and the air handling units are started.

Indication of smoke or high temperature in the control room or relay room supply ducts or emergency filter system discharge duct will actuate alarms in the control room.

We have reviewed the design of the control building area ventilation system and conclude that it meets the requirements set forth in Criterion 19 of the General Design Criteria with regard to the capability to operate the plant from the control room during normal and accident conditions, that it meets the single-failure criterion and, therefore, is acceptable.

9.4.2 Auxiliary and Radwaste Area Ventilation System

The safety-related portions of the auxiliary and radwaste area ventilation system consist of the auxiliary building pump room and motor control center and switchgear areas cooling systems. The systems are designed to maintain a suitable room air temperature for personnel and equipment and to minimize the radioactive release to the atmosphere by recycling air within the plant. The systems consist of a total of seven full capacity air handling and cooling units, three serving the three charging/safety injection pump rooms, two serving the two residual heat removal spray pump rooms and two serving the two safety-related motor control centers and switchgear areas. This design assures that at least one of the essential room cooling systems is available considering a single failure. The systems are seismic Category I and the cooling units are powered from separate Class 1E essential emergency power supplies. The essential pump room cooling units are automatically started when their corresponding pump is energized and the motor control center and switchgear area cooling units start on receipt of a safety injection or loss of offsite power signal. The cooling units are separated and are protected from tornado missiles.

Based on our review of the design of the auxiliary and radwaste area ventilation system, we conclude that it meets the single-failure criterion and the guidelines of Regulatory Guide 1.29 and is, therefore, acceptable.

9.4.3 Fuel Handling Building Ventilation System

The function of the fuel handling building ventilation system is to maintain a suitable environment for equipment operation and to limit potential radioactive release to the atmosphere during normal operation and postulated fuel handling accident conditions. The non-safety auxiliary building main supply system provides the normal fuel handling building air supply and consists of one full capacity air handling unit.

If radiation levels in the fuel handling building rise above set limits due to a fuel handling accident or other cause, a radiation monitor in the exhaust duct will send an alarm to the control room.

Based on our review of the design of the fuel handling building ventilation system, we conclude that it meets the single failure criterion and the guidelines of Regulatory Guide 1.29 and, therefore, is acceptable.

9.4.4 Intermediate Building Ventilation Systems

The safety-related portions of the intermediate building ventilation system consist of the engineered safety features switchgear rooms and "speed" switch/evacuation panel rooms cooling systems, battery room system, and intermediate building pump room cooling systems. The systems are designed to maintain a suitable room air temperature for personnel and equipment. Each system is designed to seismic Category I requirements and is powered from the Class 1E essential emergency power supplies. The systems are protected from tornado missiles.

Each of the two engineered safety feature switchgear rooms is served by one full-capacity air handling unit. All three "speed" switch rooms and the two evacuation panel rooms are served by two full-capacity air handling units. These provisions assure that at least one of the essential room cooling systems is available in the event of a single failure. These air handling units are started by the safety injection or loss of offsite power signals.

The ventilation system for the battery rooms consists of two independent redundant full-capacity supply air handling units and two full-capacity redundant exhaust fans. Each supply and exhaust system serves all three battery rooms and battery charger rooms. This assures that adequate air is provided in the event of a single failure of a system component. The normal supply air is drawn from a common outside air intake structure. Additional free air is returned to the air handling units from the battery charger rooms. Both air handling units and exhaust fans operate continuously during all normal, shutdown, and emergency conditions to prevent the accumulation of battery gases as well as maintaining suitable room ambient temperatures. Trip of a safety-related battery room air handling unit fan motor or exhaust fan motor causes an alarm in the control room.

The intermediate building pump room cooling systems consist of two full-capacity air handling units for the two service water booster pump areas and two full-capacity air handling units for the three emergency feedwater pump areas. One air handling unit serves each essential motor-driven pump. The turbine-driven emergency feedwater pump area is served by both emergency feedwater pump area air handling units. This assures that adequate cooling air is available in the event of a single failure of a system component. These air handling units are automatically started when their respective pump is energized.

Based on our review of the design of the intermediate building ventilation systems, we conclude that they meet the single-failure criterion and the guidelines of Regulatory Guide 1.29 and are therefore acceptable.

9.4.5 Miscellaneous Building Ventilation and Cooling Systems

The safety-related portions of the miscellaneous building ventilation and cooling systems consist of the diesel generator building ventilation system, the service water pumphouse ventilation system, and the chilled water system.

The diesel generator building ventilation system provides outside air to maintain the diesel generator room, diesel generator electric equipment room, and diesel generator cable-pipe-basement area temperatures during emergency conditions while the diesels are operating. The diesel generator building ventilation system is seismic Category I and consists of two redundant trains. Each train serves one of the two diesel generator rooms and includes two 50-percent-capacity supply fans, thus assuring adequate air in the event of a single failure of a system component. System air is relieved through roof vents to the outside. The system is automatically placed in operation upon receipt of a corresponding diesel engine start signal. The system may also be started and stopped manually from the control room. The fans are connected to the Class 1E emergency bus supplied by their respective diesel generator. The system, including the air intakes and relief air roof vent openings, is protected from tornado missiles.

The service water pumphouse ventilation system maintains the safety-related service water system pump/screen room and switchgear rooms temperatures to permit continuous operation of the pumps. The service water pumphouse ventilation system is seismic Category I and consists of two 100 percent capacity supply fans, thus assuring adequate air in the event of a single failure of a system component. The system is provided with a common supply and return header and individual outside air intakes and relief air vents. Either fan operates continuously during operation, and both fans are automatically placed in operation on receipt of a safety injection or loss of offsite power signal. The system may also be started and stopped manually from the control room. The fans are connected to separate Class 1E emergency power supplies. The system is located in a seismic Category I structure and protected from tornado missiles.

The chilled water system provides 45-degree Fahrenheit chilled water to various safety-related and non-safety-related area air handling unit cooling coils and pump coolers. It consists of three 100 percent capacity water chillers and three 100 percent capacity chilled water pumps. The pumps and chillers are connected to two separate and redundant chilled water trains. These provisions assure an adequate chilled water supply in the event of a single failure. The safety-related portions of the system including the pumps and chillers are seismic Category I and are isolated from non-safety-related portions by either two check valves in series or by fail closed double isolation valves which close on receipt of a safety injection signal. Seismic Category I service water provides cooling water to remove heat from the condensers or the chillers. One of the three pumps and chillers operates continuously during normal operation, and all chillers and pumps are automatically placed in operation on receipt of a safety injection or loss of offsite power signal. The system may also be started and stopped manually from the control room. The chillers, pumps, and individual chilled water train isolation valves are connected to separate Class 1E emergency power supplies. One pump and its corresponding chiller is a swing pump and is connected to both emergency power busses. The system is located in a seismic Category I structure and is protected from tornado missiles.

Based on our review of the design of the miscellaneous building ventilation and cooling systems, we conclude that they meet the single failure criterion and the guidelines of Regulatory Guide 1.29 and therefore are acceptable.

The safety-related portions of the chilled water system are in conformance with Criterion 44 of the General Design Criteria regarding the ability to transfer heat from safety-related cooling units, and therefore, the chilled water system is acceptable.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

We have reviewed the Virgil C. Summer Nuclear Station, Unit 1 Fire Protection Evaluation - Fire Hazards Analysis Report submitted by the applicant by letter dated August 19, 1977, including Revisions 1 through 5. The reevaluation was in response to our request to evaluate the fire protection program against the guidelines of Appendix A to Branch Technical Position (BTP) ASB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." As part of our review, we visited the facility to examine the relationship of safety-related components, systems, and structures in specific plant areas to both combustible materials and to associated fire detection and suppression systems. The overall objective of our review was to assure that in the event of a fire at the facility, personnel and the plant equipment would be adequate to safely shut down the reactor, to maintain the facility in a safe shutdown condition, and to minimize the release of radioactivity to the environment.

Our review included an evaluation of the automatic and manually operated water and gas fire suppression systems, the fire detection systems, fire barriers, fire doors and dampers, fire protection administrative controls, fire brigade size and training, and the Technical Specifications related to fire protection.

We have concluded that the fire protection program of the facility with the proposed improvements, is adequate and meets Criterion 3 of the General Design Criteria. We consider that the fire detection and suppression systems, the barriers between fire areas, administrative procedures for control of combustibles and ignition sources, and the trained onsite fire brigade with the capability to extinguish fires manually will provide adequate protection against a fire. Our consultants, Gage-Babcock and Associates, Inc., participated in the review of the fire protection program and in the preparation of this section of the Safety Evaluation Report, and concur with our findings.

Fire Protection Systems Description and Evaluation

The water supply system consists of two fire pumps separately connected to a buried, 12-inch pipe loop around the facility. The fire pumps are rated at 2500 gallons per minute at 125 pounds per square inch, gauge head; one is motor driven and the other is diesel engine driven. The water supply source is the Monticello Reservoir. Two water intakes are provided for each fire pump. At our request, valves will be installed on all fire hydrant laterals to facilitate hydrant maintenance without requiring that part of the fire protection loop be closed.

A separate 20 gallons per minute pressure maintenance pump (jockey pump) maintains the system pressure at 110 pounds per square inch. If the water supply system pressure falls to 95 pounds per square inch then the motor driven fire pump starts. The diesel pump automatically actuates if the header pressure falls to 85 pounds per square inch. The fire pumps are located in the circulating water pumphouse and are separated by a three-hour barrier. Separate alarms are provided in the control room to monitor pump operation, prime mover availability, or failure of a fire pump to start. A low header pressure alarm also sounds in the control room. The power supply associated with the control signal which automatically starts the fire pumps is supplied by the Class 1E electrical system. Both the fire pumps and their controllers are Underwriters' Laboratories listed.

The fire suppression system requiring the greatest water demand for areas containing or exposing safety related equipment or circuits is the cable spreading room sprinkler system. This water flow requirement is 1,250 gallons per minute and, coupled with 500 gallons per minute for hose streams and 1,000 gallons per minute for the diesel generator cooling systems creates a total water demand of 2,750 gallons per minute. Since the system can deliver 2,750 gallons per minute with a single fire pump, the water supply system is adequate and is, therefore, acceptable.

The automatic/manual sprinkler systems and the hose stations are connected to interior water supply headers. The interior water supply system is fed from two separate connections to the underground supply loop with appropriate valves to perform maintenance or to prevent a single break from impairing the entire distribution system. The water supply valves to the sprinklers are electrically supervised with alarms in the control room. All other fire protection valves controlled by a key locking procedure. Also, actuation of any water fire suppression system will cause a fire pump to start on a low header pressure signal. The low pressure alarm and a pump running signal indicate in the control room. In addition, the automatic sprinkler systems have water flow alarms which indicate in the control room. The automatic sprinkler systems, e.g., wet pipe sprinkler system, preaction sprinkler systems deluge, and water spray systems, are, or will be, designed to the requirements of National Fire Protection Association (NFPA) Standard No. 13, "Standard for Installation of Sprinkler Systems," and NFPA Standard No. 15, "Standard for Water Spray Fixed Systems."

The areas that have been or will be equipped with water suppression systems include the following:

Portions of the turbine building
Control building - all floor areas except control room and elevation 482 feet
Diesel generator cubicles
Diesel fire pump room
Drumming station* and compactor area (manual)
Auxiliary building*

- Zone 1, Area near residual heat removal system spray pump room cooling unit A
- Zone 2, charging pump room A
- Zone 3, recirculation valve room
- Zone 4, charging pump room cooling units room
- Zone 5, northeast general floor area near open ceiling hatch
- Zone 6, truck bay
  backup heater transformer area
  hallway, south end
- Zone 9, hallway, south end

Reactor building purge exhaust system charcoal filter plenum (manual)
Reactor building charcoal cleanup system filter plenum (manual)
Auxiliary building gas treatment filters
Auxiliary building charcoal exhaust system filter plenum (manual)
Fuel handling building charcoal exhaust system filter plenum (manual)
Control room emergency filter plenum (manual)

Intermediate building*

- Fire area IB-9, chilled water pump room
- Zone 5, general floor area
- Fire area IB-10, battery room ventilation equipment room
- Zone 7 and fire area IB-11, service water booster pumps area cooling equipment rooms A and B
- Fire area IB-15A, switchgear room
- Zone 10, east penetration access area
- Zone 12, general floor area, areas near redundant cable interaction
- Fire areas IB-16 and IB-17, switchgear cooling unit room
- Fire area IB-19, speed switch room cooling unit room B

Service water pump house*

- Zone 2, operating floor
- Fire area SWPH-4, ventilation equipment room

Manual hose stations will be located throughout the facility to assure that an effective hose stream can be directed to any safety related area in the facility. Additional hose will be added to present standpipe hose stations or additional standpipe hose stations will be installed as follows:

---

*Sprinkler system installed at our request

Control building

All hose station hose lengths will be increased to 100 feet.
One additional hose station will be installed at the west end of elevation 463 feet.

Auxiliary building

Zone 9 - all hose station hose lengths will be increased to 100 feet.

Service water pump house

At least one standpipe hose station with 100 feet of hose will be provided.

These systems are or will be consistent with the requirements of NFPA Standard No. 14, "Standpipe and Hose Systems" for sizing, spacing, and pipe support requirements.

We have reviewed the design criteria and the bases for the water suppression systems. We conclude that these systems, with the changes indicated, meet the guidelines of Appendix A to Branch Technical Position ASB 9.5-1 and are, therefore, acceptable.

Total flooding carbon dioxide systems are provided for the relay room and the computer rooms and they are actuated by heat detection systems. The carbon dioxide systems are designed to achieve after a 30-second delay, a 50 percent concentration.

We have reviewed the design criteria and bases for the carbon dioxide fire suppression systems. We conclude that these systems satisfy the provisions of Appendix A to Branch Technical Position ASB 9.5-1 and are, therefore, acceptable.

The fire detection systems consist of the detectors, associated electrical power supplies, and the annunciation panels. The types of detectors used are ionization (products of combustion) and thermal (heat sensors). Fire detection systems give an audible and visual alarm which annunciates in the plant control room. Local audible and/or visual alarms are also provided. The fire detection systems are connected to the emergency power supply. Fire detection systems will be installed in all areas having safety related equipment. This includes the control room area, the new and spent pool storage areas, and areas of cable concentration.

The fire detection systems will be installed according to NFPA Standard 72D, "Standard for the Installation, Maintenance, and Use of Proprietary Protection Signalling Systems." Those fire detection systems which are used to actuate suppression systems will be upgraded to a Class A system defined in NFPA Standard 72D.

We have reviewed the fire detection systems to ensure that fire detectors are adequate to provide detection and alarm of fires that could occur. These systems will be installed with due consideration for the use of detector spacings less than those recommended for smooth, unobstructed ceilings. We have also reviewed the fire detection system's design criteria to ensure that they conform to the applicable sections of NFPA Standard 72D. We conclude that the design and the installation of the fire detection systems meet the guidelines of Appendix A to Branch Technical Position ASB 9.5-1 and are, therefore, acceptable.

Other Items Related To the Station Fire Protection Program

All concrete floors, walls and ceilings enclosing fire areas that contain safe shutdown equipment are fire rated at a minimum of three hours. Where required, structural steel framing will be provided a three-hour fire rating by the use of spray on fire proofing. Such areas include Zone 3 (east penetration access area) and Zone 5 (general floor area, including the mezzanine above the emergency feedwater pumps) of the intermediate building. Floor and ceiling openings in the service water booster pump area cooling equipment room A and between the switchgear cooling unit rooms A and B will be provided with three-hour fire rated closures.

In the control building and the service water pump house, the applicant has used a drywall-on-steel-stud construction as a claimed three-hour fire-rated barrier separating rooms in safety-related areas. The steel studs are seismically designed and faced on both sides with three layers of 5/8 inch type-X gypsum. The applicant will provide test data to verify three-hour construction or will upgrade the walls to three-hour rated construction.

The applicant has verified to our satisfaction that the fire rating of the fire stop seals used in the penetrations for cable trays, conduits and piping is three hours.

Based on our review and the applicant's commitment, we conclude that the fire barriers and barrier penetrations are, or will be, in accordance with the guidelines of Appendix A to Branch Technical Position ASB 9.5-1 and are, therefore, acceptable.

We have reviewed the placement of fire doors and dampers to assure proper fire rating has been provided.

Fire doors are kept in a closed position and are controlled by administrative procedures, which is acceptable to us. Fire doors carry a Underwriters' Laboratories label except for certain pressure and bullet resistant doors. The pressure and bullet doors are acceptable to us in the areas in which they are used.

The applicant has provided three-hour ventilation fire dampers for most of the three-hour wall ceiling/floor assemblies. Certain locations have two 1-1/2-hour fire dampers. These cases were analyzed and found acceptable where the fire load was small and the estimated fire duration was well below the damper rating. In other areas, three-hour rated dampers will be provided.

Based on our review, we conclude that the fire doors and dampers provided are, or will be, in accordance with the guidelines of Appendix A to Branch Technical Position ASB 9.5-1 and are, therefore, acceptable.

Alternate Shutdown

At our request, the applicant performed a detailed fire hazards analysis of each plant area. This fire hazards analysis included consideration of the potential effects of a transient exposure fire on equipment and cables required for safe shutdown. The results of the detailed fire hazards analysis are presented in Revision 5. An alternate shutdown system was proposed for the control, cable spreading, relay rooms, and the basement and the intermediate cable chases. Two independent shutdown panels are provided and are located in the intermediate building (elevation 436) and are separated by a fire wall. A fire in either the control room or spreading rooms would not jeopardize operation of the alternate shutdown panels nor would a fire in the panels cause malfunctions in the control room or the cable spreading room. In a like manner separation will be provided so that a fire in any cable chase in the control building will not affect both the control room and the alternate shutdown panels. Therefore, a single fire event in any of the above areas will not impair mutually redundant safe shutdown systems of Division I and II simultaneously.

We conclude that the installation of an alternate shutdown system prior to fuel loading will preclude the possibility of a single fire event in the control room, cable spreading room, relay rooms, and selected cable chases from impairing mutually redundant safe shutdown systems simultaneously. Therefore, we find the applicant's commitment to be acceptable. We will review the design of the alternate shutdown system and report the results of our evaluation in a supplement to this Safety Evaluation Report.

Fire Protection for Specific Areas

The control room fire area is separated from the balance of the plant by three-hour fire rated ceiling/floor assemblies, one three-hour fire rated wall and three drywall-on-steel-stud walls discussed previously. Support areas within the control room fire area, including offices, storage rooms, and a kitchen area, are separated from the control room by noncombustible partitions which extend from the floor to the suspended ceiling.

There is no automatic fire suppression for this area. Smoke detection will be provided for the entire control room fire area, in the ventilation system ducts, and in the main control board and any other cabinet which contains redundant safe shutdown circuits. Standpipe hose stations and portable extinguishers are provided for manual fire suppression activities.

The control room support areas will be separated from the control room by a one-hour fire rated wall which will extend from the floor slab to the ceiling slab, above the suspended ceiling, or an automatic sprinkler system will be provided in the support areas. Also, as discussed previously, the applicant has installed an emergency shutdown panel so that alternate shutdown capability exists independent of the control room. With the improvements indicated, the fire protection for the control room meets the guidelines of Appendix A to Branch Technical Position 9.5-1 and is, therefore, acceptable.

The upper and lower cable spreading rooms are separated from the balance of the plant by three-hour fire rated ceiling/floor assemblies, one three-hour wall and the drywall-on-steel-stud walls discussed previously. Access to these areas are from two separately located enclosed stairwells.

An automatic, pre-action sprinkler system, with water spray on individual cable trays, is installed in the upper and lower cable spreading rooms. These fire suppression systems serve as the primary fire extinguishing system. Additional backup is provided by standpipe systems with increased hose lengths as indicated previously and portable extinguishers. Portable fans are available for smoke venting. In addition, installed smoke detectors will initiate an early warning alarm in the control room.

We were initially concerned that a fire could affect redundant shutdown systems in either of the cable spreading rooms. However, as discussed previously, the applicant has installed an emergency shutdown panel so that alternate shutdown capability exists independent of the cable spreading rooms. The fire protection for both of the cable spreading rooms with the substantiation of the walls as three-hour rated barriers and the alternate shutdown capability, meets the guidelines of Appendix A to Branch Technical Position ASB 9.5-1 and is, therefore, acceptable.

The east and northeast cable chases in the control building contain varying amounts of cables in trays, including safety-related cable. The cable chases are separated from other areas of the facility by three-hour fire rated ceiling/floor assemblies, the three-hour fire rated exterior wall of the control building, and the drywall-on-steel-stud walls discussed previously. Access to these cable chases are through the personnel and laboratory area for the basement chases, through the relay room and future relay room for the intermediate chases, and through the control room and future control room for the main floor chases.

An automatic, pre-action sprinkler system, with water spray on individual cable trays, is installed in each cable chase. These fire suppression systems serve as the primary fire extinguishing system. Additional backup is provided by standpipe systems, with increased hose lengths as indicated previously, and portable extinguishers. Portable fans are available for smoke venting, and installed smoke detectors will alarm in the control room.

We were initially concerned that a fire could affect redundant shutdown system circuits in both basement chases and the intermediate chase. However, the applicant will install an alternate shutdown capability independent of any single cable chase. The fire protection of the cable chases, with the modifications indicated above, meets the guidelines of Appendix A to Branch Technical Position ASB 9.5-1 and is, therefore, acceptable.

One potential fire hazard inside containment is associated with the three reactor coolant pumps because of the potential for oil spraying under pressure. Each reactor coolant pump is located in its own cubicle. To prevent an oil fire, the applicant will provide an engineered oil containment and collection system in each reactor coolant pump cubicle. The system will confine a pressurized oil spray from a rupture at any point in the lube oil system. The oil collection system will be seismically supported.

Reactor protection channel instrumentation is separated into each quadrant and routed through different penetrations. The two residual heat removal system suction isolation valves are located on opposite sides from each other, thereby precluding a single fire damaging both valves. Any redundant cable division routed within 20 feet of each other will be protected by having one of the divisions enclosed in a 1-1/2 hour fire rated barrier, to be installed prior to fuel loading.

Reactor building fire protection features include: hose stations; smoke detectors; and fire extinguishers.

We have reviewed the applicant's fire hazards analysis for the areas inside the reactor building and conclude that the fire protection with the improvements indicated above, meets the guidelines of Appendix A to Branch Technical Position ASB 9.5-1 and is, therefore, acceptable.

The component cooling water pumps are located in the intermediate building. There are a total of three component cooling pumps. A cross-connection is provided between units. All three pumps are located in the same fire area.

At our request, the applicant has agreed to provide the following fire protection provisions to protect against an exposure fire:

1. Area smoke detection system,

2. A sprinkler system that will provide coverage on the pumps and extend at least 15 feet beyond each pump.

3. A one-hour fire rated cable barrier on one division if redundancy is compromised by separation less than 20 feet of clear open space (no combustibles), and

4. A 10 foot high radiant heat shield wall constructed of drywall between pumps B and C. Only one pump is needed for safe shutdown. Therefore, even assuming a loss of two component cooling pumps, cooling can be provided by using the single remaining pump.

We have reviewed the component cooling pump area and find the fire protection provisions, with the proposed modifications, are in accordance with Appendix A to Branch Technical Position ASB 9.5-1 and are, therefore, acceptable.

All three heating ventilation and air conditioning system chilled water pumps are located in one room. At our request, the applicant has agreed to provide the following fire protection features:

1. An automatic sprinkler system

2. Radiant shield walls of one-hour construction between all three pumps to divide the room into three areas

3. A fire detection system

4. A one-hour fire rated barrier for cable from one division which passes through the pump area for another division

5. Curbs between the pump areas

Only one pump is needed for safe shutdown. Therefore, even assuming a loss of two heating, ventilation and air conditioning system chilled water pumps, safe shutdown is assured by using the remaining pump. We have reviewed the heating, ventilation and air conditioning system chilled water pump area and find the fire protection provisions, with the proposed modifications, are in accordance with Appendix A to Branch Technical Position ASB 9.5-1 and are, therefore, acceptable.

A number of plant areas have physical arrangements wherein redundant divisions of cable/conduits and equipment are not separated by fire-rated barriers and, therefore, could be vulnerable to a single, transient fire event. Originally, the applicant was relying solely on administrative controls to preclude a fire event from taking place in affected areas. At our request, the applicant agreed, in Revision 4, to provide an automatic water suppression system in these areas and, in addition, to provide a one-hour fire rated barrier for one of the divisions where the redundant circuits or equipment are separated horizontally by less than 20 feet of clear, open space without intervening combustibles. The barrier will completely enclose one of the redundant divisions. Those areas that will have one-hour fire rated barriers in addition to the automatic water suppression system include the following:

Control building

Fire area CB-18, main floor chase, east
Fire area CB-20, main floor chase, northeast

Auxiliary building

Zone 1, subbasement
Zone 2, charging pump room A
Zone 3, recirculation valve room
Zone 4, charging pump room cooling units room
Zones 5 and 6, northeast area and truck bay, redundant cable on separate floors with open floor hatch
Zone 6, backup heater transformer area hallway, south end
Zone 9, hallway, south end

Intermediate building

Fire area IB-9, chilled water pump room
Zone 5, general floor area
Fire area IB-10, battery room ventilation equipment room
Zone 7 and fire area IB-11, service water booster pump area cooling equipment rooms A and B
Fire area IB-15A, switchgear room
Zone 10, east penetration access area
Zone 12, general floor area
Fire areas IB-16 and IB-17, switchgear cooling unit room
Fire areas IB-19, speed switch room cooling unit room

Service water pump house

Fire area SWPH-4, ventilation equipment room

Reactor building

We have reviewed the areas containing redundant divisions of equipment and cable and conclude that with the modifications, the fire protection meets Appendix A to ASB 9.5-1 and is, therefore, acceptable.

The applicant's fire hazards analysis addresses other plant modifications not specifically discussed in this Safety Evaluation report. The applicant is installing: additional eight-hour battery pack emergency lighting in all areas which could be manned for safe shutdown, including the control room, and in access and egress routes to all fire areas; portable extinguishers, to include permanent placement of extinguishers in the charging pump cooling units room, the switchgear cooling units rooms, the battery ventilation equipment rooms, inside containment, the service water pumphouse, and also water extinguishers in the control room, cable spreading room, and the electrical equipment rooms; hose stations; portable smoke blowers; smoke detectors for the control room area, control room ventilation plenums, and the new/spent fuel pool area; and fire rated fire stop materials for replacement of any combustible expansion joints. With these proposed modifications the affected areas will be in accordance with the guidelines of Appendix A to Branch Technical Position ASB 9.5-1, and are, therefore, acceptable.

Administrative Controls, Fire Brigade, and Technical Specifications

The administrative controls for fire protection consists of the fire protection organization, the fire brigade training, the controls over combustibles and ignition source, the prefire plans and procedures for fighting fires and quality assurance. In the fire protection program reevaluation, the applicant compared the adminisrative controls to our supplemental guidance "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," dated June 14, 1977. The applicant has to our satisfaction demonstrated that their administrative controls and training procedures meet our supplemental staff guidelines. The applicant will implement the administrative controls and procedures before fuel loading.

The applicant will have a five-man fire brigade, which meets our guidelines, and, therefore, is acceptable.

We conclude that the fire brigade equipment and training conform to the recommendations of the National Fire Protection Association, to Appendix A to Branch Technical Position ASB 9.5-1, and to our supplemental staff guidelines and, therefore, are acceptable.

The applicant has committed to follow our Standard Technical Specifications on fire protection.

Conclusion

We conclude that a fire occurring in any area of the Virgil C. Summer Nuclear Station, Unit 1 with all proposed modifications accomplished will not prevent the unit from being brought to a controlled safe cold shutdown. Further, such a fire would not cause the release of significant amounts of radiation.

We find that the fire protection program for the facility with the improvements and modifications to which the applicant has committed in the Final Safety Analysis Report to be implemented prior to fuel loading will meet the guidelines contained in Appendix A to Branch Technical Position ASB 9.5-1, meets Criterion 3 of the General Design Criteria and is, therefore, acceptable.

On May 23, 1980, the Commission issued a Memorandum and Order (CLI-80-21) which states that: "The combination of the guidance contained in Appendix A to BTP 9.5-1 and the requirements set forth in this rule define the essential elements for and acceptable fire protection program at nuclear power plants docketed for Construction Permits prior to July 1, 1976, for demonstration of compliance with General Design Criterion 3 of Appendix A to 10 CFR Part 50." On October 27, 1980, the Commission approved a rule concerning fire protection. The rule and its Appendix R were developed to establish the acceptable fire protection requirements necessary to resolve certain areas of concern in contest between the staff and licensees of plants operating prior to January 1, 1979.

Although this fire protection rule does not apply to Virgil C. Summer Nuclear Station, Unit 1, based on our review and evaluation and the applicants' commitments, we conclude that the protection program for this facility will meet the following three issues identified in Appendix R.

1. Section III.G., Fire Protection of Safe Shutdown Capability

2. Section III.J., Emergency Lighting

3. Section III.O., Oil Collection System for Reactor Coolant Pump.

The implementation schedule will be in accordance with the requirements of the rule.

Based on these commitments and our evaluation, we conclude that the Virgil C. Summer Nuclear Station, Unit 1 fire protection program will meet all the requirements of Appendix R to 10 CFR Part 50 when the committed modifications have been completed.

9.5.2 Communication Systems

The communication systems are designed to provide reliable intraplant and plant-to-offsite communications under both normal plant operation and accident conditions.

The communication systems provided for the facility include: a) page/party public address system, b) a maintenance communications system, c) a redundant paging system for selected areas, d) a private telephone system, and e) a plant-to-offsite communications system.

The page/party public address system consists of a system of speakers, permanent stations and portable handset locations throughout the plant. There are two subsystems to the main page/party system: a) the fuel handling page/party system and b) a system of page/party line access jacks for use with portable handset units. The public address system is powered from an uninterruptible battery backed 60 hertz static inverter power supply system vital bus and is capable of operation with noise levels up to the 110 to 115 decibels range.

The maintenance communications systems contains a network of plug-in jacks in selected areas of the plant and is completely isolated from all other communications systems. The system is powered from the non-Class 1E power systems bus. This system is normally used for the maintenance and calibration of equipment.

The redundant paging system consists of handsets, speakers, amplifiers and accessories similar to the page/party system. This system is completely independent of all other communication systems and is limited to areas where communications are required under emergency operating conditions. The power for the redundant paging system is from a Class 1E standby diesel generator bus.

The private telephone system consists of switchboard equipment, power supply, and commercial type telephone handsets installed throughout the station for local and offsite communications. Backup control systems are provided for critical switching and signaling equipment so that if one component fails, the entire system is not lost. The telephone system has a reliable power supply with battery backup power for use during loss of normal station power.

The plant to offsite communications system operates through the private telephone system in conjunction with the commercial telephone landline system, as well as through a microwave receiving and transmitting system. Both the landline and microwave systems operate through the private telephone system switchboard. The microwave system consists of solid state telemetry equipment owned and operated by the applicant. The telephone landlines are fed from offsite sources by the commercial telephone company.

The scope of review included assessment of the number and types of communications systems provided, assessment and adequacy of the power sources, and verification of functional capability of the communications system under all conditions of operation.

The basis for acceptance in the staff review was conformance of the design criteria and bases and design of the installed communication systems to the acceptance criteria of Section 9.5.2 of the Standard Review Plan. Other basis for acceptance was conformance to industry standards, and the ability of the systems to provide effective communications from diverse means within the facility during normal and emergency conditions under maximum potential noise levels.

Based on our review, we conclude that the installed communication systems at the plant conform to the above cited standards, criteria and design bases, they can perform their design functions and are therefore acceptable.

9.5.3 Lighting System

The lighting system for the facility is designed to provide adequate lighting in all areas of the station indoors and outdoors and consists of normal, essential and emergency lighting systems. The design is based on illumination levels that provide good light distribution with adequate light intensities based on several standards including the South Carolina Building Codes and the Handbook of the Illuminating Society of America.

The normal lighting subsystem used throughout the attended areas of the plant is 120-volt alternating current lighting which is supplied from the non-Class 1E alternating current distribution system. Illumination is from incandescent, fluorescent and high intensity discharge light sources. The normal lighting system for the outdoor areas, including the roadway, perimeter security, area and switchyard lighting, is supplied from the 480-volt single phase non-Class 1E alternating current power supply.

The essential lighting subsystem consists of those portions of the normal alternating current lighting system which are powered from the Class IE alternating current system. The essential lighting system operates in conjunction and supplements the normal lighting system where a more reliable source of illumination is required for critical tasks or for access or egress. Upon loss of the normal lighting, the essential lighting is provided for access in the turbine building, auxiliary building, diesel generator building water treatment building, intermediate building, and control building. Essential lighting is also provided with direct current power backup, for continuation of critical activities in the control room, relay room, computer room, critical electrical distribution areas of the control building, evacuation panel and emergency safety features switchgear rooms, and the diesel generator rooms.

The emergency lighting subsystem provides lighting throughout essential plant operation areas for operation of safety-related equipment, building egress, and safety of plant personnel in the event of loss of the normal and essential lighting systems. The emergency lighting subsystem is powered from the Class 1E 125-volt direct current power station batteries equipped with battery chargers and only becomes energized upon the loss of normal lighting in the reactor building or essential lighting to the diesel generator or control room. In remote areas not served by the station battery, emergency lighting is provided by portable self-contained battery-powered units. Circuits for the emergency lighting are installed in two independent trains.

The scope of our review of the lighting systems included assessment of the number and types of lighting systems provided, assessment and adequacy of the power sources for the normal and emergency lighting systems, and verification of functional capability of the lighting system under all conditions of operation.

The basis for acceptance in our review was conformance of the design bases and criteria, and design of the lighting systems and necessary auxiliary supporting systems to the acceptance criteria of Section 9.5.3 of the Standard Review Plan. Other bases for acceptance were conformance to industry standards and the ability to provide effective lighting in all areas of the facility under all conditions of operations.

Based on our review, we conclude the various lighting systems provided at the facility are in conformance with the above cited standards, criteria and design basis, they can perform their design function, and are therefore acceptable.

9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System

Emergency Diesel Engine Auxiliary Support Systems (General)

There are two emergency diesel generators for the facility and each diesel engine has the following auxiliary support systems which are addressed in detail in the Safety Evaluation Report sections indicated below:

1. Fuel oil storage and transfer system (section 9.5.4.2)

2. Cooling water system (section 9.5.5),

3. Starting system (section 9.5.6),

4. Lubrication system (section 9.5.7) and

5. Combustion intake and exhaust system (section 9.5.8)

This section of the Safety Evaluation Report applies to all the above systems.

The diesel generator and its auxiliary support systems, except for the buried fuel oil storage tanks and a portion of the connecting fuel transfer piping up to the diesel generator building wall, are housed in a seismic Category I diesel generator building structure which provides protection from the effects of tornadoes, tornado missiles, and floods. The buried portions of the fuel oil storage and transfer system are also protected from tornadoes, tornado missiles and floods. Therefore, the requirements of Criterion 2 of the General Design Criteria, "Design Bases of Protection Against Natural Phenomena," Criterion 4 of the General Design Criteria, "Environmental and Missile Design Bases," recommendations of Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants" and Regulatory Guide 1.117, "Tornado Design Classification," are met.

The diesel engine and its engine-mounted and separately skid-mounted portions of the auxiliary support systems piping and components normally furnished with the diesel generator package are designed to seismic Category I requirements and follow the guidelines of the Diesel Generator Manufacturers Association (DEMA) standards. The diesel engine, and its mounted auxiliary support systems piping and components conform to the requirements of Regulatory Guide 1.9, "Selection, Design and Qualification of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Plants," IEEE Standard 387-1977, "Standard Criteria Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations," which endorses the Diesel Engine Manufacturers Association (DEMA) standard, and the quality assurance requirements of Appendix B to 10 CFR Part 50.

The applicant will perform preoperational and startup tests of the diesel engine auxiliary support systems in accordance with recommendations of Regulatory Guide 1.68, "Initial Test Programs for Water Cooled Reactor Power Plants."

The design of the diesel engine auxiliary support systems are evaluated with respect to the recommendations of Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," in Section 3.6 of this Safety Evaluation Report.

The adequacy of the fire protection for the emergency diesel generator and associated auxiliary support systems with respect to the recommendations of Branch Technical Position ASB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," is evaluated in Section 9.5.1 of this Safety Evaluation Report.

The designs of the diesel generator auxiliary support systems also have been evaluated with respect to the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Generator Reliability." This report made specific recommendations on increasing the reliability of nuclear power plant emergency diesel generators. Information requests concerning these recommendations were transmitted to the applicant on January 29, 1980. The applicant responded in amendments 16 and 19 to the Final Safety Analysis Report and subsequent telephone conversations stating how they meet or will meet the recommendations of NUREG/CR-0660.

We have reviewed these responses and have determined that conformance to the recommendations is as follows:

Recommendation	Conformance	Safety Evaluation Report Section
1. Moisture in air starting system	Yes	9.5.6
2. Dust and dirt in diesel generator room	Yes	9.5.8
3. Turbocharger gear drive problem	Not Applicable	-
4. Personnel training	Yes	9.5.4.1
5. Automatic prelube	Partial	9.5.7
6. Testing, test loading and preventive maintenance	Yes	9.5.4.1
7. Improve the identification of root cause of failures	Yes	9.5.4.1
8. Diesel generator ventilation and combustion air systems	Yes	9.5.8
9. Fuel storage and handling	Yes	9.5.4.2
10. High temperature insulation for generator	*	9.5.4.1
11. Engine cooling water	Yes	9.5.5
12. Concrete dust control	Yes	9.5.4.1
13. Vibration of instruments	Partial	9.5.4.1

*Explicit conformance is considered unnecessary by the NRC staff in view of the equivalent reliability provided by the design, margin, and qualification testing requirements that are normally applied to emergency standby diesel generators.

On the basis of our review, we have concluded that there is sufficient assurance of diesel generator reliability to warrant unrestricted plant operation through the first refueling period. However, to assure long-term reliability of the diesel generator installations, we require that the following design and procedural modifications be implemented prior to the first refueling.

1. This item is discussed in Section 9.5.7 of this Safety Evaluation Report.

2. Vibration of instruments and controls: The applicant stated that two control panels are furnished with the engine skid, one control relay panel is mounted at the generator end of the diesel generator skid and the second panel, an engine gauge board including pressure switches, is located at the engine end of the skid and mounted on vibration isolation devices. The applicant also stated that the diesel generator package (including the control panels) is seismically qualified.

   Mounting the panels on vibration isolation devices and seismically testing them as part of the diesel generator skid package does not qualify this equipment with controls and monitoring instrumentation for continuous operation under severe vibrational stresses, unless the skid-mounted panels and equipment have been specifically developed, tested, and qualified for these conditions.

   We require the applicant to either provide test results and results of analyses which validate that the skid-mounted control panels and mounted equipment have been developed, tested, and qualified for operation under severe vibrational stresses encountered during diesel engine operation or floor mount the control panels presently furnished with the diesel generator separate from the skid on a vibration-free floor area.

   The present diesel generator design meets the requirements of Criteria 17 and 21 of the General Design Criteria. Upon completion of the above changes and modifications, the design of the diesel generator and its auxiliary systems will also be in conformance with recommendations of NUREG/CR-0660 for enhancement of diesel generator reliability and the related NRC staff guidelines and criteria. We therefore conclude that this will provide reasonable assurance of diesel generator reliability through the design life of the facility.

Emergency Diesel Engine Fuel Oil Storage and Transfer System

The design function of the emergency diesel engine fuel oil storage and transfer system is to provide a separate and independent fuel oil supply train for each diesel generator, and to permit operation of the diesel generator at engineered safety feature load requirements for a minimum of seven days without replenishment of fuel.

There are two emergency diesel generators for the single unit plant. Each diesel engine fuel oil storage and transfer system consists of a 550-gallon per day tank sufficient to power the diesel engine at rated load for approximately one and a half hours, a 52,000-gallon storage tank sufficient to power the diesel engine at maximum continuous load conditions for seven days, two alternating current power motor-driven transfer pumps powered from the associated diesel and the associated piping, valves, instrumentation, and controls.

Each diesel engine fuel oil storage and transfer system is independent and physically separated from the other system. A single failure within any one of the two systems will affect only the associated diesel generator. Therefore, the requirements for Criterion 17 of the General Design Criteria, as related to the capability of the fuel oil system to meet independence and redundancy criteria are met.

The diesel fuel oil storage and transfer system piping and components are designed to seismic Category I requirements and also designed, fabricated and tested in accordance with ASME Section III, Class 3 (Quality Group C), requirements as recommended by Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water- Steam- and Radioactive-Waste Containing Components of Nuclear Power Plants," and by Regulatory Guide 1.29, "Seismic Design Qualification."

The design of the emergency diesel engine fuel oil storage and transfer system conforms to ANSI-N 195, "Fuel Oil Systems for Diesel Generators." In addition, the applicant states that the fuel oil quality and tests will be in accordance with recommendations of Regulatory Guide 1.137, "Diesel Fuel Oil Systems."

The scope of our review of the diesel engine fuel oil storage and transfer system included layout drawings, piping and instrumentation diagrams, and descriptive information in Section 9.5.4 of the Final Safety Analysis Report for the system and auxiliary support systems essential to its operation.

The basis for the acceptance in our review was conformance of the design criteria and bases and design of the diesel engine fuel oil storage and transfer system to the acceptance criteria of Section 9.5.4 of the Standard Review Plan as described above, recommendations of NUREG/CR-0660, and industry codes and standards.

Based on our review, we conclude that the emergency diesel engine fuel oil storage and transfer system is in conformance with the above cited criteria and design bases, it can perform the design safety function, it meets the recommendation of NUREG/CR-0660 and is therefore acceptable.

9.5.5 Emergency Diesel Engine Cooling Water System

The design function of the emergency diesel engine cooling water system is to maintain the temperature of the diesel engine within a safe operating range under all load conditions and to maintain the engine coolant preheated during standby conditions to improve starting reliability.

The diesel engine cooling water system is a closed cooling system. The major components of this system for each diesel engine include turbocharger air coolers, governor oil cooler, jacket water cooler, an engine-driven jacket water coolant pump, a motor-driven auxiliary jacket water coolant pump, an expansion tank (standpipe), an electric immersion heater, a thermostatic three-way valve, heat exchanger, required instrumentation, controls and alarms, and the associated piping and valves to connect the equipment. When the diesel engine is operating, the heat generated is rejected to the service water system by means of the jacket water cooler.

The emergency diesel engine cooling water system consists of two subsystems: 1) the intercooler system and 2) the jacket water system. The intercooler system cools the turbocharger air intercoolers, alternator cutboard bearing and fuel injection nozzles. The jacket water system cools the cylinder liners cylinder heads, and turbocharger cooling spaces.

During operation of the diesel engine, temperature regulation of the diesel engine coolant is accomplished automatically through action of temperature sensing three-way thermostatic valve. When the engine is idle, the engine coolant is heated by an electric heater, controlled by a thermostat to keep the engine warm and ready to accept loads within the prescribed time interval. Alarms have been provided to enable the control room operator to monitor the diesel generator while the unit is in the standby mode or in operation.

To minimize material corrosion, the diesel engine cooling water is chemically treated with corrosion inhibitors as recommended by the engine manufacturer.

There are two emergency diesel generators for the plant and each has a physically separate and independent cooling water system. Therefore the requirements of Criterion 44 of the General Design Criteria, as related to redundancy and single-failure criteria are met.

The diesel engine cooling water system including the piping, valves, and the engine cooling heat exchangers are designed, fabricated, and tested in accordance with the ASME Code Section III, seismic Category I, Class 3 (Quality Group C) requirements as recommended by Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water- Steam- and Radioactive-Waste-Containing Components of Nuclear Power Plants," and Regulatory Guide 1.29, "Seismic Design Qualification." The jacket water expansion tank is designed to seismic Category I requirements and also designed, fabricated, and tested in accordance with ASME Code Section VIII requirements and conforms to the quality assurance requirements of Appendix B to 10 CFR Part 50.

The diesel engine cooling water system conforms with Branch Technical Position ICSB-17 (PSB), as it relates to engine cooling water protective interlocks. The diesel generator system protective interlocks are discussed in Section 8.3 of this Safety Evaluation Report.

The diesel engine cooling water system has provisions to permit periodic inspection and functional testing during standby and normal modes of power plant operation as required by Criterion 45 of the General Design Criteria, "Inspection of Cooling Water System," and Criterion 46 of the General Design Criteria, "Testing of Cooling Water System."

The scope of our review of the emergency diesel engine cooling water system included layout drawings, piping and instrumentation diagrams, and descriptive information in Section 9.5.5 of the Final Safety Analysis Report for the system and auxiliary support systems essential to its operation.

The basis for acceptance in our review was conformance to the design criteria and bases and design of the emergency diesel engine cooling water system to the acceptance criteria of Section 9.5.5 of the Standard Review Plan, recommendations of NUREG/CR-0660, industry standards, and the ability of the system to maintain stable diesel engine cooling water temperature under all load conditions.

Based on our review, we conclude that the emergency diesel engine cooling water system is in conformance with the above cited criteria and design bases, it can perform the design safety function, it meets the recommendations of NUREG/CR-0660, and is therefore acceptable.

9.5.6 Emergency Diesel Engine Starting Systems

The design function of the emergency diesel engine starting system is to provide a reliable method for automatically starting each diesel generator such that the rated frequency and voltage is achieved and the unit is ready to accept required loads within 10 seconds.

There are two emergency diesel generators for the facility. Each emergency diesel generator has an independent and redundant air starting system consisting of two separate full capacity air starting subsystems each with sufficient air capacity to provide a minimum of five consecutive cold engine starts. Redundancy in starting systems is provided so that a malfunction or failure in one system does not impair the ability of the other system to start the diesel engine.

Each subsystem includes an air compressor, a receiver tank, a moisture separator with trap, a refrigerated air dryer with trap to remove water, oil, and foreign particles from the air supply to the storage tank, intake air filters, injection lines and valves, air-to-cylinder distributor and starting valves, instrumentation, controls, alarms, and the associated piping to connect the equipment. Alarms annunciate on the local panel and in the main control room to enable the operators to monitor the air pressure of the diesel generator starting air system.

The emergency diesel engine starting system including engine-mounted piping, the air receiver tanks, and the interconnecting piping between the air receiver and diesel generator skid are designed to seismic Category I requirements and also designed, fabricated, and tested in accordance with ASME Section III, Class 3 (Quality Group C), requirements as recommended by Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and by Regulatory Guide 1.29, "Seismic Design Qualification."

The scope of review of the emergency diesel engine starting system included layout drawings, piping and instrumentation diagrams, and descriptive information in Section 9.5.6 of the Final Safety Analysis Report for the system and auxiliary support systems essential to its operation.

The basis for acceptance in our review was conformance to the design criteria and bases and design of the emergency diesel engine starting system to the acceptance criteria of Section 9.5.6 of the Standard Review Plan as described above, recommendations of NUREG/CR-0660, industry standards and the ability of the system to start the diesel generator within a specified time period.

Based on our review, we conclude that the design of the emergency diesel engine starting system is in conformance with the above cited criteria and design bases, it can perform the design safety function, it meets the recommendations of NUREG/CR-0660, and is therefore acceptable.

9.5.7 Emergency Diesel Engine Lubricating Oil System

The design safety function of the emergency diesel engine lubricating oil system is to provide a supply of filtered lubrication oil to the various moving parts of the diesel engine including pistons and bearings.

Major components of the emergency diesel engine lubricating oil system include engine-driven pumps, auxiliary motor-driven pumps, a lube oil collection sump, strainers and filters, a lube oil cooler, an electric heater and thermostatic three-way valve, instrumentation, controls, and alarms, and associated piping and valves to connect the equipment. Crankcase pressure relief doors are provided for protection from crankcase explosion. Alarms and protective devices are provided to enable the control room operator to monitor the diesel generator lube oil system during standby, startup, or in operation.

The emergency diesel engine lubricating oil system is an integral part of the diesel engine and has three subsystems which circulate lube oil through the engine for lubrication and cooling when the engine is operating. The engine heat is rejected to the engine closed loop cooling system which in turn gives its heat to the service cooling water system. The three subsystems are: (1) the engine lube oil system, (2) the rocker arm lube oil system, and (3) the auxiliary lube oil system. The engine lube oil system supplies oil to all main bearings, the camshaft bearings, cam followers, fuel injection pumps, and valve push rods. The rocker arm lube system supplies lube oil to the valve rockers and upper engine parts excluding the turbocharger. This portion of the system is composed of an engine-driven pump and an alternating current motor-driven pump which operates automatically five minutes per day while the diesel is in the standby mode. No alarm is provided in the event the motor-driven pump fails. We find this unacceptable. We require that an alarm which indicates pump failure be installed by the first refueling. The auxiliary lube oil system is operated only when the diesel engine is on standby, at which time the lube oil is heated by an electric heater and circulated through the engine continuously by an alternating current motor-driven pump to improve the first-try starting reliability. This portion of the system has an alarm to indicate pump and/or heater failure.

The diesel engine lubricating oil system is designed to seismic Category I and ASME Code Section III, Class 3 requirements as recommended by Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and by Regulatory Guide 1.29, "Seismic Design Qualifications."

The diesel generator lubricating oil system conforms with Branch Technical Posotion ICSB-17 (PSB), as it relates to diesel engine lubrication system protective interlocks. The diesel generator system protective interlocks are discussed in Section 8.3 of this Safety Evaluation Report.

The scope of our review of the diesel generator lubricating oil system included piping and instrumentation diagrams, and descriptive information in Section 9.5.7 of the Final Safety Analysis Report for the system and auxiliary support systems essential to its operation.

The basis for acceptance in our review was conformance to the design criteria and bases and design of the diesel generator lubricating oil system to the acceptance criteria of Section 9.5.7 of the Standard Review Plan, recommendations of NUREG/CR-0660, industry standards, and the ability of the system to provide necessary engine lubrication during periods of operation and to maintain the engine lube oil at a temperature that improves first-try starting reliability during periods of standby.

Based on our review, we conclude that the diesel generator lubricating oil system for each diesel engine is in conformance with the above cited criteria and design bases, it can perform the design safety function, it will meet the recommendations of NUREG/CR-0660 upon completion of the above modification, and is therefore acceptable.

9.5.8 Emergency Diesel Engine Combusion Air Intake and Exhaust System

The design function of the emergency diesel engine combusion air intake and exhaust system is to supply filtered air for combustion to the engine and to dispose of the engine exhaust to atmosphere.

A separate source of combustion air for each diesel engine is taken from the diesel generator building air intakes through an air filter, intake silencer, two turbocharger compressors and intercoolers. The path of the exhaust gas discharge is through the turbocharger, exhaust silencer, and exhaust ducting to the outside of the building.

The exhaust system is separated from the air intake system to reduce the possibility of contamination of the intake air with recirculated exhaust gases. The location of the air intake structure precludes the intake of fire extinguishing agents and other noxious gases that could affect diesel generator operation.

The diesel generator intake and exhaust system components, piping, ducting and components are seismic Cateogry I and are designed in accordance with the Diesel Engine Manufacturers Association (DEMA) requirements.

The scope of review of the diesel generator intake and exhaust system included layout drawings, piping and instrumentation diagrams, and descriptive information in Section 9.5.8 of the Final Safety Analysis Report for the system and auxiliary support systems essential to its operation.

The basis for acceptance in our review was conformance of the design criteria and bases and design of the diesel generator intake and exhaust system to the acceptance criteria of Section 9.5.8 of the Standard Review Plan as described above, recommendations of NUREG/CR-0660, industry standards, and the ability of the system to provide sufficient combustion air and release of exhaust gases to enable the emergency diesel generator to perform on demand.

Based on our review, we conclude that the diesel generator intake and exhaust system for each diesel engine is in conformance with the above cited criteria and design bases, meets the recommendations of NUREG/CR-0660, can perform the design safety function, and is therefore acceptable.

10 STEAM AND POWER CONVERSION SYSTEMS

10.1 Summary Description

The steam and power conversion system is designed to remove heat energy from the primary reactor coolant loop via three steam generators and to generate electric power in the turbine-generator. After the steam passes through the high- and low-pressure turbines, the main condensers deaerate the condensate and transfer the rejected heat to the closed cycle circulating water system which uses Monticello Reservoir to dissipate the rejected heat to the atmosphere by surface evaporation, radiation, and convection. The condensate is reheated and returned as feedwater to the steam generators. The entire system is designed for the maximum expected energy from the nuclear steam supply system.

A turbine bypass system is provided to discharge up to 85 percent of the main steam flow around the turbine during transient conditions; 48.6 percent directly to the condenser and 36.4 percent to the atmosphere. This bypass capacity together with a 10 percent reactor automatic step load reduction capability is sufficient to withstand a 95 percent generator load loss without tripping the reactor or the turbine.

10.2 Turbine-Generator

The turbine-generator converts steam power into electrical power and has a turbine control and overspeed protection system. The design function of the turbine control and overspeed protection system is to control turbine action under all normal or abnormal conditions and to assure that a full-load turbine trip will not cause the turbine to overspeed beyond acceptable limits. The turbine control and overspeed protection system is therefore essential to the overall safe operation of the facility.

The turbine-generator is manufactured by the General Electric Company and is a tandem-compound type (single shaft) with one double-flow high pressure turbine and two double-flow low pressure turbines. The rotational speed is 1800 revolutions per minute and is designed for a gross generator output of 953.9 megawatts electrical at a nominal plant exhaust pressure of two inches of mercury (absolute).

The turbine-generator is equipped with an electrohydraulic control system. The electrohydraulic control system consists of an electronic governor using solid state control techniques in combination with a high-pressure hydraulic actuating system. The system includes electrical control circuits for steam pressure control, speed control, load control, and steam control valve positioning.

Overspeed protection is accomplished by three independent systems; i.e., normal speed governor, mechanical overspeed, and electric backup overspeed control systems. The normal speed governor closes the intercept valves and control valves at 102 and 104 percent of rated speed respectively. The mechanical overspeed sensor trips the turbine stop, control, and combined intermediate valves by deenergizing the hydraulic fluid systems when 110 percent of rated speed is reached. The stop valves close in 0.15 second, the control valves in 0.2 second, and the combined intermediate valves within 0.20 second.

These valves are designed to fail closed on loss of hydraulic system pressure. The electrical backup overspeed sensor will trip these same valves, when 112 percent of rated speed is reached, by independently deenergizing the hydraulic fluid system. Both of these actions independently trip the energizing trip fluid system. The overspeed trip systems can be tested while the facility is on line.

In order to protect the turbine-generator, the following signals will shut down the turbine: (1) reactor train A trip, (2) master turbine trip pushbutton, (3) shaft oil pump discharge pressure low and turbine speed greater than 1300 revolutions per minute, (4) low electrohydraulic control hydraulic pressure, (5) loss of speed signals, (6) high vibration, (7) loss of electrohydraulic control 125-volt direct current A train power and turbine speed less than 1300 revolutions per minute, (8) thrust bearing wear or low bearing oil pressure, (9) loss of stator cooling [70-second time delay], (10) exhaust hood high temperature, (11) backup overspeed trip, [112 percent of rated speed], (12) moisture separator high level [10-second time delay], (13) generator electrical fault, (14) loss of three of three feedwater pumps, (15) hi-hi steam generator level [two of three] or train A safety injection signal, (16) low pressure heater 5A or 5B or 6A or 6B hi-hi level [10-second time delay], (17) low vacuum trip [less than 20 inches of mercury] and master reset pushbutton not actuated, (18) steam generator hi-hi pressure or train B safety injection signal and 125-volt direct current B train power, and (19) reactor train B trip and 125-volt direct current B train power.

An inservice inspection program for the main steam stop and control valves and reheat valves is provided and includes: (a) dismantling and inspection of at least one main steam stop valve, one main steam control valve, and one reheat stop valve, and one reheat intercept valve, at approximately 3-1/3 year intervals during refueling or maintenance shutdowns coinciding with the inservice inspection schedule required by Section XI of the ASME Code; (b) exercising and observing, as required, the main steam stop and control, reheat stop, and intercept valves.

The applicant will include preoperational and startup tests of the turbine generator in accordance with Regulatory Guide 1.68, "Initial Test Programs for Water Cooled Power Plants."

The turbine-generator system meets the recommendations of Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment." Evaluation of protection against dynamic effects associated with the postulated rupture of piping is discussed in Section 3.6 of this Safety Evaluation Report.

The scope of our review of the turbine-generator included descriptive information in Section 10.2 of the Final Safety Analysis Report, and flow charts and diagrams. The basis for acceptance in our review was conformance of the design criteria and bases and design of the turbine-generator system to the acceptance criteria of Section 10.2 of the Standard Review Plan and industry standards.

Based on our review, we conclude that the turbine-generator overspeed protection system is in conformance with the above cited criteria and design bases, it can perform its design safety functions, and is therefore acceptable.

10.3 Main Steam Supply System

10.3.1 Design

The function of the main steam supply system is to convey steam from the steam generators to the high-pressure turbine and other auxiliary equipment for power generation. The steam produced in the three steam generators will be routed to the high-pressure turbine by three main steam lines up to the common header. Each main steam line will contain one main steam isolation valve. The portions of the main steam lines from the steam generators, through the containment, and up to and including the main steam isolation valves are Quality Group B and seismic Category I.

The main steam isolation valves are designed to close in five seconds upon receipt of a main steam isolation valve closure signal. The valves are designed to stop steam flow from either direction. Failure of one main steam isolation valve to close, coincident with a steam line break, will not result in the uncontrolled blowdown of more than one steam generator. In the event of a steam line break upstream of a main steam isolation valve and a failure of the main steam isolation valve to close on the unaffected steam generator, blowdown of the unaffected steam generator is prevented by the closure of the non-seismic Category I turbine stop valves and turbine bypass valves which serve as an acceptable backup for this accident.

Seismic Category I safety valves and power-operated relief valves are provided for each steam generator immediately outside the containment structure upstream of the main steam isolation valves. The power-operated relief valves are air-operated and fail in the closed position on loss of air supply. They are also equipped with handwheels to facilitate manual operation if required.

Based on our review, we conclude that the main steam supply system design, up to and including the main steam isolation valves, is in conformance with the single failure criterion, the position of Regulatory Guide 1.29 related to seismic design, and main steam isolation valve closure time requirements and is, therefore, acceptable.

The portion of the main steam supply system downsteam of the main steam isolation valves is not required to effect or support safe shutdown of the facility.

The main steam supply system is designed to deliver steam from the steam generators to the high-pressure turbine. The main steam and turbine steam systems provide steam to the moisture separator reheaters, main feedwater pump turbines, turbine gland sealing system, feedwater heaters, and turbine bypass system.

The scope of our review of the main steam supply system (between the main steam isolation valves and up to and including the turbine stop valves) included descriptive information in Section 10.^ of the Final Safety Analysis Report, and flow charts and diagrams. The basis for acceptance in our review was conformance of the design criteria and bases and design of main steam supply system to the acceptance criteria of Section 10.3 of the Standard Review Plan.

Based on our review, we conclude the main steam supply system between the main steam isolation valves and up to and including the turbine stop valves is in conformance with the above cited criteria and design bases, it can perform its design functions, and is, therefore, acceptable.

10.3.2 Steam and Feedwater System Materials

The mechanical properties of materials selected for Class 1, 2 and Class 3 components of the steam and feedwater systems satisfy Appendix I of Section III of the ASME Boiler and Pressure Vessel Code, and Parts A, B and C of Section II of the ASME Boiler and Pressure Vessel Code. The fracture toughness properties of the steam and feedwater systems are in accordance with the 1974 edition and applicable addenda of Section III of the ASME Code. This edition of the ASME Code states that the design specification shall determine wether fracture toughness testing is required. The applicant indicates that feedwater isolation valves, feedwater check valves, feedwater system reactor building penetration assemblies and feedwater piping were impact tested. Main steam supply system impact testing was not specified since the minimum service temperature is 327 degrees Fahrenheit at 100 pounds per square inch. These fracture toughness tests and mechanical properties required by the ASME Code provide reasonable assurance that ferritic materials will have adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture.

Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal;" Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel;" and Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel" do not apply to the steam or feedwater systems since austenitic stainless steels are not utilized in these systems.

The welding procedures used in limited access areas satisfy the recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility." The on-site cleaning and cleanliness controls during fabrication satisfy the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the requirements of ANSI Standard N45.2-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants." The precautions taken in controlling and monitoring the preheat and interpass temperatures during welding of carbon and low alloy steel components satisfy the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low-Alloy Steel."

Conformance with the cited codes, standards, and regulatory guides constitutes an acceptable basis for assuring the integrity of steam and feedwater systems, and for meeting the requirements of Criterion 1 of the General Design Criteria.

10.3.3 Secondary Water Chemistry

In late 1975, we incorporated provisions into the Standard Technical Specifications that required limiting conditions for operation and surveillance requirements for secondary water chemistry parameters. The Technical Specifications for all pressurized water reactor facilities that have been issued an operating license since 1974, contain either these provisions, or a requirement to establish these provisions after baseline chemistry conditions have been determined. The intent of the provisions was to provide added assurance that the operators of newly licensed plants would properly monitor and control secondary water chemistry to limit corrosion of steam generator tubes and the tube support plates.

In a number of instances, the Technical Specifications have significantly restricted the operational flexibility of some plants with little or no benefit with regard to limiting degradation of steam generator tubes and the tube support plates. Based on this experience and the knowledge gained in recent years, we have concluded that technical specification limits are not the most effective way of assuring that steam generator degradation will be minimized.

Due to the complexity of the corrosion phenomena involved and the state of the art as it exists today, we are of the opinion that, in lieu of specifying limiting conditions in the Technical Specifications, a more effective approach would be to institute a license condition requiring the implementation of a secondary water chemistry monitoring and control program containing appropriate procedures and administrative controls.

The required program and procedures are to be developed by applicants, with input from their nuclear steam supply system supplier or other consultants, to account for site and plant-specific factors that affect chemistry conditions in the steam generators. In our view, facility operation following such procedures would provide assurance that licensees would devote proper attention to controlling secondary water chemistry, while also providing the needed flexibility to allow them to deal effectively with an off-normal condition that might arise.

Consequently, we requested, in a letter dated August 24, 1979, that the applicant propose a secondary water chemistry program which will be referenced in a condition to the operating license. In the letter, we concluded that such an operating license condition, in conjunction with existing Technical Specifications on steam generator tube leakage and inservice inspection, would provide the most practical and comprehensive means of assuring that steam generator tube integrity would be maintained.

By letter dated May 8, 1980, the applicant provided a water chemistry monitoring and control program for the Virgil C. Summer Nuclear Power Station, Unit 1. In response to our requests for additional information, the applicant submitted supplemental information in letters dated July 14 and August 12, 1980. The program addressed the following:

1. Identification of a sampling schedule for the critical parameters and of control points for these parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions for off-control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data and (b) the sequence and timing of administrative events required to initiate corrective action.

The proposed secondary water chemistry monitoring and control program calls for monitoring critical parameters to inhibit steam generator corrosion and tube degradation. We discussed with the applicant our concern that out-of-limit water chemistry should be monitored at the discharge of the condenser hot well pump for detecting condenser leakage. We also discussed the corrective action to repair the condenser leak within 96 hours of confirming a leak, in accordance with Branch Technical Position MTEB 5-3 which is appended to Section 5.4.2.1 of the Standard Review Plan.

The applicant agreed to use the hot well discharge sampling point as the monitor for condenser leaks. In addition, he agreed to an alternate approach for meeting the 96-hour corrective action requirement of Branch Technical Position MTEB 5-3 in the event of a condenser leak. The alternate approach consists of implementing corrective action and limiting operation under transient chemistry conditions of feedwater and steam generator blowdown to less than 100 hours, less than 24 hours or less than two hours for progressively severe out-of-limit secondary water chemistry. We find this alternate approach to Branch Technical Position MTEB 5-3 acceptable since:

1. It establishes a specific continuously monitored condensate sample point for confirming a condenser leak,

2. The incorporation of feedwater impurity-time operational limits provides an early indication of impurities entering the steam generator before the entire steam generator secondary side reaches or exceeds its operational limits, and

3. It provides an effective, integrated impurity-time limit to the quantity of impurities entering the steam generator.

It should be noted that the steam generators for the facility are of the Westinghouse design having carbon steel supporting plates with drilled tube support holes. Steam generators of this design have experienced denting and cracking in operating plants. Although an effective secondary water chemistry control program can reduce the rate of tube degradation, there is no assurance that a 40-year steam generator lifetime can be achieved.

In spite of the possibility of tube cracking, we have concluded that operation of the steam generators will not constitute an undue risk to the health and safety of the public for the following reasons:

1. Primary to secondary leakage rate limits, and associated surveillance requirements have been established to provide assurance that the occurrence of tube cracking during operation will be detected and appropriate corrective action, such as tube plugging, will be taken such that any individual crack that is present will not become unstable under normal operating, transient or accident conditions.

2. Inservice inspection requirements and preventive tube plugging criteria have been established to provide assurance that the great majority of degraded tubes will be identified and removed from service before leakage develops.

We have reviewed the applicant's secondary water chemistry monitoring and control program and, based on the above evaluation, we have determined that it meets (1) the NRC staff requirements delineated in our August 24, 1979 letter; (2) positions 2 and 3 in Branch Technical Position MTEB 5-3, revision 1, (3) the acceptance criteria of Section 5.4.2.1 of the Standard Review Plan for secondary coolant purity; and (4) the requirements of Criterion 14 of the General Design Criteria as they relate to secondary water chemistry control and monitoring. Accordingly, we conclude that the applicant's secondary water chemistry monitoring and control program is acceptable.

10.4. Other Features

10.4.1 Main Condenser

The main condenser is designed to function as a heat sink for the turbine exhaust steam, turbine bypass steam, and other turbine cycle flows, and to receive and collect condensate flows for return to the steam generator. The main condenser transfers heat to the circulating water system which uses Monticello Reservoir to dissipate the rejected heat to the atmosphere by surface evaporation, radiation, and convection.

The main condenser is not required to effect or support safe shutdown of the facility or to perform in the operation of facility safety features. The main condenser has two zones and is designed to produce turbine back pressures of 2.13 and 3.00 inches of mercury absolute for the two zones when operating at rated turbine output. The main condenser design includes provision for hotwell surge storage of the condensate and feedwater systems which is enough for approximately a two-minutes supply at design conditions. Off-gas from the main condenser is processed in the main condenser air removal system.

The main condenser is designed to accept full-load exhaust steam from the main turbine and steam generator feedwater pump turbines, up to 48.6 percent of the main steam flow from the turbine bypass system, and other cycle steam flows. The main condenser is also designed to deaerate the condensate to the required water quality. Stainless steel tubes have been used to minimize corrosion and erosion of condenser tubes. The applicant will include preoperational and startup tests of the main condenser in accordance with recommendations of Regulatory Guide 1.68, "Initial Test Programs for Water Cooled Reactor Power Plants."

The scope of our review of the main condenser included layout drawings and descriptive information of the condenser in Section 10.4.1 of the Final Safety Analysis Report.

The basis for acceptance in our review was conformance of the design criteria and bases and design of the condenser to the acceptance criteria of Section 10.4.1 of the Standard Review Plan and industry standards.

Based on our review, we concluue that the main condenser is in conformance with the above cited criteria and design bases, and that it can perform its design function, and is therefore acceptable.

10.4.2 Main Condenser Evacuation System

The main condenser evacuation system is designed to establish and maintain condenser vacuum by discharging noncondensible gases from the condenser through the auxiliary building vent stack. The system is designed to Quality Group D and to a non-seismic design classification. The main condenser evacuation system consists of three mechanical vacuum pumps. Air and noncondensibles from the vacuum pump exhaust are continuously monitored by a radiation detector prior to release to the environment, via the auxiliary building ventilation exhaust system charcoal adsorbers and high efficiency particulate filters.

The scope of our review included the system capability to process radioactive gases and the design provisions incorporated to monitor and control releases of radioactive materials in gaseous effluents in accordance with Criteria 60 and 64 of the General Design Criteria. Based upon our evaluation, we find the proposed main condenser evacuation system acceptable. The basis for our acceptance has been conformance of the applicant's design, design criteria, and design bases for the main condenser evacuation system to applicable regulations and regulatory guides.

10.4.3 Turbine Gland Sealing System

The turbine gland sealing system is designed to control radioactive steam leakage from, and air inleakage into, the turbine. The components of the system are designed to Quality Group D and to a non-seismic design classification. The turbine gland sealing system consists of labyrinth seals, a steam supply system, a gland steam condenser, and steam exhauster. Steam is supplied to the labyrinth seals from an auxiliary steam supply system during startup and from the main steam system during load operations. The gland seal steam exhauster maintains a slight vacuum in the system and exhausts the noncondensibles to the atmosphere through a release point on the turbine building roof.

We have reviewed the applicant's system description and design criteria for the components of the turbine gland sealing system and have found them consistent with the criteria given in Regulatory Guide 1.26.

The basis for acceptance in our review has been conformance of the applicant's design, design criteria, and design bases for the turbine gland sealing system to the applicable regulations referenced above. Based upon our evaluation, we find the proposed turbine gland sealing system acceptance.

10.4.4 Turbine Bypass System

The turbine bypass system is designed to bypass up to 85 percent of main steam flow; 48.6 percent around the turbine to the main condenser and 36.4 percent to the atmosphere. This capacity together with a 10 percent reactor automatic step load capability is sufficient to withstand a 95 percent generator load loss without tripping the reactor or turbine. The turbine bypass system is used to control reactor pressure as follows: a) during the reactor heatup to rated pressure; b) while the turbine generator is being brought up to speed and synchronized; c) during power operation when the reactor steam generation exceeds the transient turbine steam requirements; and d) during reactor cooldown. This system is not required to perform during accident conditions.

The bypass system is composed of the following three groups of air-operated valves: 1) eight condenser dump valves, 2) three atmosphereic dump valves, and 3) three power-operated relief valves. Each valve is rated for a capacity of approximately 6.1 percent of the main steam flow at full-load pressure and temperature. Each valve is provided with a diffuser, mounted downstream of the valve, to reduce noise level. The condenser dump valves, atmospheric dump valves, and power-operated relief valves provide bypass capacities of 48.6 percent, 18.3 percent, and 18.3 percent or rated main steam flow respectively, for a total bypass capacity of 85 percent. The eight condenser dump valves are mounted between the main steam isolation valves and turbine stop valves in pairs on four manifolds which are piped to the condenser. The three atmospheric dump valves are located just downstream of the main steam isolation valves, and the three power-operated relief valves are located just upstream of the main steam isolation valves, and both discharge to the atmosphere. The turbine bypass system is not a safety-related system and is not required for facility shutdown following an accident. The turbine bypass valves are designed to fail closed upon loss of air to the air-operated valves. The eight turbine bypass valves (condenser dump valves) are designed to close on loss of main condenser vacuum.

The applicant will include preoperational and startup tests of the turbine bypass system in accordance with recommendations of Regulatory Guide 1.68,

"Initial Test Programs for Water Cooled Reactor Power Plants." The turbine bypass system can be tested while the unit is on line.

The turbine bypass system meets the recommendations of Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures in Fluid System Piping Outside Containment." Our evaluation of the protection against dynamic effects associated with the postulated rupture of piping is covered in Section 3.6 of this Safety Evaluation Report.

The scope of our review of the turbine bypass system included drawings, piping and instrumentation diagrams, and descriptive information of the system in Section 10.4.4 of the Final Safety Analysis Report.

The basis for acceptance in our review was conformance of the design criteria and bases and design of the turbine bypass system to the acceptance criteria of Section 10.4.4 of the Standard Review Plan and industry standards.

Based on our review, we conlude that the turbine bypass system is in conformance with the above cited criteria and design bases, it can perform its design function, and is, therefore, acceptable.

10.4.5 Circulating Water System

The circulating water system is designed to remove the heat rejected from the main and auxiliary condensers to the atmosphere via Monticello Reservoir. The circulating water system is not required to maintain the facility in a safe shutdown condition or mitigate the consequences of accidents.

The applicant provided the results of an analysis of the effects of possible flooding as a result of a postulated failure of the circulating water system at an expansion joint. Flooding in the turbine building and adjacent areas is detected by flood-level sensors located at two elevations in the condenser area of the turbine building. The operation of each circulating water line is monitored by pressure indicators and circulating water condenser inlet and outlet temperature indicators, which serve as a means of detecting a failure of an expansion joint at the condenser. The flood sensor located at the lowest elevation will alarm locally in the main condenser cleaning pit should the water level reach this elevation as a result of a postulated expansion joint failure. As the water level continues to rise, it reaches a second set of two groups of three level switches located in the strainer pit 10 feet above the first set of level switches. Actuation of any two switches within either group trips the circulating water pump, initiates closure of the pump discharge valves and high pressure condenser discharge valves, and alarms this action in the control room. The water level reached in the turbine building by the time the pump discharge valve is closed and circulating water flow is stopped is 13.5 feet below the penetration to the control building or intermediate building which houses safety-related equipment. There are no essential systems or components located in the turbine building.

Based on our review, we conclude that the circulating water system design is acceptable with respect to protection of safety-related components from flooding from the postulated failure of the system.

10.4.6 Condensate, Condensate Cleanup, and Feedwater Systems

The condensate cleanup system, condensate, and feedwater systems were reviewed on the basis that their failure should not result in the loss of any essential equipment and should not affect safe shutdown of the facility. They were also reviewed to assure that adequate isolation is provided for these systems where they connect to seismic Category I systems.

The portion of the feedwater system extending from and including the feedwater isolation valves outside containment to the steam generator inlets is designed to seismic Category I requirements.

We have reviewed the design of these systems and conclude that their failure will not affect safe shutdown of the facility, and, therefore, they are acceptable.

10.4.7 Emergency Feedwater System

The emergency feedwater system is designed to supply an independent source of water to the steam generators during normal plant startup, shutdown, and layup operations and in the event of loss of main feedwater supply. The major components of the emergency feedwater system are one 100-percent-capacity steam turbine-driven pump and two 50-percent-capacity motor-driven pumps. The turbine-driven pump receives steam from two of the main steam lines upstream of the main steam isolation valves and exhausts to the atmosphere. The turbine-driven pump train is available to supply emergency feedwater independent of onsite or offsite alternating current power supplies. The alternating current powered motor-operated valves at the steam supply lines to the turbine-driven emergency feedwater pump are normally full open and remain open in case of loss of alternating current power. When either of these two valves is closed, it is annunciated in the control room.

Each emergency feedwater supply line to each of the three steam generators is provided with normally-open, pneumatically-operated flow control valves which will fail open on loss of instrument air. These valves are also provided with safety-grade air accumulators to permit remote closure for isolation of a secondary system pipe break. The motor-driven pumps and associated valves and instrumentation can be powered from their corresponding emergency diesel generators in the event of a loss of offsite power. The provisions discussed above are adequate to assure the system will function in accordance with our diversity of power guidelines. The emergency feedwater pumps normally take suction from the seismic Category I condensate storage tank, which is discussed in Section 9.2.4 of this Safety Evaluation Report. The pumps can also take suction from the seismic Category I service water system which is connected to the pump suction with normally closed, remote manual, motor-operated isolation valves and serves as a redundant backup source of supply for an indefinite period of time. The provisions discussed above provide adequate flexibility and redundancy to assure proper operation of the emergency feedwater system.

The facility uses the preheat model steam generators. We have evaluated the preheat model steam generator for hydraulic instabilities (water hammer phenomenon potential). Based on these studies we have established the need for a verification test to demonstrate that no damaging water hammer will occur in the steam generator and/or feedwater system. The applicant has committed to perform a test using the standard plant operating procedures to verify that unacceptable water hammer has not occurred. The applicant has also committed to provide us with a copy of the test procedure prior to performing the test. We find the above commitments acceptable. We conclude that completion of the test without unacceptable feedwater hammer damage will satisfy our concerns pending the completion of Task Action Plan A-1 as discussed in Appendix C to this Safety Evaluation Report.

We have reviewed the adequacy of the emergency feedwater system design necessary for safe operation of the facility during normal, abnormal, and accident conditions. We conclude that the system design conforms with the diversity guidelines of our Branch Technical Position ASB 10-1 and that the system has sufficient flexibility and redundancy and is, therefore, acceptable.

11 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

The radioactive waste management systems are designed to provide for controlled handling and treatment of liquid, gaseous, and solid wastes. The liquid radioactive waste system processes wastes from equipment and floor drains, sample waste, decontamination and laboratory wastes, regenerant chemical wastes, and laundry and shower wastes. The gaseous radioactive waste system provides holdup capacity to allow decay of short-lived noble gases stripped from the primary coolant and treatment of ventilation exhausts through high efficiency particulate air and charcoal adsorbers as necessary to reduce releases of radioactive materials to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 20 and Section 50.34a of 10 CFR Part 50. The solid radioactive waste system provides for the solidification, packaging, and storage of radioactive wastes generated during facility operation prior to shipment offsite to a licensed facility for burial.

In our evaluation of the liquid and gaseous radioactive waste systems, we have considered: (1) the capability of the systems for keeping the levels of radioactivity in effluents "as low as is reasonably achievable" based upon expected radwaste inputs over the life of the facility; (2) the capability of the systems to maintain releases below the limits of 10 CFR Part 20 during periods of fission product leakage at design levels from the fuel; (3) the capability of the systems to meet the processing demands of the facility during anticipated operational occurrences; (4) the quality group and seismic design classification applied to the equipment and components and structures housing these systems; (5) the design features that will be incorporated to control the releases of radioactive materials in accordance with Criterion 60 of the General Design Criteria; and (6) the potential for gaseous release due to hydrogen explosions in the gaseous radwaste system.

In our evaluation of the solid radioactive waste treatment system, we have considered: (1) system design objectives in terms of expected types, volumes, and activities of waste processed for offsite shipment; (2) the applicant's process control program; (3) waste packaging and conformance to applicable Federal packaging regulations, and provisions for controlling potentially radioactive airborne dusts during baling operation; and (4) provisions for onsite storage prior to shipping.

In our evaluation of the process and effluent radiological monitoring and sampling systems, we have considered the system's capability: (1) to monitor all normal and potential pathways for release of radioactive materials in the environment; (2) to control the release of radioactive materials to the environment; and (3) to monitor performance of process equipment and detect radioactive material leakage between systems.

In our evaluation, we have determined the quantities of radioactive materials that will be released in liquid and gaseous effluents and the quantity of radioactive waste that will be shipped offsite to a licensed burial facility.

In making these determinations, we have considered waste flows, activity levels and equipment performance, consistent with expected normal facility operation, including anticipated operational occurrences for an assumed 30 years of normal facility operation.

The estimated releases of radioactive materials in liquid and gaseous effluents were calculated using the PWR-GALE Code described in NUREG-0017. The liquid and gaseous source terms are given in Table 11-1 and Table 11-2, respectively. The principal parameters used in these calculations are given in Table 11-3.

The source terms were used to calculate the individual and population doses in accordance with the mathematical models and guidance contained in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1. Meteorologic factors in the dose calculations were determined using the guidance in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1. The calculated individual and population doses are given in Table 11-4.

Based upon the following evaluation, we conclude that the liquid and gaseous radioactive waste treatment systems for the facility are capable of maintaining releases of radioactive materials in liquid and gaseous effluents to "as low as is reasonably achievable" levels in accordance with Section 50.34a of 10 CFR Part 50, and with Sections II.A, II.B, II.C, and II.D of Appendix I to 10 CFR Part 50.

Based upon our evaluation, as described below, we find the proposed liquid, gaseous, and solid radioactive waste systems and associated process and effluent radiological monitoring and sampling systems acceptable.

11.2 System Description and Evaluation

11.2.1 Liquid Waste Processing System

The liquid waste processing system for the facility consists of processing equipment and instrumentation necessary to collect, process, monitor and recycle and dispose of radioactive liquid wastes. The liquid radwaste system is designed to collect and process wastes based upon the origin of the waste in the facility and upon the expected levels of radioactivity. All liquid waste is processed on a batch basis to permit optimum control of releases. Prior to being released, samples are analyzed to determine the types and amounts of radioactivity present. Based upon the results of the analyses, the waste is recycled for eventual reuse in the facility, retained for further processing, or released under controlled conditions to the environment.

A radiation monitor in the discharge line will automatically terminate liquid waste discharges if radiation measurements exceed a predetermined level. A schematic diagram of the liquid waste processing system is given in Figure 11-1. The liquid waste processing system will consist of two subsystems: drain channel A and drain channel B. In addition, the boron recovery system input

TABLE 11-1

CALCULATED RELEASES OF RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS FROM VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

Nuclide	Curies per year
Corrosion and Activation Products	
Cr-51	1.1(-4)[a]
Mn-54	1(-3)
Fe-55	1.1(-4)
Fe-59	6(-5)
Co-58	5(-3)
Co-60	8.8(-3)
Zr-95	1.4(-3)
Nb-95	2(-3)
Np-239	4(-5)
Fission Products	
Br-83	4(-5)
Rb-86	2(-5)
Sr-89	2(-5)
Mo-99	2.8(-3)
Tc-99m	3(-3)
Ru-103	1.4(-4)
Ru-106	2.4(-3)
Ag-110m	4.4(-4)
Te-127m	2(-5)
Te-127	2(-5)

Nuclide	Curies per year
Fission Products	
Te-129m	9(-5)
Te-129	6(-5)
I-130	1.9(-4)
Te-131m	5(-5)
I-131	1(-1)
Te-132	9.4(-4)
I-132	3.8(-3)
I-133	5.7(-2)
I-134	1(-5)
Cs-134	2.1(-2)
I-135	8.3(-3)
Cs-136	2.7(-3)
Cs-137	3(-2)
Ba-137m	5.7(-3)
Ba-140	1(-5)
La-140	1(-5)
Ce-144	5.2(-3)
All Others	4(-5)
Total except Tritium	0.26
Tritium	360

a = Exponential notation; 1.1(-4) = 1.1×10^{-4}.

TABLE 11-2

CALCULATED RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS FROM VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

(Curies per year)

Nuclide	Waste Gas Decay Tanks	Reactor Building	Auxiliary Building	Turbine Building	Air Ejector Exhaust	Total
Kr-83m	a	1	a	a	a	a
Kr-85m	a	11	2	a	1	14
Kr-85	203	5	a	a	a	210
Kr-87	a	2	1	a	a	3
Kr-88	a	14	4	a	3	21
Kr-89	a	a	a	a	a	a
Xe-131m	3	10	a	a	a	13
Xe-131m	a	43	2	a	1	46
Xe-133	a	2500	110	a	70	2700
Xe-135m	a	a	a	a	a	a
Xe-135	a	55	7	a	4	66
Xe-137	a	a	a	a	a	a
Xe-138	a	a	1	a	a	1
Total Noble Gases						3,100
I-131	a	4.2(-2)	1.4(-2)	1.2(-3)	8.3(-3)	6.6(-2)
I-133	a	3.3(-2)	2(-2)	1.4(-3)	1.2(-2)	6.6(-2)
Tritium	-	-	-	-	-	800
C-14	7	1	a	a	a	8
Ar-41	a	25	a	a	a	25
Mn-54	4.5(-5)	2.1(-4)[b]	1.8(-4)	c	c	4.4(-4)
Fe-59	1.5(-5)	7.3(-5)	6(-5)	c	c	1.5(-4)
Co-58	1.5(-4)	7.3(-4)	6(-4)	c	c	1.5(-3)
Co-60	7(-5)	3.3(-4)	2.7(-4)	c	c	6.7(-4)
Sr-89	3.3(-6)	1.7(-5)	1.3(-5)	c	c	3.3(-5)
Sr-90	6(-7)	2.9(-6)	2.4(-6)	c	c	5.9(-6)
Cs-134	4.5(-5)	2.1(-4)	1.8(-4)	c	c	4.4(-4)
Cs-137	7.5(-5)	3.7(-4)	3(-4)	c	c	7.5(-4)

a = Less than one curie per year for noble gases and carbon-14; less than 10^{-4} curies per year for iodine.

b = Exponential notation; 2.1(-4) = 2.1×10^{-4}.

c = Less than one percent of total for this nuclide.

Table 11-3

PRINCIPAL PARAMETERS AND CONDITIONS USED IN CALCULATING RELEASES OF RADIOACTIVE MATERIAL IN LIQUID AND GASEOUS EFFLUENTS FROM VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

Reactor power level (megawatts thermal)	2900
Plant capacity factor	0.80
Failed fuel percent[a]	0.12
Primary steam	
Mass of coolant (pounds)	4.2×10^5
Letdown rate (gallons per minute)	60
Shim bleed rate (gallons per day)	1.5×10^3
Leakage to secondary system (pounds per day)	100
Leakage to containment building	b
Leakage to auxiliary building (pounds per day)	160
Frequency of degassing for cold shutdowns (per year)	2
Secondary system	
Steam flow rate (pounds per hour)	1.2×10^7
Mass of liquid/steam generator (pounds)	9.5×10^4
Mass of steam/steam generator (pounds)	8.6×10^3
Secondary coolant mass (pounds)	2.3×10^6
Rate of steam leakage to turbine area (pounds per hour)	1.7×10^3
Number of steam generators	3
Reactor building volume (cubic feet)	1.8×10^6
Annual frequency of reactor building purges (shutdown)	4
Reactor building low volume purge rate (cubic feet per minute)	1000
Reactor building atmosphere cleanup rate (cubic feet per minute)	24,000
Prepurge cleanup time (hours)	16
Iodine partition factors (gas/liquid)	
Leakage to auxiliary building	0.0075
Main condenser/air ejector (volatile species)	0.15

		Blowdown Treatment System
Liquid radwaste system decontamination factor	Iodine	1×10^3
	Cesium, Rubidium	1×10^2
	Others	1×10^3

	Boron Recovery System	Drain Channel A	Drain Channel B
Iodine	1×10^5	1×10^4	1×10^3
Cesium, Rubidium	2×10^3	1×10^5	1×10^4
Others	1×10^4	1×10^5	1×10^4

	All Nuclides Except Iodine	Iodine
Radwaste evaporator decontamination factor	10^4	10^3
Coolant radwaste system evaporator decontamination factor	10^3	10^2

Table 11-3 (Continued)

PRINCIPAL PARAMETERS AND CONDITIONS USED IN CALCULATING RELEASES OF RADIOACTIVE MATERIAL IN LIQUID AND GASEOUS EFFLUENTS FROM VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

	Anions	Cesium, Rubidium	Other Nuclides
Boron recycle feed demineralizer decontamination factor (H_3BO_3)	10	2	10
Primary coolant letdown demineralizer decontamination factor (Li_3BO_3)	10	2	10
Evaporator condensate polishing demineralizer (H^+ OH^-)	10	10	10
Mixed bed radwaste demineralizer	$10^2(10)$	2(10)	$10^2(10)$
Steam generator blowdown demineralizer	$10^2(10)$	10(10)	$10^2(10)$
Anion bed demineralizer (any system)	$10^2(10)$	1(1)	1(1)
Reactor building internal recirculation system charcoal adsorber efficiency percentage (iodine removal)			90
Condenser air removal system charcoal adsorber efficiency percentage (iodine removal)			70
Reactor building purge system charcoal adsorber efficiency percentage for high and low volume purges (iodine removal)			70
Auxiliary building exhaust charcoal adsorber efficiency percentage (iodine removal)			70
Reactor building internal recirculation and high and low volume purges and auxiliary building exhaust systems - high efficiency particulate air decontamination factor (particulate removal)			100

a = This value is constant and corresponds to 0.12 percent of the operating power fission product source term as given in NUREG-0017 (April 1976).

b = One percent per day of the primary coolant noble gas inventory and 0.001 percent per day of the primary coolant iodine inventory.

c = For two demineralizers in series, the decontamination factor for the second demineralizer is given in parentheses.

TABLE 11-4

COMPARISON OF CALCULATED DOSES TO A MAXIMUM INDIVIDUAL FROM VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1 OPERATION WITH APPENDIX I TO 10 CFR PART 50 DESIGN OBJECTIVES[a]

Criteron	RM-50-2 Objective	Appendix I Design Objective	Calculated Doses
Liquid effluents			
Dose to total body from all pathways	5 millirem per year per site	3 millirem per year	c
Dose to any organ from all pathways	5 millirem per year per site	10 millirem per year	c
Noble gas effluents (at site boundary) (one mile south)			
Gamma dose in air	10 millirad per year per site	10 millirad per year	c
Beta dose in air	20 millirad per year per site	20 millirad per year	c
Dose to total body of an individual	5 millirem per year per site	5 millirem per year	c
Dose to skin of an individual	15 millirem per year per site	15 millirem per year	c
Dose to any organ from all pathways (residence/garden) (1.2 miles east)	15 millirem per year per site	15 millren per year	c

a = Appendix I to 10 CFR Part 50 design objectives from Sections II.A, II.B, and II.C of Appendix I to 10 CFR Part 50; considers doses to maximum individual per reactor unit. From Federal Register, Volume 40, page 19442, May 5, 1975.

b = Carbon-14 and tritium have been added to this category.

c = Based on the releases given in Tables 11-1 and 11-2 of this Safety Evaluation Report and the releases and calculated doses given in the Draft Environmental Statement, we conclude that these doses will be less than the applicable RM-50-2 objectives. We will report what these doses are calculated to be in a supplement to this Safety Evaluation Report.

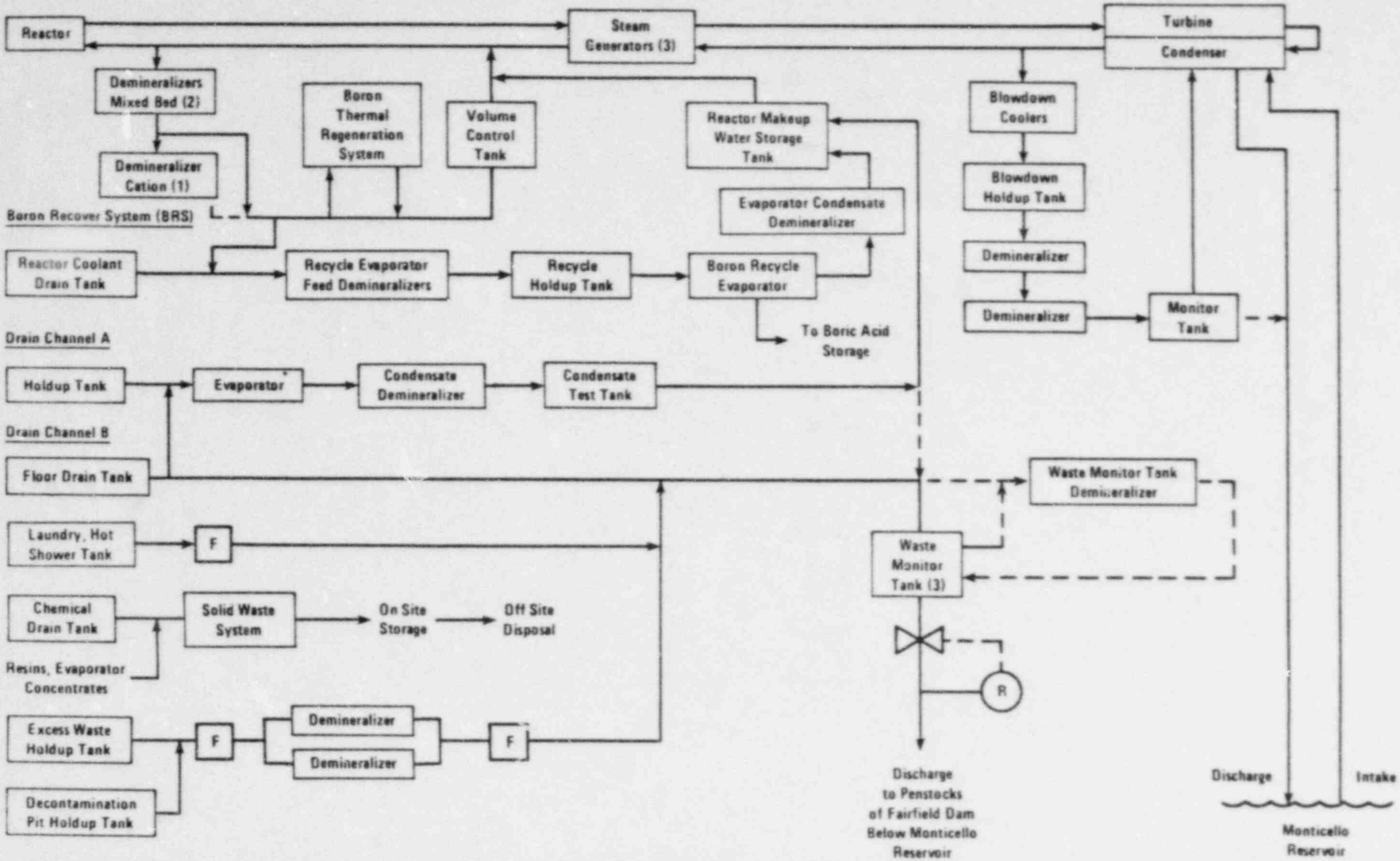


FIGURE 11-1 LIQUID RADIOACTIVE WASTE SYSTEMS V.C. SUMMER NUCLEAR PLANT.

for boron control, the boron recovery system will process a portion of the letdown flow (shim bleed) from the chemical and volume control system along with wastes collected in the reactor coolant drain tank. In our evaluation model, we assumed that a portion of the boron recovery system will be released through the liquid waste processing system for tritium control.

Nuclear Blowdown Processing System

Steam generator blowdown is processed through the nuclear blowdown processing system. The blowdown is cooled through heat exchangers to below the flash point and is piped to the blowdown holdup tank. The blowdown then passes through two mixed bed demineralizers in series and is collected in a monitor tank prior to being recycled to the main condenser hotwell or prior to being discharged to the penstocks of the Fairfield Pumped Storage Facility.

Laundry, hot shower, and decontamination wastes are normally released without treatment; the non-tritiated waste subsystem (drain channel B) will be used to treat effluents from these sources when radioactivity concentrations are in excess of predetermined levels.

Chemical and Volume Control System and Boron Recovery System

A letdown stream of approximately 60 gallons per minute of primary coolant is removed from the reactor primary coolant system for processing through the chemical and volume control system. The letdown stream is cooled through the letdown heat exchangers, reduced in pressure, processed through one of two mixed bed demineralizers, and sent to the volume control tank. A third demineralizer containing cation resins is used intermittently for lithium and cesium control.

The facility is designed to be capable of operating either as a base-load or as a load-following facility. During base-load operation, the reactor primary coolant boron concentraion is controlled during core life by bleed and feed operation, using the boron recovery system. During load-following operation, the boron concentration is controlled during core life by the boron thermal regeneration demineralizer system. At the end of core life, whether the reactor is operated in a base-load or load-following mode, boron concentration will be controlled by a boron thermal regeneration demineralizer, utilizing fresh resin.

In the applicant's Appendix I to 10 CFR Part 50 evaluation, base-load operation and use of the boron recovery system was assumed. This assumption is more conservative with respect to quantities of liquid radwastes generated than is the assumption of load-following operation using the boron thermal regeneration demineralizers; therefore, we also assumed base-load operation in our evaluation.

We estimated the boron recovery system input from the chemical and volume control system to be approximately 1440 gallons per day. Primary coolant-grade water from equipment drains and equipment leakage in containment is collected in the reactor coolant drain tank. We estimated the boron recovery system input from the reactor coolant drain tank to be approximately 300 gallons per day. The 1400-gallon per day reactor shim bleed and the 300-gallon per day reactor input from the reactor coolant drain tanks are processed through a 30 cubic foot mixed bed demineralizer and collected in a 42,000-gallon recycle holdup tank. The contents of the recycle holdup tank are then processed through a 15-gallon per minute boric acid evaporator and a 20 cubic foot anion bed condensate polishing demineralizer. A stripper column removes dissolved gases from the vapor body zone of the evaporator. The processed liquid is returned to the primary coolant system, stored in a holdup tank, or released to the penstocks of the Fairfield Pumped Storage Facility via the liquid waste processing system discharge header. In our evaluation, we assumed that approximately 10 percent of the processed liquid from the boron recovery system is released to the penstocks of the Fairfield Pumped Storage Facility.

Drain Channel A

Drain channel A processes tritiated wastes from equipment leaks and drains, valve leakoffs, pump seal leakoffs, tank overflows, and other tritiated and aerated water sources. The waste is collected in a 10,000-gallon collection tank (filled to 40 percent capacity). Liquids collected in this tank are processed through a 15-gallon per minute evaporator and a 42 cubic foot, 35-gallon per minute mixed bed demineralizer. The processed liquid is returned to the primary coolant system, stored in a holdup tank for reuse, or pumped to a waste monitor tank for monitoring and discharge. In our evaluation, we assumed that approximately 10 percent of the treated process stream from drain channel A is released to the penstocks of the Fairfield Pumped Storage Facility. The remaining 90 percent is assumed to be recycled to the reactor makeup water storage tank for reuse.

Drain Channel B

Aerated wastes and nontritiated wastes are processed through drain channel B subsystems for discharge to the environment. These wastes include floor drains, laundry and hot shower drains, and other non-reactor-grade sources. Drain channel B equipment includes one 10,000-gallon floor drain tank, a filter, one 600-gallon chemical drain tank, one 10,000-gallon laundry and hot shower tank, two 5,000-gallon waste monitor tanks, one 42-cubic foot, 35-gallon per minute waste monitor tank demineralizer, and one 15-gallon per minute waste evaporator. The wastes (excluding laundry and hot shower wastes) are collected in the floor drain tank at an input flow rate of 1,340 gallons per day. We assumed that these wastes are processed through the waste evaporator and calculated the decay time during processing to be approximately three days. The process liquids collected in the waste monitor tank are discharged into the penstocks of the Fairfield Pumped Storage Facility.

Laboratory and sample drains are segregated and collected for disposal or treatment based upon their origins. In the radiochemical laboratory, primary coolant sample wastes are drained to the chemical drain tank and subsequently solidified for disposal; rinse water and non-reactor-grade sample wastes are drained to the floor drain tank and are processed in drain channel B. In the sample room, excess purges of primary coolant samples are drained to the drain channel A waste holdup tank for recycle; non-reactor-grade sample wastes are drained to the floor drain tank and are processed in drain channel B.

Excess Liquid Waste System

An excess liquid waste processing and storage subsystem is provided to supplement the normal liquid waste processing system. The excess liquid waste system can accept excess liquid wastes from drain channel A, drain channel B, and laundry and hot shower inputs when the normal collection tanks are filled to capacity. The excess liquid waste system consists of two 10,000 gallon storage tanks, two pumps, a prefilter, two redundant 30 cubic feet, 35-gallon per minute mixed bed demineralizers, and a final filter. Excess liquid wastes processed in the excess liquid waste system are collected in a monitor tank for analysis prior to discharge.

Laundry and Hot Shower Drain System

Laundry and hot shower drains are normally released without treatment after filtration and monitoring for radioactivity. This waste is collected in a 10,000-gallon laundry and hot shower tank. The drain channel B subsystem can process these wastes should radioactivity measurements indicate activity levels above a predetermined value.

Nuclear Blowdown Processing System

The steam generator blowdown is processed through the nuclear blowdown processing system. The nuclear blowdown processing system cools the blowdown through heat exchangers, reduces the pressure and then processes the waste through one of two demineralizer trains. Each train contains a 150 cubic foot mixed bed primary demineralizer and a 90 cubic foot mixed bed polishing demineralizer in series. The treated stream is normally recycled to the main condenser hotwell with provisions for release to the main condenser cooling water discharge tunnel when the effluent stream has a concentration below a predetermined level. The steam generator blowdown treatment system has a design processing capacity of 250 gallons per minute.

Condensate Cleanup System

The secondary coolant system contains a condensate cleanup system, consisting of three powdered resin filter/demineralizers (one is a spare) sized for 50 percent of the maximum condensate flow. It will be used during startup and as required during condenser leakage. Normally, the filter/demineralizers will be backwashed 18 times per year to a 12,000 gallon backwash holdup tank. Representative samples will be taken from the tank and subjected to gamma isotopic analysis prior to batch release. If the analysis shows that the activity levels are acceptable for release, the slurry will be discharged to a settling pond. The applicant will be required to analyze both solid and liquid samples prior to controlled release to the settling pond. If release to the settling pond is not acceptable, the applicant proposes to solidify the slurry using an urea-formaldehyde system. The solid waste will be disposed at a authorized burial facility. The acceptability of urea-formaldehyde solification has not been established; but will be evaluated by the staff as part of the process control program. We consider the system design capacity to be adequate for meeting the needs of the plant during normal operation including anticipated operational occurrences and therefore acceptable.

Conformance with Federal Regulations and Branch Technical Positions

The liquid radioactive waste treatment system is located in the auxiliary building which is designed to seismic Category I criteria. The seismic design and quality group classifications and capacities of principal components considered in the liquid radwaste system evaluation are listed in Table 11-5. The seismic and quality group classifications of the liquid radioactive waste treatment system equipment are based upon criteria which were acceptable during the construction permit licensing stage, i.e., Quality Group C, seismic Category I design, for components of tritiated waste systems, and Quality Group D classification, nonseismic design for components of non-tritiated systems. Although these criteria differ from the current criteria contained in Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," we have determined that the provisions incorporated into the design of the liquid radioactive waste treatment system equipment are acceptable under the guidelines of Regulatory Guide 1.143.

The system design includes measures intended to control the release of radioactive materials due to potential overflows from indoor and outdoor storage tanks. Tank levels are monitored either locally or in the control room and high level alarms will be activated should preset levels be exceeded. Overflow provisions such as sumps, dikes and overflow lines permit the collection and subsequent processing of tank overflow. We consider these provisions to be capable of controlling the release of radioactive materials to the environment.

We have determined that during normal operation, the proposed liquid radioactive waste treatment system is capable of reducing the release of radioactive materials in liquid effluents to approximately 0.26 curie per year, excluding tritium and dissolved gases, and 360 curies per year for tritium. The calculated annual releases of radionuclides in liquid effluents are given in Table 11-2.

Using the source terms given in Table 11-2, we estimate the total body dose to an individual in an unrestricted area to be less than three millirems per year, or any organ dose to be less than 10 millirems per year, in accordance with Section II.A of Appendix I to 10 C Part 50, as shown in the comparison Table 11-4.

Also, we calculate the release of radioactive material in the liquid effluents, exclusive of tritium and dissolved gases, to be less than five curies per year, and the total body and any organ dose to be less than five millirems per year from the facility, in accordance with the option of Section II.D of Appendix I to 10 CFR Part 50 as provided in the Annex to Appendix I to 10 CFR Part 50, as shown in the comparison Table 11-4. We conclude that the liquid radioactive waste treatment system is capable of reducing liquid radioactive effluents to as low as is reasonably achievable levels in accordance with Section 50.34a of 10 CFR Part 50, Appendix I to 10 CFR Part 50, and the Annex to Appendix I to 10 CFR Part 50.

We have determined that the liquid radoactive waste treatment system is capable of reducing the release of radioactive materials in liquid effluents to concentrations below the limits in 10 CFR Part 20, during periods of fission product leakage from the fuel at design levels.

TABLE 11-5

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS CONSIDERED IN THE EVALUATION OF LIQUID AND GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEMS

Component	Number	Size	Capacity, each	Quality Group	Seismic Category
Liquid Systems					
Chemical and volume control system					
Volume control tank	1	350 cubic feet	-	C	I
Mixed bed demineralizer	2	30 cubic feet	120 gallons per minute	C	I
Cation bed demineralizer	1	20 cubic feet	60 gallons per minute	C	I
Thermal regeneration demineralizer	4	70 cubic feet	120 gallons per minute	C	I
Boron Recycle System					
ecycle Holdup tanks	2	-	42,000 gallons	C	I
ecycle evaporator feed demineralizer	2	30 cubic feet	120 gallons per minute	C	I
Recycle evaporator condensate demineralizer	1	20 cubic feet	35 gallons per minute	D	-
Recycle evaporator package	1	-	15 gallons per minute	C	I
Drain Channel A System					
Waste holdup tank	1	-	10,000 gallons	C	I
Waste evaporator	1	-	15 gallons per minute	C	I
Evaporator condensate demineralizer	1	30 cubic feet	35 gallons per minute	D	-
Evaporator condensate tank	1	-	5,000 gallons	D	-
Drain Channel B System					
Floor drain tank	1	-	10,000 gallons	D	-
Waste monitor tank demineralizer	1	30 cubic feet	35 gallons per minute	D	-

TABLE 11-5 (Continued)

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS CONSIDERED IN THE EVALUATION OF LIQUID AND GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEMS

Component	Number	Size	Capacity, each	Quality Group	Seismic Category
Excess Waste System					
Demineralizer	2	30 cubic feet	35 gallons per minute	D	-
Steam Generator Blowdown and Other Systems					
Primary demineralizer	2	150 cubic feet	-	D	-
Polishing demineralizer	2	90 cubic feet	-	D	-
Blowdown holdup tank	1	-	13,000 gallons	D	-
Condensate demineralizers (Powdex	3	-	4,411 gallons per minute	D	-
ackwash holdup tank	1	-	12,000 gallons	D	-
Gaseous Systems					
Gaseous Waste Processing System					
Waste gas compressors	2	40 standard cubic feet per minute	150 pounds per square inch, gauge	C	I
Catalytic recombiners	2	-	40 standard cubic feet per minute	C	I
Gas decay tanks	8	600 cubic feet	150 pounds per square inch, gauge	C	I

11.2.2 Gaseous Radioactive Waste Treatment System

The gaseous radioactive waste and facility ventilation exhaust systems are designed to collect, store, process, monitor, recycle, and/or discharge potentially radioactive gaseous wastes which will be generated during normal operation of the plant. The systems consist of equipment and instrumentation necessary to reduce releases of radioactive gases and particulates to the environment. The principal sources of gaseous waste are the effluents from the gaseous waste management system, condenser vacuum pumps, and ventilation exhausts from the reactor containment, auxiliary (including the fuel handling area) and turbine buildings.

The principal system for treating gaseous wastes will be the gaseous waste processing system. The gaseous waste processing system collects and stores the hydrogenated fission product gases stripped from the volume control tank, the boron recovery system evaporator, and the reactor coolant drain tank. Ventilation exhaust air from the containment, auxiliary building fuel handling area, and the condenser air removal system is processed through high efficiency particulate air filters and charcoal adsorbers prior to release to the environment. Ventilation exhaust air from the turbine building will be released to the environment without treatment.

The gaseous radioactive waste treatment and ventilation exhaust systems are shown schematically in Figure 11-2.

Gaseous Waste Processing System

The gaseous waste processing system is designed to collect and process gases stripped from the volume control tanks, the boron recovery system evaporator, and the reactor drain tanks. The gases are pumped and compressed through a recombiner into pressurized storage tanks for decay. Redundant 40 standard cubic feet per minute capacity compressors are provided for this purpose. There are eight storage tanks included in the gaseous waste processing system with a design pressure of 150 pounds per square inch, gauge and a volume of 600 cubic feet in each.

We calculated a holdup time of 90 days based upon the tank volume and operating pressure. Our evaluation assumed one tank held in reserve for back-to-back shutdowns, one tank in process of filling, and the remaining filled tanks held for decay. The discharges from the decay tanks will be passed through a high efficiency particulate air filter and charcoal adsorber prior to release to the environment. A radiation monitor will automatically terminate the discharge if radiation levels exceed a predetermined value in the discharge line.

Containment Ventilation System

Radioactive gases are released inside the containment when primary system components are opened or when primary system leakage occurs. During normal operation, the reactor building is purged continuously with a flow rate of 1000 cubic feet per minute. In our evaluation, we have included four shutdown purges per year. Prior to purging, the containment atmosphere is recirculated for 16 hours through the reactor building charcoal cleanup system which includes high efficiency particulate air filters and charcoal adsorbers. We assumed

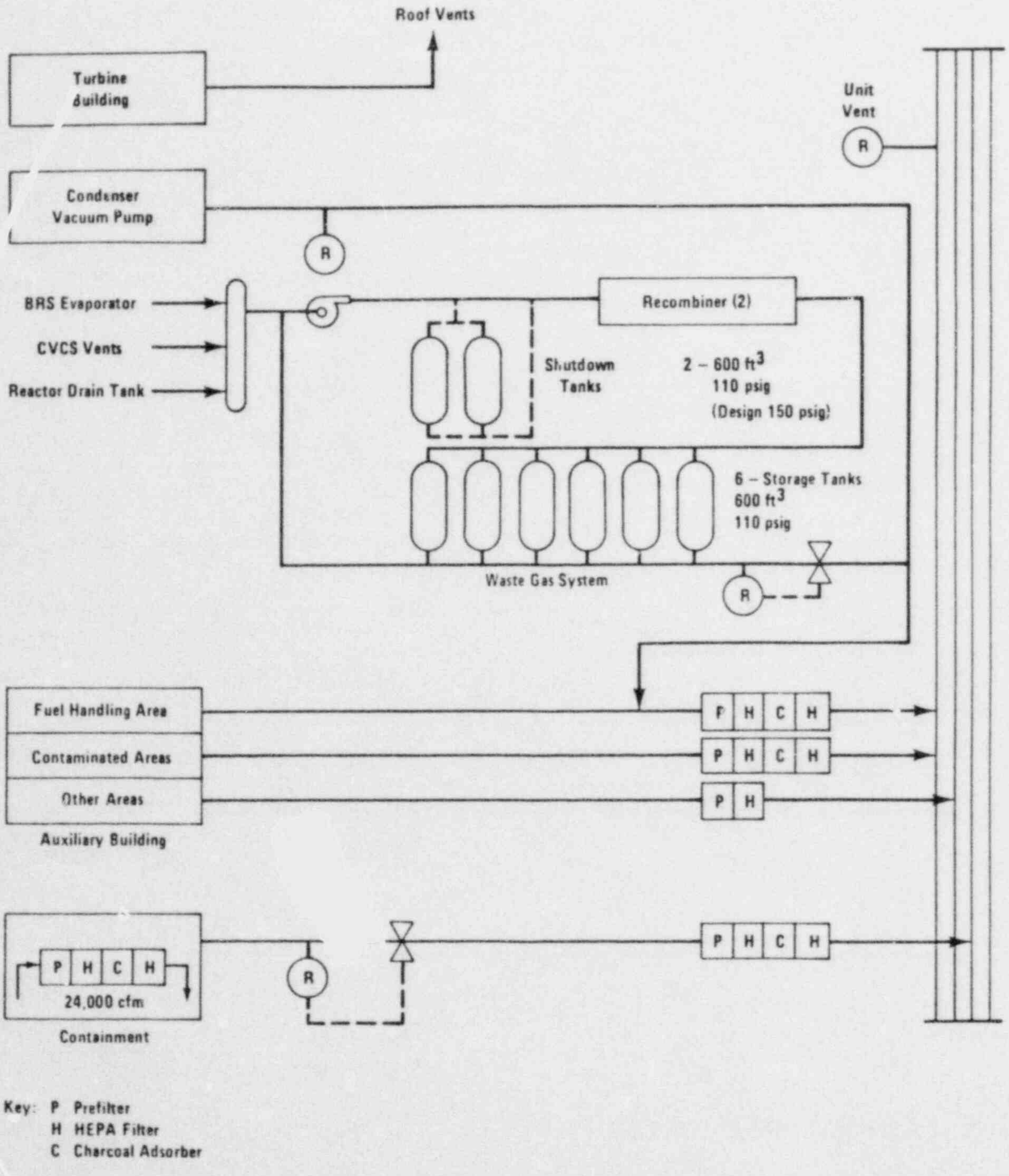


FIGURE 11-2 GASEOUS WASTE SYSTEM AND VENTILATION SYSTEM V.C. SUMMER NUCLEAR PLANT.

radionuclide removal during the recirculation phase to be based upon a flow rate of 24,000 cubic feet per minute, a mixing efficiency of 70 percent, and an iodine decontamination factor of 10 for charcoal adsorbers. We assumed that the reactor building purge exhausts are released to the environment through high efficiency particulate air filters and two inch thick charcoal adsorbers. Reactor building purge exhaust radioactivity monitors will automatically isolate the purge system upon detection of a radioactivity concentration above a predetermined level.

Ventilation Releases From Other Buildings

Radioactive materials will be introduced into the plant atmosphere due to leakage from equipment transporting or handling radioactive materials. We estimated that 160 pounds of primary coolant per day will leak to the auxiliary building with an iodine partition factor of 0.075. Small quantities of radionuclides will be released to the turbine building atmosphere based upon an estimated 1700 pounds per hour of steam leakage. Our calculations assumed that effluents from the auxiliary building, including the fuel handling area, will be processed through high efficiency particulate air filters and two inch thick charcoal adsorbers prior to release to the environment. The effluents from the turbine building will be released directly to the environment without treatment.

Main Condenser Vacuum Pump

Offgas from the main condenser vacuum pumps contains radioactive gases as a result of primary-to-secondary system leakage. In our evaluation, we assumed a primary-to-secondary system leak rate of 100 pounds per day. Noble gases and iodine are contained in the steam generator leakage and released to the environment through the main condenser vacuum pumps in accordance with the partition factors listed in Table 11-3. The vacuum pump exhaust is released to the environment through high efficiency particulate air filters and charcoal adsorbers.

Conformance with Federal Regulations and Branch Technical Positions

The seismic design and quality group classifications and capacities of the principal equipment in the gaseous waste processing system are listed in Table 11-5. We find the applicant's gaseous waste management system meets or exceeds the guidelines of Regulatory Guide 1.143 and is, therefore, acceptable.

The gaseous waste processing system is located in the auxiliary building which is a seismic Category I structure. We have compared the design, testing and maintenance of the high efficiency particulate air filters and charcoal adsorbers installed in normal ventilation exhaust systems with the guidelines of Regulatory Guide 1.140, Revision 1 (October, 1979) and conclude the system is acceptable.

The gaseous waste processing system is designed to prevent a hydrogen explosion. The gaseous waste processing system is monitored by dual hydrogen and oxygen analyzers with provision for automatic termination of oxygen feed if the oxygen concentration downstream of the recombiners exceeds 60 parts per million.

We find the applicant's proposed gaseous waste treatment and plant ventilation systems are capable of reducing the release of radioactive materials in gaseous effluents to approximately 3,100 curies per year for noble gases, 0.066 curie per year for iodine-131, 800 curies per year for tritium, eight curies per year for carbon-14, and 0.0042 curies per year for particulates. The calculated annual releases of radionuclides in gaseous effluents are given in Table 11-2.

Using the source terms given in Table 11-2, we estimate the annual air dose in an unrestricted area to be less than 10 millirads for gamma radiation and 20 millirads for beta radiation. We estimate the annual individual external doses from gaseous effluents in an unrestricted area to be less than 5 millirems to the total body and 15 millirems to the skin. We estimate the annual dose in an unrestricted area from all applicable pathways due to release of radioiodine and radioactive material in particulate form and tritium and carbon-14 to be less than 15 millirems to any organ. The off-site doses are in accordance with Section II.B and II.C of Appendix I to 10 CFR Part 50 (see Table 11-4).

Rather than perform an individualized cost-benefit analysis required by Section II.D of Appendix I to 10 CFR Part 50, the applicant elected to show conformance with the numerical design objectives specified in the September 4, 1975, amendment to Appendix I to 10 CFR Part 50 (RM-50-2). As shown in Table 11-2, the calculated quantity of iodine-131 released in gaseous effluents is less than one curie per year. Also, as shown in Table 11-4, the calculated doses for the Virgil C. Summer Nuclear Station, Unit 1 are less than the dose design objectives set forth in RM-50-2 and therefore satisfy the requirements of Section II.D of Appendix I to 10 CFR Part 50.

We conclude that the gaseous waste treatment and ventilation systems are capable of reducing releases of radioactive materials in gaseous effluents to "as low as is reasonably achievable" levels in accordance with Section 50.34a of 10 CFR Part 50 and Appendix I to 10 CFR Part 50.

We have determined that the proposed gaseous radioactive waste treatment system and ventilation exhaust systems are capable of reducing the release of radioactive materials in gaseous effluents to concentrations below the limits of 10 CFR Part 20 during periods of fission product leakage from the fuel at design levels.

11.2.3 Solid Radioactive Waste Treatment System

The solid radioactive waste treatment system is designed to process two general types of solid wastes: "wet" solid wastes which require solidification prior to shipment, and "dry" solid wastes which require packaging and, in some cases, compaction prior to shipment to a licensed burial facility.

The solid wastes consist mainly of spent filter cartridges, demineralizer resins, chemical samples, and evaporator bottoms which contain radioactive materials removed from liquid streams during processing. Wet solid wastes are combined with urea-formaldehyde solidification agent and catalyst in containers (50 cubic foot containers and 55-gallon drums) to form a solid matrix. The containers are subsequently sealed and placed in a shield, as required, for offsite shipment.

Dry solid wastes, consisting mainly of ventilation air filtering medium (charcoal), contaminated clothing, paper, rags, laboratory glassware, and tools, are packaged in 55-gallon drums.

Wet Solid Wastes

The principal sources of spent resins are eight 30 cubic foot and two 20 cubic feet liquid radwaste system demineralizers, four 70 cubic foot purification and deborating demineralizers, four steam generator blowdown purification demineralizers, and one spent fuel pool demineralizer. Spent resins from the demineralizers are collected and stored in two storage tanks in the primary and secondary system. Prior to packaging, the resin is sluiced to a 50 cubic foot container and dewatered before solidification. The resin beads are solidified by the addition of liquid waste, urea-formaldehyde agent and catalyst. Concentrated evaporatory waste is collected in a 5,000-gallon waste evaporator concentrate tank. Chemical samples are stored in a 600-gallon chemical drain tank. These wastes are mixed in an in-line mixer with a urea-formaldehyde solution. The mixture is pumped to a 50 cubic foot container where a catalyst is added to polymerize the urea-formalydehyde solution.

The applicant has described a process control program based upon sampling and analysis of each batch of waste to be solidified. The correct mix ratio of waste, solidification agent, and catalyst is selected upon evaluation of the analysis and upon the basis of solidification data supplied by the vendors of the packaging equipment and solidification agents. A small specimen of waste is then mixed with solidification agent and catalyst in the selected ratios to verify soldification; the actual solidification of the batch of waste will take place only after successful solidification of the specimen is demonstrated. The applicant will be required to submit his complete process control program for staff review and approval at the time of submission of the radiological effluent technical specifications.

On the basis of our evaluation and on recent data from operating pressurized water reactor plants, we have determined that approximately 15,000 cubic feet per unit of wet solid wastes, containing approximately 860 curies of activity, will be long-lived fission and corrosion products, mainly cesium-137, cobalt-58, cobalt-60 and iron-55.

Dry Solid Wastes

Dry solid wastes will be packaged in 55-gallon drums. Compressible wastes such as clothing, paper and rags will be compressed prior to packaging. During the baling operation, the drum and compacting mechanisms are enclosed. The enclosure is vented to the auxiliary building atmosphere through a high efficiency particulate air filter by a blower to reduce the potential for airborne radioactive dust. We estimate the dry solid wastes will total 10,000 cubic feet per year with a total activity content of five curies per year.

Conformance with Federal Regulations and Branch Technical Positions

The solid radioactive waste treatment system is housed in the auxiliary building and conforms to the design, construction and testing criteria of Regulatory Guide 1.143. The auxiliary building is designed to seismic Category I criteria.

In addition, the solid radwaste system provides for waste storage in accordance with Branch Technical Position ETSB 11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants." Storage facilities include an area in the auxiliary building for approximately 20 shipping containers (50 cubic feet each of high-level waste, and 60 55-gallon drums of low-level waste. The space provides approximately 30 days storage for high-level wastes and 90 days storage for low-level wastes. We find the storage capacity adequate for meeting the demands of the facility for normal operation.

The solidification of radioactive waste of the Virgil C. Summer Nuclear Station, Unit 1 will be performed in accordance with a process control program and, therefore, conforms with Branch Technical Position ETSB 11-3.

On the basis of our evaluation of the solid radioactive waste treatment system, we conclude that the system design will accommodate the wastes expected during normal operations, including anticipated operational occurrences.

The packaging and shipping of all wastes will be in accordance with the applicable requirements of 10 CFR Parts 20 and 71, and 49 CFR Parts 170-178. From these findings, we conclude that the solid radioactive waste treatment system is acceptable.

* 11.3 Process and Effluent Radiological Monitoring and Sampling Systems

The process and effluent radiological monitoring and sampling systems are designed to provide information concerning radioactivity levels in systems throughout the facility, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in discharges from the facility to the environment. The liquid and gaseous effluent streams are continuously monitored and sampled for radioactivity. Monitors on liquid and gaseous effluent release lines automatically terminate discharges should radiation levels exceed a predetermined value. Table 11-6 indicates the proposed location, number, type, and sensitivity of each continuous monitor. Systems which are not amenable to continuous monitoring or for which detailed isotopic analyses are required are sampled and analyzed in the plant laboratory. The sampling system provides representative liquid and gaseous samples to effectively monitor the operation of the facility, and provides isotopic analysis for determining the radioactive materials in liquid and gaseous effluents. Sample points are located at each tank in the liquid radwaste treatment system for sampling tank contents both before and after each processing step. In the gaseous radwaste treatment system, sample points are located at each gas decay tank and in the facility vent.

We have reviewed the locations and types of effluent and process monitoring and sampling provided. Based upon the facility design and upon the continuous monitoring locations and sampling locations, we have concluded that all normal and potential release pathways are monitored. We have also determined that the sampling and monitoring provisions are adequate for detecting radioactive material leakage to normally uncontaminated systems and for monitoring facility processes which affect radioactivity releases. On this basis, we consider the monitoring and sampling provisions to meet the requirements of Criteria 60, 63, and 64 of the General Design Criteria and the guidelines of Regulatory

TABLE 11-6

PROCESS AND EFFLUENT MONITORS

Stream Monitored	Type Detector	Range*
Component cooling water	scintillation	2×10^{-6} to 2×10^{-2} microcuries per milliliter
Steam generator blowdown process monitor	scintillation	2×10^{-6} to 2×10^{-2} microcuries per milliliter
Spent fuel cooling water	scintillation	2×10^{-6} to 2×10^{-2} microcuries per milliliter
Liquid waste effluent**	scintillation	2×10^{-6} to 2×10^{-2} microcuries per milliliter
Boron recycle system discharge**	scintillation	2×10^{-6} to 2×10^{-2} microcuries per milliliter
Nuclear blowdown waste effluent**	scintillation	2×10^{-6} to 2×10^{-2} microcuries per milliliter
Turbine room sump monitor**	scintillation	2×10^{-6} to 2×10^{-2} microcuries per milliliter
Liquid waste plant discharge**	scintillation	2×10^{-6} to 2×10^{-2} microcuries per milliliter
Steam generator blowdown discharge**	scintillation	2×10^{-6} to 2×10^{-2} microcuries per milliliter

TABLE 11.6 (Continued)

PROCESS AND EFFLUENT MONITORS

Stream Monitored		Type Detector	Range*
Gases			
Main plant vent discharge	gas:	beta scintillation	2×10^{-6} to 2×10^{-2} microcuries per cubic centimeter
	particulate:	beta scintillation	10^{-11} to 10^{-7} microcurie per cubic centimeter
	iodine:	beta scintillation	3×10^{-11} to 10^{-7} microcuries per cubic centimeter
Reactor building purge exhaust**	gas:	beta scintillation	2×10^{-6} to 2×10^{-2} microcuries per cubic centimeter
	particulate:	beta scintillation	10^{-11} to 10^{-7} microcuries per cubic centimeter
	iodine:	scintillation	3×10^{-11} to 10^{-7} microcuries per cubic centimeter
Condenser exhaust	gas:	scintillation	4×10^{-6} to 4×10^{-2} microcuries per cubic centimeter
Waste gas discharge**	gas:	beta scintillation	2×10^{-4} to 2×10^{0} microcuries per cubic centimeter

*All liquid and gaseous effluent streams will be monitored in accordance with the guidelines of Regulatory Guide 1.21.

**Terminates discharge or diverts discharge to holdup system when the radioactivity level exceeds a predetermined value.

Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants".

11.4 Evaluation Findings

In our evaluation, we have calculated releases of radioactive materials in liquid and gaseous effluents for normal operation including anticipated operational occurrences based upon expected radwaste inputs over the life of the facility.

In our evaluation, we determined that the applicant's proposed design of the liquid and gaseous waste treatment systems satisfies the design objectives of Appendix I to 10 CFR Part 50.

We conclude that the liquid and gaseous radioactive waste treatment systems will reduce radioactive materials in effluents to "as low as is reasonably achievable" levels in accordance with Section 50.34a of 10 CFR Part 50 and are, therefore, acceptable.

We have considered the potential consequences resulting from reactor operation with a one percent operating power fission product source term and determined that under these conditions, the concentrations of radioactive materials in liquid and gaseous effluents in unrestricted areas will be a small fraction of the limits specified in 10 CFR Part 20.

We have considered the capabilities of the radioactive waste treatment systems to meet the anticipated demands of the facility due to anticipated operational occurrences and have concluded that the liquid, gaseous and solid waste system capacities, and design flexibilities are adequate to meet the anticipated needs of the facility.

We have reviewed the applicant's quality assurance provisions for the radioactive waste treatment systems, the quality group classification used for system components, the seismic design applied to the design of the gaseous waste processing system, and the seismic design applied to the design of structures housing these systems. The design of the radioactive waste treatment system and the structures housing the system meet the acceptance criteria as set forth in Regulatory Guide 1.143.

We have reviewed the provisions incorporated into the applicant's design to control the releases of radioactive materials in liquids due to inadvertent tank overflows and conclude that the measures proposed by the applicant are consistent with our acceptance criteria as set forth in Regulatory Guide 1.143.

Our review of the radiological process and effluent monitoring system included the provisions of sampling and monitoring all normal and potential effluent discharge paths in conformance with Criterion 64 of the General Design Criteria, for providing automatic termination of effluent releases and assuring control over releases of radioactive materials in effluents in conformance with Criterion 60 of the General Design Criteria and Regulatory Guide 1.21, for sampling and monitoring facility waste process streams for process control in conformance with Criterion 63 of the General Design Criteria, for conducting sampling and analytical programs in conformance with the guidelines of Regulatory Guide 1.21, and for monitoring process and effluent streams during postulated accidents. The review included piping and instrument diagrams and process flow diagrams for the liquid, gaseous and solid radioactive waste treatment systems, ventilation systems, and the location of monitoring points relative to effluent release points. We conclude that the applicant's radiological process and effluent monitoring systems are acceptable.

Based upon the foregoing evaluation, we conclude that the proposed radwaste treatment and monitoring systems are acceptable. The basis for acceptance has been conformance of the applicant's design, design criteria, and design bases for the radioactive waste treatment and monitoring systems to the applicable regulations and regulatory guides referenced above, as well as the NRC staff technical positions and industry standards.

12 RADIATION PROTECTION

This section presents an evaluation of the adequacy of the radiation protection design features and the health physics program at the facility to control radiation exposures within the limits of 10 CFR Parts 20 and 50. We have evaluated the proposed radiation protection program presented in Chapter 12 of the Final Safety Analysis Report. The radiation protection measures incorporated at the facility are intended to "ensure that internal and external radiation exposures to station personnel, contractor, and the general population due to station conditions, including anticipated operational occurrences, will be within applicable limits, and furthermore, will be as low as is reasonably achievable."

The basis of our acceptance of the applicant's radiation protection program is that doses to personnel will be maintained within the limits of 10 CFR Part 20, "Standards for Protection Against Radiation." The applicant's radiation protection designs and program features are also consistent with the guidelines of Regulatory 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," Revision 3. Some of the radiation protection measures which the applicant will use at the facility include: location of radioactive components in separately shielded cubicles; use of remote handling equipment, ventilation systems des .ned for easy access and service to minimize doses during maintenanc .econtamination, filter changes; use of permanent radiation monitoring s .tems: and training of personnel in radiation protection. The applicant's use of these and other radiation protection features will help assure that occupational radiation exposures are maintained as low as is reasonably achievable, both during facility operation and during decommissioning.

On the basis of our review of the Final Safety Analysis Report, we conclude that the radiation protection measures incorporated in the design will provide reasonable assurance that occupational doses will be maintained as low as is reasonably achievable and below the limits of 10 CFR Part 20. These radiation design features are also consistent with the guidelines of Regulatory Guide 8.8.

12.1 Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)

The applicant has provided in the Final Safety Analysis Report a management commitment to assure that the facility is designed, constructed, and operated in a manner consistent with Regulatory Guides 8.8, 8.10, "Operating Philosophy for Maintaining Occupational Exposures As Low As Is Practicable," Regulatory Guide 1.8, "Personnel Selection and Training," Revision 1, and 10 CFR Part 20. The "as low as is reasonably achievable" philosophy was applied during the initial design of the facility. Since then, the applicant has continued to review, update, and modify the facility design during the ensuing design and construction phases. Onsite inspections are conducted to check the shielding and piping layout design. The objective of these design reviews and inspections is to assure that the personnel exposures at the facility will be maintained as low as is reasonably achievable.

The Health Physics Supervisor has the responsibility to assure that radiation exposures are maintained as low as is reasonably achievable. That individual is responsible for developing the radiation protection training program, the radiation surveillance program, and health physics procedures to assure that exposures of all personnel are kept within the limits of 10 CFR Part 20 and are as low as is reasonably achievable. These procedures and programs are developed incorporating Regulatory Guide 8.2, "Guide for Administrative Practice in Radiation Monitoring," Regulatory Guides 1.8, 8.8, 8.10 and 10 CFR Pa t 20. Therefore, the policy considerations are acceptable.

To reduce radiation exposures, the applicant has incorporated general considerations into the design to reduce 1) the need to enter radiation fields, 2) the time of exposure when entry is necessary, and 3) the dose rate during exposure. These general considerations are implemented by specific radiation protection design guidelines. Also, information gained from the applicant's study of operating experience from power reactors is factored into the design. Finally, design reviews are performed by radiation protection personnel to ensure that occupational radiation exposures will be as low as is reasonably achievable. These design considerations are consistent with the guidance of Regulatory Guide 8.8. Therefore, the design considerations are acceptable.

Since the construction permit stage, the applicant has incorporated the following facility and equipment design features at the facility to satisfy the design objectives and minimize radiation exposures.

1. A Permaili shield was added to prevent neutron streaming to an area outside the secondary shield.

2. The gap between the reactor building and fuel handling building was shielded to prevent streaming when spent fuel assemblies are transferred from the reactor building to the fuel handling building.

3. The wall separating the mixed bed demineralizers in the chemical and volume control system was increased in thickness to lower the potential dose from the adjacent demineralizer.

4. The sample sink was relocated to make it possible to put the sample lines and the sample vessels in a shielded chase.

5. The solid radwaste area was modified to provide separate storage areas for high- and low-level waste.

6. The decontamination area in the hot machine shop was increased and additional equipment was added to provide more capability for equipment and tool decontamination.

7. The original design included a provision for processing blowdown through the cycle makeup demineralizers. This provision was deleted to preclude the possibility of a significant radiation source occurring in an unlimited access area.

8. Monitors were relocated to a lower radiation area.

These design considerations in developing operational procedures to maintain exposures as low as is reasonably achievable are acceptable. The applicant has committed to include in the operational procedures measures for reducing exposure and the criteria for implementation of those measures consistent with the guidance of Regulatory Guide 8.8. We conclude that the operating and maintenance personnel at the facility will follow specific plans and procedures in order to assure that as low as is reasonably achievable goals are achieved in the operation of the nuclear station and, therefore, the program is acceptable.

12.2 Radiation Sources

Sections 12.1.3 and 12.2.3 of the Final Safety Analysis Report describes the sources of contained and airborne radioactivity used as inputs for the dose assessment and for the design of the shielding and ventilation systems. The methods and bases used by the applicant to estimate the source terms are also described. A more detailed description of the source term development is presented in Chapter 11 of the Final Safety Analysis Report.

The location and strengths of the contained radiation sources which must be shielded or included in the dose assessment are provided. The bases for the source terms meet our acceptance criteria as described below. The fission product source terms are based on a failed fuel fraction of 0.01. The coolant and corrosion activation product source terms are based on measurements at operating pressurized water reactors, and they are consistent with American National Standard N237-1976, "Source Term Specification." Neutron and prompt gamma source terms are based on reactor core physics calculations and operating reactor experience. The contained radiation source terms presented are comparable to estimates by other applicants with pressurized water reactor designs and are acceptable.

The reactor core is the primary source of radiation in the containment, emitting neutrons and gamma rays. The reactor coolant system is the next highest source of radiation in the containment. The reactor coolant contains fission products from fuel clad defects and activation and corrosion products. Of these radiation sources, nitrogen-16 is the predominant activity in the reactor coolant pumps, steam generators, and the reactor coolant piping. In buildings other than the containment, the primary sources of personnel exposure are fission products, activation and corrosion products, and spent fuel assemblies in the fuel building. The shielding used to protect personnel from these sources is based on fission source terms for full-power operation with one percent fuel cladding defects. Other parameters used, as well as a complete description of source term development, are contained in Chapter 11 of the Final Safety Analysis Report. The source terms presented are comparable to estimates by other applicants with similar designs, and we conclude they are acceptable.

The applicant has provided a tabulation of the normal expected radioactive concentrations in all of the applicable regions due to equipment leakage. The bases for these leakage calculations are in accordance with Regulatory Guide 1.112, "Calculation of Releases of Radioactivity Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors." The ventilation system will route air from areas of low potential contamination to areas of increasing potential airborne contamination. The amount of uncontrolled exfiltration from an area will be minimized by exhausting a greater volumetric flow than is supplied to an area. The resulting expected airborne isotopic concentrations in all applicable regions will be well below the maximum permissible concentrations for occupational workers. We conclude that the radiation sources for the facility are acceptable and in compliance with the criteria of Section 12.2 of the Standard Review Plan.

12.3 Radiation Protection Design Features

Sections 12.0, 12.1.1, 12.1.2, 12.1.4, 12.2.1, 12.2.2, and 12.2.4 of the Final Safety Analysis Report describe the features which are included in the radiation protection design of the plant to maintain occupational exposures as low as is reasonably achievable. Separate descriptions are presented for the categories of facility design features, shielding, ventilation, and area radiation and airborne monitoring instrumentation.

The applicant has provided evidence that the dose accumulating functions performed by workers have been considered in the facility design. Features have been included in the design to help maintain exposure as low as is reasonably achievable in the performance of those functions. These features will facilitate access to work areas, reduce or allow the reduction of source intensity, reduce the time required in radiation fields, and provide for portable shielding and remote handling tools. The applicant's facility design features are consistent with the guidelines of Regulatory Guide 8.8. Therefore, we conclude that the facility design features are acceptable. The applicant has provided five radiation zones as a basis for classifying occupancy and access restrictions on various areas. On this basis, maximum design dose rates are established for each zone and used as input for shielding of the respective zones.

The areas inside the restricted area are divided into a number of radiation dose rate zones for design purposes. The areas that will have to be occupied on a predictable basis during normal operations and anticipated occurrences are zoned such that exposures will be below the limits of 10 CFR Part 20 and will be as low as is reasonably achievable. The zoning system and access control features also meet the posting and entry requirements of Section 20.203 of 10 CFR Part 20. Therefore, we conclude that the design dose rate zone system is acceptable.

Several features are included in the plant design and operational program to minimize the buildup of activated corrosion products--a major contributor to occupational dose. The use of high cobalt, hard facing wear materials in the primary system has been limited to those places where it is necessary. Steam generators use Inconel tubing with a cobalt content less than 0.1 percent maximum to minimize the cobalt-60 source in the primary system. Valve and pipe connections have been designed to minimize this buildup.

Control of chemistry in the primary system will also minimize this buildup. Therefore, we conclude that the applicant has given acceptable consideration to the inclusion of design features to minimize the buildup of activated corrosion products.

In response to our question, the applicant has included features in the radiation protection design specifically for the purpose of maintaining doses as low as is reasonably achievable during decommissioning. However, many of the features included in the design to reduce doses during operation will also help reduce doses during decommissioning. The applicant estimates that specification and limitation on cobalt content in equipment components will serve to limit radiation doses from crud buildup during both operation and subsequent decommissioning. We estimate that the collective occupational dose due to decommissioning will be of the same order of magnitude as annual doses due to operation. Therefore, we conclude that the applicant has given acceptable consideration to the issue of personnel exposure during decommissioning.

The shielding was designed to meet the requirements of the radiation dose rate zone system discussed above. The applicant's shielding design methods, including the use of source terms, cross-section data, shield and source geometries, and radiation transport calculational schemes, are consistent with generally accepted practice. We checked several of the shields and drawings presented by the applicant to ensure that the shield design is acceptable. Also, the shield design and construction will be consistent with the guidelines of Regulatory Guide 8.8 and Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants."

The ventilation system is designed to assure that airflow will be from areas of low potential for airborne radioactivity to areas of higher potential and then to filters or vents. Also, the system will maintain concentrations of airborne radioactivity in normally occupied areas within the limits of 10 CFR Part 20 and 10 CFR Part 50. The ventilation filter trains are designed to allow exposures to be maintained as low as is reasonably achievable during servicing, consistent with the guidance of Regulatory Guide 8.8. Therefore, we conclude that the ventilation system radiation protection design features are acceptable.

Detectors for the area radiation monitoring system will be located in normally occupied areas which have the potential for radiation fields in excess of the maximum design radiation dose rate. The detectors are designed to cover the expected and maximum design dose rates and dose rates due to anticipated operational occurrences. The monitors will have readout and annuciation in the control room. The monitors will also have variable alarm setpoints and local audible alarms. The detectors will be calibrated at least annually. Therefore, we conclude that the area radiation monitoring system design is acceptable.

The applicant has provided area radiation monitors around the fuel storage areas to meet the requirements of Section 70.24 of 10 CFR Part 70 and to be consistent with the guidance of Regulatory Guide 8.12, "Criticality Accident Alarm Systems."

The applicant will rely on the area radiation monitoring system and portable radiation monitoring instruments to assess the radiation hazard to personnel in areas which may be accessed during the course of an accident. The area monitors will receive backup power from the diesel generators. The portable instruments will be placed to be readily accessible to personnel responding to an emergency. The portable instruments will be designed with a sufficient instrument range for use in the event of an accident. The area radiation monitoring system will include a high range monitor which will be positioned to monitor the containment. The monitor will be inside a capped containment wall penetration and protected from the potentially severe environment inside containment during an accident. Based on the shielding provided by penetration cap and containment wall, the monitor may be used to infer dose rates inside containment. We conclude that the accident radiation monitoring system is acceptable.

Our acceptance criterion in the Standard Review Plan for airborne radioactivity monitoring systems states that air should be sampled at normally occupied locations where airborne radioactivity may exist. The airborne radioactivity monitoring system as described in the Final Safety Analysis Report meets our ten-hour maximum permissible concentration criterion. The applicant has provided a sequential monitor capable of detecting airborne particulates and iodine radioactivity which alarms in the control room. We conclude that the airborne radioactivity monitoring system is acceptable.

12.4 Dose Assessment

The estimates of annual man-rem exposure are based on conservatively assumed radiation sources, design shielding, calculated design dose rates, and manpower level, taking into account expected functions and occupancy times, and a working year of 2000 hours. Based on expected dose rates and occupancy times, expected airborne radioactivity concentrations, and estimates of the time and manpower necessary to perform the various tasks involved in plant operation, the applicant estimates average annual occupational radiation exposure at the facility to be about 400 man-rem. This information is consistent with experiences from operating reactors and information presented in NUREG-75/032, "Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1974." The dose assessment discussed above includes a breakdown of the annual man-rem doses associated with major functions; operations, maintenance (including special maintenance), refueling, security, radwaste handling and inservice inspection. This descriptive information meets the minimum information needs set forth in Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light Water Reactor Power Plants--Design Man-Rem Estimates." Therefore, we find the bases for the facility's exposure estimates acceptable and consistent with the acceptance criteria in Section 12.4 of the Standard Review Plan.

The applicant has estimated the potential exposures of individual workers to airborne radioactivity in various parts of the station. These estimates are quite low; in most cases the estimates are only a few percent of the allowable exposures given in Section 20.103 of 10 CFR Part 20. These estimates are comparable to estimates presented by other applicants with pressurized water reactor designs. Therefore, we conclude that the assessments of exposure to airborne radioactivity are acceptable.

12.5 Health Physics Program

Sections 12.1.5, 12.2.5, and 12.3 of the Final Safety Analysis Report describe the applicant's health physics program. The description includes the radiation protection organization, equipment, instrumentation, and facilities, and the procedures for radiation protection.

The applicant's organization will include health physics professionals and technicians. The Health Physics Supervisor will have the responsibility for implementing the health physics program and maintaining exposures as low as is reasonably achievable. The organizational aspects of the program are consistent with the guidance of Regulatory Guide 8.8.

The applicant's radiation protection facilities will include portable instrument calibration and storage areas, personnel and equipment decontamination areas, change room, access control point, health physics station, counting room, radiochemical laboratory, and a whole-body counting area. A variety of personnel monitoring devices and personnel protection equipment will also be provided. Protective clothing, respiratory protection devices, and personnel dosimeters will be included in the available equipment. The health physics equipment, instrumentation, and facilities are consistent with the guidance of Regulatory Guide 8.8 and are therefore acceptable.

The applicant has described the procedures which will be used to implement the radiation protection program. The procedures described are for access control, radiation work permits, radiation surveys, personnel monitoring, bioassay, radiation protection training, contamination control, methods of maintaining exposures as low as is reasonably achievable, and reviews of the effectiveness of the health physics program. The applicant has committed to providing a personnel neutron dosimetry program which will be consistent with Regulatory Guide 8.14, "Personnel Neutron Dosimeters," Revision 1. The procedures as described are consistent with the guidance of Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, and Regulatory Guide 8.8, and meet the requirements of 10 CFR Part 20.

In Amendment 20 to the Final Safety Analysis Report, the applicant specified that the Health Physics Supervisor reports directly to the Assistant Plant Manager as specified in our "Criteria for Utility Management and Technical Competence," dated July 17, 1980, and in Regulatory Guide 8.8. In addition, a health physics technician will be onsite at all times as specified in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." We conclude that the organization for health physics meets our criteria and is acceptable.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Qualifications

Sections 13.1, 13.2, and 13.3 of the Standard Review Plans cover the corporate management and technical support provided for operations including the educational background and experience of individuals holding management and supervisory positions; the structure, functions, and responsibilities of the onsite organization established to operate and maintain the plant including shift manning requirements; and the qualifications of the applicant's plant personnel.

Our evaluation of these matters is included in Section 22 of this Safety Evaluation Report under items I.A.1.3, Shift Manning, and I.B.1.2, Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants.

13.2 Training Program

The plant manager has overall responsibility for training of personnel within the Virgil C. Summer Nuclear Station, Unit 1. The training programs will be implemented at the facility under the general direction of the administrative supervisor. The overall nuclear training program is conducted in conjunction with contracted training from Westinghouse to meet the guidelines of Regulatory Guide 1.8. All applicants for cold license examinations will meet the guidelines as stipulated in ANSI N18.1, 1971.

The training program for licensed personnel will consist of preparatory training, fundamental nuclear technology, nuclear power plant simulator training, operating pressurized water reactor observation, design lecture series, and onsite training. The preparatory training has been conducted by the applicant while the remaining phases will or have been conducted by Westinghouse. Pressurized water reactor observation training was conducted at the Zion plant and onsite training includes a Westinghouse training coordinator in residence at the Virgil C. Summer Nuclear Station, Unit 1.

Specialists and non-licensed personnel have received or are scheduled to receive contracted training in nuclear engineering, instrumentation and control, plant chemistry, and maintenance engineering. All members of the plant staff will receive general employee training in the areas of radiation control, and safety, emergency plan and procedures, security plan, and industrial safety.

The plant manager is responsible for assuring that the required fire protection training is performed. Scheduling and documenting the training is the responsibility of the nuclear training coordinator. Fire protection training will be administered to the fire brigade, fire protection staff and general employees. Periodic drills and practice sessions will be conducted to evaluate the effectiveness of the fire protection training. Instruction, including radiation principles and practices, will be administered to the offsite fire departments that participate in onsite emergencies.

The information submitted relative to the training programs is satisfactory to give reasonable assurance that qualified individuals will be available for safe operation of the facility.

13.3 Emergency Planning

See Appendix F of this Safety Evaluation Report.

13.4 Review and Audit

Section 13.4 of the Standard Review Plan addresses the applicant's plans for conducting reviews of operating phase activities that are important to safety. These activities will be conducted by the applicant's plant staff supervisors with special audits under the responsibility of the assistant plant manager. Independent reviews will be conducted by the Nuclear Safety Review Committee and independent audits by the quality assurance organization.

Our evaluation of the review and audit function is included in Section 22 of this Safety Evaluation Report under item I.B.1.2, Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants.

13.5 Procedures

Actions concerning structures, systems, and components of the Virgil C. Summer Station, Unit 1, that are safety-related will be conducted in accordance with written and approved procedures. The applicant has committed to conforming to Regulatory Guide 1.33. Operating procedures are to be prepared using ANSI 18.7-1976 as a guide.

All procedures except maintenance and modification and surveillance test procedures are scheduled to be completed three months prior to fuel loading. Fire protection measures will include the provisions made in the "Fire Protection Evaluation Report," Gilbert Associates, Inc. dated July 1977.

The information submitted relative to these subjects is satisfactory at the operating license stage of review. For additional discussion of our review of emergency operating procedures, refer to Section 22 of this Safety Evaluation Report.

13.6 Industrial Security

The applicant has submitted a physical security plan for the protection of the facility against potential acts of radiological sabotage. The staff has reviewed this document, entitled "Virgil C. Summer Nuclear Station Physical Security Plan," Amendment 2, dated September 1980, and Amendment 3, dated October 1980, against the requirements of Section 73.55 of 10 CFR Part 73 and has determined that the plan is acceptable.

The identification of vital areas and measures used to control access to these areas, as described in the plan, may be subject to amendments in the future based on a confirmatory evaluation of the facility to determine those areas where acts of sabotage might cause a release of radionuclides in sufficient quantities to result in dose rates equal to or exceeding 10 CFR Part 100 limits.

The applicant's physical security plan is being withheld from public disclosure in accordance with Section 2.790(d)(1) of 10 CFR Part 2.

14 INITIAL TEST PROGRAM

The applicant's initial test program is divided into three phases: acceptance tests, preoperational tests, and startup tests. Acceptance tests are essentially equipment checkouts that are performed prior to turning over the equipment to the facility operations staff. Preoperational tests generally are system level tests and are conducted prior to fuel loading to demonstrate that structures, systems, and components meet performance requirements. Startup tests consist of fuel loading and following activities (precritical tests, initial criticality, low power tests, and power ascension tests) that demonstrate that the facility will operate in accordance with design and is capable of responding as designed to anticipated transients and postulated accidents as specified in the Final Safety Analysis Report. Our review concentrated on the last two phases, i.e., preoperational and startup tests.

The applicant's organization and staff for performing the initial test program were reviewed. An adequate number of appropriately qualified personnel are assigned to develop test procedures, conduct the tests, and review the results of the tests. Plant staff personnel are utilized to maximize the training benefits of the test program.

The test procedures were developed using input from the nuclear steam supply system vendor, the architect-engineer, the applicant's engineering staff, and other equipment suppliers and contractors as needed. The applicant's review of operating experiences at similar facilities was also factored into the development of the test procedures.

The tests are being conducted using approved test procedures. Administrative controls assure that (1) test prerequisites are met, (2) necessary data sheets and other documentation are completed, and (3) necessary modifications to the test procedures are appropriately reviewed prior to implementation. Administrative controls also assure that any modifications or repairs that are identified as a result of testing are implemented properly and that necessary retesting is performed.

The results of each test are reviewed for technical adequacy and completeness by qualified personnel including the nuclear steam supply system vendor and the architect-engineer as appropriate. Preoperational test results are reviewed prior to fuel loading and the startup test results from each activity or power level will be reviewed prior to proceeding to the next activity or power level.

Normal plant operating, emergency, and surveillance procedures are used in performing the initial test program, thereby verifying the correctness of the procedures to the extent practical.

In planning for the initial test program, the applicant allowed adequate time to conduct all preoperational tests and startup tests. The scheduled sequence for performing the startup tests established that systems required to prevent, limit or mitigate the consequences of postulated accidents will be tested prior to exceeding 25 percent of rated power and that the safety of the facility will not be totally dependent on the performance of untested systems, structures, and components. The applicant stated that test procedures would be available for the NRC's Office of Inspection and Enforcement review at least 30 days prior to the expected performance date of the test and at least 90 days prior to fuel loading.

We reviewed the abstract of each test procedure presented in Chapter 14 of the Final Safety Analysis Report. We verified that there are test abstracts for those structures, systems, components, and design features that:

1. Will be used for shutdown and cooldown of the reactor under normal conditions and for maintaining the reactor in a safe condition for an extended shutdown period.

2. Will be used for shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions.

3. Will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications.

4. Are classified as engineered safety features or will be relied on to support or assure the operations of engineered safety features within design li'

5. Are assume. to function or for which credit is taken in the accident analysis of the facility, as described in the Final Safety Analysis Report, and

6. Will be used to process, store, control, or limit the release of radioactive materials.

We also reviewed the test objectives, prerequisites, test methods, and acceptance criteria of each test abstract in sufficient detail to establish that the functional adequacy of the structures, systems, components, and design features will be demonstrated.

We reviewed the initial test program's conformance with applicable Regulatory Guides including 1.20 (May 1976), "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," 1.41 (March 1973), "Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments," 1.52 (July 1976), "Design, Testing, and Maintenance Criteria for Engineered Safety Feature Atmosphere Cleanup System, Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," 1.68 (November 1973), "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," 1.68.2 (July 1978), "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants," 1.79 (September 1975, "Preoperational

Testing of Emergency Core Cooling Systems for Pressurized Water Reactors," 1.80 (June 1974). "Preoperational Testing of Instrument Air Systems," and 1.108 (August 1977), "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants."

Based on the above, we have concluded that the initial test program described in the application meets the acceptance criteria of Section 14.2 of the Standard Review Plan and will demonstrate the functional adequacy of facility structures, systems, and components. We also have concluded that the initial test program meets the test requirements of Criterion 1 of the General Design Criteria and Section XI of Appendix B to 10 CFR Part 50.

Refer to Section 22 of this Safety Evaluation Report for a discussion of additional test requirements resulting from the TMI-2 accident.

15 ACCIDENT ANALYSIS

15.1 General

The applicant has performed safety analyses to evaluate the capability of the Virgil C. Summer Nuclear Station Unit 1 to withstand normal and abnormal operational transients and a broad spectrum of postulated accidents without undue risk to the health and safety of the public. The events considered include all relevant types discussed in the Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2. The postulated events have been classified by the applicant with respect to evaluation criteria as follows:

Condition I: Normal Operation and Operational Transients -- events which are expected frequently or regularly in the course of power operation, refueling maintenance, or maneuvering of the facility.

Condition II: Faults of Moderate Frequency -- events that at worst result in a reactor trip with the facility being capable of return to operation.

Condition III: Infrequent Faults -- events that are very infrequent during the life of the facility and may result in fuel damage which could preclude the resumption of immediate operation.

Condition IV: Limiting Faults -- events which are not expected to occur, but are postulated because their consequences would include the potential for release of significant amounts of radioactive material.

The applicant's classification of events analyzed is itemized in Table 15-1 of this Safety Evaluation Report. The input parameters and analytical techniques used for transients and accidents are discussed in Sections 15.1.1 and 15.1.2 of this Safety Evaluation Report.

15.1.1 Input Parameters for Transient and Accident Analyses

We have reviewed the assumptions and parameters employed in the transient and accident analyses. Reactor protection system trip set points and the assumed trip delay times used in the analyses are tabulated in Table 15-2 of this Safety Evaluation Report. The analyses assumed a time of 2.3 seconds to reach 85 percent of the control rod travel. This will be verified during the pre-operational testing program.

The events initiated at full power were assumed to start at the "guaranteed nuclear steam supply system thermal output" power of 2785 thermal megawatts plus allowance of two percent for errors in steady-state core power. For events where adequacy of the containment and the engineered safety features equipment is involved, the applicant has used an initial nuclear steam supply system power output of 2900 thermal megawatts plus allowance for error. These latter analyses were for the events of loss of non-emergency alternating current power to the facility auxiliaries, loss of normal feedwater flow, feedwater system pipe break, steam generator tube rupture, and loss-of-coolant accidents resulting from postulated small and large reactor coolant system pipe ruptures. The assumed initial nuclear steam supply system thermal power

TABLE 15-1

CATEGORIES OF TYPICAL TRANSIENTS AND FAULTS

Condition I - Normal Operation and Operational Transients

- Reactor startup
- Reactor shutdown
- Refueling operations
- Power operation

Condition II - Faults of Moderate Frequency

- Uncontrolled control rod assembly bank withdrawal while the reactor is subcritical or at power
- Partial loss of forced reactor coolant flow
- Startup of an inactive reactor coolant loop
- Turbine trip
- Loss of normal feedwater
- Loss of offsite power
- Uncontrolled boron dilution
- Control rod assembly misalignment
- Excessive load increase
- Accidental depressurization of reactor coolant system
- Accidental depressurization of main steam system
- Inadvertent operation of emergency core cooling system during power operation

Condition III - Infrequent Faults

- Improper loading of a fuel assembly
- Complete loss of forced reactor coolant flow
- Minor secondary system pipe break
- Single control rod assembly withdrawal at full power
- Waste gas decay tank rupture
- Loss of reactor coolant from small break

Condition IV - Limiting Faults

- Control rod ejection
- Fuel handling accident
- Steam generator tube rupture
- Major secondary system pipe rupture
- Loss-of-coolant accident
- Single reactor coolant pump locked rotor or broken shaft

TABLE 15-2

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSIS

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delay (seconds)
Power range high neutron flux, high setting	118 percent	0.5
Power range high neutron flux, low setting	35 percent	0.5
Overtemperature temperature differential	Variable (see Figure 15.1-1 of the Final Safety Analyses Report	6.0*
Overpower temperature differential	Variable (see Figure 15.1-1 of the Final Safety Analysis Report	6.0*
High pressurizer pressure	2420 pounds per square inch, gauge	2.0
Low pressurizer pressure	1860 pounds per square inch, gauge	2.0
Low reactor coolant flow (from loop flow detectors)	87 percent of loop flow	1.0
Reactor coolant pump undervoltage trip**	70 percent of nominal	1.5
Turbine trip	Not applicable	1.0
Low steam generator level	Six percent of narrow range span between 0 and 20 percent nominal load and increasing linearly to 49 percent of span at 100 percent of nominal load	2.0
Low-low steam generator level	0	2.0
High steam generator level trip of the feedwater pumps and closure of feedwater system valves, and turbine Trip	83 percent of narrow range level span	2.0

*Total time delay, including resistance temperature detector time response and trip circuit channel electronics delay, from the time the temperature in the coolant loops exceed the trip setpoint until the control rods are free to fall.

**An additional trip on underfrequency is provided to trip the reactor on an underfrequency condition resulting from frequency disturbances on the power grid.

output levels used in Virgil C. Summer Nuclear Station, Unit 1 transient and accident analyses are in accordance with the requirements of Regulatory Guide 1.49 "Power Levels of Nuclear Power Plants."

The applicant has selected the most adverse conditions of core life with respect to reactivity coefficients (moderator temperature coefficient and the Doppler coefficient), control rod worths, and local power peaking factors. The applicant has stated that no credit was taken for non-safety-grade systems to mitigate the consequences of any accident presented in Chapter 15 of the Final Safety Analysis Report. Furthermore, for some transients, the analyses were performed where the operation of non-safety-grade systems was assumed when such operation gave more conservative results, or for the purpose of showing a comparison of results with and without such operation. The above assumptions used in the analyses are acceptable.

Upon our request, the applicant provided a systematic functional analyses of components required for each Chapter 15 event analyzed. These analyses were summarized in Amendment 8 of the Final Safety Analysis Report in the form of block diagrams called Accident Sequence Diagrams. These diagrams identify for each event: (a) the safety systems required to function to provide the safety actions necessary to mitigate the consequences of the transient or accident, (b) required operator actions, and (c) safetyrelated information readouts and controls utilized by the operator to analyze and control the transient or accident. We have reviewed the summary diagrams and conclude they are acceptable.

The effects of a new fuel rod pressure design criterion have been considered for the safety analysis of Condition III and IV events. The effects were addressed in a Westinghouse generic Topical Report WCAP-8963, which was reviewed and approved by the NRC staff in a safety evaluation report dated May 9, 1978 in memorandum from D. F. Ross, Jr. to D. B. Vassallo. The applicant has stated that considering this new design criterion, there would only be a small percentage increase in the number of fuel rods which could fail for Condition III and IV events, and that Westinghouse has determined that the dose consequences of the accidents remain essentially unchanged and well below the consequences of the loss-of-coolant accident. We have concluded that the increased fuel rod pressure will not result in a significant number of additional departure from nucleate boiling events during Condition III and IV events.

The effect of fuel rod bow on the departure from nucleate boiling heat flux was not included in the safety analyses. As discussed in Section 4.4 of this Safety Evaluation Report, a penalty on the allowable enthalpy hot channel factors will be included in the Technical Specifications to correct for the fuel rod bow effect on departure from nuclear boiling as a function of burnup. This penalty factor provides assurance that the minimum departure from nucleate boiling values predicted for the anticipated transients will not violate the fuel design limit of 1.30.

In response to our request, the applicant has provided information which identifies operator actions needed and the time involved to mitigate the transient for each event analyzed in Chapter 15 of the Final Safety Analysis Report. The applicant indicated that the only event which requires the operator's action within 20 minutes is the main steamline break accident. The time at which operator action is required is after 10 minutes in case of steamline break to limit cooldown and repressurization of the primary system.

Following a main steamline break there is no specific time at which operator action is required to obtain acceptable results for the core integrity analysis. This response is acceptable to the staff.

15.1.2 Analytical Techniques

Most of the analytical techniques used for the Virgil C. Summer Nuclear Station, Unit 1 accident and transients analyses have been reviewed and approved. Those for which we have not completed our review are described in the following topical reports:

1. WCAP-7907 LOFTRAN Code Description

2. WCAP-7908 A FACTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod

3. WCAP-9227 Main Steamline Break Sensitivity Studies

4. WCAP-9230 Report on the Consequences of a Postulated Main Feedline Rupture

5. WCAP-7998 BLKOUT Code Description

6. WCAP-7909 MARVEL Code Description

The analytical methods used for postulated transients and accidents are normally reviewed on a generic basis. Our review at this time and indicates that there is reasonable assurance that the conclusions based on the analyses presented in the Final Safety Analysis Report will not be appreciably altered by the completion of the analytical methods review. If the final approval of the methods indicates revisions to the analyses are required, the applicant will be required to implement the results of such changes.

15.2 Transients

A number of transients can be expected to occur with moderate frequency as a result of equipment malfunctions or operator error in the course of refueling and power operation during the facility lifetime. Such transients meet the criteria of Condition II in the evaluation and classification presented by the applicant.

We have compared the Condition II events of Table 15-1 of this Safety Evaluation Report to typical anticipated operational occurrances normally considered for safety reviews as specified in the Standard Review Plan. We have noted that the applicant has chosen the "Complete Loss of Reactor Coolant Flow" to be classified as a Condition III event--an infrequent event, and we disagree with the applicant's classification. According to the Standard Review Plan this event should be considered as a moderate frequency transient and we have evaluated the consequences accordingly.

Our basic acceptance criteria for the review of the submitted transients were as follows:

1. Pressure in the reactor coolant and main steam systems should not exceed 110 percent of design pressure (Section III of the ASME Boiler and Pressure Vessel Code).

2. Clad integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio of 1.30 throughout the transient will satisfy the 95/95 criterion. (The 95/95 criterion provides a 95 percent probability, at a 95 percent confidence level, that no fuel rod in the core experiences a departure from nucleate boiling).

3. Transients will not lead to more serious conditions (assuming other independent faults have not occurred).

All of the transients which are expected to occur with moderate frequency can be grouped to cause the following process disturbances: increase in heat removal by the secondary system; decrease in heat removal by the secondary system; core flow decrease; core reactivity increase; reactor coolant inventory increase; and reactor coolant inventory decrease.

The applicant was requested to provide a summary table demonstrating that the departure from nucleate boiling ratio and reactor coolant pressure occuring for the limiting moderate frequency transients were within licensing limits. In response to the staff's concerns, the applicant demonstrated that the limiting departure from nucleate boiling ratio was above 1.3 and the limiting overpressurization transient was below 2575 pounds per square inch, absolute.

It is therefore concluded that the facility does conform to present regulations with regard to moderate frequency events, as described in Section 15 of the Final Safety Analysis Report.

15.2.1 Increase in Heat Removal by the Secondary System Events

An unplanned increase in heat removal by the secondary system that might be expected to occur with moderate frequency can be caused by feedwater system or pressure regulatory malfunctions or excessive increase in secondary steam flow or the inadvertent opening of a steam generator safety or relief valve. All of these postulated transients have been reviewed. The transients were evaluated by the applicant using mathematical models described in Westinghouse topical reports WCAP-7907 and WCAP-7909. The analytical techniques requiring review completion by the staff are identified in Section 15.1.2 of this Safety Evaluation Report. The parameters used as input to these models were reviewed and found to be suitably conservative. The results of the analysis for the transients showed that cladding integrity was maintained by assuring that the minimum departure from nucleate boiling ratio did not decrease below 1.30 and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressures.

The Final Safety Analysis Report indicates that the most limiting increase in heat removal event with regard to core thermal margins is the increase in feedwater flow. Similarly, the most limiting event for pressure in the reactor coolant system is the excessive increase in secondary steam flow. Analyses indicate that no significant pressure excursion would occur for this category of events. All transients in this category meet our acceptance criteria with regard to core thermal limits and pressure limits.

15.2.2 Decrease in Heat Removal by the Secondary System Events

A number of plant transients can result in an unplanned decrease in heat removal by the secondary system. Those that might be expected to occur with moderate frequency are turbine trip, loss of external load, loss of condenser vacuum, loss of non-emergency alternating current power to the station auxiliaries, and loss of normal feedwater flow. All of these postulated transients have been reviewed. The transients were evaluated by the applicant using mathematical models described in the Westinghouse Topical Reports WCAP-7907 and 7898. Review status of these reports is provided in Section 15.1.2 of this Safety Evaluation Report.

For the loss of feedwater event analysis, no credit was taken for the pressurizer and the steam generator power-operated relief valves. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analyses for these transients showed that cladding integrity was maintained by assuring that the minimum departure from nucleate boiling ratio did not decrease below 1.30 and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of their design pressures.

15.2.3 Decrease in Reactor Coolant System Flow Rate Events

Several types of occurrences in the facility can result in an unplanned decrease in reactor coolant flow rate. The ones that might be expected to occur with moderate frequency during the life of the facility are a partial loss of coolant flow caused by reactor coolant pump trip(s) or a complete loss of forced reactor coolant flow that may result from the simultaneous loss of electrical power to all pumps. For the partial loss of forced reactor coolant flow transient evaluation, two cases have been analyzed--loss of one pump with three loops in operation and loss of one pump with two loops in operation. The two cases analyzed for the complete loss of forced reactor coolant flow were the loss of three pumps with three loops in operation and the loss of two pumps with two loops in operation. These postulated transients have been reviewed and the most limiting decrease in flow event with regard to core thermal margins and pressure within the reactor coolant was the complete loss of reactor coolant flow transient. All of the transients were evaluated by the applicant using mathematical models described in the Westinghouse Topical Reports WCAP-7907, WCAP-7908, WCAP-7956, and WCAP-7973. The analytical techniques requiring review completion by the NRC staff are identified in Section 15.1.2 of this Safety Evaluation Report. The values of the parameters used for input to this model were reviewed and found to be suitably conservative. The results of the analysis of the complete loss of reactor coolant flow transient showed that cladding integrity was maintained by assuring that the minimum departure from nucleate boiling ratio did not decrease below 1.30 and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressures.

15.2.4 Core Reactivity Insertion Events

There are a number of transients that may occur with moderate frequency which can cause unplanned core reactivity insertions. These transients include startup of an inactive reactor coolant loop at an incorrect temperature which would result in increased core flow and thereby an increase in core reactivity, and an uncontrolled boron dilution incident. The mathematical models used in the evaluation of core reactivity insertion events are described in the Westinghouse Topical Reports WCAP-7907, 7908, 7956, and 7980. The analytical techniques requiring review completion by the staff are identified in Section 15.1.2 of this Safety Evaluation Report.

Reactivity can be added to the core by adding primary grade water to the reactor coolant system via the makeup portion of the chemical and volume control system. Various chemical and volume control system malfunctions which could lead to an unplanned boron dilution incident have been reviewed. The applicant has analyzed postulated boron dilution transients starting from plant conditions of startup, power operation (automatic and manual), hot standby, cold shutdown, and refueling. An additional criterion was imposed for these transients. From the time an alarm makes the operator aware of unplanned moderator dilution, the following minimum intervals must be available before a complete loss of shutdown margin occurs:

1. During refueling: 30 minutes

2. During startup, cold shutdown, hot standby, and power operation: 15 minutes

As a result of the above requirements, during refueling operations valves FCV-113B, FCV-168A, 8439, 8441, and 8454 will be locked closed. This will minimize the potential for boron dilution through the chemical and volume system. The applicant was requested to evaluate the the potential for a boron dilution accident caused by dilution sources other than the chemical and volume control system and has determined that the analysis presented for inadvertent opening of a valve in the boron thermal regeneration system bounds all potential sources of inadvertent dilution under all modes of operation. Also, the applicant has stated that the facility design is such that it is not vulnerable to a boron dilution event with the sodium hydroxide tank as the source. We have reviewed the appropriate diagrams and agree with the applicant.

The results of the events analyzed showed that for the limiting case the operator has 84.4 minutes to take corrective action if the incident occurred during refueling; more than 56 minutes if at startup; and about 62.3 minutes if at power in manual or automatic control mode. However, times for the operator to take corrective action if the event occurs from cold shutdown or hot standby have not been addressed. The predicted time intervals for the events analyzed meet the acceptance criterion. The most severe unplanned boron dilution event analyzed occurs at power and results in a minimum departure from nucleate boiling ratio of greater than 1.30 and reactor coolant and main steam system pressures of less than 110 percent of design.

The results of the startup of an inactive reactor coolant loop at an incorrect temperature transient showed the minimum departure from nucleate boiling ratio remained above 1.30 throughout the transient.

In summary, none of the transients with regard to core thermal margins, due to an unplanned core reactivity insertion event, results in a minimum departure from nucleate boiling ratio greater than 1.30 or reactor coolant and main steam system pressures of less than 110 percent of design. Since all the acceptance criteria have been met, we conclude that the facility design is acceptable with respect to transients resulting from core reactivity anomalies provided the results of the boron dilution events from hot standby and cold shutdown are submitted prior to issuance of the operating license and satisfy the acceptance criteria.

15.2.5 Decrease in Reactor Coolant Inventory Event

An event which can result in a decrease of reactor coolant inventory with an expected moderate frequency is an inadvertent opening of a pressurizer safety or relief valve. The applicant has informed us that this analysis is documented in Westinghouse Topical Report WCAP-9600. However, the staff will require that the applicant document its analysis in Section 15.2.5 of the Final Safety Analysis Report.

15.2.6 Increased Reactor Coolant Inventory Event

Events that can result in an increase of reactor coolant inventory with an expected moderate frequency are inadvertent operation of the emergency core cooling system, and chemical and volume control system malfunctions.

The increase in reactor coolant inventory event was evaluated with a mathematical model described in the Westinghouse Topical Report WCAP-7907. The review status of that report is provided in Section 15.1.2 of this Safety Evaluation Report.

Two different scenarios of an inadvertent emergency core cooling system operation event were considered: (1) reactor trip occurs at the same time spurious injection from the emergency core cooling system occurs, and 2) the reactor protection system produces a trip later in the transient. The analyses assumed inadvertent borated water injection into the cold legs by two charging pumps while at power. Because of the low shutoff head of the low head safety injection pumps and the low pressure of the accumulator, no flow is injected into the reactor coolant system from these systems, and therefore they provide no contribution to the event at high pressures.

The results of the inadvertent operation of emergency core cooling system during power operation showed that the minimum departure from nucleate boiling ratio would not decrease from the initial value throughout the transient, thus assuring that cladding integrity would be maintained, assuming the operator will terminate high pressure safety injection in accordance with the termination criteria before the reactor coolant system pressure reaches the set points of safety and relief valves.

Malfunctions of the chemical and volume control system that can lead to an inventory increase are discussed in Section 15.2.4 of this Safety Evaluation Report under the boron dilution event of the reactivity insertion category.

15.3 Postulated Accidents

We have reviewed the postulated events with regard to the facility design bases. These events have been classified by the applicant to be Condition III and IV events and are itemized in Table 15-1 of this Safety Evaluation Report and discussed below. The specific criteria we used by the staff in evaluating the consequences of the postulated accidents are:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures.

2. The potential for core damage should be evaluated on the basis that it is acceptable if the minimum departure from nucleate boiling ratio remains above 1.30. If the departure from nucleate boiling ratio falls below this value, fuel damage (rod perforation) should be assumed unless it can be shown, based on an acceptable fuel damage model, that no fuel failure results. If fuel damage is calculated to occur, it should be of sufficiently limited extent so that the core will remain in place and geometrically intact with no loss of core cooling capability.

3. Any activity release must be such that the calculated doses at the site boundary are within the guidelines of 10 CFR Part 100.

15.3.1 Feedwater System Piping Breaks

The analyses and effects of feedwater line breaks inside containment, during various modes of operation, and with or without offsite power, have been reviewed. The limiting feedwater line rupture was assumed to be a doubleended rupture located between the steam generator and the main feedwater line check valve. Since the feedwater line rupture has the potential of reducing the capability of the secondary system to remove the heat generated by the core, an emergency feedwater system is provided to assure that adequate feedwater will be available to remove decay heat, to prevent overpressurization of the reactor coolant system and to prevent uncovering of the reactor core. The analysis indicates that 55.1 minutes after reactor trip the core decay heat and pump heat decrease below the emergency feedwater system capacity, which is sufficient to allow liquid in the reactor coolant system to cover the core at all times, to prevent overpressurization of the reactor coolant system.

The mathematical models used in the accident evaluation are described in the Westinghouse Topical Report WCAP-7907. The review status of this report is identified in Section 15.1.2 of this Safety Evaluation Report. The results of the analysis of the feedwater line break accident showed that the pressures in the reactor coolant system and the main steam system remain below 110 percent of the respective design pressures and the analysis has shown that the effects on the reactor coolant system for this accident are less severe than for the postulated steam line break. The minimum departure from nuclceate boiling ratio would remain above 1.30 throughout the accident. Although the results presented for a major feedwater line break meet the acceptance criteria, a generic sensitivity analysis that supports the applicant's selection of the most limiting feedwater line break has been submitted to us. These sensitivity studies are described in the Westinghouse Topical Report WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture." The status of this report is indicated in Section 15.1.2 of this Safety Evaluation Report. We will require that the applicant comply with any changes to WCAP Reports 7907 and 9230 that result from completion of the staff review of these reports.

15.3.2 Spectrum of Steam Piping Failures

The analyses and effects of steam line break accidents, on any location, during various modes of operation and with or without offsite power, have been reviewed. The applicant has stated that since the steam generators are provided with safety-grade integral flow restrictors with a 1.4 square foot throat area. Any rupture with a break greater than 1.4 square feet, regardless of location would have the same effect on the nuclear steam supply system as the 1.4 square foot break. The accident which resulted in the most severe consequences was the 1.4 square-foot steam line rupture analyzed at zero thermal power and with offsite power available. In the analysis it was assumed that the most reactive rod cluster assembly was stuck in its fully withdrawn position and a single failure in the engineered safety features occurred concurrent with the accident. The single failure selected was in the high head safety injection system. The mathematical models used in the evaluation are described in the Westinghouse Topical Report WCAP-7909 and WCAP-7956. These reports are still under staff review and we will require that the applicant comply with any changes resulting from our review of these reports.

The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the steam line break accidents showed no predicted fuel damage and no loss of core cooling capability. The maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressures. The applicant has stated that sensitivity studies were performed in determining the effect of initial reactor coolant flow on fuel thermal margins following a main steam line break accident. These studies show that the results are relatively insensitive to reactor coolant flow. The applicant has referenced WCAP-9227 for supporting steam line break information. The status of this report is indicated in Section 15.1.2 of this Safety Evaluation Report.

The NRC staff position as documented in NUREG-0138 with regard to the steam line break accident permits reliance on non-safety valves downstream of the main steam line isolation valves to prevent the blowdown of a second steam generator in the event of the failure of a safety-grade main steam isolation valve associated with one of the intact steam generators. However, there are numerous lines with valves located along the length of each steam line between the main steam isolation valve and the turbine stop valves (i.e., feedwater turbine supply, by-pass lines to condenser, steam extraction lines). In response to our question, the applicant, in Table 10.3-3 of the Final Safety Analysis Report, has listed all valves located along the steamlines between the main steam isolation valves and the turbine stop valves and the status of the valve operation during a postulated main steam break accident. The applicant indicated that only several valves are manually operated and will remain open after a reactor trip. The applicant has indicated that the assumed total steam release from the unaffected steam generators during a main steam line break accident, as listed in Table 15.4-23 of the Final Safety Analysis Report, is based on energy balance calculations following the steam line break accident independent from the flow path in the main steam system. However, the applicant confirmed that the total steam flow through the open valves downstream of the main steam isolation vlaves is less than the amount assumed in Table 15 & 23 of the Final Safety Analysis Report. Alos, these open valves downstream of the mainstream isolation valves can be manually closed if a main steam isolation valve fails to close during main steam isolation. We find this acceptable.

The Westinghouse methodology used for steam line break analyses is still under generic review. Since the TMI-2 accident, new criteria have been developed, which require the tripping of the reactor coolant pumps during emergency core cooling system initiation. The conseuqnce of tripping the reactor coolant pumps during the transient has not been analyzed by the applicant. The staff, however, believes that the analyses conducted, with and without offsite power, bounds the consequence of tripping the reactor coolant pumps during the postulated accident. This will be verified during our generic review of the Westinghouse methodology.

15.3.3 Reactor Coolant Pump Rotor Seizure

The analyses and effects of an instantaneous seizure of a rotor or an instantaneous break of a shaft of a reactor coolant pump during any allowed mode of operation have been reviewed. The applicant has stated that since the time of reactor trip for both of these events would be nearly identical, the consequences of both of these events would be about the same. The mathematical models used in the evaluation are described in the Westinghouse Topical Reports WCAP-7907, WCAP-7908, and WCAP-7973. The review status of these reports is documented in Section 15 1.2 of this Safety Evaluation Report. The parameters used as input to this model were reviewed and found to be suitably conservative. Analyses of the locked rotor event were performed with three and two loops operating. The most limiting event with regard to thermal margin, was for initial three-loop operation.

The results of this analysis showed that less than 10 percent of the fuel rods were predicted to experience departure from nucleate boiling and that the peak clad temperature reached was 1955 degrees Fahrenheit. Fuel damage is minimal and no loss of core cooling capability will result. The radiological consequences of this event have been analyzed and are reported in Section 15.4 of this Safety Evaluation Report. The maximum reactor coolant system pressures for the initial three and two loop operation were 2675 pounds per square inch absolute and 2726 pounds per square inch absolute, respectively. below 110 percent of the design pressure (2750 pounds per square inch absolute). Also, the maximum pressure within the main steam system did not exceed 110 percent of the design pressure. We find, the analysis submitted by the applicant acceptable.

15.3.4 Spectrum of Piping Breaks Within the Reactor Coolant Pressure Boundary

The applicant has analyzed the performance of the emergency core cooling system in accordance with the requirements of Section 50.46 of 10 CFR Part 50. The analyses considered a spectrum of postulated break sizes and locations and were performed with the evaluation model described in the Appendix K to 10 CFR Part 50. We have reviewed this information and our evaluation is contained in Section 6.3 of this Safety Evaluation Report.

15.3.5 Anticipated Transients Without Scram

In a pressurized water reactor, the anticipated transients which require prompt action to shut down the reactor in order to avoid plant damage and possible offsite effects can be classified in two groups: those that isolate the reactor from the heat sink, and those in which the heat sink is maintained. (A list of these transients is included in Appendix IV to Volume II of NUREG-0460, April, 1978.) In general, the consequences of both of these types of events are an increase in reactor power or system pressure, or both. In Section 6.3 of NUREG-0460, Volume I, potentially unacceptable consequences of anticipated transients without scram events for pressurized water reactors are indicated to include (1) pressure rises that could threaten the integrity of the reactor coolant pressure boundary, (2) loss of core cooling, and (3) leakage of radioactive material from the facility.

In NUREG-0460, we concluded that for plants which fall within the envelope of the Westinghouse generic anticipated transients without scram analyses, the anticipated transient without scram acceptance criteria will not be violated if the actuation circuitry of turbine trip and auxiliary feedwater systems which are relied upon to mitigate anticipated transient without scram consequences are sufficiently reliable and are separate and diverse from the reactor protection system. Additionally, the functionability of valves required for long-term cooling following the postulated anticipated transient without scram events has to be demonstrated.

We issued requests for the industry to supply generic analyses to confirm the anticipated transient without scram mitigation capability described in Volume 3 of NUREG-0460. The staff evaluation of these reports was issued for comment as NUREG-0460, Volume 4, in March, 1980.

We presented our recommendations on anticipated transients without scram to the Commission in September, 1980, including the recommendations for modifications contained in Volume 4 of NUREG-0460. The Commission will determine the required modifications to resolve anticipated transients without scram concerns as well as the required schedule for implementation of such modifications. The facility will of course, be subject to the Commission decision in this matter.

The following discusses the bases for operation of the Virgil C. Summer Nuclear Station, Unit 1 while final resolution of anticipated transients without scram is before the Commission.

In NUREG-0460, Volume 3, we state: "The staff has maintained since 1973 (for example, see pages 69 and 70 of WASH-1270) and reaffirms today that the present likelihood of severe consequences arising from an ATWS event is acceptably small and presently there is no undue risk to the public from ATWS. This conclusion is based on engineering judgment in view of: (a) the estimated arrival rate of anticipated transients with potentially severe consequences in the event of scram failure; (b) the favorable operating experience with current scram systems; and (c) the limited number of operating reactors." In view of these considerations and our expectation that the necessary plant modifications will be implemented in one to four years following a Commission decision on anticipated transients without scram, we have generally concluded that pressurized water plants can continue to operate because the risk from anticipated transient without scram events in this time period is acceptably small. To further reduce the risk from anticipated transients without scram events during the interim period before completing the plant modifications determined by the Commission to be necessary, we have required that the following steps be taken:

1. Develop emergency procedures to train operators to recognize an anticipated transient without scram event, including consideration of scram indicators, rod position indicators, flux monitors, pressurizer level and pressure indicators, pressurizer relief valve and safety valve indicators, and any other alarms annunciated in the control room with emphasis on alarms not processed through the electrical portion of the reactor scram system.

2. Train operators to take actions in the event of anticipated transients without scram events, including consideration of manually scramming the reactor, prompt actuation of the auxiliary feedwater sytem to assure delivery to the full capacity of this system, and initiation of turbine trip. The operator should also be trained to initiate boration by actuation of the high pressure safety injection system to bring the facility to a safe shutdown condition.

We consider these procedural requirements an acceptable basis for interim operation of the facility based on our understanding of the plant response to postulated anticipated transients without scram events.

In response to our requirements on operator training and emergency procedures, the applicant has submitted emergency operating procedures which include provisions for postulated anticipated transients without scram events. These procedures have been reviewed and modified and are now consistent with the currently accepted Westinghouse guidelines. We conclude that the actions taken to reduce the risk from anticipated transients without scram events are adequate to support interim operation of the facility to 100 percent of rated power.

15.3.6 Conclusions

On the basis of our review of the Virgil C. Summer Nuclear Station, Unit 1 accident and transient analysis, we conclude that, the consequences of normal and anticipated transients and postulated accidents are acceptable. However, the applicant will be required to submit analyses for the boron dilution events from hot standby and cold shutdown prior to issuance of a full power license.

15.4 Radiological Consequences of Accidents

The applicant has analyzed the offsite radiological consequences for postulated accidents based on a maximum core thermal power level of 2900 megawatts for the Virgil C. Summer Nuclear Power Plant, Unit 1. We have reviewed the accident analysis presented in the Final Safety Analysis Report and have performed independent calculations of the offsite radiological consequences resulting from a loss-of-coolant accident, a fuel handling accident, a rod ejection accident, a steam line break accident, a steam generator tube rupture accident, a locked rotor accident, and a waste gas decay tank accident. These evaluations are discussed in separate subsections below, and the results are presented in

Table 15-2 of this Safety Evaluation Report. We also considered in our evaluation the offsite doses resulting from leakage from the emergency core cooling system following the postulated loss-of-coolant accident.

15.4.1. Loss-of-Coolant Accident

We have evaluated the postulated loss-of-coolant accident using assumptions consistent with Regulatory Guide 1.4, "Assumptions Used For Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors" (given in Table 15.3 of this Safety Evaluation Report). The calculated doses shown in Table 15-4 of this Safety Evaluation Report are based upon the assumed automatic switchover of the spray system from the injection mode to the recirculation mode as discussed in Section 6.5.2 of this Safety Evaluation Report. Our analysis of the loss-of-coolant accident indicates that both the zero to two hour dose at the exclusion area boundary and the zero to 30 day dose at the low population zone boundary would not exceed the guideline values of 10 CFR Part 100.

15.4.2 Fuel Handling Accident

We have evaluated the radiological consequences of a fuel handling accident involving dropping of fuel assembly in the fuel handling building using the assumptions in Table 15-5 of this Safety Evaluation Report which are consistent with Regulatory Guide 1.25, "Assumptions used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." We assumed that the postulated accident occurred 100 hours after shutdown, following a long period of oeration 2900 megawatts thermal power. This is the earlier time permitted by the Standard Technical Specifications. The fuel handling building ventilation system contains redundant engineered safety feature grade charcoal filters for processing releases from the spent fuel pool prior to releases to the environment. The applicant has estimated that a maximum of 314 fuel rods would be damaged if a fuel assembly was dropped on top of the storage racks.

We have reviewed the fuel handling area ventilation system as described in the Final Safety Analysis Report for its ability to draw a negative pressure during fuel handling operations in order to prevent exfiltration. The applicant has initially planned to draw down the fuel handling building and then run the supply and exhaust fans at the same flow rates. We have reviewed this mode of operation and concluded that the ventilation system, as designed, may not produce the required degree of control to maintain the negative pressure in the fuel building following a postulated fuel handling accident.

A preliminary test using only the exhaust fans has indicated that the system will produce a negative pressure in the building in excess of minus 0.125-inch water gauge so that at moderate wind speeds, the direct release to the atmosphere by exfiltration will be prevented and the radioactive gases will be pulled through the filters in the exhaust system. Our analysis indicates that under the proposed method of operation (i.e., without direct exfiltration), the doses will be far below the guideline values of 10 CFR Part 100.

TABLE 15-3

ASSUMPTIONS USED TO ESTIMATE RADIOLOGICAL CONSEQUENCES DUE TO A POSTULATED LOSS OF COOLANT ACCIDENT

Power level, megawatts thermal	2900
Operating time, years	3
Primary containment leak rate, percent per day	0.2 to 24 hours 0.1 after 24 hours
Fraction of core inventory available for leakage from containment, percent:	
noble gases	100
iodine	25
Primary containment free volume, cubic feet	1.84×10^{6}
Iodine form fractions, percent	
elemental	91
organic	4
particulate	5
Spray removal rates, per hour	
elemental	10
particulate	0.207
Fraction of primary containment unsprayed, percent	25
Relative concentrations, seconds per cubic meter	
0-2 hours at 1609 meters	3.3×10^{-4}
0-8 hours at 4827 meters	4.1×10^{-5}
8-24 hours at 4827 meters	2.6×10^{-5}
24-96 hours at 4827 meters	1.0×10^{-5}
96-720 hours at 4827 meters	2.6×10^{-6}
Control room free volume, cubic feet	226,040
Fitered recirculation flow rate, cubic feet per minute	19,600
Recirculation filter efficiencies, percent	95 for all species
Control room unfiltered infiltration rate, cubic feet per minute	10

TABLE 15-3
(Continued)

ASSUMPTIONS USED TO ESTIMATE
RADIOLOGICAL CONSEQUENCES DUE TO A
POSTULATED LOSS OF COOLANT ACCIDENT

Control room filtered air pressurization rate, cubic feet per minute	400
Duration of accident, days	30 days
Breathing rate of operators in control room for the course of the accident, cubic meters per second	3.47×10^{-4}
Relative concentration, seconds per cubic meter	
0-8 hours	2.65×10^{-3}
8-24 hours	1.71×10^{-3}
24-96 hours	5.78×10^{-4}
96-720 hours	1.12×10^{-4}
Iodine partition factor	634

TABLE 15-4

ACCIDENT DOSE ANALYSIS

	Two-Hour Dose to Thyroid (rem)	Two-Hour Dose to to Whole Body (rem)
Steam generator tube failure		
Case 1	226	<1
Case 2	4.0	<1
Steam line break		
With 5 percent fuel failure	210	<1
With 1 microcurie per gram coolant concentration	4.7	<1
Reactor coolant pump		
Locked rotor	33	<1
Control rod assembly ejection		
Case 1	69	<1
Case 2	14	<1
Fuel handling accident in the spent fuel pool area	12	<1
Waste gas tank failure	--	<0.5

	Doses, rem Thyroid	Whole Body
Loss-of-coolant accident		
0-2 hours (exclusion area boundary including containment purge dose of one rem to thyroid	158	4.0
0-30 days (low population zone)		
Containment leakage	70	1.3
Emergency core cooling system leakage	58	<1
Total for loss of coolant accident (0-30 days)	128	1.3
0-30 days control room operators	13.1	4.6

TABLE 15-5

ASSUMPTIONS USED IN THE ANALYSIS OF FUEL HANDLING ACCIDENT DOSES IN THE SPENT FUEL POOL AREA

Power level, megawatts thermal	2,900
Power peaking factor	1.65
Shutdown time, hours	100
Number of fuel rods assumed failed	314
Number of fuel rods in core	41,450
Fraction of Inventory in failed pins released to Pool:	
Noble gases, percent	10
Iodine, percent	10
Fraction of iodine in pool released from pool, percent (based upon a minimum of 23 feet of water above the top of the fuel rods in the storage racks)	1
Fuel building filter efficiencies:	
Elemental iodine, percent	90
Organic iodine, percent	70
Iodine distribution above pool:	
Elemental, percent	75
Organic, percent	25
Relative concentrations, seconds per cubic meter	
0-2 hours at exclusion area boundary	3.3×10^{-4}
0-2 hours at low population zone boundary	4.1×10^{-5}

A preoperational test will be performed after the installation of additional equipment in the fuel handling area, to determine whether a negative pressure of 0.125-inch water gauge can be obtained and maintained in the fuel handling area, as described in Amendment 19 to the Final Safety Analysis Report. The applicant will report these results to the NRC staff as verification of the correct operation of the system. The applicant has also provided a technical specification stating that the fuel handling area will be maintained at a negative 0.125-inch water gauge during all fuel handling operations.

Based upon the proposed operation of the ventilation system during fuel handling, we conclude that the design and operation of the facility, with respect to the fuel handling accident in the fuel handling area, will limit the estimated radiological consequences following a postulated fuel handling accident to values less than the acceptance criteria of Section 15.7.4 of the Standard Review Plan.

With regard to a postulated fuel handling accident inside containment, the applicant has provided the results of an analysis of the radiological consequences of such an accident using assumptions which are comparable to those given in Regulatory Guide 1.25 mentioned above. We have independently evaluated the potential for releases of radioactivity should this accident occur. During refueling operations, the containment atmosphere is exhausted through non-seismic ductwork and charcoal filters to the plant vent. Although no quantitative credit was given for this filtration system (since the filters and the ductwork are not seismically designed), this filtration system provides an additional margin of safety.

In addition, the applicant has provided seismic, safety-grade instrumentation in the area of the refueling pool to rapidly detect any release of radioactivity, should the postulated accident occur, and to generate the isolation signal to the reactor building purge system isolation valves should abnormal radiation levels be detected.

We have reviewed the proposed design and conclude that the ventilation system, as described above, including the instrumentation to rapidly detect a release of radioactivity and initiate closure of the purge isolation valves provides sufficient assurance that the offsite doses would be no greater than those calculated by the staff for the fuel handling accident in the spent fuel pool area and presented in Table 15-5 of this Safety Evaluation Report.

15.4.3 Control Rod Ejection Accident

For the postulated control rod ejection accident, a mechanical failure of a control rod mechanism housing is assumed such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position. This results in the release of radioisotopes to the environment through both the containment building and the secondary system. Although the resulting doses at the exclusion area boundary in case of an actual accident would be the composite of the doses computed for releases via the containment buidling and through the secondary system, the individual doses presented in Table 15-4 of this Safety Evaluation Report for Cases 1 and 2 assume that all the activity is released through the specific release pathway identified for each case. The evaluation of radiological consequences has been performed by the NRC staff using the recommendations of Regulatory Guide 1.77 and a conservative description of the facility response to the accident. The assumptions used to determine the potential consequences from releases through the containment and through the secondary system are presented in Table 15-6 of this Safety Evaluation Report. The calculated doses are listed in Table 15-4 of this Safety Evaluation Report and are well within the guidelines of 10 CFR Part 100. Technical specification limits on primary-secondary coolant leakage assure that the potential doses can be maintained well within the exposure guidelines of 10 CFR Part 100.

15.4.4 Steamline Break Accident

Both we and the applicant have evaluated the radiological consequences of a postulated steamline break accident occurring outside containment and upstream of the main isolation valve. Although the contents of the secondary side of the affected steam generator would be vented initially to the atmosphere as an elevated release, we have conservatively assumed that the entire release throughout the course of the accident is released under ground level conditions. During the course of the accident, the shell side of the affected steam generator was assumed to stay dry since emergency feedwater flow to the affected steam generator would be blocked off under the conditions of this accident. Due to the dry-out condition, all iodine transported to the secondary side by leakage (one gallon per minute) was assumed available for release to the atmosphere with no reduction due to holdup or attenuation in determining the acceptability of the design. We have evaluated three scenarios. For the first case, the applicant has indicated that departure from nucleate boiling may occur in five percent of the fuel if the most reactive control rod is assumed to remain fully withdrawn. As a second case, as a result of the power and pressure transient, we assumed that an iodine spike occurred in which the iodine release rate from fuel to coolant is increased by a factor of 500. Prior to the accident, the plant was assumed to be operating at a primary coolant activity level of one microcurie per gram. The third case assumed the primary coolant activity to be at the transient full power technical specification limit of 60 microcuries per gram of dose equivalent I-131.

Experience has shown that, for Cases 1 and 3 (both of which are compared to dose guidelines of 10 CFR Part 100) Case 1 is limiting when significant fuel failure is assumed. Case 2 is compared to a small fraction of the guidelines of 10 CFR Part 100. Our calculated doses and assumptions are presented in Tables 15-4 and 15-7 respectively, of this Safety Evaluation Report.

Our analysis has shown that technical specification limits on primary and secondary coolant activities and primary to secondary leak rate will limit potential doses to small fractions of the exposure guidelines of 10 CFR Part 100. The potential doses shown in Table 15-4 are within the exposure guidelines of 10 CFR Part 100 even if the accident should occur with a preaccident iodine spike or assuming additional fuel failures occur during the accident as a result of the most reactive control rod remaining fully withdrawn.

15.4.5 Steam Generator Tube Rupture Accident

A non-mechanistic guillotine break of a steam generator tube is postulated to occur when the reactor is at power. The initial leak rate of primary coolant through each end of the broken tube is initially high and gradually decreases

TABLE 15-6

ASSUMPTIONS USED IN ANALYSIS OF CONTROL ROD ASSEMBLY EJECTION ACCIDENT

Assumptions Common to Both Cases

1. Thermal power level of 2900 megawatts.
2. Ten percent of iodine and noble gas inventory in gap of failed fuel.
3. Release of total gap activity in failed fuel to containment building.
4. Standard ground level release meteorology and dose conversion factors.

Assumptions for Case 1*

1. Ten percent fuel failed in transient.
2. Fifty percent plate-out of radioactive iodines.
3. Containment building sprays are not initiated.
4. Containment building leak rate of 0.20 percent per day for 24 hours and one-half of this value thereafter.

Assumptions for Case 2**

1. Ten percent fuel with clad failures after accident.
2. One-quarter percent fuel melted after accident.
3. One hundred percent of noble gases and 25 percent of iodines contained in melted fuel instantaneously released to reactor coolant system.
4. Pressure equalization between primary and secondary systems reached in 175 seconds.

*Assumes all releases from the containment

**Assumes all releases through the secondary system

TABLE 15-7

ASSUMPTIONS USED IN ANALYSIS OF STEAMLINE BREAK ACCIDENT

Initial thermal power of 2900 megawatts.

Five percent of fuel failed during transient, 10 percent of activity in gap for the case where the most reactive control rod remains fully withdrawn.

The preexisting iodine concentration of one microcurie per gram is assumed when all control rods are fully inserted. Beginning at time=0, the release rate of iodine from defective fuel is assumed to increase by a factor of 500.

Offsite power is not available.

Primary-to-secondary leak rate at one gallon per minute for two hours, all leakage occurring in the steam generator associated with the failed steam line.

Steam line failure outside containment upstream of the isolation valve.

as the pressure difference between the reactor vessel and the steam generator is reduced. This leak rate is larger than the maximum capacity of the charging pumps to maintain inventory and of the pressurizer heaters to maintain pressure. The resultant reactor and turbine trip terminate power output to the grid. This disturbance to the grid is assumed to cause loss of offsite power to the facility.

With loss of offsite power, plant cooldown is effected by a combination of the operation of automatic safety valves and manual atmospheric relief valves. Diagnosis of the accident is achieved by observing annunciations of the condenser high radiation alarm, steam generator feedwater/steam flow mismatch, decreasing pressurizer pressure and level, and increasing level in one steam generator. We assumed, conservatively, that it would take the operator 30 minutes from the onset of tube rupture to diagnose the accident and isolate the affected steam generator.

The applicant has calculated that 48,000 pounds of steam will be released from the affected steam generator before it can be isolated. In evaluating the consequences of this accident, we have conservatively assumed that all of the steam released was primary coolant. All of the activity in the coolant is assumed to become airborne. This highly conservative but simplified analysis was performed to determine the need for a more detailed mechanistic analysis identified in the Standard Review Plan.

Two cases have been analyzed, one assuring the primary coolant activity level is at the steady state Technical Specification limit and the other at the transient full power limit. The assumptions used in our analyses are listed in Table 15-4 of this Safety Evaluation Report. Technical specification limits on primary and secondary coolant activities will limit potential doses to small fractions of the exposure guidelines of 10 CFR Part 100. The potential doses shown in Table 15-4 of this Safety Evaluation Report are within the exposure guidelines of 10 CFR Part 100 even if the accident were to occur with a preaccident iodine spike.

15.4.6 Reactor Coolant Pump Locked Rotor

The applicant has indicated that a reactor coolant pump locked rotor accident could lead to 10 percent of the fuel rods violating departure from nucleate boiling criteria. We assume such departure to lead to fuel clad failure, releasing 10 percent of the radioiodine and noble gas inventory in that fraction of the fuel. The primary coolant system remains intact and radioiodine releases are through steam generator tube leakage to the secondary system. Should the main turbine condensers not be available during recovery from this accident, it is assumed that steam is dumped to the environment by a combination of the operation of the safety valves and manual atmospheric relief valves. The applicant's analysis indicates that the steam generator relief and safety valves will be utilized for a total period of eight hours after the accident to maintain reactor cooling. We have assumed that the technical specification steam generator tube leakage is occurring to one generator for the full eight hours and that this generator was initially at the secondary coolant technical specification limit for radioiodine. The other assumptions we have utilized as similar to those used for Case 2 of the control rod assembly ejection accident shown in Table 15-6 of this Safety Evaluation Report.

TABLE 15-8

ASSUMPTIONS USED IN ANALYSIS
OF CONTROL ROD ASSEMBLY EJECTION ACCIDENT

Assumptions Common to Both Cases

Initial thermal power level of 2900 megawatts.

Primary-to-secondary leak rate of one gallone per minute to unaffected steam generator.

Initial secondary coolant activity of 0.1 microcuries per gram.

Pressure equalized between primary and secondary systems in 30 minutes.

Loss of offsite power.

Assumptions for Case 1*

Tube failure occurred during recovery from an earlier event which increased primary coolant activity to 60 micorcuries per gram of I-131 equivalent.

All of the iodine in 48,500 points of primary coolant and 14 curies of secondary system iodine inventory are volitized in the steam and released to the environment.

Assumptions for Case 2**

Tube failure occurred during operation at normal maximum primary coolant contamination limits, one micorcurie per gram of I-131 equivalent.

Source Spike factor after accident is assumed to be 500.

The dose as shown in Table 15-9 of this Safety Evaluation Report is an acceptably small fraction of the 10 CFR Part 100 guidelines.

15.4.7 Liquid Tank Failure Accident

The consequences of component failures for components located outside the reactor containment, which could result in releases of liquids containing radioactive materials to the environs, were evaluated. Our evaluation considered: (1) the radionuclide inventory in each component assuming a one percent operating power fission product source term; (2) a component liquid inventory equal to 80 percent of its design capacity; (3) the mitigating effects of the facility design including overflow lines and the location of storage tanks in curbed areas designed to retain spillage; and (4) the effects of site geology and hydrology.

The applicant has incorporated provisions in the design to retain releases from liquid overflows as discussed in Section 11.2.1 of this Safety Evaluation Report.

We determined that there are no groundwater users down-gradient from potential liquid releases due to liquid tank failures.

In the event of a spill, we postulated liquid flow directly to groundwater beneath the plant and transport to the Broad River. The nearest users which could be affected would be those using the Broad River for drinking water.

Based upon our evaluation, the potential tank failure resulting in the greatest quantity of activity released to the environment is failure of the 5,000-gallon waste evaporator concentrates holdup tank. In our evaluation, we have determined the liquid transit time for the leakage to the nearest user to be 11.1 years and the minimum dilution to be 1.9×10^6. Considering the leakage, groundwater dilution, river dilution and transit time, the calculated radionuclide concentrations in the Broad River result in values that are small fractions of the limits of Table II, Column 2, of Appendix B to 10 CFR Part 20 for unrestricted areas. Based upon the foregoing evaluation, we conclude that the provisions incorporated in the applicant's design to mitigate the effects of component failures involving contaminated liquids are acceptable.

16 TECHNICAL SPECIFICATIONS

The Technical Specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the NRC. The finally approved Technical Specifications will be made a part of the operating license. Included will be sections covering definitions, safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

The Technical Specifications for this facility will be based on, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," (NUREG-0452, Revision 2). This document has been updated from earlier revisions as a result of continued discussion with Westinghouse and other licensees with Westinghouse pressurized water reactors.

We have worked with the applicant and have prepared a draft of the Technical Specifications for the Virgil C. Summer Nuclear Station, Unit 1. On the basis of our review to date, we conclude that normal plant operation within the limits of the Technical Specifications will not result in offsite exposures in excess of the 10 CFR Part 20 limits. Furthermore, the limiting conditions for operation and surveillance requirements will assure that necessary engineered safety features will be available in the event malfunctions within the facility.

17 QUALITY ASSURANCE

17.1 General

The description of the quality assurance program for the operations phase of the Virgil C. Summer Nuclear Station, Unit 1, is contained in Section 17.2 of the Final Safety Analysis Report through Amendment 22. Our evaluation of the quality assurance program is based upon a detailed review of this information and discussion with representatives from the applicant. We assessed the applicant's quality assurance program description to determine its compliance with the requirements of Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," the Standard Review Plan, Section 17.2, Rev. 0 dated November 24, 1975, "Quality Assurance During the Operations Phase," and the applicable regulatory guidance listed in Table 17-1 of this Safety Evaluation Report.

17.2 Organization for the Quality Assurance Program

The organizational structure responsible for the operation of the facility and for the establishment and execution of the quality assurance program for the operations phase is shown in Figure 17-1. The Executive Vice President, Operations, is responsible for the administration of safe and efficient production of electric power for the facility. The Vice President, Group Executive, Nuclear Operations provides direction for quality assurance services and is responsible for developing and specifying the operational quality assurance program policy and assigning sufficient authority to organizations to assure these objectives will be attained. The Vice-President and Group Executive, Nuclear Operations who reports to the Executive Vice President, Operations is responsible for assuring an independent quality assurance function in order to comply with Appendix B to 10 CFR Part 50. Responsibility for quality assurance administration has been assigned to the Manager of Quality Assurance, who reports to the Group Manager, Quality Assurance and assures proper implementation of the operational quality assurance program. The operational quality assurance program description in Section 17.2 of the Final Safety Analysis Report provides a description of how the 18 criteria of Appendix B to 10 CFR Part 50 will be fulfilled. Implementation of the quality assurance program description will be carried out by the operational quality assurance plan and corresponding quality assurance procedures. The operational quality assurance plan is developed, maintained, implemented, and controlled by the Manager of Quality Assurance. The Group Manager, Quality Assurance has the responsibility to provide administrative control, coordination, and evaluation of the operational quality assurance program.

The Director of Surveillance Systems and onsite quality assurance staff will consist of approximately ten people and can be supplemented during major modifications or other periods requiring additional quality assurance coverage. The Director of Surveillance Systems and staff are responsible for the overall program effectiveness at the facility, and report to the Manager of Quality

TABLE 17-1

REGULATORY GUIDANCE APPLICABLE TO
QUALITY ASSURANCE PROGRAM

1. Regulatory Guide 1.28 (Revision 0 - June 1972), "Quality Assurance Program Requirements (Design and Construction)."

2. Regulatory Guide 1.30 (Revision 0 - August 1972), "Quality Assurance Requirements for Installation, Inspection, and Testing of Instrumentation and Electric Equipment."

3. Regulatory Guide 1.33 (Revision 1 - January 1977), "Quality Assurance Program Requirements (Operation)."

4. Regulatory Guide 1.37 (Revision 0 - March 1973), "Qualtiy Assurance Requirements for Cleaning of Fludi Systems and Associated Components of Water-Cooled Nuclear Power Plants."

5. Regulatory Guide 1.38 (Revision 0 - March 1973), "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants."

6. Regulatory Guide 1.39 (Revision 2 - September 1977), "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."

7. Regulatory Guide 1.58 (Revision 0 - August 1973), "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel."

8. Regulatory Guide 1.64 (Revision 2 - June 1976), "Quality Assurance Requirements for the Design of Nuclear Power Plants."

9. Regulatory Guide 1.74 (Revision 0 - February 1974), "Quality Assurance Terms and Definitions."

10. Regulatory Guide 1.88 (Revision 2 - October 1976), "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records."

11. Regulatory Guide 1.94 (Revision 1 - April 1976), "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants."

12. Regulatory Guide 1.116 (Revision 0-R - June 1976), "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems."

TABLE 17-1 (Continued)

REGULATORY GUIDANCE APPLICABLE TO
QUALITY ASSURANCE PROGRAM

13. Regulatory Guide 1.123 (Revision 1 - July 1977), "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants."

14. Regulatory Guide 1.44 (Revision 0 - January 1979), "Auditing of Quality Assurance Programs for Nuclear Power Plants."

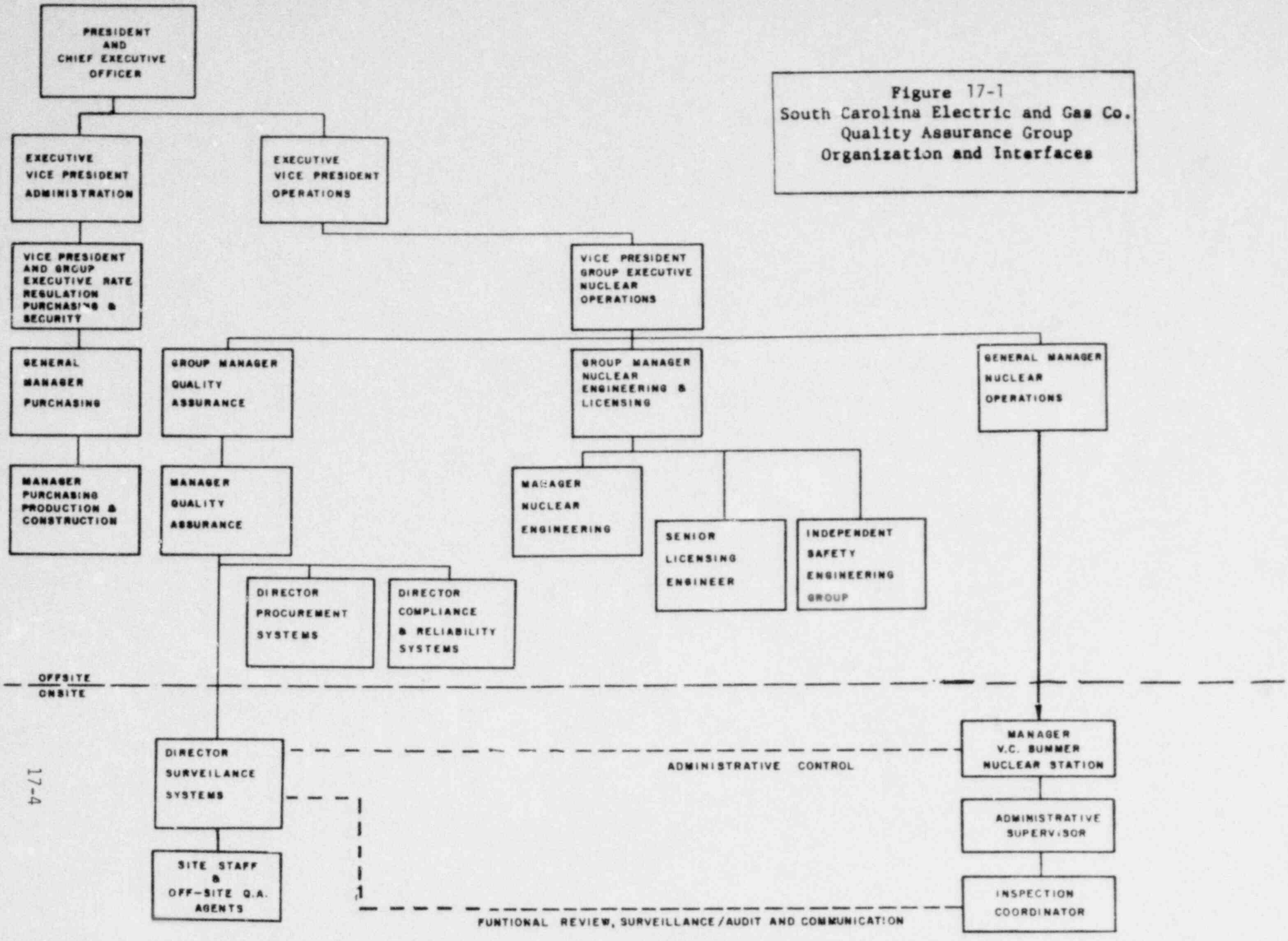


Figure 17-1
South Carolina Electric and Gas Co.
Quality Assurance Group
Organization and Interfaces

Assurance. Inspection of safety-related work at the facility is the responsibility of the Inspection Coordinator who reports to the Administrative Supervisor and maintains a line of communication with the Director of Surveillance Systems with respect to quality matters. The Inspection Coordinator has a staff of nine personnel qualified in the disciplines of instrumentation and control, mechanical engineering, and nondestructive examination who perform inspections of all operational activities relative to their respective disciplines.

The quality assurance organization has the authority to: (1) identify quality problems, (2) initiate, recommend, or provide solutions, (3) verify implementation of solutions, and (4) stop unsatisfactory work or further processing of unsatisfactory material. The quality assurance organization is responsible for: (1) reviewing and approving quality-related documents, e.g., instructions, procedures, drawings, and specifications, (2) performing vendor quality assurance prequalifications, (3) assuring that design and procurement documents contain quality requirements, (4) surveillance and auditing of vendors, (5) documenting and reporting to management nonconformances discovered during surveillance or audit, (6) assuring corrective actions are effective and accomplished in a timely manner, and (7) auditing of maintenance and operation activities.

The Executive Vice-President, Operations, is responsible for the administration of the operation of the facility in a safe manner. Reporting to him is the Vice-President and Group Executive, Nuclear Operations, assisted by the Manager, Nuclear Operations who exercises managerial control over the operation, maintenance, and modification activities of the facility. This responsibility is coordinated through the Manager, Virgil C. Summer Nuclear Station, Unit 1, who exercises managerial and supervisory control over all facility personnel to implement and coordinate company policy with regard to operation of the facility. Disputes between quality assurance and quality control personnel and other organizations which cannot be resolved are referred to higher management for resolution.

17.3 Quality Assurance Program

The quality assurance program for the operations phase is structured to be in accordance with Appendix B to 10 CFR Part 50 and with the provisions of the NRC regulatory guidance shown in Table 17-1. The operational quality assurance program is implemented via the operational quality assurance plan and procedures manuals. These documents, coupled with the quality assurance program description and Technical Specifications form the foundation from which the overall quality assurance function is formulated. Together, they describe how the requirements of Appendix B to 10 CFR Part 50 are satisfied. The Manager, Quality Assurance is responsible for the development, implementation, and control of the quality assurance plan. Each organizational manager is responsible for identifying the activities affecting quality and for assuring that these activities are adequately described by procedures. Quality-related procedures will be reviewed and concurred with by the quality assurance organization.

The quality assurance program for the facility requires that quality assurance documents encompass detailed controls for: (1) translating codes, standards, and regulatory requirements into tchnical specifications, procedures, and instructions, (2) developing, reviewing, and approving procurement documents, including changes, (3) prescribing all quality-affecting activities by documented instructions, procedures, or drawings, (4) issuing and distributing approved documents,

(5) qualifying and certifying quality assurance and quality control personnel, (6) purchasing items and services, (7) identifying materials, parts, and components, (8) performing special processes, (9) inspecting and/or testing material, equipment, processes or services, (10) calibrating and maintaining measuring and test equipment, (11) handling, storing, and shipping of items, (12) identifying the inspection, test, and operating status of safety-related items, (13) identifying and dispositioning nonconforming items, (14) correcting conditions adverse to quality, (15) preparing and maintaining quality assurance records, and (16) auditing of activities which affect quality.

The indoctrination and training program established by the South Carolina Electric and Gas Company quality assurance organization assures that personnel performing activities affecting quality are knowledgeable in quality assurance requirements, implementing procedures, and instructions and that they have competence and skill in the performance of their quality-related activities.

Quality is verified through review, surveillance, inspection, testing, checking, and audit of quality-related activities. The quality assurance program requires that quality verification and inspections be performed by individuals who are not directly responsible for performing the actual work activity. Inspections are performed with procedures, instructions, and/or checklists reviewed and concurred with by the South Carolina Electric and Gas Company quality assurance organization.

The South Carolina Electric and Gas Company quality assurance organization is responsible for the establishment and implementation of the audit program. Audits are performed in accordance with established procedures by qualified personnel not having direct responsibilities in the areas being audited. Audits will be performed by the Quality Assurance Department to evaluate all aspects of the quality assurance program including the effectiveness of implementation.

The quality assurance program requires documentation of audit results and review by the person having responsibility in the area audited to determine and take corrective action.

Follow-up audits are performed to determine that nonconformances are corrected and that the corrective action precludes repetitive occurrences. Audit reports, which indicate performance trends and the effectiveness of the quality assurance program, are prepared and issued to responsible mangement for review and assessment.

17.4 Conclusion

Based on our detailed review and evaluation of the quality assurance program description contained in Section 17.2 of the Final Safety Analysis Report, we conclude that:

1. The organizations and persons performing quality assurance functions have the required independence and authority to carry out the quality assurance program effectively without undue influence from those directly responsible for cost and schedules.

2. The quality assurance program describes requirements, procedures, and controls that, when properly implemented, comply with the requirements of Appendix B to 10 CFR Part 50 and with the criteria contained in Section 17.2 of the Standard Review Plan.

3. Except as noted below, the quality assurance program covers activities affecting structures, systems, and components important to safety as identified in the Final Safety Analysis Report.

Accordingly, we conclude that the applicant's description of the quality assurance program with the exception of our review of the applicant's response to our position on those systems, structures, and components which should be under the control of the quality assurance program, is in compliance with applicable NRC regulations. We will report the resolution of this matter in a supplement to this Safety Evaluation Report.

18 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In its letter of November 15, 1972, the Advisory Committee on Reactor Safeguards (Committee) indicated that certain matters would require further attention and resolution during construction of the Virgil C. Summer Nuclear Station, Unit 1. These items were addressed in Supplement No. 1 to our Safety Evaluation Report dated January 12, 1973.

Certain of these matters are addressed further in this Safety Evaluation Report, as identified below. References are given to sections in this report related to the construction of the facility for further discussion.

Service Water Pond Dams	Section 2.6
Emergency Core Cooling System	Section 6.3
Incore Instrumentation	Section 22.0
Turbine Missiles	Section 3.5
Pipe Whip from Postulated High Energy Pipe Breaks	Section 3.6
Protection for Postulated Rupture of a Main Steamline - Outside Containment	Section 3.6
Densification of Fuel Pellets	Section 4.3
Anticipated Transients Without Scram	Section 15.3.5
Issues Generic to Light Water Reactors	Appendix C
Seismicity in the Vicinity of Charleston South Carolina	Section 2.5

The operating license application for the proposed facility is being reviewed by the Committee. We intend to issue a supplement to this Safety Evaluation Report after the Committee's report on this application is available. The supplement will append a copy of the Committee's report and will address the significant comments made by the Committee, and will address the significant comments made by the Committee, and will also describe steps taken by us to resolve any issues raised as a result of the Committee's review.

19 COMMON DEFENSE AND SECURITY

The applicant has stated that the activities to be conducted will be within the jurisdiction of the United States and that all the directors and principal officers of the applicant are citizens of the United States. The applicant is not owned, dominated, or controlled by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data that might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes is involved. For these reasons, and in the absence of any information to the contrary, we find that the activities to be performed will not be inimical to the common defense and security.

20 FINANCIAL QUALIFICATIONS

The NRC's regulations which relate to financial data and information required to establish financial qualifications for applicants for a facility operating license are embodied in Section 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50. To assure that we have the latest information to make a determination of the financial qualifications of applicants, it is our current practice to review this information during the later stages of our review of an application. We are continuing our review of the applicant's financial qualifications and will report the results of our evaluations in a supplement to this Safety Evaluation Report.

21 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

21.1 General

Pursuant to the financial protection and idemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the NRC has issued regulations in 10 CFR Part 140. These regulations set forth the NRC's requirements with regard to proof of financial protection by, and idemnification of, licenses for facilities such as power reactors under 10 CFR Part 50.

21.2 Preoperational Storage of Nuclear Fuel

The NRC's regulations in 10 CFR Part 140 require that each holder of a construction permit under 10 CFR Part 50, that is also the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintained financial protection in the amount of $1,000,000 and execute an indemnity agreement with the NRC Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

The applicant has furnished the NRC proof of financial protection in the amount of $1,000,000 in the form of a Nuclear Energy Liability Insurance Association Policy (Nuclear Energy Liability Policy, Facility Form No. FN-226). Further, the applicant has executed an Indemnity Agreement with the NRC effective as of the date of its preoperational fuel storage license. The applicant has paid the annual indemnity fee applicable to preoperational fuel storage.

21.3 Operating Licenses

Under the NRC's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for the Virgil C. Summer Nuclear Station, Unit 1 (which has a rated capacity in excess of 100,000 electrical kilowatts), is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which amount is currently $140 million.

Accordingly, a license authorizing operation of the Virgil C. Summer Nuclear Plant, Unit 1 will not be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended so that the policy limits have been increased, to meet the requirements of the NRC's regulations for reactor operation. Similarly, operating licenses will not be issued until an appropriate amendment to the present indemnity agreement has been executed. The applicant will be required to pay an annual fee for operating license indemnity as provided in our regulations.

On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the operating license, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to an operating license, including those as to proof of financial protection in the requisite amount and as the execution of an appropriate indemnity agreement with the NRC.

22 TMI-2 REQUIREMENTS

22.1 Introduction

The accident at Three Mile Island (TMI) Unit 2 resulted in requirements which were developed from the recommendations of several groups established to investigate the accident. These groups include the Congress, the General Accounting Office, the President's Commission on the Accident at Three Mile Island, the NRC Special Inquiry Group, the NRC Advisory Committee on Reactor Safeguards, the Lessons-Learned Task Force and the Bulletins and Orders Task Force of the NRC Office of Nuclear Reactor Regulation, the Special Review Group of the NRC Office of Inspection and Enforcement, the NRC Staff Siting Task Force and Emergency Preparedness Task Force, and the NRC Offices of Standards Development and Nuclear Regulatory Research. The report NUREG-0660 entitled "NRC Action Plan Developed as a Result of the TMI-2 Accident" (Action Plan) was developed to provide a comprehensive and integrated plan for the actions now judged necessary by the NRC to correct or improve the regulation and operation of nuclear facilities. The Action Plan was based on the experience from the TMI-2 accident and the recommendations of the investigating groups.

In the development of the Action Plan (NUREG-0660), the NRC has transformed the recommendations of the investigating groups into discrete scheduled tasks that specify changes in its regulatory requirements, organization, or procedures. Some actions to improve the safety of operating plants were judged to be necessary before an action plan could be developed, although they were subsequently included in the Action Plan. Such actions came from the bulletins and orders issued by the NRC immediately after the accident, the first report of the Lessons-Learned Task Force issued in July 1979, and the recommendations of the Emergency Preparedness Task Force. Before these immediate actions were applied to operating plants they were approved by the Commission.

Our review of TMI-2 requirements is based on NUREG-0694, "TMI-Related Requirements for New Operating Licenses." The Virgil C. Summer Nuclear Station, Unit 1 was reviewed for comformance with the NRC regulations and the TMI-2 requirements.

The TMI-related requirements and actions for new operating licenses as specified in NUREG-0694 are of four types: (1) those required to be completed by the applicant prior to receiving a fuel loading and low-power testing license, (2) those required to be completed by the applicant prior to receiving a license to operate at appreciable power levels up to full power, (3) those the NRC will take prior to issuing licenses, either for fuel loading and low power testing or for full power operation, and (4) those required to be completed by an applicant prior to a specified date.

This section of the Safety Evaluation Report addresses the applicant's implementation of the TMI-related requirements for the facility. The applicant has provided a report, "Response to TMI-2 Action Plan," by its letter dated June 20, 1980, that gives its initial response to our requirements. During our review, we met with the applicant in our offices and at the facility. The applicant has amended its initial response as a result of our review. Meeting results and applicant's letters relevant to our review are discussed in applicable sections of this Safety Evaluation Report.

Each of the following sections corresponds to one of the four parts of NUREG-0694. Section 22.2 addresses fuel loading and low-power testing requirements. Section 22.3 addresses full power requirements. Section 22.4 addresses NRC actions. Section 22.5 addresses dated requirements. All of the requirements of NUREG-0694 are addressed.

On October 31, 1980, in a letter from D. Eisenhut to all licensees of operating plants, and applicants for operating licenses and holders of construction permits, the staff issued NUREG-0737, "Clarification of TMI Action Plan Requirements". NUREG-0737 added new requirements to those already issued in NUREG-0694. In addition, NUREG-0737 provided additional clarification of many of the NUREG-0694 requirements. This section of the Safety Evaluation Report reflects our evaluation of the NUREG-0694 requirement and, to the extent practicable, addresses the clarification of these items as provided in NUREG-0737. This Safety Evaluation Report does not address those additional items required by NUREG-0737. The applicant has recently amended its Final Safety Analysis Report to address the additional requirements of NUREG-0737. We will evaluate the applicant's responses and report our evaluation in a supplement to this Safety Evaluation Report.

A summary of the differences between NUREG-0737 and NUREG-0694 is provided below:

Item	Shortened Title	Change
I.A.1.1	Shift Technical Advisor	New Requirements
I.A.1.3	Shift Manning	Clarification
I.A.2.1	Immediate Upgrading of RO and SRO Training and Qualification	Clarification
I.A.2.3	Administration of Training Program	Clarification
I.A.3.1	Revise Scope and Criteria for Licensing Examinations	New Requirements
I.C.1	Short Term Accident and Procedure Review	Clarification
I.C.5	Review and Revise Procedures	Clarification
I.C.6	Verify Correct Performance of Operating Activities	New Requirements
I.D.2	Plant Safety Parameter Display Console	Clarification

Item	Shortened Title	Change
II.B.1	Reactor Coolant System Vents	Clarification
II.B.2	Plant Shielding	Clarification
II.B.3	Post-Accident Sampling	Clarification
II.D.1	Relief and Safety Valve Test Requirements	New Requirements
II.D.3	Valve Position Indication	Clarification
II.E.3.1	Emergency Power for Pressurizer Heaters	Clarification
II.E.4.1	Dedicated Hydrogen Penetrations	Clarification
II.E.4.2	Containment Isolation Capability	New Requirements
II.F.1	Accident Monitoring Instrumentation	Clarification
II.F.2	Instrumentation for Detection of Inadequate Core Cooling	Clarification
II.K.2	Orders on B&W Plants	New Requirements
II.K.3	Final Recommendations, B&O Task Force	New Requirements
III.D.1.1	Measure Leak Rates and Establish Program to Keep Leakage Rates ALARA	Clarification
III.D.3.3	In-Plant Iodine Radiation Monitoring	Clarification
II.D.3.4	Control Room Habitability	Clarification

22.2 Fuel-Loading and Low-Power Testing Requirements

I.A.1.1 Shift Technical Advisor

Requirement

A technical advisor to the shift supervisor shall be present on all shifts and available to the control room within ten minutes. Although minimum training requirements have not been specified, shift technical advisors should enhance the accident assessment function at the facility.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.2.1.b, and letters of September 27 and November 9, 1979.)

Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The shift technical advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the facility for transients and accidents. The shift technical advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the shift technical advisors that pertain to the engineering aspects of assuring safe operation of the facility, including the review and evaluation of operating experience.

Clarification

1. Due to the similarity in the requirements for dedication to safety, training and onsite location and the desire that the accident assessment function be performed by someone whose normal duties involve review of operating experiences, our preferred position is that the same people perform the accident and operating experience assessment functions. The performance of these two functions may be split if it can be demonstrated that the persons assigned the accident assessment role are aware, on a current basis, of the work being done by those reviewing operating experience.

2. To provide assurance that the shift technical advisor will be dedicated to concern for the safety of the plant, our position has been that shift technical advisors must have a clear measure of independence from duties associated with the commercial operation of the plant. This would minimize possible distractions from safety judgments by the demands of commercial operations. We have determined that, while desirable, independence from the operations staff of the plant is not necessary to provide this assurance. It is necessary, however, to clearly emphasize the dedication to safety associated with the shift technical advisor position both in the shift technical advisor job description and in the personnel filling this position. It is not acceptable to assign a person, who is normally the immediate supervisor of the shift supervisor, to shift technical advisor duties as defined herein.

3. It is our position that the shift technical advisor should be available within ten minutes of being summoned and therefore should be onsite. The onsite shift technical advisor may be in a duty status for periods of time longer than one shift, and therefore, asleep at some times, if the ten minute availability is assured. It is preferable to locate those doing the operating experience assessment onsite. The desired exposure to the operating plant and contact with the shift technical advisor (if these functions are to be split) may be able to be accomplished by a group, normally stationed offsite, with frequent on-site presence. We do not intend, at this time, to specify or advocate a minimum time on site.

4. The implementation schedule for the shift technical advisor requirements is to have the shift technical advisor on duty by January 1, 1980 or fuel loading date, whichever is later, and to have shift technical advisors who have completed all training requirements, on duty by January 1, 1981. While minimum training requirements have not been specified for January 1, 1980, the shift technical advisors on duty by that time should enhance the accident and operating experience assessment function at the facility.

Additional clarification regarding the shift technical advisor requirements was provided in our letters dated September 5, 1980 and October 31, 1980 (NUREG-0737). These letters discussed the long-term shift technical advisor program and requested additional information on current and long-term shift technical advisor training programs from licensees by January 1, 1981 and from applicants on a schedule consistent with the NRC review schedule.

Discussion and Conclusions

The applicant has committed to provide a technical advisor to the shift supervisor who will be present on all shifts and available to the control room within ten minutes. The shift technical advisor's primary duty will be to support the diagnosis of off-normal events and to advise the shift supervisor on actions to terminate or mitigate the consequences of such events. The shift technical advisor will report to the technical support supervisor who is responsible for directing the activities of the plant technical support emergency group.

The shift technical advisors in training as of the July 8-10, 1980 site visit all hold degrees in engineering. Their training is to include facility operations with emphasis on transient and accident analyses, and facility response. It will also include simulator manipulation.

At this stage of review, it appears that the applicant is satisfying our requirement for providing shift technical advisors. In accordance with our clarification letters of September 5, 1980, and October 31, 1980 the applicant must provide a description of their shift technical advisor training to demonstrate conformance with our October 30, 1979 letter and a description of their long-term shift technical advisor program. We will review this information following receipt of their response to the clarification letter. We will further review the current status of the shift technical advisor program during the forthcoming site visit of the IE/NRR management review group. We will report further on this matter in a supplement to this Safety Evaluation Report.

I.A.1.2 Shift Supervisor Administrative Duties

Requirement

Review the administrative duties of the shift supervisor and delegate functions that, detract from or, are subordinate to the management responsibility for assuring safe operation of the facility to other personnel not on duty in the control room.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.2.1a, Item (4), and letters of September 27 and November 9, 1979.)

Position

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the facility under all conditions while on shift and that clearly establishes the shift supervisor's command duties.

2. Procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other management personnel. Particular emphasis shall be placed on the following:

   a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The principle shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

   c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function that the shift supervisor is to provide for assuring safety.

4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management

responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

Discussion and Conclusions

In Amendment 18 to the Final Safety Analysis Report, the applicant committed to the issuance of a corprorate management directive that establishes the command duties of the shift supervisor and emphasizes the shift supervisor's primary management responsibility for safe operation of the plant.

The applicant's senior facility and corporate management personnel have conducted a review of the administrative duties of shift supervisor and have added administrative personnel to the operating group in order to relieve the shift supervisor of routine duties.

The duties, responsibilities, and authority of the shift supervisor and control room operators have been defined in Administrative Procedure AP-518. We will review this procedure during the IE/NRR management review group site visit to assure that it adequately describes the duties, responsibilities and authority of these individuals, including emphasis on the need for the shift supervisor to remain in overall charge and not to become totally involved in any single operation when multiple operations are required in the control room. In addition, it must also adequately describe the mechanism for relief of the shift supervisor and the provisions for assumption of the command function in the event of a temporary absence of the shift supervisor from the control room.

The applicant states that on-the-job training and classes emphasize the responsibility for safe operation and management functions of the shift supervisor. We find that the applicant is in the process of completing actions to meet the requirements of TMI Action Plan Item I.A.1.2. We will review these actions during the forthcoming site visit of the IE/NRR management review group and we will report further on this matter in a supplement to this Safety Evaluation Report. It will assure that the required procedures are in effect prior to issuance of an operating license.

I.A.1.3 Shift Manning

Requirement

The minimum shift crew for a unit shall include three operators, plus an additional three operators when the unit is operating. Shift staffing may be adjusted at multi-unit stations to allow credit for operators holding licenses on more than one unit.

In each control room, including common control rooms for multiple units, there shall be at all times a licensed reactor operator for each reactor loaded with fuel and a senior reactor operator licensed for each reactor that is operating. There shall also be onsite at all times, an additional relief operator licensed for each reactor, a licensed senior reactor operator who is designated as shift supervisor, and any other licensed senior reactor operators required so that their total number is at least one more than the number of control rooms from which a reactor is being operated.

Administrative procedures shall be established to limit maximum work hours of all personnel performing a safety-related function to no more than 12 hours of continuous duty with at least 12 hours between work periods, no more than 72 hours in any seven-day period, and no more than 14 consecutive days of work without at least two consecutive days off.

These requirements shall be met before fuel loading.

Position

Assure that the necessary number and availability of personnel to man the operations shifts have been designated by the licensee. Administrative procedures should be written to govern the movement of key individuals about the plant to assure that qualified individuals are readily available in the event of an abnormal or emergency situation. This should consider the recommendations on overtime in NUREG-0578. Provisions should be made for an aide to the shift supervisor to assure that, over the long term, the shift supervisor is free of routine administrative duties.

Clarification

A position was provided in a July 31, 1980 letter from the NRC Director, Division of Licensing, to all applicants for operating licenses and licensees of operating plants which stated the NRC's interim criteria for shift staffing and limitations on use of overtime. The position was further modified in NUREG-0737, "Clarification of TMI Action Plan Requirements," issued on October 31, 1980. The updated position is an follows:

> "At any time a licensed nuclear unit is being operated in Modes 1-4 for a pressurized water reactor (power operation, startup, hot standby, or hot shutdown respectively) or in Modes 1-3 for a boiling water reactor (power operation, startup, or hot shutdown respectively), the minimum shift crew shall include two licensed senior reactor operators, one of whom shall be designated as the shift supervisor, two licensed reactor operators, and two unlicensed auxiliary operators. For a multi-unit station, depending upon the station configuration, shift staffing may be adjusted to allow credit for licensed senior reactor operators and licensed reactor operators to serve as relief operators on more than one unit; however, these individuals must be properly licensed on each such unit. At all other times, for a unit loaded with fuel, the minimum shift crew shall include one shift supervisor who shall be a licensed senior reactor operator, one licensed reactor operator, and one unlicensed auxiliary operator."

Adjunct requirements to the shift staffing criteria stated above are as follows:

a. A shift supervisor with a senior reactor operator's license, who is also a member of the plant supervisory staff, shall be onsite at all times when at least one unit is loaded with fuel.

b. A licensed senior reactor operator shall, at all times, be in the control room from which a reactor is being operated. The shift supervisor may from time to time act as relief operator for the licensed senior reactor operator assigned to the control room.

c. For any plant with more than one reactor containing fuel, the number of licensed senior reactor operators onsite shall, at all times, be at least one more than the number of control rooms from which the reactors are being operated.

d. In addition to the licensed senior reactor operators specified in a., b., and c. above, for each reactor containing fuel, a licensed reactor operator shall be in the control room at all times.

e. In addition to the operators specified in a., b., c., and d. above, for each control room from which a reactor is being operated, an additional licensed reactor operator shall be on site at all times and available to serve as relief operator for that control room. As noted above, this individual may serve as relief operator for each unit being operated from that control room, provided the individual holds a current license for each unit.

f. Auxiliary (non-licensed) operators shall be properly qualified to support the unit to which assigned.

g. In addition to the staffing requirements stated above, shift crew assignments during periods of core alterations shall include a licensed senior reactor operator to directly supervise the core alterations. This licensed senior reactor operator may have fuel handling duties but shall not have other concurrent operational duties.

Staffing requirements shall be completed by fuel load for operating license applicants.

Licensees of operating plants and applicants for operating licenses shall include in their administrative procedures (required by license conditions) provisions governing required shift staffing and movement of key individuals about the facility. These provisions are required to assure that qualified personnel to man the operational shifts are readily available in the event of an abnormal or emergency situation.

These administrative procedures shall also set forth a policy, the objective of which is to operate the plant with the required staff and develop working schedules such that use of overtime is avoided, to the extent practicable, for the plant staff who perform safety-related functions (e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, instrumentation and control technicians and key maintenance personnel).

IE Circular No. 80-02, "Nuclear Power Plant Staff Work Hours," dated February 1, 1980 discusses the concern of overtime work for members of the plant staff who perform safety-related functions.

The staff recognizes that there are diverse opinions on the amount of overtime that would be considered permissible and that there is a lack of hard data on the effects of overtime beyond the generally recognized normal eight-hour working day, the effects of shift rotation, and other factors. NRC has initiated studies in this area. Until a firmer basis is developed on working hours, the administrative procedures shall include as an interim measure the following guidance, which generally follows that of IE Circular No. 80-02. In the event that overtime must be used (excluding extended periods of shutdown for a refueling, major maintenance or major plant modifications) the following overtime restrictions shall be followed:

(1) An individual shall not be permitted to work more than 12 hours straight (not including shift turnover time).

(2) There should be a break of at least 12 hours (which can include shift turnover time) between all work periods.

(3) An individual shall not work more than 72 hours in any seven day period.

(4) An individual shall not work more than 14 consecutive days without having two consecutive days off.

However, recognizing that circumstances may arise requiring deviation from the above restrictions, such deviation may be authorized by the plant manager or his deputy, or higher levels of management in accordance with published procedures and with appropriate documentation of the cause.

If a reactor operator or senior reactor operator has been working more than 12 hours during periods of extended shutdown (e.g., at duties away from the control board), such individuals shall not be assigned shift duty in the control room without at least a 12-hour break preceding such an assignment.

NRC encourages the development of a staffing policy that would permit the licensed reactor operators and senior reactor operators to be periodically assigned to other duties away from the control board during their normal tours of duty.

If a reactor operator is required to work in excess of eight continuous hours, he shall be periodically relieved of primary duties at the control board, such that periods of duty at the board do not exceed about four hours at a time.

The guidelines on overtime do not apply to the shift technical advisor provided he or she is provided sleeping accommodations and a 10-minute availability is assured.

Operating license applicants shall complete these administrative procedures before fuel loading. Development and implementation of the administrative procedures at operating plants will be reviewed by the Office of Inspection and Enforcement beginning 90 days after July 31, 1980.

Discussion and Conclusions

Submittals from the applicant regarding the status of personnel currently in training for operator licenses indicate that 29 candidates are from the operating group. Additional candidates represent the training and technical support groups. The applicant has advised us that their intention is to provide sufficient licensed operators to permit operation with one shift devoted to training.

On December 30, 1980 the applicant submitted their plans for developing and maintaining a sufficient number of licensed operators for this facility. Additional clarification was provided in a letter dated January 22, 1981. The plans cover a five year operating period and include assumed attrition rates and license examination failure rates. The applicant's forecast shows a four shift rotation through June 1981 at which time a five shift rotation begins. The stated purpose of the fifth shift is to provide for ongoing operator training and the applicant plans to add a sixth operating shift at such time that there are sufficient operating personnel.

The applicant's plans indicate a reasonable margin between available and required operators over the forecasted time period and we conclude that their training program should provide sufficient licensee operators to staff the plant without a need for routine overtime.

During the IE/NRR site visit for management review, the staff will review the status of the administrative procedures relating to the shift manning position and will report the results in a supplement to this Safety Evaluation Report. IE will assure that procedures are in place prior to issuance of an operating license.

I.A.3.1 Revise Scope and Criteria for Licensing Examinations

Requirement

All reactor operator license applicants shall take a written examination with a new category dealing with the principles of heat transfer and fluid mechanics, a time limit of nine hours, and a passing grade of 80 percent overall and 70 percent in each category.

All senior reactor operator license applicants shall take the reactor operator examination, an operating test, and a senior reactor operator written examination with a new category dealing with the theory of fluids and thermodynamics, a time limit of seven hours, and a passing grade of 80 percent overall and 70 percent in each category.

These requirements shall be met before fuel loading. (See letter of March 28, 1980.)

Discussion and Conclusions

The applicant will comply with the requirement that the applicants for operator licenses will be required to grant permission to the NRC to inform their facility management regarding the results of examinations. In addition with each application, a notarized "consent to release information" form will be submitted granting permission to release examination results to the South Carolina Electric & Gas Company.

The applicant has submitted an outline of the training in heat transfer, fluid flow, thermodynamics and mitigation of accidents for their initial training and requalification program in Section 13.2 of the Final Safety Analysis Report. Also included is the revised examination criteria for accelerated training consistent with the new passing grade for issuance of licenses.

The modification to the requalification program covering training in specific reactivity control manipulation for startup, normal, abnormal and emergency operations has also been submitted by the applicant in Section 13.2 of the Final Safety Analysis Report.

Based on the information submitted by the South Carolina Electric & Gas Company, we conclude that the applicant has satisfied the requirements of item I.A.3.1.

I.B.1.2 Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants

Requirement

The licensee organization shall comply with the findings and requirements generated in an interoffice NRC review of licensee organization and management. The review will be based, in part, on an NRC document entitled "Draft Criteria for Utility Management and Technical Competence." The first draft of this document was dated February 25, 1980. The current draft was issued for interim use and public comment in September 1980 as NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources". These draft guidelines address the organization, resources, training, and qualifications of plant staff, and management (both on site and off site) for routine operations and the resources and activities (both on site and off site) for accident conditions.

The licensee shall establish a group that is independent of the plant staff, but is assigned on site to perform independent reviews of plant operational activities and that has a capability for evaluation of operating experiences at nuclear power plants.

Organizational changes are to be implemented on a schedule to be determined prior to fuel loading.

Position

Corporate management of the utility-owner of a nuclear power plant shall be sufficiently involved in the operational phase activities, including plant modifications, to assure a continual understanding of plant conditions and safety considerations. Corporate management shall establish safety standards for the operation and maintenance of the nuclear power plant. To these ends, each utility-owner shall establish an organization, parts of which shall be located on site, to: perform independent review and audits of plant activities; provide technical support to the plant staff for maintenance, modifications, operational problems, and operational analysis, and aid in the establishment of programmatic requirements for plant activities.

The licensee shall establish an integrated organizational arrangement to provide for the overall management of nuclear power plant operations. This organization shall provide for clear management control and effective lines of authority and communication between the organizational units involved in the management, technical support, and operation of the nuclear unit. The key characteristics of a typical organizational arrangement are:

a. Integration of all necessary functional responsibilities under a single responsible head.

b. The assignment of responsibility for the safe operation of the nuclear power plant(s) to an upper-level executive position.

Utility management shall establish a group, independent of the plant staff, but assigned on site, to perform independent reviews of plant operational activities. The main functions of this group will be to evaluate the technical adequacy of all procedures and changes important to safe operation of the facility, and to evaluate and assess the operating experience and performance of the facility.

Discussion and Conclusion

The interoffice (IE/NRR) NRC review of the applicant's organization and management has not taken place yet and therefore no conclusions can be made on organization and management adequacy. A status report is provided here to cover the staff review to date since a significant amount of staff and applicant effort has been spent in this area.

On July 8-10, 1980 NRC staff members met with representatives of the applicant in both their corporate offices and in the plant to discuss their qualifications to operate the facility. The applicant was in the process of a corporate reorganization and the visit provided an opportunity for to review both the proposed organization, and the qualifications of their key personnel against our draft criteria. The review was accomplished by means of presentations by the applicant followed by staff interviews with key personnel on the corporate and plant staffs. An exit meeting was held to permit the NRC staff to present its concerns. The major concerns presented were as follows:

1) The corporate engineering group should be relocated, organizationally, under the Vice President and Group Executive for Nuclear Operations, the individual responsible for all activities related to nuclear plants. (It is noted that he was assigned no responsibility for activities related to non-nuclear plants.)

2) A corporate nuclear training function should be established to follow corporate training and to provide a cognizance of plant training to the Vice President and Group Executive for Nuclear Operations.

3) A corporate nuclear maintenance function should be established.

4) The size of the engineering organization could result in too much reliance on outside assistance.

5) Administrative procedures are required to address responsibilities and information flow. and responsibilities. Of particular importance was those procedures addressing the interface between the plant and corporate staffs.

6) There appeared to be insufficient hands-on operating experience with large pressurized water reactors in the operating organization.

7) The Onsite Independent Safety Engineering Group and Corporate Review Group should start functioning as soon as possible to enable them to gain experience in the final construction and pre-operational testing phases.

8) Management should assure that all levels of plant staffing understand and are committed to making the shift supervisor/shift technical advisor interface function properly.

On September 9, 1980 the applicant documented their response to the staff concerns listed above. Their response provided a satisfactory approach to all items but final implementation must still be reviewed at the time of the site visit for management review.

This item will be addressed, in its entirety, in a supplement to this Safety Evaluation Report.

I.C.1 Accident Analysis and Procedure Revision

Requirement

Analyze small-break loss-of-coolant accidents over a range of break sizes, locations, and conditions (including some specified multiple equipment failures) and inadequate core cooling due to both low reactor coolant system inventory and the loss of natural circulation to determine the important phenomena involved and expected instrument indications. Based on these analyses, revise as necessary, emergency procedures and training.

These requirements shall be met before fuel loading. (See NUREG-0578, Sections 2.1.3b and 2.1.9, and letters of September 27 and November 9, 1979.)

Position

Analyses, procedures, and training addressing the following are required:

1. Small-break loss-of-coolant accidents;

2. Inadequate core cooling; and

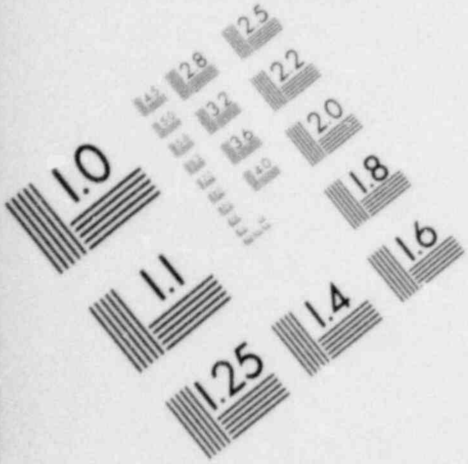
3. Transients and accidents.

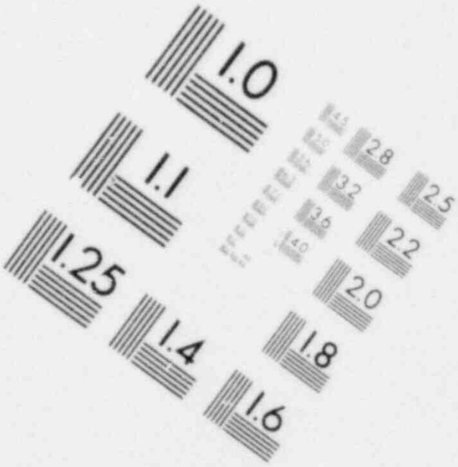
Analysis of small breaks greviously required by the Bulletins and Order Task Force must be completed. In addition, pretest calculations of some of the Loss of Fluid Test facility small-break tests shall be performed as means to verify the analyses performed in support of the small-break emergency procedures and in support of an eventual long-term verification of compliance with Appendix K to 10 CFR Part 50.

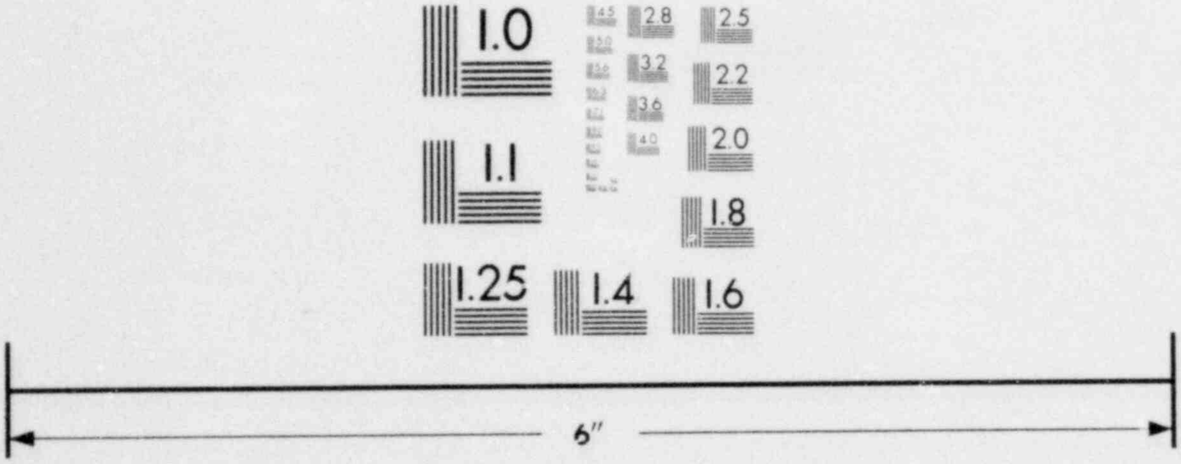
IMAGE EVALUATION
TEST TARGET (MT-3)

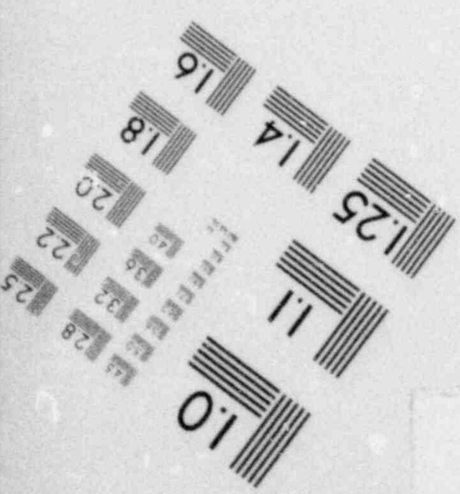
6″

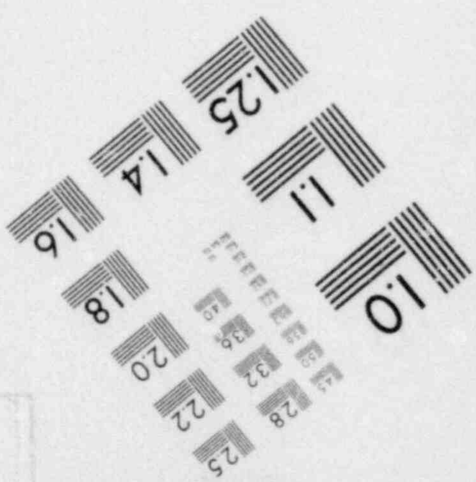
IMAGE EVALUATION TEST TARGET (MT-3)











In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required - loss-of-coolant accident with forced flow, and without forced flow).

2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3b of NUREG-0578).

The analyses of transients and accidents shall include the design basis events specified in Section 15 of the Final Safety Analysis Report The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. For the present, these analyses need not address passive failures or multiple system failures. In the recent analysis of small break loss-of-coolant accidents, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to the water level being below the top of the core for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of the loss of reactor coolant to the extent that the water level in the reactor vessel drops below the top of the core, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. Analyses performed by the nuclear steam supply system vendors will be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparisons to data, including Loss of Fluid Test small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

Discussion and Conclusions

This item requires analyses, procedure guidelines, emergency procedures, and operator training related to small-break loss-of-coolant accidents, inadequate core cooling, and transients and non-loss-of-coolant accidents.

Westinghouse submitted analyses for small-break accidents in Topical Report WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS System"; June 1979. Emergency procedure guidelines were than developed from these analyses by the Westinghouse Plant Owners Group. These guidelines were reviewed and approved by the staff in November, 1979. The staff review of these analyses and guidelines was performed by the Bulletin and Orders Task Force as is documented in their report on Westinghouse reactors, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611, January, 1980 (Appendix IX, Section 2.2). We have reviewed the design features of the Virgil C. Summer Nuclear Station, Unit 1 and conclude that the review and approval of the small-break loss-of-coolant accident analyses and guidelines apply in total to the Virgil C. Summer Nuclear Station, Unit 1.

The emergency operating procedures for reactor trip, safety injection, loss of coolant accident (including small breaks), steam generator tube rupture, loss of feedwater, inadequate core cooling and natural circulation have been reviewed and rewritten in conformance with the guidelines prepared by the Westinghouse Owners Group. The procedure for anticipated transients without scram was also reviewed and is discussed in Section 15.3.5 of this Safety Evaluation Report. The Battelle Pacific Northwest Laboratories assisted the staff in these reviews and the walk-through discussed for item I.C.8 in Section 22.2 of this Safety Evaluation Report.

The procedures have been revised based on our comments and include early verification of emergency core cooling system equipment operation and information sufficient to evaluate for degradation. A number of adjustments were made to clarify the intended operator action and to remove the vagueness of some action. The applicant will use the comments and corrections as a guide to upgrade their remaining emergency operating procedures. Based on our review of the applicant's emergency operating procedures, we find they are consistent with the guidelines for Westinghouse plants. Acceptability of the emergency operating procedures for full power operation of the Virgil C. Summer Nuclear Station, Unit 1 is discussed in item I.C.8 in Section 22.2 of this Safety Evaluation Report.

I.C.2 Shift Relief and Turnover Procedures

Requirement

Revise plant procedures for shift relief and turnover to require signed checklists and logs to assure that the operating staff (including auxiliary operators and maintenance personnel) possess adequate knowledge of critical plant parameter status, system status, availability, and alignment.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.2.1c, and letters of September 27 and November 9, 1979.)

Position

The licensee shall review and revise as necessary the facility procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

    a. Assurance that critical facility parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

    b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console. What to check, and criteria for acceptable status, shall be included on the checklist.

    c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement. (This shall be recorded as a separate entry on the checklist.)

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by itself could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check, and criteria for acceptable status, shall be included on the checklist); and,

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedures (for example, periodic independent verification of system alignments).

Discussion and Conclusions

Shift relief and turnover requirements are described in Amendment 18 to the Final Safety Analysis Report. Checklists are provided for the oncoming and offgoing control room foreman and the oncoming shift supervisor to complete and sign. These checklists provide assurance that actual facility parameters are within allowable limits and that required systems are available and are in proper alignment for the prevention and mitigation of operational transients. Systems and components that are in a degraded mode of operation permitted by the Technical Specifications shall be listed, and time in degraded mode are compared with Technical Specifications action statements. Auxiliary operator checklists include any equipment under maintenance or test that could degrade a system or initiate an operational transient and shall include criteria for acceptable status. The operations supervisor will make unannounced audits of shift relief to evaluate the effectiveness of shift relief and turnover.

We find that the applicant is in the process of completing actions to meet the requirements of TMI Action Plan item I.C.2. We will review these actions during the forthcoming site visit of the IE/NRR management review group and will report further on this issue in a supplement to the Safety Evaluation Report. IE will assure that the required procedures are in effect prior to issuance of an operating license.

I.C.3 Shift Supervisor Responsibilities

Requirement

Issue a corporate management directive that clearly establishes the command duties of the shift supervisor and emphasizes the primary management responsibility for safe operation of the plant. Revise plant procedures to clearly define the duties, responsibilities and authority of the shift supervisor and the control room operators.

These requirements shall be met before fuel loading. (See NUREG-0578, Section 2.2.1a, Items 1, 2, and 3, and letters of September 27 and November 9, 1979.)

Discussion and Conclusion

For our evaluation of this matter, refer to discussion of Item I.A.1.2 in this section of this Safety Evaluation Report.

I.C.4 Control Room Access

Requirement

Revise plant procedures to limit access to the control room to those individuals responsible for the direct operation of the plant, technical advisors, specified NRC personnel, and to establish a clear line of authority, responsibility, and succession in the control room.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.2.2a, and letters of September 27 and November 9, 1979.)

Position

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access; and

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The licensee shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

Discussion and Conclusions

In Amendment 18 to the Final Safety Analysis Report, the applicant states that administrative procedures have been issued which establish:

1. The authority and responsibility of the person in charge of the control room to limit access, and

2. A clear line of authority and responsibility in the control room in the event of an emergency.

We find that the applicant has taken actions to meet the requirements of TMI Action Plan item I.C.4. We will review these actions during the forthcoming site visit of the IE/NRR management review group will report further on this matter in a supplement to this Safety Evaluation Report. IE will assure that the required procedures are in effect prior to issuance of an operating license.

I.C.5 Procedures for Feedback of Operating Experience to Plant Staff

Requirement

Review and revise, as necessary, procedures to assure that operating experiences are fed back to operators and other personnel.

This requirement shall be met before fuel loading.

Position

Each licensee shall review its administrative procedures to assure that operating experience originating from both within and outside the organization is continually provided to operators and other operations personnel and is incorporated into training programs. Operating reactors will complete this action by September

1980. Operating license applicants will complete by this action by September 1980 or prior to fuel loading.

Discussion and Conclusion

By letter dated July 18, 1980, the applicant provided a status report for NUREG-0660/0694 items. The applicant stated that procedures were being developed to cover feedback of operating experiences. On October 31, 1980, we issued NUREG-0737, "Clarification of TMI Action Plan Requirements", which included guidance and clarification for this item. During their site visit, the IE/NRR management review group will review the status of these procedures and will report on this item in a supplement to this Safety Evaluation Report. IE will assure that the required procedures are in effect prior to issuance of an operating license.

I.C.7 Nuclear Steam Supply System Vendor Review of Procedures

Requirement

Obtain nuclear steam supply system vendor review of low-power testing procedures to further verify their adequacy.

This requirement must be met before fuel loading.

Discussion and Conclusions

The applicant has advised us in a letter from T. Nichols to H. Denton, dated December 2, 1980 that Westinghouse comments on most of the emergency operating, power ascension, and low power physics test procedures have been received and resolved. The applicant also committed in the December 2, 1980 letter to complete this review and address any Westinghouse comments on the remainder of the procedures before power operation. We find this acceptable.

I.D.1 Control Room Design

Requirement

Perform a preliminary assessment of the control room to identify significant human factors deficiencies and instrumentation problems and establish a schedule approved by the NRC for correcting deficiencies.

This requirement shall be met before fuel loading.

Position

As part of the NRC staff actions following the TMI-2 accident, we require that all licensees and applicants for operating licenses conduct a detailed control room design review. We expect these detailed control room design reviews to be initiated within the next several months and completed by the end of 1982. In addition, as an interim measure, we require those applicants for operating licenses who are unable to complete this detailed control room design reviews prior to fuel loadings, to make a preliminary design assessment.

Discussion and Conclusions

As a result of these requirements, South Carolina Electric & Gas Company submitted its preliminary control room design assessment to the NRC staff for its review and evaluation.

The control room design review/audit was performed by five persons from the NRC staff who were assisted by two human factors consultants, from Biotechnology, Inc., and the National Bureau of Standards. The NRC review/audit was conducted onsite during the period August 25-29, 1980.

The control room design review included an evaluation of control room layout, the adequacy of the information provided, the arrangement and identification of important controls and instrumentation displays, the usefulness of the audio and visual alarm systems, the information recording and recall capability, lighting, and other considerations of human factors that have an impact of operator effectiveness. This review was performed by means of detailed inspection of the control panels, interviews with operators, and observation and videotaping of operators as they walked through selected emergency procedures.

Many of the human factors design deficiencies noted during the review/audit were previously identified in the South Carolina Electric & Gas Company's preliminary assessment report.

Although our review identified a number of human factors design deficiencies, we found in general that the control room was designed to promote effective operator actions. However, we believe that a number of human factors design improvements are needed in the control room prior to issuance of an operating license. Certain improvements will enhance the operators detection and response capability and will lessen the possibility of operator error during stressful operating conditions to permit safe operation of the facility.

A list of the more significant human factors related design deficiencies observed in the control room during the review/audit is contained in the NRC's "Human Factors Engineering Control Room Design Review" draft report, dated October 9, 1980.

We discussed these deficiencies with the applicant in a meeting held on October 22, 1980. The applicant has developed a program plan which is intended to resolve the deficiencies identified by the staff and by the applicant's consultant, the Essex Corporation.

This program plan which is identified in a letter from T. C. Nichols, Jr. to H. R. Denton, dated November 12, 1980 consists of two phases. Phase 1 is concerned with the identification of solutions to the deficiencies identified by the staff and the development of an implementation schedule for modifications. Any deficiencies for which solutions are not presented will be documented and justified. Phase 1 is scheduled for completion by mid-January 1981.

Based on the adequacy and acceptability of the plan received, we will report our findings in a supplement to this Safety Evaluation Report.

Phase 2 is a longer term program and will include the remaining portions of the one year full control room human engineering evaluation, the review of the design concept for the technical support center, and the review of a new control room computer console.

The applicant has committed to submit phase 2 findings at the end of the detailed control room design review as defined in NUREG-0660 (Item I.D.1).

I.G.1 Training During Low-Power Testing

Requirement

Define and commit to a special low-power testing program approved by NRC to be conducted at power levels no greater than five percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training.

This requirement shall be met before fuel loading.

Position

The TMI Task Action Plan states that new operating licensees will conduct a set of low power tests to increase the capability of shift crews and assure training in plant evolutions and off-normal events. Near-term operating license facilities will be required to develop and implement intensified exercises during the low power testing program. This may involve the repetition of startup tests on different shifts for training purposes.

Prior to issuance of a low power license, each applicant must commit to conduct a low-power test program similar to that conducted at Sequoyah Unit 1 and North Anna Unit 2.

The low-power test program conducted at Sequoyah Unit 1 consisted of nine tests, eight of which involve natural circulation in the reactor coolant system at low power conditions, but at normal, or nearly normal operating pressures and temperatures.

The specific tests proposed are:

1. Natural circulation test;

2. Natural circulation with simulated loss of offsite alternating current power;

Discussion and Conclusions

NUREG-0694 requires license applicants to "define and commit to a special low power testing program approved by NRC to be conducted at power levels no greater than five percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training."

In a letter to the applicant dated November 14, 1980, the staff outlined an acceptable program for compliance with this requirement. This letter stated that the program should provide the following:

> "Each licensed reactor operator (RO or SRO who performs RO or SRO duties, respectively) should experience the initiation, maintenance and recovery from natural circulation mode, using nuclear heat to simulate decay heat. Operators should be able to recognize when natural circulation has stabilized, and should be able to control saturation margin, RCS pressure, and heat removal rate without exceeding specified operating limits.
>
> These tests should demonstrate the following plant characteristics: length of time required to stabilize natural circulation, core flow distribution, ability to establish and maintain natural circulation with or without onsite and offsite power, and the ability to uniformly borate and cool down to hot shutdown conditions using natural circulation. The latter demonstration may be performed using decay heat following power ascension and vendor acceptance tests, and need only be performed at those plants for which the test has not been demonstrated at a comparable prototype plant."

In letters dated October 31, 1980 and December 22, 1980, the applicant committed to perform a series of tests to satisfy the above requirements. These tests are described briefly:

Natural Circulation Demonstration

The reactor will be operating at approximately three percent power, all reactor coolant pumps tripped, and natural circulation established. After natural circulation has stabilized, the pressurizer heaters will be turned off and the pressurizer cooldown rate determined and subcooling meter accuracy verified. Auxiliary spray, charging and letdown flow adjustments, and steam dump flow will be used to familiarize the operators with their effects on reactor pressure, temperature, level, and subcooling margin during natural circulation decay heat removal conditions. This test will be performed by each shift.

Natural Circulation with Simulated Loss of Offsite Power

The reactor will be operating at approximately one percent power. Loss of offsite power will be simulated by de-energizing appropriate equipment as necessary within the constraints of (1) public health and safety and (2) minimizing the risk to plant equipment. The steam generators will be fed using the emergency feedwater system. This test and those that follow will be performed once.

Natural Circulation with Simulated Loss of Offsite and Onsite Alternating Current Power

This test was performed at Sequoyah with the reactor critical at one percent power to simulate decay heat. Since that time, the staff has approved the option of performing this test with the reactor shutdown using operating reactor coolant pumps to simulate decay heat input. Loss of all alternating current power is simulated by de-energizing as many buses and as much equipment as possible within the constraints of safety and possible equipment damage. The turbine-driven emergency feedwater pump is used to feed the steam generators during the test.

Natural Circulation from Stagnant Conditions

The reactor will be critical at hot zero power, the reactor coolant pumps tripped and the steam generators isolated. After coolant flow has ceased, reactor power is increased to approximately three percent and natural circulation established. This test has been performed at the Sequoyah facility. For subsequent applications, including the Virgil C. Summer Nuclear Station, Unit 1 application, this test was waived and operator training is provided on a simulator, which has been programmed, using data from the Sequoyah test, to accurately duplicate the responses demonstrated at Sequoyah.

Boron Mixing and Cooldown Under Natural Circulation

Natural circulation will be established at approximately three percent reactor power. A normal boration will be initiated and three percent power maintained by rod withdrawal. Pressurizer auxiliary spray will be used to facilitate mixing. After approximately 100 parts per million boration and verification of mixing by analysis, the reactor will be cooled down to 450 degrees Fahrenheit at constant power. For plants performing this test subsequent to Sequoyah, the staff has approved postponement of the test until sufficient decay heat is available to permit testing with the reactor shut down, and at a time that does not cause excessive economic impact, or the test may be deleted if analysis verifies that results of the test at a similar plant are applicable. For this facility, the applicant will provide an analysis to attempt to justify deletion of the test; if deletion is not adequately justified, it will be performed during the first fuel cycle, using decay heat.

It is the conclusion of the staff that the special low power test program proposed by the applicant meets the requirements of Item I.G.1 of the TMI Task Action Plan subject to the staff review of the detailed test procedures and a safety analysis to be submitted to the staff at least four weeks prior to performing the tests.

The requirements of Item I.G.1 shall be met prior to exceeding five percent power. Upon completion of these tests the applicant shall provide NRC (IE and NRR) with a test report. This report shall identify any acceptable criteria listed in the detailed test procedures, which were not met.

II.B.4 Training for Mitigating Core Damage

Requirement

Develop a training program to instruct all operating personnel in the use of installed systems, including systems that are not engineered safety features, and instrumentation to monitor and control accidents in which the core may be severely damaged.

This requirement shall be met before fuel loading.

Position

The staff requires that the applicant develop a program to assure that all operating personnel are trained in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. The training program shall include the following topics.

A. Incore Instrumentation

1. Use of fixed or movable incore detectors to determine extent of core damage and geometry changes.

2. Use of thermocouples in determining peak temperatures; methods for extended range readings; methods for direct readings at terminal junctions.

B. Excore Nuclear Instrumentation

Use of excore nuclear instrumentation for determination of void formation; void location basis for excore nuclear instrumentation response as a function of core temperatures and density changes.

C. Vital Instrumentation

1. Instrumentation response in an accident environment; failure sequence (time to failure, method of fa lure); indication of reliability (actual vs. indicated level).

2. Alternative methods for measuring flows, pressures, levels, and temperatures.

   a. Determination of pressurizer level if all level transmitters fail.

   b. Determination of letdown flow with a clogged filter (low flow).

   c. Determination of other reactor coolant system parameters if the primary method of measurement has failed.

D. Primary Chemistry

1. Expected chemistry results with severe core damage; consequences of transferring small quantities of liquid outside containment; importance of using leak-tight systems.

2. Expected isotopic breakdown for core damage; for clad damage.

3. Corrosion effects of extended immersion in primary water; time to failure.

E. Radiation Monitoring

1. Response of process and area monitors to severe damages; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector output expected accuracy of detectors at different locations; use of detectors to determine extent of core damage.

2. Methods of determining dose rate inside containment from measurements taken outside containment.

F. Gas Generation

1. Methods of hydrogen generation during an accident; other sources of gas (xenon, krypton); techniques for venting or disposal of non-condensibles.

2. Hydrogen flammability and explosive limit; sources of oxygen in containment or reactor coolant system.

Discussion and Conclusions

South Carolina Electric & Gas Company will conduct a training program that meets all the requirements as stated above. This training program will also become a part of the requalification program.

Attendance is required for all personnel in the Operations Department at the Virgil C. Summer Nuclear Station, Unit 1. This includes licensed operators, licensed senior operators and non-licensed operators. Personnel identified by the Emergency Plan as qualified to become emergency directors are required to attend. All shift technical advisors and nuclear training coordinators are also required to attend. An examination will be given to all personnel attending the program, and any person scoring less than 80 percent will be required to review the material and be reexamined until a grade of 80 percent is achieved.

Based on the foregoing, we have concluded that the South Carolina Electric & Gas Company training and requalification program meets our requirements for training personnel in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged.

Discussion and Conclusions

II.D.1 Relief and Safety Valve Test Requirements

Requirement

Describe a test program and schedule for testing to qualify reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.2, and letters of September 27 and November 9, 1979.)

Position

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

Clarification

1. Expected operating conditions can be determined through the use of analysis of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports."
2. This testing is intended to demonstrate valve operability under various flow conditions, that is, the ability of the valve to open and shut under the various flow conditions should be demonstrated.
3. Not all valves on all plants are required to be tested. The valve testing may be conducted on a prototypical basis.
4. The effect of piping on valve operability should be included in the test conditions. Not every piping configuration is required to be tested, but the configurations that are tested should produce the appropriate feedback effects as seen by the relief or safety valve.
5. Test data should include data that would permit an evaluation of discharge piping and supports if those components are not tested directly.

Discussion and Conclusions

The applicant has stated that it will participate in the Electric Power Research Institute/Nuclear Safety Analysis Center program to conduct performance testing of pressurized water reactor relief and safety valves and associated piping and supports. The applicant has referenced the proposed program, "Program Plan for the Performance Verification of PWR Safety/Relief Valves and System," for the performance testing of these valves.

The Electric Power Research Institute program plan for the performance testing of pressurized water reactor safety and relief valves was initially submitted to the NRC in December, 1979. Revision 1 to this plan was submitted in July, 1980 for NRC staff review and comments. The NRC staff has completed its review of this program plan and finds it acceptable with comments which are being forwarded to the pressurized water reactors owners group for resolution and incorporation in the test program.

The applicant has committed to participate in the Electric Power Research Institute Program and therefore, meets the requirements of item II.D.1 to the extent practicable at this time. We believe that this commitment provides adequate assurance that the requirement for performance testing of the safety and relief valves will be satisfied. Our basis for accepting this commitment is that preliminary discussions with the Electric Power Research Institute indicate that our comments to the test program can be resolved and the requirements of NUREG-0578 can therefore be met. The applicant has committed to provide a qualification test report by December 1, 1981. We require that the applicant document those submittal and additional test requirements specified for this item in NUREG-0737.

The applicant's response to the performance testing requirement for pressurized water reactor relief and safety valves is acceptable. We will report the final results of this review in a supplement to this Safety Evaluation Report on completion of testing.

II.D.3 Relief and Safety Valve Position Indication

Requirement

Install positive indication in the control room of relief and safety valve position derived from a reliable valve position detection device or a reliable indication of flow in the valve discharge pipe.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.3a, and letters of September 27 and November 9, 1979.)

Position

Reactor coolant system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

Clarification

1. The basic requirement is to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken.

2. The valve position should be indicated in the control room. An alarm should be provided in conjunction with this indication.

3. The valve position indication may be safety grade. If the position indication is not safety grade, a reliable single-channel direct indication powered from a vital instrument bus may be provided if backup methods of determining valve position are available and are discussed in the emergency procedures as an aid to operator diagnosis and action.

4. The valve position indication should be seismically qualified consistent with the component or system to which it is attached. If the seismic qualification requirements cannot be met feasibly by January 1, 1980, a justification should be provided for less than seismic qualification and a schedule should be submitted for upgrade to the required seismic qualification.

5. The position indication should be qualified for its appropriate environment (any transient or accident which would cause the relief or safety valve to lift). If the environmental qualification program for this position indication will not be completed by January 1, 1980, a proposed schedule for completion of the environmental qualification program should be provided.

Discussion and Conclusions

The power-operated relief valves of the pressurizer are provided with limit switches to indicate the valves' open/closed position in the control room. The limit switches are NAMCO EA-180 limit switches which are seismically and environmentally qualified. An acoustical type sensor is provided downstream of each safety valve to detect flow through the safety valves. The sensor activates an alarm in the control room when flow is detected. Control room indication is provided to enable the plant operator to determine which valve is open. We conclude that the design satisfies the above position and is acceptable.

II.E.1.2 Auxiliary Feedwater Initiation and Indication

Requirement

Install a control-grade system for automatic initiation of the auxiliary feedwater system that meets the single-failure criterion, is testable, and is powered from the emergency buses, and control-grade indication of auxiliary feedwater flow to each steam generator that is powered from emergency buses.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.7a and b, and letters of September 27 and November 9, 1979.)

Position

To improve the reliability of the auxiliary feedwater system the staff is requiring licensees to upgrade the system where necessary to assure timely automatic initiation when required. The system upgrade was to proceed in two phases. In the short term, as a minimum, control-grade signals and circuits are to be used to automatically initiate the auxiliary feedwater system. This control-grade system is required to meet the following requirements: from NUREG-0578, Section 2.1.7.a

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

3. Testability of the initiating signals and circuits shall be a feature of the design.

4. The initiating signals and circuits shall be powered from the emergency buses.

5. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

6. The alternating current powered motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the auxiliary feedwater system from the control room.

In the long term, these signals and circuits are to be upgraded in accordance with safety-grade requirements. Specifically, in addition to the above requirements, the automatic initiation signals and circuits must have independent channels, use qualified components, have system bypassed/inoperable status features, and conform to control system interaction criteria, as stipulated in IEEE Standard 279.

In addition to the above automatic initiation requirements, the capability to ascertain the actual performance of the auxiliary feedwater system from the control room must be provided. For Westinghouse plants, this is accomplished by a combination of auxiliary feedwater flow indication and steam generator wide range level indication in the control room.

In the short term, the auxiliary feedwater system flow and steam generator level indication is to meet control-grade requirements. Specifically, these flow and level instrument channels must be powered from the vital instrument buses, testability of these channels must be a feature of the design, and the instrumentation indicating the performance of the auxiliary feedwater system (flow and wide range level indication for each steam generator) must satisfy the single-failure criterion. For the long term, to adequately determine the performance of the auxiliary feedwater system sufficient safety-grade instrumentation (specifically steam generator wide range level) must be provided.

Discussion and Conclusions

See discussion of item II.E.1.2 in Section 22.3 of this Safety Evaluation Report.

II.E.4.1 Containment-Dedicated Penetrations

Requirement

Provide a design of the containment isolation system for external recombiners or purge systems for post-accident combustible gas control, if used, that is dedicated to that service only and meets the single-failure criterion.

Review and revise, if necessary, the procedures for use of the combustible gas control system following an accident resulting in a degraded core and release of radioactivity into the containment.

This requirement shall be met before fuel loading. (See NUREG-0578, Sections 2.1.5a and 2.1.5c, and letters of September 27 and November 9, 1979.)

Position

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that satisfy the redundancy and single failure requirements of Criterion 54 and 56 of the General Design Criteria and that are sized to satisfy the flow requirements of the recombiner or purge system.

Clarification

1. This requirement is only applicable to those plants whose licensing basis includes requirements for external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere.

2. An acceptable alternative to the dedicated penetration is a combined design that is single-failure proof for containment isolation purposes and single-failure proof for operation of the recombiner or purge system.

3. The dedicated penetration or the combined single-failure proof alternative should be sized such that the flow requirements for the use of the recombiner or purge system are satisfied.

4. Components necessitiated by this requirement should be safety grade.

5. A description of required design changes and a schedule for accomplishing these changes should be provided by January 1, 1980. Design changes should be completed by January 1, 1981.

Discussion and Conclusion

Because internal recombiners are used at the facility, the requirement for dedicated penetrations for external recombiners is not applicable.

II.F.1 Additional Accident Monitoring Instrumentation

Requirement

Provide procedures for estimating noble gas, radioiodine, and particulate release rates if the existing effluent instrumentation goes off the scale.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.8b, and letters of September 27 and November 9, 1979.)

Position for Noble Gas Effluent Monitor

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other regulatory guides, which will be promulgated in the near term.

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

1. Noble gas effluent monitors with an upper range capacity of 10^5 microcuries per cubic centimeter (Xe-133) are considered to be practical and should be installed in all operating plants.

2. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable) concentrations to a maximum of 10^5 microcuries per cubic centimeter (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of 10.

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error.

A human factors analysis should be performed taking into consideration:

(1) the use of this information by an operator during both normal and abnormal plant conditions;
(2) integration into emergency procedures;
(3) integration into operator training; and
(4) other alarms during an emergency and the need for prioritization of alarms.

Clarification for Noble Gas Effluent Monitor

NUREG-0578, Section 2.1.8b provided the basic requirements for this item. Letters dated September 27 and November 9, 1979, provided clarification and NUREG-0660, Item II.F.1 provided the action plan for additional accident monitoring instrumentation by specifying noble gas effluent radiological monitor requirements. Additional clarification was provided by letters dated September 5, and October 31, 1980.

The following guidelines were established in the clarification letters:

(1) Applicants shall provide continuous monitoring of high-level, post-accident releases of radioactive noble gases from the plant. Gaseous effluent monitors shall meet the requirements specified in Table 22-1. Typical plant effluent pathways to be monitored are also given in Table 22-1.

TABLE 22-1

HIGH-RANGE NOBLE GAS EFFLUENT MONITORS

REQUIREMENTS - Capability to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. All potential accident release paths shall be monitored.

PURPOSE - To provide the plant operator and emergency planning agencies with information on plant releases of noble gases during and following an accident.

DESIGN BASIS MAXIMUM RANGE

Design range values may be expressed in Xe-133 equivalent values for monitors employing gamma radiation detectors or in microcuries per cubic centimeter of air at standard temperature and pressure (STP) for monitors employing beta radiation detector (Note: 1 Rad/per hour at 1 foot = 6.7 curies Xe-133 equivalent for point source.) Calibrations with a higher energy source are acceptable. The decay of radionuclide noble gases after an accident (i.e., the distribution of noble gases changes) should be taken into account.

10^5 microcuries per cubic centimeter	- Undiluted containment exhaust gases (e.g., reactor building purge of a pressurized water reactor, drywell purge through the standby gas treatment system of a boiling water reactor).
	- Undiluted condenser air removal system exhaust of a pressurized water reactor
10^4 microcuries per cubic centimeter	- Diluted containment exhaust gases (e.g., > 10:1 dilution, as with auxiliary building exhaust air).
	- Reactor building (secondary containment) exhaust air of a boiling water reactor.
	- Secondary containment exhaust air of a pressurized water reactor.
10^3 microcuries per cubic centimeter	- Buildings with systems containing primary coolant or offgases (e.g., auxiliary buildings of a pressurized water reactor
	- Steam safety valve discharge and atmospheric steam dump valve discharge of a pressurized water reactor
10^2 microcuries per cubic centimeter	- Other release points (e.g., radwaste buildings, fuel handling/storage buildings).

TABLE 22-1 (Cont'd)

REDUNDANCY	-	Not required; monitoring the final release point of several discharge inputs is acceptable.
SPECIFICATIONS	-	None; sampling design criteria per ANSI N13.1.
POWER SUPPLY	-	Vital instrument bus or dependable backup power supply to normal alternating current power.
CALIBRATION	-	Calibrate monitors using gamma detectors to Xe-133 equivalent (1 Rad per hour at 1 foot = 6.7 curies Xe-133 equivalent for point source). Calibrate monitors using beta detectors to Sr-90 or similar long lived beta isotope of at least 0.2 million electron volts.
DISPLAY	-	Continuous and recording as equivalent Xe-133 concentrations or microcuries per cubic centimeter of actual noble gases.
QUALIFICATION	-	The instruments shall provide sufficiently accurate responses to perform the intended function in the environment to which they will be exposed during accidents.
DESIGN CONSIDERATIONS	-	Offline monitoring is acceptable for all ranges of noble gas concentrations. Inline (induct) sensors are acceptable for 10^2 microcuries per cubic centimeter to 10^5 microcuries per cubic centimeter noble gases. For less than 10^2 microcuries per cubic centimeter, offline monitoring is recommended. Upsteam filtration (prefiltering to remove radioactive iodines ad particulates) is not required; however, design should consider all alternatives with respect to capability to monitor effluents following an accident. For external mounted monitors (e.g., main steam line of a pressurized water reactor), the thickness of the pipe should be taken into consideration in accounting for low-energy gamma radiation.

(2) The monitors shall be capable of functioning both during and following an accident. System designs shall accommodate a design-basis release and then be capable of following decreasing concentrations of noble gases.

(3) Offline monitors are not required for the pressurized water reactor secondary side main steam safety valve and dump valve discharge lines. For this application, externally mounted monitors viewing the main steam line upstream of the valves are acceptable with procedures to correct for the low energy gammas the external monitors would not detect. Isotopic identification is not required.

(4) Instrumentation ranges shall overlap to cover the entire range of effluents from normal through accident conditions.

The design description shall include the following information:

a. System/method description, including:

i. Instrumentation to be used including range or sensitivity, energy dependence or response, calibration frequency and technique, and vendor's model number, if applicable;

ii. Monitoring locations (or points of sampling), including description of methods used to assure representative measurements and background radiation correction;

iii. Location of instrument readout(s) and method of recording, including description of the method or procedure for transmitting or disseminating the information or data;

iv. Assurance of the capability to obtain readings at least every 15 minutes during and following an accident, and;

v. The source of power to be used.

b. Description of procedures or calculational methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown.

(5) This requirement applies to all operating reactors and applicants for operating license. Implementation must be completed by January 1, 1982.

(6) License applicants should have available for review the final design description of the as-built system, including piping and instrument diagrams, together with either (1) a description of procedures for system operation and calibration, or (2) copies of procedures for system operation and calibration. Changes to Technical Specifications will be required. License applicants are requested to submit the above details in accordance with the proposed review schedule, but in

no case less than four months prior to the issuance of an operating license. A post-implementation review will be performed.

Discussion and Conclusions for Noble Gas Effluent Monitor

In Amendments 18 through 22 and letters dated August 28, 1980 and November 6, 1980, South Carolina Electric & Gas Company provided information to satisfy our requirements for noble gas effluent monitoring. Monitors for radioactive effluents currently installed at the facility are designed to detect and measure releases associated with normal reactor operations and anticipated operational occurrences. Such monitors are required to operate in radioactivity concentrations approaching the minimum concentration detectable with "state-of-the-art" sample collection and detection methods. These monitors comply with the criteria of Regulatory Guide 1.21 with respect to releases from normal operations and anticipated operational occurrences.

Radioactive gaseous effluent monitors designed to operate under conditions of normal operation and anticipated operational occurrences do not have sufficient dynamic range to function under release conditions associated with certain types of accidents. Criterion 64 of the General Design Criteria requires that effluent discharge paths be monitored for radioctivity that may be released from postulated accidents.

The potential gaseous effluent release points at the facility consist of the main plant vent exhaust, the reactor building purge exhaust and the atmospheric steam relief discharge pipes. Before fuel loading, the applicant has committed to install mid/high-level noble gas monitors for monitoring noble gas releases via the main plant vent exhaust, reactor building purge vent exhaust and the steam dump/safety valves. The monitors will be designed to meet the requirements and satisfy the characteristics given in Table 22-1.

Based on our review of the applicant's description of the noble gas effluent monitoring system, we conclude that the system can meet the requirements of item II.F.1 provided off-line monitors (shielded appropriately to minimize interference by background radiation) are utilized to monitor effluent concentrations in the lower ranges, and procedures are available for converting instrument readings to concentrations and release rates for reactor building purge releases.

A post-implementation review will be performed of the installed system, detailed drawings and procedures for systems operation and our evaluation will be provided in a supplement to this Safety Evaluation Report.

Position for Sampling and Analysis of Plant Effluents

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other regulatory guides, which will be promulgated in the near-term.

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring or radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factors analysis should be performed taking into consideration:

(1) the use of this information by an operator during both normal and abnormal plant conditions,
(2) integration into emergency procedures,
(3) integration into operator training, and
(4) other alarms during emergency and need for prioritization of alarms.

Clarification for Sampling and Analysis of Plant Effluents

NUREG-0578, Section 2.1.8b provided the basic requirements for this item. Letters dated September 27 and November 9, 1979, provided clarification; however, NUREG-0660 inadvertently omitted the requirement for additional accident-monitoring by sampling and analysis of plant effluents. Additional clarification was provided by letters dated September 5 and October 31, 1980.

The following guidelines were established in the clarification letters.

(1) Applicants shall provide continuous sampling of plant gaseous effluents for post-accident releases of radioactive iodines and particulates to meet the requirements of Table 22-2. Applicants shall also provide onsite laboratory capabilities to analyze or measure these samples. This requirements should not be construed to prohibit design and development of radioiodine and particular monitors to provide online sampling and analysis for the accident condition. If gross gamma radiation measurement techniques are used, then provisions shall be made to minimize noble gas interference.

(2) The shielding design basis is given in Table 22-2. The sampling system design shall be such that plant personnel could remove samples, replace sampling media and transport the samples to the onsite analysis facility with radiation exposures that are not in excess of the criteria of Criterion 19 of the General Design Criteria of five-rem whole-body exposure and 75 rem to the extremities during the duration of the accident.

(3) The design of the systems for the sampling of particulates and iodines should provide for sample nozzle entry velocities which are approximately isokinetic (same velocity) with expected in-duct or in-stack air velocities. For accident conditions, sampling may be complicated by a reduction in stack or vent effluent velocities to below design levels, making it necessary to substantially reduce sampler intake flow rates to achieve the isokinetic condition. Reductions in air flow may well be beyond the capability of available sampler flow controllers to maintain isokinetic conditions; therefore, the staff will accept flow control devices which have the capability of maintaining isokinetic

TABLE 22-2

SAMPLING AND ANALYSIS OF MEASUREMENT OF HIGH-RANGE RADIOIODINE AND PARTICULATE EFFLUENTS IN GASEOUS EFFLUENT STREAMS

EQUIPMENT - Capability to collect and analyze or measure representatives samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. The capability to sample and analyze for radioiodine and particulate effluents is not required for secondary main steam safety valve and dump valve discharge lines of a pressurized water reactors.

PURPOSE - To determine quantitative release of radioiodines and particulates for dose calculation and assessment.

DESIGN BASIS SHIELDING ENVELOPE - 10^2 microcuries per cubic centimeter of gaseous radioiodine and particulates, deposited on sampling media; 30 minutes sampling time, average gamma energy of 0.5 million electron volts.

SAMPLING MEDIA

- Iodine > 90 percent effective adsortion for all forms of gaseous iodine.
- Particulates > 90 percent effective retention of 0.3 micron diameter particles.

SAMPLING CONSIDERATIONS

- Representative sampling per ANSI N13.1-1969.
- Entrained moisture in effluent stream should not degrade adsorber.
- Continuous collection required whenever exhaust flow occurs.
- Provisions for limiting occupational dose to personnel incorporated in sampling systems, in sample handling and transport, and in analysis of samples.

ANALYSIS

- Design of analytical facilities and preparation of analytical procedures shall consider the design basis sample.
- Highly radioactive samples may not be compatible with generally accepted analytical procedures; in such cases, measurement of emissive gamma radiations and the use of shielding and distance factors should be considered in design.

conditions with variations in stack or duct design flow velocity of ± 20 percent. Further departure from the isokinetic conditions need not be considered in the design. Corrections for anisokinetic sampling conditions, as provided in Appendix C to ANSI 13.1-1969 may be considered on an ad hoc basis.

(4) Effluent streams which may contain air with entrained water, e.g., air ejector discharge, shall have provisions to assure that the adsorber is not degraded while providing a representative sample, e.g., heaters.

(5) This requirement applies to all operating reactors and applicants for operating license. Implementation must be completed by January 1, 1982.

(6) License applicants should have available for review, the final design description of the as-built system, including piping and instrument diagrams together with either (1) a description of procedures for system operation and calibration, or (2) copies of procedures for system operation and calibration. Changes to Technical Specifications will be required. License applicants should submit the above details in accordance with the proposed review schedule, but in no case less than four months prior to the issuance of an operating license. A post-implementation review will be performed.

Discussion and Conclusions for Sampling and Analysis of Plant Effluents

In a letted dated November 6, 1980, the applicant has provided information to satisfy the above requirements. Monitors for radioactive effluents currently installed at the facility are designed to detect and measure releases associated with normal reactor operations and anticipated operational occurrences. Such monitors are required to operate in radioactivity concentrations approaching the minimum concentration detectable with "state-of-the-art" sample collection and detection methods. These monitors comply with the criteria of Regulatory Guide 1.21 with respect to releases from normal operations and anticipated operational occurrences.

The potential gaseous effluent release points at the Virgil C. Summer Nuclear Station, Unit 1 to be considered for iodines and particulates are the main plant vent and reactor building purge exhausts. The applicant proposes to utilize the existing isokinetic probes under normal and accident conditions. Charcoal adsorbers which are used to collect iodines under normal conditions are replaced by silver zeolite cartridges for collecting iodines under accident conditions. The charcoal adsorbers will be purged of noble gases before they are taken to the in-plant laboratory for the analysis of iodine deposited on the adsorbers. There are provisions for transporting silver zeolite cartridges in shielded containers to the in-plant laboratory for analysis. The applicant will be required to make appropriate corrections for anisokinetic sampling conditions as provided in Appendix C to ANSI 13.1-1969 on an ad hoc basis.

We find that the radioiodine and prticulate effluent sampling and analysis systems the applicant proposes to implement can meet our requirements provided (1) appropriate corrections for anisokinetic sampling conditions are applied, and (2) sufficient shielding is provided for the sampling media utilized for sampling main plant vent and reactor building purge exhausts.

A post-implementation review of the installed system, detailed drawings, and procedures for systems operation will be performed and an evaluation will be provided in a supplement to the Safety Evaluation Report.

Position for Containment Pressure Indication

A continuous indication of containment pressure should be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five pounds per square inch, gauge for all containments.

Clarification for Containment Pressure Indication

The containment pressure indication shall meet the design provisions of Regulatory Guide 1.97 including qualification, redundancy and testability.

Discussion and Conclusions for Containment Pressure Indication

The applicant has committed to install two separate containment pressure measuring systems and a recording system to record one channel. The system is capable of measuring containment pressure from 10 to 190 pounds per square inch, absolute and meets the design provisions of IEEE 323-1971. The applicant has verified that the containment pressure measuring system meets the design provisions of Regulatory Guide 1.97. Therefore, we conclude that the applicant's response to date concerning this item is acceptable.

Equipment has been delivered for the containment pressure measuring and recording system and is scheduled to be installed by January 1, 1981.

Position for Containment Water Level Indication

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for pressurized water reactors and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for pressurized water reactors and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For boiling water reactors, a wide range instrument shall be provided and cover the range from the bottom to five feet above the normal water level of the suppression pool

Clarification for Containment Water Level Indication

1. The narrow range sump level instrument shall monitor the normal containment sump level vice the containment emergency sump level.

2. The wide range [illegible]ainment water level instruments shall meet the requirements of the propos[illegible] revision to Regulatory Guide 1.97," Instrumentation for Light-Water Cooled Nuclear Power Plant to Assess Plant Conditions During and Following an Accident."

3. The narrow range containment water level instruments shall meet the requirements of Regulatory Guide 1.89," Qualification of Class IE Equipment of Nuclear Power Plants."

4. The equivalent capacity of the wide range pressurized water reactor level instrument has been changed from 500,000 gallons to 600,000 gallons to assure consistency with the proposed revision to Regulatory Guide 1.97. It should be noted that this measurement capability is based on recent plant designs. For older plants with smaller water capacities, licensees may propose deviations from this requirement based on the available water supply capability at their plant.

5. The containment water level indication shall be installed by January 1, 1981.

Discussion and Conclusions for Containment Water Level Indication

The applicant plans to use the presently installed instruments for narrow range and wide range measurement of containment water level. The narrow range instruments cover the range in the containment sump from elevation 400 feet 6 inches to 413 feet while the wide range instruments cover from the containment floor from elevation 413 feet to 425 feet. The maximum calculated water level in the reactor building is 418 feet 6 inches, corresponding to an amount of 522,500 gallons. The wide range instruments and the narrow range instruments satisfy the provisions of IEEE 323-1971. The applicant has verified that the wide range instruments satisfy the provisions of Regulatory Guide 1.97 while the narrow range instruments meet the provisions of Regulatory Guide 1.89.

The containment water level indications are installed.

Position for Containment Hydrogen Indication

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10 percent hydrogen concentration under both postive and negative ambient pressure.

Clarification for Containment Hydrogen Indication

1. The containment hydrogen indication shall meet the design provisions of Regulatory Guide 1.97 including qualification, redundancy, and testability.

2. The containment hydrogen indication shall be installed by January 1, 1981.

Discussion and Conclusion for Containment Hydrogen Indication

The applicant has committed to install hydrogen indication monitors capable of measuring hydrogen concentrations between 0 and 10 percent. The hydrogen indicators are installed. The conceptual design and target implementation schedule satisfy our requirements for this item. The applicant has to verified that the hydrogen monitoring system meets the design provisions of Regulatory Guide 1.97.

The hydrogen indicators at the plant are installed.

Position for Containment Radiation Monitor

In containment radiation-level monitors with a maximum range of 10^8 rad per hour shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

Clarification for Containment Radiation Monitor

1. Provide two radiation monitor systems in containment which are documented to meet the requirements of Table II.F.1-3 of NUREG-0737.

2. The specification of 10^8 rad per hour in the above position was based on a calculation of postaccident containment radiation levels that include both particulate (beta) and photon (gamma) radiation. A radiation detector that responds to both beta and gamma radiation cannot be qualified to post loss-of-coolant accident containment environments but gamma-sensitive instruments can be so qualified. In order to follow the course of an accident, a containment monitor that measures only gamma radiation is adequate. The requirement was revised in the October 30, 1979 letter to provide for a photon-only measurement with an upper range of 10^7 rad per hour.

3. The monitors shall be located in containment(s) in a manner as to provide a reasonable assessment of area radiation conditions inside containment. The monitors shall be widely separated so as to provide independent measurements and shall "view" a large fraction of the containment volume. Monitors should not be placed in areas which are protected by massive shielding and should be reasonably accessible for replacement, maintenance, or calibration. Placement high in a reactor building dome is not recommended because of potential maintenance difficulties.

4. For boiling water reactor Mark II containments, two such monitoring systems should be inside both the primary containment (drywell) and the secondary containment.

5. The monitors are required to respond to gamma photons with energies as low as 60 thousand electron volts and to provide an essentially flat response for gamma energies between 100 thousand electron volts and three thousand electron volts million electron volts, as specified in Table II.F.1-3 of NUREG-0737. Monitors that use thick shielding to increase the upper range will underestimate post-accident radiation levels in containment by several orders of magnitude because of their insensitivity to low energy gammas and are not acceptable.

Discussion and Conclusion for Containment Radiation Monitor

The South Carolina Electric and Gas Company has installed two "high range containment monitors" in containment. One monitor located in penetration 309 will be shielded from the containment atmosphere by the penetration cover. Thus it will not be sensitive to low energy gamma radiation from noble gases and does not meet the position of NUREG-0578 to be unshielded. We informed the applicant that either a second containment monitor or a monitor in a penetration whose reading can be correlated to actual containment radiation levels would be acceptable. The applicant has committed to provide in a future amendment justification for locating the monitor in a containment penetration and how to correlate a reading from the shielded monitor to the acutal radiation level inside containment. The applicant has also committed to install an additional unshielded monitor to satisfy the requirement of Item II.F.1, if the justification for the shielded monitor is unacceptable to NRC. We will report on the final resolution of this matter in supplement to this Safety Evaluation Report.

II.F.2 Inadequate Core Cooling Instruments

Requirement

Develop procedures to be used by operators to recognize inadequate core cooling with currently installed instrumentation in pressurized water reactors. Install a primary coolant saturation meter. Provide a description of any additional instruments or controls needed to supplement installed equipment to provide unambiguous, easy-to-interpret indication of inadequate core cooling, procedures for use of this equipment, analyses used to develop these procedures, and a schedule for installing this equipment.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.3b, and letters of September 27 and November 9, 1979.)

Positions

Criterion 13 of the Criteria General Design Criteria requires instrumentation to monitor variables "...for accident conditions as appropriate to assure adequate safety." In the past, Criterion 13 of the General Design Criteria was not interpreted to require instrumentation to directly monitor water level in the reactor vessel as an indicator of the adequacy of core cooling. The instrumentation available on some operating reactors that could indicate inadequate core cooling was generally included in the reactor design to perform other functions.

During the TMI-2 accident, a condition of low water level in the reactor vessel and inadequate core cooling existed and was not recognized for a long period of time. This problem was the result of a combination of factors including an insufficient range of existing instrumentation, inadequate emergency procedures, inadequate operator training, unfavorable instrument location (scattered information), and perhaps insufficient instrumentation.

The purpose of this review of the TMI-2 short-term recommendations is to evaluate the implementation of the post-TMI inadequate core cooling indication requirements described in NUREG-0578 as follows:

1. Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure

development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of NUREG-0578).

In addition, each pressurized water reactor shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that it not to be used exclusive of other related plant parameters.

2. Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Clarification of the Position for Existing Instrumentation

1. The analysis and procedures addressed in paragraph one above will be reviewed and should be submitted to the NRC for review.

2. The purpose of the subcooling meter is to provide a continuous indication of margin to saturated conditions. This is an important diagnostic tool for the reactor operators.

3. Redundant safety-grade temperature inputs from each hot leg (or use of multiple core exit thermocouples) are required.

4. Redundant safety-grade system pressure measures should be provided.

5. Continuous display of the primary coolant saturation conditions should be provided.

6. Each pressurized water reactor should have safety-grade calculational devices and display (minimum of two meters) or a highly reliable single-channel environmentally qualified and testable system plus a backup procedure for use of steam tables. If the plant computer is to be used, its availability must be documented.

7. In the long term, the instrumentation qualifications must be required to be upgraded to meet the requirements of Regulatory Guide 1.97 "Instrumentation for Light Water Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1980.

8. In all cases appropriate steps (electrical, isolation, etc.) must be taken to assure that the addition of the subcooling meter does not adversely impact the reactor protection or engineered safety features systems.

9. Table 22-3 provides a definition of information required on the subcooling meter. (Note: Table 22-3, has been completed by applicant, and provides the required information.)

TABLE 22-3

INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information displayed (T-Tsat, Tsat, Pressure, etc.)	P-Psat subcooled T-Tsat superheat
Display Type (analog, digital, cathode ray tube)	Analog and digital
Continuous or on demand	Analog is continuous Digital is on demand
Single or redundant display	Redundant
Location of display	Main control board
Alarms (include setpoints)	See "A" below
Overall uncertainty (degrees Fahrenheit, pounds per square inch)	Digital four degrees Fahrenheit for thermocouples; three degrees Fahrenheit for resistance temperature detectors Analog five degrees Fahrenheit for thermocouple; five degrees Fahrenheit for resistance temperature detectors
Range of display	See "B" below
Qualifications (seismic, environmental, IEEE 323)	Being qualified to applicable requirements of IEEE-323-71 based on instruments meeting required response spectra.

Calculator

Type (process computer, dedicated digital or analog calculator)	Dedicated digital
If process computer is used specify availability (percent of time)	Not used
Single or redundant calculators	Redundant
Selection logic (highest temperature, lowest pressure)	Highest temperature for resistance temperature detector or thermocouple, lowest pressure
Qualifications (seismic, environmental, IEEE-323)	None at present

TABLE 22-3 (Cont'd)

Calculational technique (steam tables, functional fit, ranges)	Functional fit ambient to critical point
Input	
Temperature (resistance temperature detector or thermocouples)	Resistance temperature detector, thermocouple and tempeature reference
Temperature (number of sensors and locations)	Eight thermocouples per channel two hot leg and two cold leg resistance temperature detectors per channel
Range of temperature sensors	0-700 degrees Fahrenheit for resistance temperature detectors 0-1650 degrees Fahrenheit for thermocouples (see note 1)
Uncertainty* of temperature sensors (See note 2)	Thermocouples ± three degrees Fahrenheit resistance temperature detector - IEEE 323-74 and IEEE 344-75
Pressure (specify instrument used)	All are Barton model except one diverse wide range - later. One wide range per channel
Range of Pressure sensor	Wide range 0-3000 pounds per square inch Narrow range 1700-2500 pounds per square inch
Uncertainty* of pressure sensors (pounds per square inch) (See note 3)	± three percent span
Qualification (seismic, environmental, (IEEE 323)	IEEE 323-1974
Backup Capability	
1. Availability of temperature and pressure	Temperature and pressure readings are available on the control board and from the computer for use in determining subcooled margin manually.
2. Availability of steam tables	Saturated steam tables are available in the main control room. Emergency operating procedures contain instructions and curves for verifying subcooled conditions.

TABLE 22-3 (Cont'd)

3. Training and Operators	Operators will receive appropriate training on the use of the subcooling monitor and othermethods to determine required subcooling conditions.
4. Procedures	Emergency operating procedures contain instructions for determining subcooling conditions.

*Uncertainties must address condition of forced flow and natural circulation.

Note 1 - Calibration unit range 0-2300 degrees Fahrenheit.

Note 2 - Normal accuracy for thermocouple since they are not qualified. Accident accuracy for resistance temperature detectors does not include channel inaccuracies.

Note 3 - Estimated accuracy at conditions expected for small break loss-of-coolant accident based on 10 percent of span accuracy for large break loss-of-coolant accident environment. Does not include channel inaccuracies.

"A" - Caution - 25 degrees Fahrenheit subcooled for resistance temperature detector; 15 degrees Fahrenheit subcooled for thermocouple.
- Alarm - 0 degrees Fahrenheit subcooled for resistance temperature detector and thermocouples.

"B" - Calibrated region - 1000 pounds per square inch subcooled to 2000 degrees Fahrenheit superheat overall - never off scale.

Discussion and Conclusions

Description of Existing Instrumentation

We have reviewed the adequacy of the proposed inadequate core cooling monitoring system concept for detection and early warning of inadequate core cooling, and the functional performance and reliability requirements for the inadequate core cooling monitoring system hardware. We also reviewed th› human factors aspects of the inadequate core cooling information display and the operator actions to prevent inadequate core cooling or to restore core cooling.

We performed the review based on the applicant's response to Section II.F.2 of TMI Action Plan submitted on July 18, December 4, and December 15, 1980.

Subcooling Monitoring

The core subcooling monitor provides continuous monitoring of the margin to saturation in the reactor core (i.e., the amount of subcooling) on the main control board. The core subcooling monitor utilizes inputs from the hot leg resistance temperature detectors, reactor coolant system pressure sensors, and selected incore thermocouples. A summary of information required for the subcooling monitor was provided in Table 22-3. The subcooling monitor has been installed and will be fully operational prior to fuel load in response to NRC requirements of NUREG-0737. This system has temperature inputs from resistance temperature detectors (two hot and two cold legs per channel), incore thermocouples (eight per channel) and temperatue reference for the incore thermocouples. Two pressure inputs are taken from the reactor coolant system.

Margin to saturation is available on the main control board based on both auctioneered high hot leg temperature and on auctioneered high incore thermocouples. In addition to the main control board indication, alarms are provided on the main control board to indicate development of off-normal conditions and to indicate the approach to loss of normal core subcooling.

The core subcooling monitor is designed and qualified to the requirements of IEEE-323-1971. Sections 3.10 and 3.11 of the Final Safety Analysis Report provide additional details of the seismic and environmental qualification.

Incore Thermocouple Monitor

The primary means for monitoring thermocouple temperatures is the plant process computer. The computer constantly monitors all 51 incore thermocouple temperature values over a range of 70 degrees Fahrenheit to 2200 degrees Fahrenheit. A spatially oriented core map can be printed on operator demand giving the temperature at each core exit thermocouple location. This process takes less than ten minutes. A printed list of all incore thermocouple temperatures can be obtained in less than five minutes. When any value exceeds preset alarm limits (700 degrees Fahrenheit hi and 1200 degrees Fahrenheit hi-hi) the computer prints an alarm message on the alarm typewriter and on the control board cathode ray tubes. Up to 51 of the thermocouples can be trended by the computer with output ont he trend typewriter. Four of the 51 incore thermocouples may be selected for trending on computer trend recorders located on the main control board. Up to 30 thermocouple values may be selected for continuous display on either control room cathode ray tube. The computer also is capable of determining and displaying the highest thermocouple value and the average of all thermocouple values on the cathode ray tube. The trend and alarm typewriters are located on the computer operators console in the control room. The two cathode ray tubes are located on the center section (reactor panel) of the main control board. Trend and display selections are controlled from the computer operators console.

The second means of monitoring incore thermocouple temperature is the core subcooling monitor system. Each channel of the subcooling monitor receives inputs from eight thermocouples (two per core quadrant per channel, for a total of 16 thermcouples). A digital readout of any of the 16 single thermocouple temperatures may be obtained at the subcooling monitor panel located in the control room. The upper limit of the readout if 2300 degrees Fahrenheit.

The third means available for monitoring thermcouple temperature is the incore thermocouple readout meter located in the incore instrumentation panel in the control room. Any of the 51 incore thermocouples may be selected by toggle switch positioning and read on the analog readout. The readout range is 100 - 700 degrees Fahrenheit.

If the manual readout should go off scale high, or problems exist with any of these systems, thermocouple temperature may be measured directly by connecting a millivolt potentiometer to any of the thermocouple inputs in the back of the incore instrumentation panel in the control room.

In the event that the margin to saturation decreases to less than 15 degrees Fahrenheit as indicated by thermocouple input to the subcooling monitor, the "core subcooling alarm" annunciator actuates and monitoring of the incore thermocouples is initiated in accordance with annunciator response procedures.

If any five exit incore thermocouples indicate a temperature greater than or equal to 1200 degrees Fahrenheit, action is initiated in accordance with the inadequate core cooling emergency operating procedures.

<u>Staff Evaluation of Existing Inadequate Core Cooling Instrumentation and Procedures for Power Level Up to 100 Percent</u>

The Westinghouse Owner's Group, of which the applicant is a member, has performed analyses as required by TMI Task item I.C.1 to study the effects of inadequate core cooling. These analyses were provided to the NRC Bulletins and Orders Task Force for review on October 31, 1979. As part of the submittal made by the Owner's Group, an instruction to restore core cooling during a small loss-of-coolant accident was included. This instruction provides the basis for procedure changes and operator training required to recognize the existence of inadequate core cooling and restore core cooling based on existing instrumentation. These guidelines were reviewed and approved by the staff in November, 1979.

The applicant has submitted the inadequate core cooling procedure (EOP-14) for staff review. Based on our review of the applicant's emergency operating procedures, we find they are consistent with the guidelines for Westinghouse plants. Therefore, we find the procedures to be acceptable for full power operation.

The staff has also reviewed the design of the core subcooling meter and incore thermocouples systems. We will require the applicant to submit the documentation required by NUREG-0737, which includes an evaluation of the core subcooling monitor instrumentation and incore thermocouple system prior to full power operation.

The staff concludes that the existing inadequate core cooling instrumentation and procedures proposed by the applicant are acceptable for fuel load and operation up to five percent power.

The subcooling meter display will be reviewed and our evaluation will be reported in a supplement to this Safety Evaluation Report.

Prior to full power operation, we will require:

(1) An acceptable evaluation report on the conformance of the existing final inadequate core cooling instrumentation to the requirements of II.F.2 Attachment 1 and Appendix B to NUREG-0737.

(2) A description of the computer functions associated with inadequate core cooling monitoring and functional specification for relevant software in the process computer and in the subcooling meter calculators. The reliability of the process computer must be addressed.

(3) An updated description and status report on the planned modifications for the subcooling meters, if necessary, based on our human factors review.

Clarification of the Position for Additional Instrumentation

A clarification of requirements for additional inadequate core cooling instrumentation was provided in the H. Denton letter (dated October 1979) to all Operating Nuclear Power Reactor Applicants and Licensees and in "Clarification of TMI Action Plan Requirements" from H. Denton to Commissioners (dated October 22, 1980) as follows:

(1) Design of new instrumentation should provide an unambiguous indication of inadequate core cooling. This may require new measurements or a synthesis of existing measurements which meet design criteria (item 7).

(2) The evaluation is to include reactor-water-level indication.

(3) Licensees and applicants are required to provide the necessary design analysis to suppor the proposed final instrumentation system for inadequate core cooling and to evaluate the merits of various instruments to monitor water level and to monitor other parameters indicative of core-cooling conditions.

(4) The indication of inadequate core cooling must be unambiguous in that it should have the following properties:

(a) It must indicate the existence of inadequate core cooling caused by various phenomena (i.e., high-void fraction-pumped flow as well as stagnate boil-off); and,

(b) It must not erroneously indicate inadequate core cooling because of the presence of an unrelated phenomenon.

(5) The indication must give advance warning of the approach of inadequate core cooling.

(6) The indication must cover the full range from normal operation to complete core uncovery. For example, water-level instrumentation may be chosen to provide advance warning to two-phase level drop to the top of the core and could be supplemented by other indicators such as incore and core-exit thermocouples provided that the indicated temperatures can be correlated to provide indication of the existence of inadequate core cooling and to infer the extent of core uncovery. Alternatively, full-range level instrumentation to the bottom of the core may be employed in conjunction with other diverse indicators such as core-exit thermocouples to preclude misinterpretation due to any inherent deficiencies or inaccuracies in the measurement system selected.

(7) All instrumentation in the final inadequate core cooling system must be evaluated for conformance to Appendix B, "Design and Qualification Criteria for Accident Monitoring Instrumentation," as clarified or modified by the provisions of items 8 and 9 that follow. This is a new requirement.

(8) If a computer is provided to process liquid-level signals for display, seismic qualification is not required for the computer and associated hardware beyond the isolator or input buffer at a location accessible for maintenance following an accident. The single-failure criteria of item 2, Appendix B, need not apply to the channel beyond the isolation device if it is designed to provide 99 percent availability with respect to functional capability for liquid-level display. The display and associated hardware beyond the isolation device need not be Class 1E, but should be energized from a high-reliability power source which is battery backed. The quality assurance provisions cited in Appendix B, item 5, need not apply to this portion of the instrumentation system. This is a new requirement.

By January 1, 1981, the licensee shall provide a report detailing the planned instrumentation system for monitoring of inadequate core cooling. The report should contain the necessary information, either by inclusion or by reference to previous submittals including pertinent generic reports, to satisfy the requirements which follows:

(1) A description of the proposed final system including:

(a) a final design description of additional instrumentation and displays;

(b) a detailed description of existing instrumentation systems (e.g., subcooling meters and incore thermocouples), including parameter ranges and displays, which provide operating information pertinent to inadequate core cooling considerations; and

(c) a description of any planned modifications to the instrumentation systems described in item 1.b above.

(2) The necessary design analysis, including evaluation of various instruments to monitor water level, and available test data to support the design described in item 1 above.

(3) A description of additional test programs to be conducted for evaluation, qualification, and calibration of additional instrumentation.

(4) An evaluation, including proposed actions, on the conformance of the inadequate core cooling instrument system to this document, including Attachment 1 and Appendix B. Any deviations should be justified.

(5) A description of the computer functions associated with inadequate core cooling monitoring and functional specifications for relevant software in the process computer and other pertinent calculators. The reliability of non-redundant computers used in the system should be addressed.

(6) A current schedule, including contingencies, for installation, testing and calibration, and implementation of any proposed new instrumentation or information displays.

(7) Guidelines for use of the additional instrumentation, and analyses used to develop these procedures.

(8) A summary of key operator action instructions in the current emergency procedures for inadequate core cooling and a description of how these procedures will be modified when the final monitoring system is implemented.

(9) A description and schedule commitment for any additional submittals which are needed to support the acceptability of the proposed final instrumentation system and emergency procedures for inadequate core cooling.

Discussion and Conclusions for Additional Instrumentation to Detect Inadequate Core Cooling

South Carolina Electric & Gas Company, in their December 4, 15, and 30, 1980 responses to TMI Action Plan Item II.F.2 of NUREG-0737, has committed to install a redundant Westinghouse designed reactor vessel level instrumentation system in the Virgil C. Summer Nuclear Station, Unit 1.

The reactor vessel level system provides a direct reading of reactor vessel level on the main control board which can be used by the operator in conjunction with the core subcooling monitor to identify the possibility of inadequate core cooling conditions. Reactor vessel level is also utilized to indicate the need to vent non-condensible gases from the reactor vessel head.

The Westinghouse reactor vessel level instrumentation system utilizes two sets of differential pressure cells to measure reactor vessel level. The narrow range reactor vessel level instrumentation system instrument provides an indication of reactor vessel water level from the bottom of the reactor vessel to the top of the reactor vessel when zero or one reactor coolant pump is operating. The narrow range instrument also measures the reactor core and internals pressure drop, and therefore provides an indication of the relative void content or density of the circulating fluid, when only one reactor coolant pump is operating. When more than one reactor coolant pump is operating, the narrow range instrument reading will be off scale.

The wide range reactor vessel level instrumentation system instrument provides an indication of reactor core, internals, and outlet nozzle pressure drop for any combination of operating reactor coolant pumps. Comparison of the measured pressure drop with the normal, single phase pressure drop provides an approximate indication of the relative void content or density of the circulating fluid. The wide range instrument monitors vessel level on a continuous basis.

The reactor vessel level instrumentation system is designed and qualified to the requirements IEEE-323-1971. Sections 3.10 and 3.11 of the Final Safety Analysis Report provide details of the seismic and environmental qualification.

The applicant stated in their December 4, 1980 letter that the procurement of the reactor vessel level instrumentation system is complete, the installation is in progress and the testing will be conducted in March, 1981.

The applicant also submitted a December 30, 1980 letter with a "Summary Report, Westinghouse Reactor Vessel Level Instrumentation System for Monitoring Inadequate Core Cooling" in response to the concern for vessel level instrumentation given in NUREG-0737 Part II.F.2. This report is under review by the staff.

The staff has reviewed the applicant's submittals dated December 4, December 15, and December 30, 1980 and has concluded that the full power operation license should not be issued until all the documentation required by NUREG-0737 is received and found to be acceptable. After the installation, testing, and calibration of the Westinghouse reactor vessel level instrumentation system, the final approval of the level system will be reviewed according to the requirements in Section II.F.2 of NUREG-0737. We will provide our evaluation in a supplement to this Safety Evaluation Report.

II.G Emergency Power for Pressurizer Equipment

Requirement

Motive and control components of the power-operated relief valves and associated block valves and the pressurizer level indication shall be capable of being supplied from the offsite power source or from the emergency power buses when offsite power is not available.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.1, and letters of September 27 and November 9, 1979.)

Position

Consistent with satisfying the requirements of Criteria 10, 14, 15, 17 and 30 of the General Design Criteria for the event of loss of offsite power, the following positions shall be implemented:

1. Motive and control components of the power-operated relief valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

2. Motive and control components associated with the power operated relief valve block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the power operated relief valves and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

4. The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

Clarification

1. While the prevalent consideration from TMI lessons learned is being able to close the power operated relief valves and block valves, the design should retain, to the extent practicable, the capability to open these valves.

2. The motive and control power for the block valve should be supplied from an emergency power bus different from that which supplies the power operated relief valves.

3. Any changeover of the power operated relief valve and block valve motive and control power from the normal offsite power to the emergency onsite power is to be accomplished manually in the control room.

4. For those designs where instrument air is needed for operation, the electrical power supply requirement should be capable of being manually connected to the emergency power sources.

Discussion and Conclusions

The control power for the solenoids of the three power operated relief valves is from safety grade sources which would be available if offsite power is not available. Two of the solenoids are powered from one 125 volt direct current Class 1E bus and the third one is powered from the redundant 125 volt direct current Class 1E bus. The motive power for two of the power operated relief valves is nitrogen supplied from accumulators which have been sized for 150 strokes and the third one is pneumatically operated from the instrument air system. In the event of loss of offsite power, the instrument air starting system required to actuate the third power operated relief valve can be supplied from the emergency diesel generator.

The three block valves are motor operated valves energized from two redundant emergency 480 volt buses which are powered automatically from their respective diesel generator upon loss of offsite power. Two of the valves are powered from one bus and the third is connected to the other redundant bus.

The motive and control power supplied to the power operated relief valves and their associated block valves is through safety grade devices.

Three redundant channels of pressurizer level instrumentation are provided. These channels are powered from the vital instrument buses which are capable of being powered from the diesel generators upon loss of offsite power.

We have reviewed the above information and conclude it is in accordance with our positions and is acceptable.

II.K.1 IE Bulletins on Measures to Mitigate Small-Break Loss-of-Coolant Accidents Loss of Feedwater Accidents*

The following requirements shall be met before fuel loading.

C.1.5 Requirement

Review all valve positions, positioning requirements, positive controls and related test and maintenance procedures to assure proper engineered safety feature functioning. (See Bulletin 79-06A, Item 8, 79-06B, Item 7, 79-08, Item 6.)

Discussion and Conclusion

The applicant's response to this requirement indicates that a series of procedures (administrative, operating, maintenance, testing) has been developed to provide control of valve alignments, positioning requirements, and positive control requirements to assure proper functioning of engineered safety features.

The plant system operating procedures for safety-related systems include valve checklists that specify initial valve alignment requirements for system startup. Valve manipulations required during the conduct of the procedure are specified in the procedures so that the system is correctly aligned at all times for the current plant mode of operation.

Surveillance test procedures have been developed which require verification of correct valve alignments by either visual observation or flow verification. The test procedures include a sign-off verification at the completion of the test by an independent operator to assure that the valves have been restored to the system procedure designated status.

Administrative procedures have been developed to control any alterations of safety-related systems for test or maintenance. They insure that Technical Specification limiting conditions for operation are maintained or appropriate corrective action is taken. The procedures require that an independent operator verify that the valve alignment is returned to the designated status at the completion of maintenance activities.

Requirements for positive controls (locks on valves or electrical breakers) are specified in the Technical Specifications and plant procedures. An

*Table C.1 of NUREG-0660 lists all the requirements given in IE bulletins.

administrative procedure is developed to control such locks and to verify on a periodic basis that they have not been tampered with.

We find the above applicant's procedures covering review and verification of the operational status of engineered safety feature valving adequately addresses the concerns raised in this item.

C.1.10 Requirement

Review and modify, as required, procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known. low-pressure setpoint is reached regardless of the pressurizer level. (See IE Bulletin 79-06A and Revision 1, Item 3.)

Discussion and Conclusions

Section 7.2.1.1.2 of the Final Safety Analysis Report indicates that the reactor trip and safety injection are initiated on pressurizer low pressure without coincident signal from pressurizer low-level. We conclude that the design meets the requirements of item II.K.1 (C.1.17).

II.K.3 Final Recommendations of Bulletins & Orders Task Force

The following requirements shall be met before fuel loading.

C.3.9 Requirement

For Westinghouse-designed reactors, modify the pressure integral derivative controller, if installed on the power-operated relief valve to eliminate spurious openings of the power-operated relief valve.

Clarification

The Westinghouse recommended modification is to raise the interlock bistable trip setting to preclude derivative action from opening the power-operated relief valve. Some licensees have proposed setting the derivative action setting to zero and thereby eliminate it from consideration. Either modification is acceptable to the staff.

Discussion and Conclusions

The applicant, in accordance with the Westinghouse recommendation, has modified the facility design by incorporating a rate-time constant in the proportional integral derivative controller of zero seconds. This, in effect, removes the derivative action from the controller which decreases the likelihood of opening the power-operated relief valve since the actuation (opening) signal will not be sensitive to the rate of change of the pressurizer pressure.

We find that the applicant has satisfied this TMI Action Plan item which calls for the elimination of spurious openings of the power operated relief valve caused by the derivative feature of the proportional integral derivative controller.

C.3.10 Requirement

For Westinghouse-designed reactors, if the anticipatory reactor trip upon turbine trip is modified so that it will be bypassed at power levels less than 50 percent, rather than below 10 percent as in current designs, demonstrate that the probability of a small-break loss-of-coolant accident resulting from a stuck-open power operated relief valve is not significantly changed by this modification.

Clarification

It has been proposed for some Westinghouse facilities that the anticipatory trip-bypass be raised from less than 10 percent reactor power to less than 50 percent. Where such a modification is proposed, item C.3.10 of NUREG-0660 requires applicants to demonstrate that the probability of a small-break loss of coolant accident resulting from a stuck open power operated relief valve is not significantly changed by such a modification.

The design of the facility features an anticipatory reactor trip upon turbine trip which is interlocked with the P-7 setpoint to prevent a reactor trip at power levels less than 10 percent. No modification to this design feature has been proposed or made.

Discussion and Conclusions

This issue is not applicable to the facility, since its design does not incorporate the modification to the anticipatory reactor trip turbine trip.

C.3.11 Requirement

Demonstrate that the power-operated relief valve installed in the plant has a failure rate equivalent to or less than the valves for which there is an operating history.

Discussion and Conclusion

This issue is not applicable to the facility since it does employ power-operated relief valves for which there is an operating history.

C.3.12 Requirement

For Westinghouse-designed reactors, confirm that there is an anticipatory reactor trip on turbine trip.

Discussion and Conclusions

The facility has an anticipatory reactor trip on turbine trip. (Refer to discussion of item II.K.3.10 in this Safety Evaluation Report.) Therefore, we find that this requirement has been satisfied.

III.A.1.1 Upgrade Emergency Preparedness

Requirement

Comply with Appendix E, "Emergency Facilities," to 10 CFR Part 50, Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," and for the offsite plans, meet essential elements of NUREG-75/111 or have a favorable finding from Federal Emergency Management Agency.

This requirement shall be met before fuel loading.

Discussion and Conclusions

See Section 13.3 of this Safety Evaluation Report.

III.A.1.2 Upgrade Emergency Support Facilities

Requirement

Establish an interim onsite technical support center separate from, but close to, the control room for engineering and management support of reactor operations during an accident. The center shall be large enough for the necessary utility personnel and five NRC personnel, have direct display or callup of plant parameters, and dedicated communications with the control room, the emergency operations center, and the NRC. Provide a description of the permanent technical support center.

Establish an onsite operational support center separate from, but with communications to, the control room for use by operations support personnel during an accident.

Designate a near-site emergency operations facility with communications with the plant to provide evaluation of radiation releases and coordination of all onsite and offsite activities during an accident.

These requirements shall be met before fuel loading. (See NUREG-0578, Sections 2.2.2.b, 2.2.c, and letters of September 27 and November 9, 1979 and April 25, 1980.)

Discussion and Conclusions

See Section 13.3 of this Safety Evaluation Report.

III.D.3.3 Inplant Radiation Monitoring

Requirement

Provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.8c, and letters of September 27 and November 9, 1979.)

Clarification

Use of Portable Versus Stationary Monitoring Equipment

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments for the following reasons:

a. The physical size of the auxiliary/fuel handling building precludes l ating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

b. Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

c. Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

d. The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high dose rate areas.

Iodine Filters and Measurement Techniques

a. The following are short-term recommendations and shall be implemented by January 1, 1980 or fuel loading date, whichever is later. The licensee shall have the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler with attached single channel analyzer. The single channel analyzer window should be calibrated to the 365 thousand electron volts of iodine-131. A representative air sample shall be taken and then counted for iodine-131 using the single channel analyzer. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too much activity collected on the sampling cartridge.

b. For Section 22.5, Dated Requirements, we require that by January 1, 1981, the licensee shall have the capability to remove the sampling cartridge to a low background, low contamination area for further analysis. This area should be ventilated with clean air containing no airborne radionuclides which may contribute to inaccuracies in analyzing the sample. Here, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble bases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples and effluent charcoal samples under accident conditions.

Discussion and Conclusions

The applicant has fixed process r[illegible] monitors, portable continuous air monitors, and grab air samplers that ca[illegible]tilize silver zeolite cartridges in the event of an unplanned gas release in[illegible]ving high concentrations of noble gases. Lithium detectors and a gas flow proportional counter can also be utilized to determine the quantity of radioiodine collected on a filter cartridge.

Portable single channel analyzers with sodium iodide detectors are available as back-ups if other instruments are inoperable. The applicant has not provided any information on how the sample will be counted in the low background counting facility. This area must be identified prior to January 1, 1981.

Health physics procedures for determining iodine concentration in noble gas environments are available. Health physics personnel are trained on action levels requiring the use of silver zeolite cartridges and the use of portable single channel analyzers and sodium iodine detectors.

The fuel load requirement is met. The applicant must identify the low background area for counting of samples to be in use after January 1, 1981.

22.3 Full-Power Requirements

I.C.7 NSSS Vendor Review of Procedures

Requirement

Obtain nuclear steam supply system vendor review of power-ascension test and emergency procedures to further verify their adequacy.

This requirement must be met before issuance of a full-power license.

Discussion and Conclusions

For our evaluation of this matter refer to item I.C.7 in Section 22.2 of this Safety Evaluation Report.

I.C.8 Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants

Requirement

Correct emergency procedures, as necessary, based on the NRC audit of selected plant emergency operating procedures (e.g., small-break loss-of-coolant accident, loss of feedwater, restart of engineered safety features following a loss of alternating current power, steamline break, or steam-generator tube rupture.

This action will be completed prior to issuance of a full-power license.

Position

Audit emergency operating procedures to provide a sense of the adequacy of the emergency procedures and review the training related to the symptoms of the postulated transients.

Conduct an in-depth review of selected emergency procedures in item I.C.1. Based on that review, observe a simulator walk-through of the selected procedures (with shift crew and shift technical advisor); observe a plant walk-through for one of the emergency procedures (observe shift crew, shift technical advisor, communications with technical support center operation and operational support center operation, etc.); and make findings on preparedness for the accidents covered by the selected procedures.

Discussion and Conclusion

Representatives of the staff and Battelle Pacific Northwest Laboratories completed a review of the emergency operating procedures and met with the applicant on October 29-30, 1980 to discuss our comments. A simulator walk-through of the selected emergency operating procedures at the Zion simulator was conducted November 3-4, 1980.

The procedures for reactor trip, safety injection, loss of reactor coolant (including small breaks), loss of feedwater flow, and steam generator tube rupture were followed for simulated accidents. The procedures used incorporate the latest comments from the staff and from Westinghouse and provided generally satisfactory instructions for the control of the accidents. During the simulations, it was observed that the procedures did not adequately address the shift technical advisor interface and, did not include instructions on isolating non-faulted steam generators if a faulted steam generator could not be identified. It was further noted that the applicant's operator training program should provide more instructions on administrative controls and handling deviations from operating the procedures. The applicant agreed with each of these observations and has appropriately modified their procedures and training program. The loss of coolant accident procedure was used during the plant walk-through on December 17, 1980 to monitor operator actions, administrative controls of documentation in the control room, and the agreement between the control panel procedure nomenclature. Some minor changes in the procedures were noted and corrected by the applicant. The administrative control of documents in the control room appeared to be adequate and the control panel/procedure nomenclature was consistent. The applicant stated that as the control panel was modified, there would be a continuing effort to revise and correct all of the operating procedures.

Based on our observations of the simulator and plant walk-through using the applicant's procedures and with the further corrections identified above, we find the procedures acceptable for operation at power levels up to 100 percent of rated power. Further revisions required by Task Action Plan items I.C.1(3). Transients and Accidents, and I.C.9 Long Term Program for Upgrading of Procedures, may require future revisions to the emergency operating procedures. We also require that the remainder of emergency operating instructions be revised in accordance with our comments on the procedures reviewed and that the operators be briefed on the revisions with 30 effective full power days of operation. The Office of Inspection and Enforcement will verify that these requirements are satisfied.

I.G.1 Training During Low-Power Testing

Requirement

Supplement operator training by completing the special low-power test program. Tests may be observed by other shifts or repeated on other shifts to provide training to the operators.

This requirement shall be met before issuance of a full-power license.

Discussion and Conclusions

Our evaluation of this matter will be provided in a supplement to this Safety Evaluation Report.

II.B.1 Reactor Coolant System Vents

Requirement

Provide a description of the design of reactor coolant system and reactor vessel head high point vents that are remotely operable from the control room and supporting analyses.

This requirement shall be met before issuance of a full-power license. See letters of September 27 and November 9, 1979.

Position

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents which can be remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the reactor coolant system which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the events shall conform to the requirements of Appendix A to 10 CFR Part 50. The vent system shall be designed with sufficient redundancy to assure a low probability of inadvertent or irreversible actuation.

Each licensee shall provide the following information concerning the design and operation of the high point vent system:

(1) Submit a description of the design, location, size and power supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of Section 50.46 of 10 CFR Part 50.

(2) Submit procedures and supporing analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

Clarification
General

(1) The important safety function enhanced by this venting capability is core cooling. For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of noncondensible gas which could interfere with core cooling.

(2) Procedures addressing the use of the reactor coolant system vents should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be directed toward achieving a substantial increase in the plant being able to maintain core cooling without loss of containment integrity for events beyond the design basis. The use of vents for accidents within the normal design basus must not result in a violation of the requirements of Section 50.44 or Section 50.46 of 10 CFR 50.

(3) The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with events in a fairly broad spectrum of sizes. The criteria for sizing a vent can be developed in several ways. One approach, which may be considered, is to specify a volume of noncondensible gas to be vented and in a specific venting time. For containments particularly vulnerable to failure from large hydrogen releases over a short period of time, the necessity and desirability for contained venting outside containment must be considered (e.g., into a decay gas collection and storage system).

(4) Where practical, the reactor coolant system vents should be kept smaller than the size corresponding to the definition of loss of coolant accident in Appendix A to 10 CFR Part 50. This will minimize the challenges to the emergency core cooling system since the inadvertent opening of a vent smaller than the loss of coolant accident definition would not require emergency core cooling system actuation, although it may result in leakage beyond Technical Specification limits. On pressurized water reactors, the use of new or existing lines whose smallest orifice is larger than the loss of coolant accident definition will require a valve in series with the vent valve that can be closed from the control room to terminate the loss of coolant accident that would result if an open vent valve could not be reclosed.

(5) A positive indication of valve position should be provided in the control room.

(6) The reactor coolant vent system shall be operable from the control room.

(7) Since the reactor coolant system vent will be part of the reactor coolant pressure boundary, all requirements for the reactor coolant pressure boundary must be met, and, in addition, sufficient redundancy should be incorporated into the design to minimize the probability of an inadvertent actuation of the system. Administrative procedures, may be a viable option to meet the single-failure criterion. For vents larger than the loss of coolant definition, an analysis is requred to demonstrate compliance with Section 50.46 of 10 CFR Part 50.

(8) The probability of a vent path failing to close, once opened, should be minimized; this is a new requirement. Each vent must have its power supplied from an emergency bus. A single failure within the power and control aspects of the reactor coolant vent system should not prevent isolation of the entire vent system when required. On boiling water reactors, block valves are not required in lines with safety valves that are used for venting.

(9) Vent paths from the primary system to within containment should go to those areas that provide good mixing with containment air.

(10) The reactor coolant vent systems (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) shall be seismically and environmentally qualified in accordance with IEEE Standard 344-1975 as supplemented by Regulatory Guides 1.92 and 1.100 and Sections 3.9.2 and 3.10 of the Standard Review Plan. Environmental qualifications are in accordance with the May 23, 1980 Commission Order and Memorandum (CLI-80-21).

(11) Provisions to test for operability of the reactor coolant vent system should be a part of the design. Testing should be performed in accordance with Subsection IWV of Section XI of the ASME Code for Category B valves.

(12) It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

(a) The use of this information by an operator during both normal and abnormal plant conditions,

(b) Integration into emergency procedures,

(c) Integration into operator training, and

(d) Other alarms during emergency and need for prioritization of alarms.

Pressurized Water Reactor Vent Design Considerations

(1) Each pressurized water reactor licensee should provide the capability to vent the reactor vessel head. The reactor vessel head vent should be capable of venting noncondensible gas from the reactor vessel hot legs (to the elevation of the top of the outlet nozzle) and cold legs (through head jets and other leakage paths).

(2) Additional venting capability is required for those portions of each hot leg that cannot be vented through the reactor vessel head vent or pressurizer. It is impracticable to vent each of the many thousands of tubes in a U-tube steam generator; however, the staff believes that a procedure can be developed that assures sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the reactor coolant system. Such operating procedures should incorporate this consideration.

(3) Venting of the pressurizer is required to assure its availability for system pressure and volume control. These are important considerations, especially during natural circulation.

The licensee shall provide the following information on the reactor vent system for staff review:

(1) The information requested in items 1 and 2 under Position;

(2) A discussion of the design with respect to conformance to the design criteria discussed under Clarification, including deviations, if any, with adequate justification for such deviations; and,

(3) Supporting information including logic diagrams, electrical schematics, piping and instrumentation diagrams, test procedures, and technical specifications.

Discussion and Conclusion

In response to the requirements of this item from NUREG-0699, the applicant, in Section 5.5.15 of the Final Safety Analysis Report (Amendment 18), provided a system description and simplified diagram of the proposed reactor coolant system vent. However, the information provided is not sufficient for our evaluation. We have requested the applicant to provide all information necessary for staff review of this item, as that specified in the above. We will report our evaluation of this item in a supplement of this Safety Evaluation Report.

II.B.2 Plant Shielding

Requirement

Provide (1) a radiation and shielding design review that identifies the location of vital areas and equipment in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by radiation during operations following an accident resulting in a degraded core, and (2) a description of the types of corrective actions needed to assure adequate access to vital areas and protection of safety equipment.

This requirement shall be met before issuance of a full-power license. See NUREG-0578, Section 2.1.6b and letters of September 27 and November 9, 1979.

Discussion and Conclusion

The applicant's plant shielding design report was reviewed to evaluate the ability to have access to vital areas necessary to operate essential systems required after a loss-of-coolant accident with significant core damage.

The systems designed to function after an accident include the residual heat removal system, safety injection system, reactor building spray systems, reactor coolant system, post-accident hydrogen removal system, nuclear sampling system, gaseous radwaste system and portions of the chemical and volume control system involving the high head and seal water injection sections. The remainder of the chemical volume and control system was excluded because it is isolated and because its use in a post-accident situation would be unacceptable.

Calculation of source terms and estimated dose rates used for shielding design are based on Regulatory Guides 1.4, 1.7 and the guidelines of Criterion 19 of the Criteria. The applicant has provided "radiation" maps that shows access routes to vital areas, to be used as an administrative guide in the control of access and reduction of personnel exposure during the course of an accident.

Vital area which require continuous or frequent occupancy in order to control, monitor, and evaluate the accident were identified. In addition, the applicant identified potential maintenance activities that might become necessary during recovery and determined when after an accident such maintenance would be possible. For the vital areas the applicant has provided a person-rems time, distance and personnel occupacy study.

The need for plant modifications in seven areas was identified. The areas are intermediate building pipe chase, boron injection tank, auxiliary building elevation 425 feet, charging pump cooling units, cubicle 12-09 of auxiliary building, residual heat removal system spray pump rooms, cooling units and sampling system. The applicant has committed that these modifications will be completed prior to issuance of the operating license or before January 1, 1982, whichever is later. Information on the post-accident sampling and analysis systems must be provided prior to full power operation. The applicant has committed that these systems will be designed to assure the limits of 10 CFR Part 20 are not exceeded.

On the basis of our review, we have concluded with the exception of the sampling and analysis information, that the applicant has performed a radiation and shielding design review for vital area access in accordance with Action Plan Item II.B.2.

II.B.3 Post-Accident Sampling

Requirement

Provide (1) a design and operational review of the reactor coolant and containment atmosphere sampling line system to determine the capability of personnel to promptly obtain a sample under accident conditions without incurring radiation exposure in excess of three and 18 3/4 rem to the whole body or extremeties, respectively. Accident conditions should assume a Reguatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

This requirement shall be met before issuance of a full-power license. See NUREG-0578, Section 2.1.8a, and letters of September 27 and November 9, 1979.

Position

A design and operational review of the reactor coolant and containment atmosphere sampling line system shall be performed to determine the capability of personnel to promptly obtain (less than one hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of three and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than two hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).

Clarification

The following items are clarifications of requirements identified in NUREG-0578, NUREG-0660, or the September 13, 1979, October 30, 1979, September 5, 1980 and October 31, 1980 clarification letters.

(1) The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samles. The combined time allotted for sampling and analysis should be three hours or less from the time a decision is made to take a sample.

(2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the three hour time frame established above, quantification of the following:

   (a) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases; iodines and cesiums, and nonvolatile isotopes);

   (b) hydrogen levels in the containment atmosphere;

   (c) dissolved gases (e.g., hydrogen), chloride (time allotted for analysis subject to discussion below), and boron concetration of liquids; and

   (d) alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

(3) Reactor coolant and containment atmosphere sampling during post-accident conditions shall not require an isolated auxiliary system (e.g., the letdown system or reactor water cleanup system) to be placed in operation in order to use the sampling system.

(4) Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or hydrogen gas in reactor coolant samples is considered adequate. Measuring the concentration is recommended, but is not mandatory.

(5) The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall rovide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within four days. The chloride analysis does not have to be done onsite.

(6) The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of Criterion 19 of the General Design Criteria (i.e., five rem whole body, 75 rem extremities). Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to Criterion 19 of the General Design Criteria (October 30, 1979 letter from D. G. Eisenhut to all licenses.

(7) The analysis of primary coolant samples for boron is required for pressurized water reactors. (Note that Revision 2 of Regulatory Guide 1.97, when issued, will likely specify the need for primary coolant boron analysis capability at boiling water reactor plants.)

(8) If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for seven days following onset of the accident and at least one sample per week until the accident condition no longer exists.

(9) The licensee's radiological and chemical sample analysis capability shall include provisions to:

   (a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately one microcurie per gram to 10 curies per gram.

(b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of two). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

(10) Accuracy, range and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

(11) In the design of the post-accident sampling and analysis capability, consideration should be given to the following items:

(a) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the reactor coolant system or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post-accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmoshere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be retured to containment or to a closed system.

(b) The ventilation exhaust from the sampling station should be filtered with charcoal adsorbers and high efficiency particulate air filters.

(12) If gas chromatography is used for reactor coolant analysis, special provisions (e.g., pressure relief and purging) shall be provided to prevent high pressure argon from entering the reactor coolant.

(13) This requirement applies to all operating reactors and applicants for operating licenses. Installation should take place by January 1, 1982.

(14) Operating license applicants - Provide a description of the implementation of the position and clarification including piping and instrumentation diagrams, together with either (a) a summary description of procedures for sample collection, sample transfer or transport, and sample analysis, or (b) copies of procedures for sample collection, sample transfer or transport, and sample analysis in accordance with the proposed review schedule but in no case less than four months prior to the issuance of an operating license. A post-implementation review will be performed.

Discussion and Conclusions

By Amendments 18 through 22 and letters dated August 28, 1980 and November 6, 1980, the applicant has committed to install a post-accident sampling system for obtaining reactor coolant and containment atmosphere samples under degraded core accident conditions without excessive exposure, by January 1, 1981. The system will be located in the auxiliary building with shielded panels for liquid and gas sampling. Provisions are included for remote control of sampling components, ventilation air purging, pumps, dilution services and drains. On-line analysis will be provided for chloride, hydrogen, oxygen, pH and boron and conductivity. Backup grab sampling capability will be provided. Provisions are also included for preventing argon gas from entering into the reactor coolant system during analyses for oxygen and hydrogen. The post-accident sampling system will be used for both normal and post-accident operations. The applicant should provide the piping and instrumentation diagrams and location figure(s) at least four months prior to the issuance of an operating license.

We find the proposed post-accident sampling and analysis system to be installed by January 1, 1981, can meet the intent of the II.B.3 requirement; however, after receipt of the applicant's description of the as-built system, we will perform a post-implementation review and provide our completed evaluation in a supplement to this Safety Evaluation Report.

II.B.4 Training for Mitigating Core Damage

Requirement

Complete the training of all operating personnel in the use of installed systems to monitor and control accidents in which the core may be severely damaged.

This requirement shall be met before issuance of a full-power license.

Discussion and Conclusions

Refer to our evaluation of item II.G.4 in Section 22.2 of this Safety Evalution Report.

II.E.1.1 Auxiliary Feedwater System Reliability Evaluation

Requirement

(1) Provide a simplified auxiliary feedwater system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for auxiliary feedwater system failure following a main feedwater transients, with particular emphasis on potential failures resulting from human errors, common causes, single point vulnerability, and test and maintenance outage.

(2) Provide an evaluation of the auxiliary feedwater system using the acceptance criteria of Section 10.4.9 of the Standard Review Plan.

(3) Describe the design basis accident and transients and corresponding acceptance criteria for the auxiliary feedwater system.

(4) Based on the analyses performed, modify the auxiliary feedwater system, as necessary.

These requirements shall be met before issuance of a full-power license.

Discussion and Conclusions

The Three Mile Island Unit 2 (TMI-2) accident and subsequent investigations and studies highlighted the importance of the auxiliary feedwater system in the mitigation of transients and accidents. As part of our assessment of the TMI-2 accident and related implications for operating plants, we evaluated the AFW systems for all operating plants having nuclear steam supply systems designed by Westinghouse (NUREG-0611) or Combustion Engineering (NUREG-0635). Our evaluations of these system designs are contained in the above documents along with our recommendations for each plant and the concerns which led to each recommendation. The objectives of the evaluation were to: (1) identify necessary changes in the auxiliary feedwater system design or related procedures at the operating facilities in order to assure the continued safe operation of these plants, and (2) to identify other system characteristics of the auxiliary feedwater systems which, on a long term basis, may require system modifications. To accomplish these objectives we:

(1) Reviewed plant specific auxiliary feedwater system designs in light of current regulatory requirements and,

(2) Assessed the relative reliability of the various auxiliary feedwater systems under various loss of feedwater transients (one of which was the initiating event of TMI-2) and other postulated failure conditions by determining the potential for auxiliary feedwater system failure due to common causes, single point vulnerabilities, and human error.

In accordance with the requirements of Item II.E.1.1 of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," we have included the following results of the auxiliary feedwater system review in this Safety Evaluation Report.

1. We have applied the generic results and recommendations from the above described reviews for operating plants to the auxiliary feedwater system for Virgil C. Summer Nuclear Station, Unit 1.

2. In a letter dated November 5, 1980, the applicant provided a document entitled, "Emergency Feedwater System Reliability Assessment, GAI Report No. 2203, Revision 1." This report evaluated the auxiliary feedwater system reliability for the three postulated transient and accident scenarios, identified for study in our March 10, 1980 letter, utilizing fault tree methodology and the NRC-approved failure rate data base. Results of the above study indicate that the auxiliary feedwater system for this facility is ranked in the high relity range for Case 1, loss of main feedwater and Case 2, loss of offsite power and in the medium to high reliability for Case 3, loss of all alternating current ac power. Dominant contributors to auxiliary feedwater system unreliability were also identified. We conclude that the applicant has satisfactorily complied with the reliability study requirements of our March 10, 1980 letter, and the auxiliary feedwater system reliability assessment is acceptable.

3. We have reviewed the applicant's deterministic comparison of the auxiliary feedwater system against Section 10.4.9 of the Standard Review Plan and Branch Technical Position ASB 10-1, and find that the auxiliary feedwater system design is in compliance. Environmental qualification of the auxiliary feedwater system is being reviewed by the Equipment Qualification Branch as a separate item, and will be reported in a supplement to this Safety Evaluation Report

4. We have reviewed the applicant's response to our request in Enclosure 2 of our letter dated March 10, 1980, regarding the design basis for auxiliary feedwater system flow requirements. The applicant provided this information in a letter dated August 15, 1980. We conclude that the applicant's design basis for auxiliary feedwater system flow requirements is acceptable.

We conclude that the implementation of the following recommendations identified from the above reviews have improved the reliability of the auxiliary feedwater system for this facility.

Implementation of Our Recommendations

A. Short Term Recommendations

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one auxiliary feedwater system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Technical Specifications; i.e., 72 hours and 12 hours, respectively.

   In response, the applicant indicated in a letter dated August 15, 1980, that the proposed Technical Specification, Section 3.7.1.2 applies. This specification limits the plant operation with one auxiliary feedwater system train out of service to 72 hours and the subsequent action time to 12 hours. We conclude that this Technical Specification is in compliance with our recommendation and is, therefore, acceptable.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the auxiliary feedwater system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all auxiliary feedwater system flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See recommendation GL-2 for the longer-term resolution of this concern.

   In a letter dated August 15, 1980, the applicant responded to this recommendation by stating that the primary auxiliary feedwater system water passes through a single normally locked open valve in the common suction piping to the auxiliary feedwater pumps. This valve is provided with a limit switch which is alarmed in the control room when it is not in the full open position. In addition, the applicant will incorporate periodic inspections into the surveillance requirements of the Technical Specifications to verify the valve position. In accordance with our requirements, we will assure that these inspections be performed monthly. The applicant has also committed to remove the hand wheel from this valve to provide further protection against its inadvertent closure. Based upon the above commitments, we conclude that the applicant's response is acceptable.

3. Recommendation GS-3 - The licensee has stated that it throttles auxiliary feedwater flow to avoid water hammer. The licensee should reexamine the practice of throttling auxiliary feedwater system flow to avoid water hammer.

   The licensee should verify that the auxiliary feedwater system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100 percent power. In cases where this reevaluation results in an increase in initial auxiliary feedwater system flow, the licensee should provide sufficient information to demonstrate that the required initial auxiliary feedwater system flow will not result in plant damage due to water hammer.

In response, the applicant indicated in a letter dated August 15, 1980, that throttling of the auxiliary feedwater system to avoid water hammer will not be utilized. Based on the applicant's response, we conclude that recommendation GS-3 is not applicable to this facility.

4. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of auxiliary feedwater system supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   (1) The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the auxiliary feedwater system pumps against self-damage before water flow is initiated.

   (2) The case in which the primary water supply is being depleted. The procedures for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

   In response to this recommendation, the applicant indicated in a letter dated August 15, 1980, that plant procedures provide criteria for transfer to the alternate water source for both the case where primary water supply is not initially available and the case where the primary water supply is being depleted. We conclude that the applicant's response is acceptable pending verification of the plant procedures by the Office of Inspection and Enforcement.

5. Recommendation GS-5 - The as-built plant should be capable of providing the required auxiliary feedwater system flow for at least two hours from any one auxiliary feedwater pump train, independent of any alternating current power source. If manual auxiliary feedwater system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in a manual on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the auxiliary feedwater system is needed. See recommendation GL-3 for the longer-term resolution of this concern.

In response to this recommendation, the applicant indicated in a letter dated August 15, 1980, that the turbine-driven pump is capable of being automatically initiated and operated independent of any alternating current power source for at least two hours. Essential controls, valve operators, and other supporting systems associated with the turbine-driven auxiliary feedwater system pump train are indpendent of alternating current power. This independence extends to the lube oil cooler which receives cooling water from the pump discharge (recirculation line). We have reviewed the applicant's response and conclude that the provisions available in the existing auxiliary feedwater system at the facility meet the requirements outlined in this recommendation and are, therefore, acceptable.

6. Recommendation GS-6 - The licensee should confirm flow path availability of an auxiliary feedwater system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implmented to require an operator to determine that the auxiliary feedwater system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary auxiliary feedwater system water source to the steam generators. The flow test should be conducted with auxiliary feedwater system valves in their normal alignment.

In a letter dated August 15, 1980, the applicant responded to this recommendation, stating that South Carolina Electric & Gas Company procedures require that the auxiliary feedwater system flow path be verified after it has been out of service for periodic testing or maintenance.

The applicant has further committed to include initial determination and second (independent) operator verification of proper valve alignment in plant procedures as required by the first part of this recommendation. We conclude that the first part of this recommendation is satisfied pending formal documentation of this commitment, and pending verification of the plant procedures by the Office of Inspection and Enforcement.

In addition, we note that the auxiliary feedwater system is used to supply feedwater to the steam generators during normal plant startup, shutdown, and layup operations. Therefore, the availability of an auxiliary feedwater flow path from the primary water source (condensate storage tank) to the steam generators is automatically verified for that flow path during normal plant startup. We, therefore, conclude that the second part of this recommendation is satisfied.

7. Recommendation GS-7 - The licensee should verify that the automatic start auxiliary feedwater system signals and associated circuitry are safety grade. If this cannot be verified, the auxiliary system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

(1) The design should provide for the automatic initiation of the auxiliary feedwater system flow.

(2) The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

(3) Testability of the initiation signals and circuits shall be a feature of the design.

(4) The initiation signals and circuits should be powered from the emergency buses.

(5) Manual capability initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

(7) The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the auxiliary feedwater system from the control room.

In response to this recommendation, the applicant stated in a letter dated August 15, 1980, that the auxiliary feedwater system is designed so that automatic initiation signals and circuits are redundant and meet safety-grade requirements. Our evaluation is provided in Section 7 of this Safety Evaluation Report.

Recommendation GS-8 - The licensee should install a system to automatically initiate auxiliary feedwater system flow. This system need not be safety-grade; however, in the short-term, it should meet the criteria listed below, which are similar to Item 2.1.7.a of NUREG-0578. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements, as indicated in Recommendation GL-2.

(1) The design should provide for the automatic initiation of the auxiliary feedwater system flow.

(2) The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

(3) Testability of the initiating signals and circuits should be a feature of the design.

(4) The initiating signals and circuits should be powered from the emergency buses.

(5) Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The alternating current powered motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

(7) The automatic initiation signals and circuits should be designed so that their failure will not result in the loss of manual capability to initiate the auxiliary feedwater system from the control room.

The present design provides for automatic initiation of auxiliary feedwater system flow. Recommendation GS-7 verifies automatic initiation of this system. Therefore, we conclude that this recommendation is not applicable to the facility.

B. Additional Short-Term Recommendations

1. Recommendation - The licensee should provide redundant level indication and low level alarms in the control room for the auxiliary feedwater system primary water supply, to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity auxiliary feedwater system pump is operating. We have reviewed the applicant's response to this recommendation and conclude that it is acceptable.

2. Recommendation (This recommendation has been revised from the original recommendation in NUREG-0611) - The licensee should perform a 48-hour endurance test on all auxiliary feedwater system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 48-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

In a letter dated November 5, 1980, the applicant provided information concerning the auxiliary feedwater pump endurance tests in response to our request for information. The applicant ran both motor-driven pumps for 73 hours continuously. The turbine-driven pump was run for approximately 37½ hours. It was then restarted and run for an additional 35½ hours. Bearing/bearing oil temperatures were monitored on all pumps for the duration of the test and temperature design limits were not exceeded. Pump room ambient conditions were within acceptable limits during the test. Pump room vibration limits were not exceeded, however, because of slightly rough horizontal readings on the "A" pump motor, additional bracing will be added to both the "A" and "B" pump motors. In addition, a new thrust bearing will be installed on the "B" pump to correct slightly rough vibration readings. Upon completion of these modifications, new vibration measurements will be taken to assure that vibration conditions have been improved. Based on the above, we conclude that the applicant's response is acceptable.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

   In a letter dated August 15, 1980, the applicant responded to this recommendation by stating that the auxiliary feedwater system design includes safety-grade, redundant indication of auxiliary feedwater flow to each steam generator in the control room. The auxiliary feedwater flow instrument channels are powered from the emergency buses. We conclude that this is acceptable and our evaluation is provided in Section 7 of this Safety Evaluation Report.

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on auxiliary feedwater system train, and there is only one remaining auxiliary feedwater system train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the auxiliary feedwater system train from the test mode to their operational alignment.

   In response to this recommendation, the applicant by letter dated August 15, 1980, indicated that the facility does not require the realignment of local manual valves to conduct periodic tests on one auxiliary feedwater system. The auxiliary control valves may be operated from the control room to isolate the auxiliary feedwater pumps for periodic testing. In addition, there are three auxiliary feedwater trains available. We conclude that this recommendation is not applicable to the facility.

C. Long-Term Recommendations

1. Recommendation GL-1 - For plants with a manual starting system, the licensee should install a system to automatically initiate the auxiliary feedwater system flow. This system and associated automatic initiation signals should be designed and installed to meet safety-grade requirements. Manual auxiliary feedwater system start and control capability should be retained with manual start serving as backup to automatic auxiliary system initiation.

   Because the applicant's response to Recommendation GS-7 states that the auxiliary feedwater system design already includes automatic start, Recommendation GL-1 is not applicable.

2. Recommendation GL-2 - Licensees with plant designs in which all (primary and alternate) water supplies to the auxiliary feedwater systems pass through valves in a single flow path should install redundant paralled flow paths (piping and valves).

   Licensees with plants in which the primary auxiliary feedwater system water supply passes through valves in a songle flow path, but the alternate auxiliary feedwater system water supplies connect to the auxiliary feedwater system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

   In response to this recommendation, the applicant indicated, in a letter dated August 15, 1980, that the auxiliary feedwater system alternate water supply (service water system) connects to the auxiliary feedwater system pump suction piping downstream of the single normally locked-open valve in a single flow path from the primary water source. Automatic opening of the motor-operated valves from the alternate water supply upon low pump suction pressure will be provided. However, the applicant at a subsequent meeting expressed concern that this automatic opening of valves to the alternate source may not occur in time to prevent auxiliary feedwater pump damage should the common suction valve be inadvertently left closed. We notified the applicant that we would require a test of this plant feature to assure that pump damage would not result prior to affecting the transfer source (service water system) in the event that the common suction supply valve from the primary water source (condensate storage tank) was left closed. In lieu of performing this test, the applicant has committed in a letter dated December ? 1980, to provide a manually-operated valve in parallel with the existing valve from the primary water source. Control room indication (audible alarm) will be provided for this valve which will annunciate when it is not in the full open position similar to that provided for the existing valve. The surveillance requirements of the Technical Specifications will incorporate monthly periodic inspections of this valve similar to those for the existing valve. In addition, the new valve will be locked open and will have its hand wheel removed to further protect against its inadvertent closure. These features will also be implemented on the existing valve. Based on the above commitments, we conclude that the applicant's response is acceptable.

3. Recommendation GL-3 - At least one auxiliary feedwater system pump and its associated flow path and essential instrumentation should automatically initiate auxiliary feedwater system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion or direct current power to alternating current power is acceptable.

   In response to this recommendation, the applicant indicated in a letter dated August 15, 1980, that the turbine-driven auxiliary feedwater pump and its associated flow path and essential instrumentation automatically initiate auxiliary feedwater system flow and is capable of being operating independent of any alternating current power source for at least two hours. We have reviewed this response, and confirm that the turbine-driven auxiliary feedwater pump train is available to supply emergency feedwater independent of onsite or offsite alternating current power supplies. Based on our review, we conclude that the applicant's response is acceptable.

4. Recommendation GL-4 - Licensees having plants with unprotected normal auxiliary feedwater system supplies should evaluate the design of their auxiliary feedwater systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure, or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

   As indicated in a letter dated August 15, 1980, the applicant will provide automatic switchover of auxiliary feedwater pump suction supply to the alternate source (service water) on low suction pressure as would result upon failure of the primary water source (condensate storage tank) due to tornado missiles.

   We note that the primary water supply to the auxiliary feedwater system pumps, the condensate storage tank, is designed to satisfy seismic Category I requirements but is not protected against the effects of tornado missiles. However, its loss as a result of tornado missiles does not affect auxiliary feedwater system function since the tornado-missiles protected alternate source (service water system) serves as a sufficient backup supply of emergency feedwater. As noted under recommendation GL-2, we stated that we would require that the applicant test the automatic suction supply transfer feature to assure that auxiliary feedwater pump damage will not result prior to affecting the switchover.

In lieu of performing this test, the applicant submitted a calculation in a letter dated December 2, 1980 which indicates that 37 seconds of suction supply is available to the auxiliary feedwater system pumps assuming failure of the exposed condensate storage tank and suction line at grade due to tornado missiles. Transfer to the service water system supply can be accomplished in 20 seconds thereby assuring adequate suction supply prior to pump damage. Based on the above, we conclude that the applicant's response is acceptable.

5. Recommendation GL-5 - The licensee should upgrade the auxiliary feedwater system automatic initiation signals and circuits to meet safety-grade requirements.

   In response to this recommendation, the applicant indicated in a letter dated August 15, 1980, that the present auxiliary feedwater system automatic initiation signals and circuits are safety grade. We conclude that this is acceptable and our evaluation is reported in Section 7 of this Safety Evaluation Report.

II.E.3.1 Emergency Power for Pressurizer Heaters

Requirement

Install the capability to supply from emergency power buses a sufficient number of pressurizer heaters and associated controls to establish and maintain natural circulation in hot standby conditions.

This requirement shall be met before issuance of a full-power license. See NUREG-0578, Section 2.1.1 (Ref. 4), and letters of September 27 (Ref. 23) an November 9, 1979 (Ref. 24).

Position

Consistent with satisfying the requirements of Criteria 10, 14, 15, 17, and 20 of the General Design Criteria for the loss of offsite power event, the following positions shall be implemented:

(1) The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation during hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

(2) Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

(3) The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

(4) Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Clarification

(1) Redundant heater capacity must be provided, and each redundant heater or group of heaters should have access to only one Class 1E division power supply.

(2) The number of heaters required to have access to each emergency power source is that number required to maintain natural circulation in the hot standby condition.

(3) The power sources need not necessarily have the capacity to provide power to the heaters concurrently with the loads required for loss-of-coolant accident.

(4) Any changeover of the heaters from the normal offsite power to emergency onsite power is to be accomplished manually in the control room.

(5) In establishing procedure to manually load the pressurizer heaters onto the emergency power sources, careful consideration must be given to:

   (a) which engineered safety feature loads may be appropriately shed for a given situation;

   (b) reset of the safety injection actuation signal to permit the operation of the heaters; and

   (c) instrumentation and criteria for operator use to prevent overloading a diesel generator.

(6) The Class 1E interfaces for main power and control power are to be protected by safety-grade circuit breakers (see also Regulatory Guide 1.75).

(7) Being non-Class 1E loads, the pressurizer heaters must be automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal (see item 5.b above).

Discussion and Conclusions

The applicant has determined based on Westinghouse calculations, that the total heat loss from the primary coolant system, under hot standby conditions, is 100 kilowatts. On this basis, they have determined that a minimum of 100 kilowatts of pressurizer heaters should be available from an assured power source within one hour after loss of offsite power to establish and maintain natural circulation at hot standby conditions. The facility design provides two backup heater groups each rated at 37 kilowatts. Each group of heaters consists of 10 banks of 53.7 kilowatt heaters. Each backup heater group has an individual 480 volt switchgear. Each 480 volt switchgear is powered from a separate 480 volt bus and a separate independent diesel generator upon loss of offsite power.

The pressurizer heaters are automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal. The design provides the capability to connect the heaters to the emergency buses provided the load is available from the diesel generator. Procedures for manually loading the pressurizer heaters on to the emergency power sources following a safety injection actuation signal or a loss of offsite power are available to the operator. Manual connection of the pressurizer heaters is accomplished by closing the modeled case circuit breaker in the pressurizer heater 480 volt switchgear while the diesel operator load is observed from metering on the main control board. The Class 1E interfaces for main power and control power are protected by safety-grade circuit breakers.

We conclude that the emergency power supply requirements for pressurizer heaters isconsistent with our position and is acceptable. Verification of the adequacy of the applicant's procedures will be confirmed by the Office of Inspection and Enforcement.

We conclude that the emergency supply requirements for pressurizer heaters is consistent with our position and is acceptable. Verification of the adequacy of applicant's procedures will be performed by the NRC's Office of Inspection and Enforcement.

II.E.4.2 Containment Isolation Dependability

Requirement

Provide (1) containment isolation on diverse signals, such as containment pressure or ECCS actuation, (2) automatic isolation of nonessential systems (including the bases for specifying the nonessential systems), (3) no automatic reopening of containment isolation valves when the isolation signal is reset.

These requirements shall be met before issuance of a full-power license. See NUREG-0578, Section 2.1.4, and letters of September 27 and November 9, 1979.

Position

1. All containment isolation system designs shall comply with the recommendations of Section 6.2.4 of the Standard Review Plan; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

II.E.4.2 Containment Isolation Dependability

Requirement

Provide (1) containment isolation on diverse signals, such as containment pressure or emergency core cooling system actuation, (2) automatic isolation of nonessential systems (including the bases for specifying the nonessential systems), (3) no automatic reopening of containment isolation valves when the isolation signal is reset.

These requirements shall be met before issuance of a full-power license. See NUREG-0578, Section 2.1.4, and letters of September 27 and November 9, 1979.

Position

1. All conainment isolation system designs shall comply with the recommendations of Section 6.2.4 of the Standard Review Plan; i.e., that ther be diversity in the parameters sensed for the initiation of containment isolation.

2. All licensees and applicants shall give careful reconsideration to the definition of essential and non- sential systems, shall identify system determined to be essential, shal, identify each system of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to NRC.

3. All non-essential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

Clarification

1. Provide diverse containment isolation signals that satisfy safety-grade requirements.

2. Identify essential and non-essential systems and provide results to NRC.

3. Resetting of containment isolation signals shall not resut in the automatic loss of containment isolation.

Discussion and Conclusions

The containment isolation system is designed to automatically isolate the containment atmosphere from the outside environment under accident conditions.

Double barrier protection, in the form of closed systems and isolation valves, is provided to assure that no single failure will result in the loss of containment integrity.

The applicant has categorized all systems penetrating containment as being either essential or non-essential. All non-essential systems having automatic containment isolation valves, and not required for an orderly reactor shutdown or to maintain containment atmospheric conditions, are closed by a Phase A containment isolation signal. The operator will have the option of manually resetting the actuation signal and taking deliberate action to open the isolation valves of certain non-essential systems if loss-of-coolant accident conditions warrant their use.

Our review of the containment isolation system includes verification that there is diversity of parameters sensed for the initiation of containment isolation, as called for by Section 6.2.4 of the Standard Review Plan. The Phase A containment isolation system design meets their requirement. The parameters sensed for the intiation of containment isolation include high containment pressure, high differential pressure between main steam lines, pressurizer low pressure and low steam line pressure.

All containment isolation valves in non-essential systems that were originally designed to close upon receipt of an automatic isolation signal meet the Lessons Learned Task Force position on diversity. The diverse safety injection signal which is derived from the sensing of diverse parameters is provided for these valves, with the exception of the main steam isolation valves. However, diverse parameters are also sensed to initiate main steam isolation valve closure.

The design of Virgil C. Summer Nuclear Station, Unit 1 precludes automatic reopening of containment isolation valves upon reset of the isolation signal.

Since the plant meets all the requirements of TMI Action Plan Item II.E.4.2, "Containment Isolation Dependability," we conclude the isolation dependability of the containment is acceptable and that all requirements for containment isolation dependability have been satisfied.

II.K.2 Commission Orders on Babcock & Wilcox Plants

Requirements

C.2.2* For B&W-designed reactors, provide procedures and training to initiate and control auxiliary feedwater independent of the integrated control system.

C.2.9 For B&W-designed reactors, provide a failure modes and effects analysis of the integrated control system. See Commission Shutdown Order.

C.2.10 For B&W-designed reactors, install safety-grade anticipatory reactor trip for loss of feedwater and turbine trip. See Commission Shutdown Order.

*Table C.2 of the Action Plan lists all of the requirements of the Commission Orders.

C.2.13 For B&W-designed reactors, confirm by a detailed analysis of thermal-mechanical conditions in the reactor vessel during recovery from a small-break LOCA, with an extended loss of all feedwater requiring the use of the high-pressure injection system, that vessel integrity is not jeopardized. See letter of August 21, 1979.

C.2.14 For B&W-designed reactors, demonstrate that the power-operator relief valves on the pressurizer will open in less than five percent of all anticipated overpressure transients using revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle. See letter of August 21, 1979.

C.2.15 For B&W-designed reactors, analyze the effects of slug flow on once-through steam generator tubes after primary system voiding. See letter of August 21, 1979.

C.2.16 For B&W-designed reactors, evaluate the effects of reactor coolant pump damage and leakage following a small-break loss of coolant accident concurrent with a loss of offsite power that results in the loss of seal cooling. See letter of August 21, 1979.

These requirements shall be met before issuance of a full-power license.

Discussion and Conclusions

These requirements are for B&W-designed reactors and are therefore not applicable to this facility since it employs a Westinghouse designed reactor.

II.K.3 Final Recommendations of the Bulletins & Orders Task Force

The following requirement shall be met before issuance of a full power license.

C.3.3 Requirement

Assure that any failure of a power operated relief valve or safety valve to close will be reported to the NRC promptly. All challenges to the power operated relief valves or safety valves should be documented in the annual report.

Discussion and Conclusions

In Section 5.2.2.3 of the Final Safety Analysis Report (Amendment 22), the applicant committed that any failure of a power operated relief valve or safety valve to close will be reported to the NRC promptly and all challenges to the power operated relief valves or safety valves will be documented in the annual report. We find this acceptable.

III.A.1.1 Upgrade Emergency Preparedness

Requirement

Provide an emergency response plan in substantial compliance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (which may be modified as a result of public comments solicited in early 1980) except that only a description of and completion schedule for the means for providing prompt notification to the population, the staffing for emergencies in addition to that already required, and an upgraded meteorological program need be provided. NRC will give substantial weight findings on offsite plans in judging the adequacy against NUREG-0654. Perform an emergency response exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations.

This requirements shall be met before issuance of a full-power license.

Discussion and Conclusions

Refer to Section 13.3 of this Safety Evaluation Report.

III.D.1.1 Primary Coolant Sources Outside Containment

Requirement

Reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels, measure actual leak rate and establish a program to maintain leakage at as-low-as-practial levels and monitor leak rates.

This requirement shall be met before issuance of a full-power license. See NUREG-0578, Section 2.1.6a, and letters of September 27 and November 9, 1979.

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Clarification

Applicants shall provide a summary description, together with initial leak-test results, of their program to reduce leakage from systems outside containment that would or could contain primary coolant or other highly radioactive fluids or gases during or following a serious transient or accident.

(1) Systems that should be leak tested are as follows (any other plant system which has similar functions or post-accident characteristics even though not specified herein, should be included):

Residual heat removal
Containment spray recirculation
High-pressure injection recirculation
Containment and primary coolant sampling
Reactor core isolation cooling
Makeup and letdown
Waste gas (includes headers and cover gas system outside of containment in addition to decay or storage system).

Include a list of systems containing radioactive materials which are excluded from program and provide justification for exclusion.

(2) Testing of gaseous systems should include helium leak detection or equivalent testing methods.

(3) Should consider program to reduce leakage potential release paths due to design and operator deficiencies as discussed in our letter to all operating nuclear power plants regarding North Anna and related incidents, dated October 17, 1979.

(4) This requirement shall be implemented by applicants for an operating license prior to issuance of a full-power license.

Discussion and Conclusions

In Amendments 18 through 22 and letters dated August 28, 1980 and November 6, 1980, the applicant has provided a description of the leak reduction measures and the inspection program for monitoring and minimizing the leakage from the systems outside the containment that would or could contain highly radioactive fluids during serious transient or accident conditions. The applicant has provided a list of the systems that will be leak tested and has committed to provide the initial leak test results to us prior to fuel load.

The staff has reviewed the proposed leak reduction and inspection program for monitoring and minimizing the leakage and finds that the program can meet the requirements of Item III.D.1.1.

III.D.3.4 Control Room Habitability

Requirement

Identify and evaluate potential hazards in the vicinity of the site as described in Sections 2.2.1, 2.2.2, and 2.2.3 of the Standard Review Plan, confirm that operators in the control room are adequately protected from these hazards and the release of radioactive gases as described in Section 6.4 of the Standard Review Plan, and, if necessary, provide the schedule for modifications to achieve compliance with Section 6.4 of the Standard Review Plan.

This requirement shall be met prior to issuance of a full-power license.

Position

Licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19 of the General Design Criteria).

Clarification

1. All licensees must make a submittal to the NRC regardless of whether or not they met the criteria of the referenced Standard Review Plan sections. The new clarification specifies that licensees that meet the criteria of the Standard Review Plan should provide the basis for their conclusion that the requirements of Section 6.4 of the Standard Review Plan are met. Licensees may establish this basis by referencing past submittals to the NRC and/or providing new or additional information to supplement past submittals.

2. All licensees with control rooms that meet the criteria of Sections 2.2.1, 2.2.2, 2.2.3, and 6.4 of the Standard Review Plan shall report their findings regarding the specific Standard Review Plan sections as discussed below. The following documents should be used for guidance:

   a. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release";

   b. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accident Chlorine Release"; and,

   c. K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

Licensees shall submit the results of their findings as well as the basis for those findings by January 1, 1981. In providing the basis for the habitability finding, licensees may reference their past submittals. Licensees should, however, ensure that these submittals reflect the current facility design and that the information requested in Table 22-4 is provided.

TABLE 22-4

INFORMATION REQUIRED FOR CONTROL ROOM HABITABILITY EVALUATION

1. Control-room mode of operation, i.e., pressurizaton and filter recirculation for radiological accident isolation or chlorine release.

2. Control-room characteristics:

   a. air volume control room

   b. control-room emergency zone (control room, critical files, kitchen, washroom, computer room, etc.)

   c. control-room ventilation system schematic with normal and emergency air-flow rates

   d. infiltration leakage rate

   e. high efficiency particulate air filter and charcoal adsorber efficiencies

   f. closest distance between containment and air intake

   g. layout of control room, air intakes, containment building, and chlorine, or other chemical storage facility with dimensions

   h. control-room shielding including radiation streaming from penetrations, doors, ducts, stairways, etc.

   i. automatic isolation capability-damper closing time, damper leakage and area

   j. chlorine detectors or toxic gas (local or remote)

   k. self-contained breathing apparatus availability (number)

   l. bottled air supply (hours supply)

   m. emergency food and potable water supply (how many days and how many people)

   n. control-room personnel capacity (normal and emergency)

   o. potassium iodide drug supply

TABLE 22-4 (Continued)

3. Onsite storage of chlorine and other hazardous chemicals:

    a. total amount and size of container

    b. closest distance from control-room air intake

4. Offsite manufacturing, storage, or transportation facilities of hazardous chemicals:

    a. identify facilities within a five-mile radius;

    b. distance from control room;

    c. quantity of hazardous chemicals in one container;

    d. frequency of hazardous chemical transportation traffic (truck, rail, and barge)

5. Technical Specifications (refer to Standard Technical Specifications)

    a. chlorine detection system

    b. control-room emergency filtration system including the capability to maintain the control-room pressurization at 1/8-inch water gauge, verification of isolation by test signals and damper closure times, and filter testing requirements.

3. All licensees with control rooms that do not meet the criteria of the above-listed references, Standard Review Plan Sections, Regulatory Guides, and other references shall perform the necessary evaluations and identify appropriate modifications.

Each licensee submittal shall include the results of the analyses of control room concentrations from postulated accidental release of toxic gases nd control room operator radiation exposures from airborne radioactive material and direct radiation resulting from design basis accidents. The toxic gas accident analysis should be performed for all potential hazardous chemical releases occurring either on the site or within five miles of the plant site boundary. Regulatory Guide 1.78 lists the chemicals most commonly encountered in the evaluation of the control room habitability but is not all inclusive.

The design basis accident radiation source term should be for the loss-of-coolant accident containment leakage and engineered safety feature leakage contribution outside containment as described in Appendices A and B to Section 15.6.5 of the Standard Review Plan. In addition, boiling-water reactor facility evaluations should add any leakage from the main steam isolation valves (i.e., valve steam leakage, valve seat leakage, main steam isolation valve leakage control system release) to the containment leakage and engineered safetyfeature leakage following a loss of coolant accident. This should not be construed as altering the staff recommendations in Section D of Regulatory Guide 1.96 (Rev. 2) regarding main steam isolation valve leakage-control systems. Other design basis accidents should be reviewed to determine whether they might constitute a more severe control room hazard than the loss of coolant accident.

In addition to the accident analysis results, which should either identify the possible need for control room modifications to provide assurance that the habitability systems will operate under all postulated conditions to permit the control room operators to remain in the control room to take appropriate actions required by Criterion 19, the licensee should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems. Table 22-4 lists the informiation that should be provided along with the licensee's evaluation.

Applicability

This requirement applies to all operating reactors and operating licensees applicants.

Discussion and Conclusions

The staff has advised the applicant of the full power requirements for control room habitability as set forth in NUREG-0660 (May 1980), "NRC Action Plan Developed as a Result of TMI-2 Accident" and in NUREG-0694 (June 1980), "TMI-Related Requirements for New Operating Licenses."

On the basis of our review of the information presented in the Final Safety Aaalysis Report, we concluded in Section 6.4 of the Safety Evaluation Report that the control room of the facility meets the habitability requirements of Criterion 19 and the guidelines of Regulatory Guides 1.78 and 1.95. By letter dated November 25, 1980 the staff was notified by the applicant that it had independently reviewed the control room habitability systems guidance provided in Sections 2.2.1, 2.2.2, 2.2.3 and 6.4 of the Standard Review Plan, and Regulatory Guides 1.78 and 1.95, and concluded that the design of the control room is such as to assure that operators in the control room will be adequately protected against exposure to unacceptable levels of radiation during and after a design basis accident and to unacceptable levels of hazardous chemicals released on or in the vicinity of the site. The applicant concluded that no design modifications are necessary.

This conclusion is consistent with the staff's findings as stated above. We conclude that the applicant has satisfied the requirements of NUREG-0694 for a full power license with respect to control room habitability systems.

22.4 NRC Actions

I.B.2.2 Reactor Inspector At Operating Reactors

Requirement

An NRC resident inspector will be assigned to each site.

This action shall be completed before fuel loading.

Position

1. The Office of Inspection and Enforcement (IE) will implement the approved resident inspector program by recruiting, training, and assigning the resident inspectors to provide a minimum of two resident inspectors at each site where there are one or two reactors.

2. IE will place a senior resident insepctor at near-term operating plants by June 1980.

Discussion and Conclusion

An NRC resident inspector has been assigned to Virgil C. Summer Nuclear Station, Unit 1. This action is complete.

I.D.1 Control Room Design Review

Requirement

NRC review of applicant's preliminary assessment of the control room design to determine whether the assessment is adequate and identify any necessary corrections and approve the schedule for correction of the deficiencies.

This action shall be completed prior to fuel loading.

Discussion and Conclusions

See discussion of item I.D.1 in Section 22.2 of this Safety Evaluation Report.

II.B.7 Analysis of Hydrogen Control

Requirement

Reach a decision on the immediate requirements, if any, for hydrogen control in small containments and apply, as appropriate, to new operating licenses pending completion of the degraded core rulemaking in II.B.8 of the Action Plan.

This action shall be completed before issuance of a full-power license.

Discussion and Conclusion

The accident at Three Mile Island, Unit 2 resulted in a severely damaged core accompanied by the generation and release to containment of hydrogen in excess of those limits allowed in current regulations. This accident highlighted the difficulties associated with mitigated the consequences of an accident more severe than the current design basis accidents. As a consequence, item II.B.8 of the TMI Action Plan (NUREG-0660), calls for a rulemaking proceeding on consideration of degraded or melted cores in safety reviews to solicit comments. Additionally, the TMI Action Plan at Item II.B.7 discusses analysis of hydrogen control and the need for inerting small containments.

The staff action on item II.B.7 was completed with issuance of the Commission papers (SECY 80-107, -80-107A and -80-107B) which discussed the technical basis for: 1) the staff position on interim hydrogen control requirements (inerting) for small containments; and 2) continued operation and licensing of nuclear power plants pending the rulemaking proceeding. With regard to Virgil C. Summer Nuclear Station, Unit 1, which utilizes a dry type of containment, the staff position is that no additional hydrogen mitigating measures beyond the current design basis is needed, pending the rulemaking proceeding.

The first steps in the resolution of item II.B.8 will be the issuance of an advance notice of rulemaking and the issuance of interim rule. The advance notice has been transmitted to the Commission in SECY 80-357, Degraded Cooling Rulemaking. A proposed interim rule has also been prepared and was transmitted to the Commission on August 25, 1980 in SECY 80-399. Proposed Interim Amendment to 10 CFR Part 50, "Relating to Hydrogen Control and Certain Degraded Core Considerations," was published in the Federal Register on October 2, 1980. The proposed Interim Rule, in summary, addresses the following areas:

1. Requires inerting of all boiling water reactor Mark I and Mark II containments.

2. Requires all other plants to evaluate the effects of large amounts of hydrogen generation and to propose and assess mitigation techniques for control of hydrogen.

3. Codifies various Lessons Learned items to reduce the likelihood of degraded core accidents.

In addition to the efects related to the rulemaking, the staff has requested that a research program be initiated to investigate the effects of degraded/melted core accidents for generic light water reactor plant designs, and to investigate various safety systems to reduce the effects of such accidents. Additionally, the staff will seek assistance to evaluate the effectiveness of distributed ignition sources within containment on an expedited basis; i.e., within three months. The staff will, however, evaluate a spectrum of mitigation techniques to control hydrogen and reduce the impact of severely degraded core accidents as part of the safety research program discussed above.

We estimate the end date of the rulemaking proceeding to be about 1983. However, the projected end date for all the internal NRC actions identified above is January 31, 1981. The proposed Interim Rule was published in the Federal Register on October 2, 1980.

II.B.8 Degraded Core - Rulemaking

Requirement

Issue an advance notice of rulemaking on requirements for design and other features for accidents involving severely damaged cores.

This action shall be completed before issuance of a full-power license.

Discussion and Conclusions

The accident at Three Mile Island, Unit 2 resulted in a severely damaged core accompanied by the generation and release to containment of hydrogen in excess of those amounts required to be considered in current regulations. This accident highlighted the difficulties associated with mitigating the consequences of an accident more severe than the current design basis accidents. As a consequence, item II.B.8 of the TMI Action Plan (NUREG-0660), calls for a rulemaking proceeding on consideration of degraded or melted cores in safety reviews to solicit comments.

The first steps in the resolution of item II.B.8 will be the issuance of an advance notice of rulemaking and the issuance of an Interim Rule. The advance notice has been drafted and is under staff review. The Interim Rule has also been prepared and is expected to be ready for Commission consideration in the near future. The Interim Rule, in summary, addresses the following areas:

1. Requires inerting of all boiling water reactor Mark I and Mark II containments;

2. Requires owners of all other plants to evaluate the effects of large amounts of hydrogen generation and to propose and assess mitigation techniques for control of hydrogen.

3. Codifies various lessons learned to reduce the likelihood of degraded core accidents.

In addition to the efforts related to the rulemaking, the staff has requested that a research program be initiated to investigate the effects of degraded/melted core accidents for generic light water reactor plant designs, and to investigate various safety systems to reduce the effects of such accidents. As a part of this safety research, we have identified the evaluation of hydrogen control of ice condenser and boiling water reactor Mark III containments as a priority item. Additionally, the staff will seek assistance to evaluate the effectiveness of distributed ignition sources within containment on an expedited basis; i.e., within about three months. The use of ignitors within containment is currently regarded as the most promising short term hydrogen control device which could be adapted to current plant designs. The staff will, however, evaluate a spectrum of mitigation techniques to control hydrogen and reduce the impact of severely degraded core accidents as part of the safety research program discussed above.

III.A.3.1 Role of NRC in Emergency Preparedness

Requirement

More explicitly define the role of the NRC in emergency situations involving NRC licenses.

Conclusion

This action was completed in a meeting between the NRC staff and the Commission on February 6, 1980.

III.A.3.3 Communications

Requirement

Install direct dedicated telephone lines between each plant and the NRC Operations Center.

This action shall be completed prior to fuel loading.

Position

Direct dedicated telephone lines shall be installed at the plant prior to fuel load.

Discussion and Conclusions

A direct dedicated telephone line has been installed between the NRC Operations Center and the NRC Resident Inspector's Office at the Virgil C. Summer Nuclear Station, Unit 1. Direct dedicated telephone lines will be installed between the NRC Operations Center and the control room and the technical support center. This will be completed prior to fuel loading.

III.B.2 Implementation of NRC and FEMA Responsibilities

Requirement

The applicant emergency plans shall meet the requirements of Appendix E to 10 CFR Part 50 and the positions in Regulatory Guide 1.101 (March 1977). Offsite plans shall meet the essential planning elements in NUREG-75/111 and supplement 1 thereto or receive a favorable finding by the Federal Emergency Management Agency.

This requirement shall be met prior to fuel loading.

Discussion and Conclusion

Refer to Section 13.3 of this Safety Evaluation Report.

III.D.2.4 Offsite Dose Measurements

Requirement

The NRC will place approximately 50 thermoluminescent dosimeters around the site in coordination with the applicant and State environmental monitoring program.

This action shall be completed prior to issuance of a full-power license.

Position

The Office of Inspection and Enforcement will place 50 thermoluminescent dosimeters around each site in coordination with States and utilities. During normal operation, IE quarterly reports from these dosimeters will be provided to NRC, State, and Federal organizations. In the event of an accident, the dosimeters can be read at a frequency appropriate to the needs of the situation.

Discussion and Conclusions

IE states that 40 thermoluminescent dosmeters have been placed around the plant site. A program has been established with the state to collect the thermoluminescent dosimeters quarterly and send them to NRC for processing.

IV.F.1 Power-Ascension Test

Requirement

The Office of Inspection and Enforcement will monitor the power-ascension test program to confirm that safety is not compromised because of the expanded startup test program and economic costs of the delay in commercial operation.

This action shall be taken during the startup and power-ascension test program.

Position

The Office of Inspection and Enforcement should increase scrutiny of the power ascension test program to prevent any compromising of safety in view of the proposed expansion of startup test programs and the economic incentives to achieve the already delayed commercial operation of new plants.

Discussion and Conclusions

IE will monitor the power-ascension test program.

22.5 Dated Requirements

I.A.1.1 Shift Technical Advisor

Requirement

The Shift Technical Advisor shall have a technical education, which is taught at the college level and is equivalent to about 60 semester hours in basic subjects of engineering and science, and specific training in the design, function, arrangement and operation of plant systems and in the expected response of the plant and instruments to normal operation, transients and accidents including multiple failures of equipment and operator errors.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.2.1b, and letters of September 27 and November 9, 1979.)

Discussion and Conclusions

Refer to our discussion of Item I.A.1.1 in Section 22.2 of this Safety Evaluation Report.

I.A.2.1 Immediate Upgrading of Operator and Senior Operator Training and Qualification

Requirement

Applicants for senior reactor operator licenses shall have four years of responsible power plant experience, of which at least two years shall be nuclear power plant experience (including six months at the specific plant) and no more than two years shall be academic or related technical training.

Certifications that operator license applicants have learned to operate the controls shall be signed by the highest level of corporate management for plant operation. These requirements shall be met on or after May 1, 1980. (See March 28, 1980 letter.)

Revise training programs to include training in heat transfer, fluid flow, thermodynamics, and plant transients. This requirement shall be met by August 1, 1980. (See March 28, 1980 letter.)

Discussion and Conclusions

Each senior reactor operator license candidate at the Virgil C. Summer Nuclear Station, Unit 1 either meets or has equivalent background to the above requirement. Certification of applications will be reviewed and approved by the South Carolina Electric & Gas Company Vice President and Group Executive, Nuclear Operations.

The Applicant has submitted an outline of the revised training programs that include training in areas required by this action plan item.

We conclude that the applicant has satisfied the requirements of item I.A.2.1.

I.A.2.3 Administration of Training Programs for Licensed Operators

Requirement

Training instructors who teach systems, integrated responses, transient and simulator courses shall successfully complete a senior reactor operator examination.

Applications shall be submitted by August 1, 1980. (See March 28, 1980 letter.)

Instructors shall attend appropriate retraining programs that address, as a minimum, current operating history, problems and changes to procedures and administrative limitations. In the event an instructor is a licensed senior reactor operator, his retraining shall be the senior reactor operator requalification program.

Programs shall be initiated by May 1, 1980. (See March 28, 1980 letter.)

Discussion and Conclusions

The applicant intends to have all instructors teaching systems, receive a cold license senior reactor operator license. This intention is described in a letter to the staff dated July 29, 1980.

All operator license program instructors will be enrolled by the applicant in an appropriate requalification program to assure they are congnizant of current operator history, problems, and changes to procedures and administrative limitations. All senior reactor operator licensed instructors shall participate in the senior reactor operator requalification program. Based on the foregoing we have concluded that the applicant has complied with the requirements of item I.A.2.3.

I.A.3.1 Revise Scope and Criteria for Licensing Exams

Requirement

Applicants for operator licenses will be required to grant permission to the NRC to inform their facility management regarding the results of examinations.

Contents of the licensed operator requalification program shall be modified to include instruction in heat transfer fluid flow, thermodynamics, and mitigation of accidents involving a degraded core.

These requirements shall be met by May 1, 1980. (See March 28, 1980 letter.)

The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license.

This requirement shall apply to all annual requalification examinations conducted after March 28, 1980. (See March 28, 1980 letter.)

Requalification programs shall be modified to require specific reactivity control manipulations. Normal control manipulations, such as plant or reactor startups, must be performed. Control manipulations during abnormal or emergency operations shall be walked through and evaluated by a member of the training staff. An appropriate simulator may be used to satisfy the requirements for control manipulations.

This requirement shall be met by August 1, 1980. (See March 28, 1980 letter.)

Discussion and Conclusions

For our evaluation refer to item I.A.3.1 in Section 22.2 of this Safety Evaluation Report.

I.C.1 Short-Term Accident Analysis and Procedure Revision

Requirement

Analyze the design basis transients and accidents including single active failures and considering additional equipment failures and operator errors to identify appropriate and inappropriate operator actions. Based on these analyses, revise, as necessary, emergency procedures and training.

This requirement was intended to be completed in early 1980; however, some difficulty in completing this requirement has been experienced. Clarification of the scope and revision of the schedule are being developed and will be issued by July 1980. It is expected that this requirement will be coupled with Task I.C.9., Long-term Upgrading of Procedures. (See NUREG-0578, Sections 2.1.3b and 2.1.9, and letters of September 27 and November 9, 1979.)

Discussion and Conclusion

Refer to our evaluation of Item I.C.1 in Section 22.2 of this Safety Evaluation Report.

I.C.6 Procedures for Verifying Correct Peformance of Operating Activities

Requirement

This was a proposd requirement in NUREG-0660 and was formally issued by NUREG-0737. This requirement shall be met by January 1, 1981.

Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

Implementation of automatic status monitoring if required will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases--one before and one after installation of automatic status monitoring equipment, if required, in accordance with item I.D.3.

Clarification

Item I.C.6 of the U. S. Nuclear Regulatory Commission Task Action Plan (NUREG-0660) and Recommendation 5 of NUREG-0585 propose requiring that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided. An acceptable program for verification of operating activities is described below.

The American Nuclear Society has prepared a draft revision to ANSI Standard N18.7-1972 (ANS 3.2) "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." A second proposed revision to Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," which is to be issued for public comment in the near future, will endorse the latet draft revision to ANS 3.2 subject to the following supplemental provisions:

(1) Applicability of the guidance of Section 5.2.6 should be extended to cover surveillance testing in addition to maintenance.

(2) In lieu of any designated senior reactor operator the authority to release systems and equipment for maintenance or surveillance testing or return-to-service may be delgated to an on-shift senior reactor operator, provided provisions are made to ensure that the shift supervisor is kept fully informed of system status.

(3) Except in cases of significant radiation exposure, a second qualified person should verify correct implementation of equipment control measures such as tagging of equipment.

(4) Equipment control procedures should include assurance that control-room operators are informed of changes in equipment status and the effects of such changes.

(5) For the return-to-service of equipment important to safety, a second qualified operator should verify proper systems alignment unless functional testing can be performed without compromising plant safety, and can prove that all equipment, valves, and switches involved in the activity are correctly aligned.

Note: A licensed operator possessing knowledge of the systems involved and the relationship of the systems to plant safety would be a "qualified" person. The staff is investigating the level of qualification necessary for other operators to perform these functions.

For plants that have or will have automatic system status monitoring as discussed in Task Action Plan item I.D.3, NUREG-0660, the extent of human verification of operations and maintenance activities wll be reduced. However, the need for such verification will not be eliminated in all instances.

Discussion and Conclusions

The applicant has provided the necessary information to satisfy this requirement. The applicant will use a combination of administrative procedures and automatic status monitoring to assure that operating activities have been adequately verified. These procedures have been reviewed by the staff at the plant and comments were provided to the applicant. These and other comments generated within the applicant's organization will be used to update and correct the procedures upon the successful completion of this revision to the procedures, we find them accepable.

II.B.1 Reactor Coolant System Vents

Requirement

Install reactor coolant system and reactor vessel head high-point vents that are remotely operable from the control room.

This requirement shall be met before January 1, 1981. See letters of September 27 and November 9, 1979.

Discussion and Conclusions

Refer to our evaluation of Item II.B.1 in Section 22.3 of this Safety Evaluation Report.

II.B.2 Plant Shielding

Requirement

Complete modifications to assure adequate access to vital areas and protection of safety equipment following an accident resulting in a degraded core.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.1.6b, and letters of September 27 and November 9, 1979.)

Discussion and Conclusions

Refer to our evaluation of Item II.B.2 in Section 22.3 of this Safety Evaluation Report.

II.B.3 Post-Accident Sampling

Requirement

Complete corrective actions needed to provide the capability to promptly obtain and perform radioisotopic and chemical analysis of reactor coolant and containment atmosphere samples under degraded-core conditions without excessive exposure.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.1.8a and letters of September 27 and November 9, 1979.)

Discussion and Conclusions

This item is discussed in Section 22.3, Item II.B.3 of this Safety Evaluation Report. The applicant has committed to install a post-accident sampling and analysis system acceptable to us by January 1, 1981. We will perform a post-implementation review of the system and provide our completed evaluation in a supplement to this Safety Evaluation Report.

II.D.1 Relief and Safety Valve Test Requirements

Requirement

Complete tests to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

This requirement shall be met by July 1, 1981. (See NUREG-0578, Section 2.1.2 and letters of September 27 and November 9, 1979.)

Discussion and Conclusions

We will report the results of our review of this item in a supplement to this Safety Evaluation Report upon completion of testing.

II.E.1.2 Auxiliary Feedwater Initiation and Indication

Requirement

Upgrade, as necessary, automatic initiation of the auxiliary feedwater system and indication of auxiliary feedwater flow to each steam generator to safety-grade quality.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Sections 2.1.7a and b, and letters of September 27, and November 9, 1979.)

(1) Auxiliary Feedwater System Initiation

Position

As part of the Lessons Learned recommendation 2.1.7.a, the automatic initiation circuitry of the auxiliary feedwater system must be "upgraded in accordance with safety-grade requirements."

Clarification

The intent of this recommendation is to assure a reliable automatic initiation system. This objective can be met by providing a system which meets the requirements of IEEE Standard 279-1971.

Discussion and Conclusions

Our evaluation of the emergency feedwater system is presented in Section 7.4.1 of this Safety Evaluation Report. We conclude that the facility's emergency feedwater system automatic initiation satisfies the above positions and, therefore, is acceptable.

(2) Auxiliary Feedwater System Flowrate Indication

Position

As part of the Lessons Learned recommendation 2.1.7.b, the flowrate indication for the auxiliary feedwater system must be "upgraded in accordance with safety-grade requirements."

Clarification

The intent of this recommendation is to assure a reliable indication of auxiliary feedwater performance. This objective can be met by providing an overall indication system which meets appropriate design principles.

Discussion and Conclusions

Our evaluation of the emergency feedwater system is presented in Section 7.4.1 of this Safety Evaluation Report. We conclude that the facility's emergency system flow indication satisfies the above position and, therefore, is acceptable.

II.E.4.1 Containment Dedicated Penetrations

Requirement

Install a containment isolation system for external recombiners or purge systems for post-accident combustible gas control, if used, that is dedicated to that service only and meets the single-failure criterion.

This requirement shall be met before January 1, 1981. See NUREG-0578, Section 2.1.5a and c and letters of September 27 and November 9, 1979.

Position

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that satisfy the redundancy and single fialure requirements of Criterion 54 and 56 of the General Design Criteria and that are sized to satisfy the flow requirements of the recombiner or purge system.

Clarification

1. This requirement is only applicable to those plants whose licensing basis includes requirements for external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere.

2. An acceptable alternative to the dedicated penetration is a combined design that is single-failure proof for containment isolation purposes and single-failure proof for operation of the recombiner or purge system.

3. The dedicated penetration or the combined single-failure proof alternative should be sized such that the flow requirements for the use of the recombiner or purge system are satisfied.

4. Components necessitated by this requirement should be safety grade.

5. A description of required design changes and a schedule for accomplishing these changes should be provided by January 1, 1980. Design changes should be completed by January 1, 1981.

Discussion and Conclusions

Because of the internal recombiners at the facility, the requirement for dedicated penetrations for external recombiners is not applicable. The requirement for the installation of dedicated penetrations by January 1, 1981 is, therefore, not applicable to the facility.

II.F.1 Additional Accident Monitoring Instrumentation

Requirement

Install continuous indication in the control room of the following parameters:

a. Containment pressure from minus five pounds per square inch, gauge to three times the design pressure of concrete containments and four times the design pressure of steel containments;

b. Containment water level in pressurized water reactors from (1) the bottom to the top of the containment sump, and (2) the bottom of the containment to a level equivalent to 600,000 gallons of water;

   Containment water level in boiling water reactors from the bottom to five feet above the normal water level of the suppression pool;

c. Containment atmosphere hydrogen concentration from 0 to 10 volume percent;

d. Containment radiation up to 10^8 rad per hour;

e. Noble gas effluent from each potential release point from normal concentrations to 10^5 microcuries per cubic centimeter (Xe-133).

Provide capability to continuously sample and perform onsite analysis of the radionuclide and particulate effluent samples.

This instrumentation shall meet the qualification, redundancy, testability and other design requirements of the proposed revision to Regulatory Guide 1.97.

This requirement shall be met by January 1, 1981. See NUREG-0578, Section 2.1.8b, and letters of September 27 and November 9, 1979.

Discussion and Conclusions

Refer to discussion of Item II.F.1 in Section 22.2 of this Safety Evaluation Report.

II.F.2 Inadequate Core Cooling Instruments

Requirement

Install, if required, additional instruments or controls needed to supplement installed equipment in order to provide unambiguous, easy-to-interpret indication of inadequate core cooling.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.1.3b and letters of September 27 and November 9, 1979.)

Discussion and Conclusions

Refer to our evaluation of Item II.F.2 in Section 22.2 of this Safety Evaluation Report.

III.A.1.2 Upgrade Emergency Support Facilities

Requirement

Provide radiation monitoring and ventilation systems, including particulate and charcoal filters, and otherwise increase the radiation protection to the onsite technical support center to assure that personnel in the center will not receive doses in excess of five rem to the whole body or 30 rem to the thyroid for the duration of the accident. Provide direct display of plant safety system parameters and call up display of radiological parameters.

For the near-site emergency operations facility, provide shielding against direct radiation, ventilation isolation capability, dedicated communications with the onsite technical support center, and direct display of radiological and meteorological parameters.

This requirement shall be met by January 1, 1981, although the safety parameter information requirements will be staged over a longer period of time. (See NUREG-0578, Section 2.2.2b and 2.2.2c, and letters of September 27 and November 9, 1979, and April 25, 1980.)

Discussion and Conclusion

Refer to Section 13.3 of this Safety Evaluation Report.

III.D.3.3 In-Plant Radiation Monitoring

Requirement

Provide the equipment, training, and procedures to accurately measure the radioiodine concentration in areas within the plant where plant personnel may be present during an accident.

This requirement shall be met before January 1, 1981. (See NUREG-0578, Section 2.1.8c, and letters of September 27 and November 9, 1979.)

Discussion and Conclusion

In Amendment 21 to the Final Safety Analysis Report, the applicant has identified the low - background area for iodine analysis. Samples may be analyzed using the available equipment: (1) normal MCA/GeLi system located in the count room (control building elevation 412); (2) portable MCA/GeLi system which can be used at any location onsite; (3) MCA/GeLi system located at the environmental laboratory (located approximately two miles from plant); (4) portable single channel GeLi analyzer with sodium iodine detectorswhich can be used in the field at acceptable locations. The requirement background/or sensitivities for performing Iodine analysis described by the applicant meet our position in NUREG-0578 and NUREG-0660 and are, therefore, acceptable.

23 CONCLUSIONS

Based on our evaluation of the application as set forth above, it is our position that, upon favorable resolution of the outstanding matters described herein, we will be able to conclude that:

1. The application of facility license filed by the applicant dated December 10, 1976, as amended, complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1; and

2. Construction of the Virgil C. Summer Nuclear Station, Unit 1, has proceeded and there is reasonable assurance that it will be substantially completed, in conformity with Construction Permit No. CPPR-94, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

3. The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

4. There is reasonable assurance (a) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (b) that such activities will be conducted in compliance with regulations of the Commission set forth in 10 CFR Chapter 1; and

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

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

Before an operating license will be issued to the applicant for operation of the Virgil C. Summer Nuclear Station, units 1, the unit must be completed in conformity with the provisional construction permit, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power levels must be verified by the Commission's Office of Inspection and Enforcement prior to issuance of the licenses.

Further, before operating license is issued, the applicant will be required to satisfy the applicable provisions of 10 CFR Part 140.

APPENDIX A

CHRONOLOGY OF NRC STAFF RADIOLOGICAL SAFETY REVIEW

December 10, 1976	Letter from applicant forwarding for acceptance review the Final Safety Analysis Report Environmental Report and General Information and requesting extension of the construction permit
December 10, 1976	Letter from applicant forwarding security plan
December 14, 1976	Letter to applicant advising of receipt of tendered application
December 15, 1976	Letter to applicant requesting additional information in connection with request to extend construction permit completion date
December 17, 1976	Letter to applicant transmitting sample technical specifications for fire protection
January 14, 1976	Letter from applicant regarding Construction Permit completion date
February 2, 1977	Letter from applicant providing details regarding construction permit extension request
February 24, 1977	Letter to applicant advising that application acceptable for docketing
February 25, 1977	Letter to applicant concerni g new regulation on industrial security
February 25, 1977	Letter from applicant transmitting application for docketing
March 31, 1977	Letter to applicant transmitting Federal Register notice
April 1, 1977	Letter to Advisory Committee on Reactor Safeguards advising that no potential problem areas have been identified no potentially difficult novel features been identified
April 11,1977	Letter from applicant transmitting annual report for 1976
April 11, 1977	Letter to applicant concerning instrument trip setpoint values
April 12, 1977	Letter to applicant transmitting acceptance review question change

April 15, 1977	Amendment No. 1, including "State of South Carolina Peacetime Radiological Emergency Response Plan" and "South Carolina Hurricanes or a Descriptive Listing of Tropical Cyclones that have Affected South Carolina"
April 22, 1977	Letter to applicant concerning standard format for meteorological data on magnetic tape
May 4, 1977	Letter to applicant transmitting intrusion detection systems handbook
May 12, 1977	Letter from applicant regarding submittal of information concerning instrument trip setpoint values
May 24, 1977	Letter from applicant transmitting amended security plan
June 30, 1977	Letter for applicant transmitting Amendment No. 2 to the Final Safety Analysis Report
July 15, 1977	Letter from applicant transmitting revised and updated wiring and schematic package drawings
July 18, 1977	Letter to applicant transmitting "Procedure for Documentation of Deviations from the Standard Review Plan"
July 21, 1977	Letter to applicant concerning analysis of postulated main steam line break accident
July 26, 1977	Letter from applicant transmitting input for magnetic tape regarding meteorological data
August 19, 1977	Letter from applicant transmitting fire protection evaluation-fire hazards analysis report
August 19, 1977	Letter to applicant transmitting review schedule
August 29, 1977	Letter from applicant concerning electric penetrations
August 29, 1977	Letter to applicant regarding fire protection functional responsibilities
September 19, 1977	Letter to applicant transmitting notice of petition regarding physical searches and proposed regulation concerning clearances
October 18, 1977	Letter from applicant providing information regarding request for extension of construction permit completion date
October 25, 1977	Letter to applicant regarding physical security assessment models
October 31, 1977	Letter to applicant concerning diesel generator operating status indication

November 28, 1977	Letter to applicant advising of delay in implemenation of physical search requirement
December 5, 1977	Letter from applicant providing information on diesel generator operating status indication
January 9, 1978	Notice of Reconstitution of Board - I. W. Smith, Chairman
January 18, 1978	Letter to applicant regarding Operator Licensing Branch regional representative
January 25, 1978	Letter to applicant transmitting Security Plan Evaluation Report Workbook and preliminary schedule for review of plan
February 2, 1978	Letter to applicant concerning review of inservice testing program for pumps and valves
February 3, 1978	Atomic Safety and Licensing Board Memorandum and Order regarding petition from B. Bursey
February 8, 1978	Notice of Hearing published
February 21, 1978	Letter to applicant transmitting first round questions
March 6, 1978	Notice of Prehearing Conference - March 30, 1978
March 8, 1978	Letter to applicant transmitting balance of first-round questions
March 22, 1978	Atomic Safety and Licensing Board memorandum regarding call from B. Bursey to Chairman Hendrie
March 23, 1978	Atomic Safety and Licensing Board Memorandum and Order regarding petition of South Carolina Attorney General's participation in hearing as an interested state
March 30, 1978	Letter from applicant transmitting Amendment No. 3 to the Final Safety Analysis Report
April 5, 1978	Letter to applicant transmitting first round questions on quality assurance
April 10, 1978	Letter to applicant regarding safeguards meeting May 11-12, 1978 in Albuquerque, New Mexico to discuss "Implementation of 10 CFR 73.55 Requirements and Status of Research for Physical Protection of Licensed Activities in Nuclear Power Reactors against Sabotage"
April 14, 1978	Letter from applicant transmitting Amendment No. 4 to the Final Safety Analysis Report
April 19, 1978	Letter to applicant transmitting safeguards handbooks

April 21, 1978	Letter from applicant transmitting right-of-way agreement
April 21, 1978	Letter from applicant transmitting "Service Water Intake Structure Settlement Effects and Related Work"
April 21, 1978	Meeting with applicant to discuss applicant's approach to development of accident profile and attendent equipment qualification
April 21, 1978	Letter from applicant transmitting Amendment No. 5 to the Final Safety Analysis Report
April 21, 1978	Letter from applicant transmitting response to March 8, 1978 staff request for additional information
April 26, 1978	Letter from applicant transmitting 1977 Annual Financial Report in Accordance with Section 50.71(b) of 10 CFR Part 50
April 26, 1978	Letter from applicant transmitting radiological reports
April 26, 1978	Letter from applicant transmitting supplemental information to "Service Water Intake Structure Settlement Effects and Related Work"
May 4, 1978	Letter to applicant transmitting prehearing conference Memorandum and Order
May 5, 1978	Letter to applicant transmitting for comment, Draft 2 to NUREG-0219, "Nuclear Security Personnel for Power Plants, Review Plan and Acceptance Criteria for a Security Training Program"
May 19, 1978	Letter from applicant transmitting Amendment No. 6 to the Final Safety Analysis Report
June 8, 1978	Meeting with applicant to discuss second round questions and positions developed by Containment Systems Branch--Summary dated June 28, 1978
June 12, 1978	Letter to applicant transmitting Sandia Laboratory reports
June 19, 1978	Letter to applicant transmitting request for additional information concerning service water intake structure and pumphouse
July 5, 1978	Letter from applicant transmitting report, "Evaluation of Safety Related Equipment Temperature Transients During the Limiting Main Steam Line Break in Containment"
July 10, 1978	Letter to applicant transmitting second round questions and positions
July 10, 1978	Meeting with applicant to discuss design of essential service water system intake structure--Summary dated September 8, 1978

July 10, 1978	Letter from applicant transmitting "Report No. 2, Service Water Intake Structure Settlement Effects and Related Work"
July 12-14, 1978	Site visit
July 17, 1978	Letter from applicant transmitting Appendix "A" to report submitted July 10, 1978
July 18, 1978	Letter to applicant transmitting "Barrier Penetration Database" NUREG/CR-0181
July 25, 1978	Letter to applicant transmitting requests for additional information for fire protection review
July 31, 1978	Letter to applicant transmitting requests for additional information and staff postions
August 1, 1978	Letter to applicant transmitting draft Appendix I to 10 CFR Part 50 Technical Specifications
August 2, 1978	Letter to applicant transmitting NUREG-0219 regarding security training
August 3, 1978	Letter to applicant concerning fire brigade manpower requirements for operating reactors
August 7, 1978	Letter from applicant transmitting seismic qualification information
August 11, 1978	Letter to applicant concerning standard format for meteorological data on magnetic tape
August 15, 1978	Letter to applicant advising of pressurized water reactor steam generator workshop
August 25, 1978	Letter from applicant transmitting Amendment No. 7 to the Final Safety Analysis Report
August 28, 1978	Letter to applicant advising of meeting to discuss upgraded guard qualification and training requirements
August 29, 1978	Letter to applicant transmitting discussion of unresolved issues concerning industrial security
September 1, 1978	Meeting with applicant to discuss proposed emergency plan--Summary dated 9/20/78
September 1, 1978	Letter to applicant transmitting request for additional information concerning environmental qualification
September 1, 1978	Letter from applicant transmitting Amendment No. 8 to the Final Safety Analysis Report

September 11, 1978	Letter to applicant advising of revised meeting schedule concerning guard qualification
September 12, 1978	Letter to applicant transmitting requests for additional information on loose parts monitoring system, the thermal-hydraulic design, and the analysis methods for accidents and transients
September 18, 1978	Letter from applicant transmitting Amendment No. 9 to the Final Safety Analysis Report
September 18, 1978	Letter from applicant transmitting information concerning fire protection evaluation-fire hazards analysis report
September 25, 1978	Meeting with applicant to discuss information requested on environmental qualification of electric equipment--Summary dated October 16, 1978
October 26, 1978	Letter from applicant advising that fuel load date has changed from November 1979 to July 1, 1980
October 26, 1978	Letter from applicant providing information on service water intake structure settlement effects and related work (updates report submitted 7/10/78)
October 31, 1978	Letter from applicant transmitting Amendment No. 10 to the Final Safety Analysis Report
November 2, 1978	Letter from applicant providing information on seismic monitoring
November 9, 1978	Letter from applicant transmitting revised Physical Security Plan
November 16, 1978	Letter to applicant transmitting Revision 1 of Draft Radiological Effluent Technical Specifications and "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," NUREG-0133
November 20, 1978	Letter to applicant advising that Tradoc Pamphlet on security is now available at no charge
November 30, 1978	Letter from applicant transmitting Revision 2 to the Fire Protection Evaluation - Fire Hazards Analysis Report
December 1, 1978	Letter from Westinghouse transmitting affidavit for proprietary information
December 6, 1978	Letter from applicant transmitting magnetic tape of onsite meteorological data
December 7, 1978	Letter to applicant transmitting request for additional information on safety of bolted connections in linear component supports

December 27, 1978	Letter to applicant transmitting request for additional information on instrumentation and control systems and structural engineering
January 5, 1979	Letter from applicant advising of results of seismic monitoring following recent earthquake
January 16, 1979	Letter from applicant transmitting revision 4 to State Radiological Emergency Plan
January 23, 1979	Letter to applicant advising that information submitted November 2, 1978 regarding table relative to main steam line break accident information will be withheld from disclosure
January 23, 1979	Letter from applicant transmitting seismic monitoring data
January 30, 1979	Order extending construction completion date to December 31, 1980, Published 2/7/79, 44 FR 7849
January 31, 1979	Letter from applicant transmitting Amendment No. 11 to the Final Safety Analysis Report
February 6-7, 1979	Meeting with applicant to discuss outstanding issues-Summary dated March 21, 1979
February 9, 1979	Letter from applicant transmitting errata sheet for February 9, 1978 submittal
February 13, 1979	Letter from applicant transmitting guidance for preparation of offsite dose calculation manual
February 13, 1979	Letter from applicant transmitting update to "Report No. 2, Service Water Intake Structure Settlement Effects and Related Work," submitted July 19, 1978
February 15, 1979	Letter from applicant transmitting information on environmental qualification of equipment
February 15, 1979	Amendment No. 12
February 23, 1979	Letter to applicant transmitting requests for additional information
March 2, 1979	Letter to applicant transmitting "Summary of Operating Experience with Recirculating Steam Generators," NUREG-0523
March 15, 1979	Letter to applicant concerning anticipated transients without scram
March 21, 1979	Letter from applicant transmitting "July-September 1978 Seismicity Near the V. C. Summer Nuclear Station," Technical Report No. 79-4

March 22, 1979	Letter to applicant advising of June 1979 6/79 meeting to discuss implementation of physical security requirements
March 23, 1979	Letter to applicant transmitting "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactors Plants, "NUREG-0531
March 23, 1979	Letter to applicant transmitting requests for additional information
March 27, 1979	Meeting with applicant to discuss outstanding issues in instrumentation and control review--Summary dated April 27, 1979
April 18, 1979	Letter to applicant transmitting "Enhancement of Onsite Emergency Diesel Generator Reliability, "NUREG-0660
April 23, 1979	Meeting with applicant to discuss outstanding issues in instrumentation and control review
June 4, 1979	Letter from applicant transmitting Amendment No. 14 to the Final Safety Analysis Report
June 5, 1979	Meeting with applicant to discuss induced seismicity at the site
June 22, 1979	Meeting with applicant to discuss induced seismicity at the site
June 13, 1979	Meeting with utilities having construction permit and operating license applications to discuss (1) staff policies regarding the review of current construction permit and operating license applications and (2) the criteria for establishing priorities for the review of those applications--Summary dated July 2, 1979
June 13, 1979	Letter to applicant concerning instrument qualification
June 22, 1979	Meeting with applicant to discuss reservoir induced seismicity - Summary dated July 26, 1979
June 28, 1979	Letter from applicant transmitting safeguards contingency plan
June 29, 1979	Letter from applicant commenting on overall schedule and construction activities
July 10, 1979	Publication of notice of issuance of Draft Environmental Statement (44 FR 40460)
July 19, 1979	Letter to applicant transmitting security training and qualification review workbook

July 30, 1979	Letter to applicant transmitting request for updated financial information
July 27, 1979	Letter from applicant transmitting "Seismic Activity Near the V. C. Summer Nuclear Station - For the Period April-June, 1979," Technical Report No. 79-2
August 6, 1979	Atomic Safety and Licensing Board Memorandum and Order
August 17, 1979	Letter from applicant transmitting security training and qualification plan
August 21, 1979	Letter from applicant transmitting fuel handling building exfiltration analysis for a postulated fuel handling accident
August 24, 1979	Letter to applicant concerning secondary water chemistry control
August 27-29, 1979	Site review of facility for fire protection--Summary dated October 19, 1979
September 4, 1979	Letter from applicant transmitting qualification test documents for diesel generators
September 24, 1979	Letter from applicant transmitting information on financial qualifications
September 26, 1979	Letter from applicant transmitting Amendment No. 15 to the Final Safety Analysis Report
September 27, 1979	Letter to applicant concerning followup actions resulting from the accident at Three Mile Island
October 15, 1979	Letter from applicant transmitting Revision 3 to Fire Protection Evaluation-Fire Hazards Analysis Report
October 15-16, 1979	Site visit by Fuel Load Forecast Panel to assess the status of construction--Summary dated November 7, 1979
October 17, 1979	Letter to applicant concerning anticipated transients without scram
October 22, 1979	Letter to applicant transmitting request for additional information and staff requirements for alternate shutdown systems
October 25, 1979	Meeting with applicant to discuss seismic qualification of components--Summary dated December 3, 1979
October 26, 1979	Letter from applicant transmitting "Final Report, Service Water Intake Structure Settlement Effects and Related Work"

October 30, 1979	Letter to applicant transmitting requests for additional information
October 31, 1979	Site visit to orient NRC staff consultants (Pacific Northwest Laboratory) with the facility and gather information for piping audit calculation - Summary dated October 31, 1979
November 2, 1979	Letter from applicant transmitting report "Comparison of August 27, 1978, Monticello Reservoir Earthquake Response Spectra to the Operating Basis Earthquake Response Spectra for the Virgil C. Summer Nuclear Station"
November 7, 1979	Meeting with applicant to discuss staff position and requests for information on fire protection--Summary dated December 17, 1979
November 9, 1979	Letter to applicant providing clarification of lessons learned short term requirements
November 16, 1979	Letter from applicant transmitting "Seismic Activity Near the V. C. Summer Nuclear Station - For the Period July-September, 1979"
November 21, 1979	Letter to applicant concerning upgraded emergency plans
November 23, 1979	Letter to applicant transmitting "Cladding Swelling and Rupture Models for LOCA Analysis," NUREG-0630, Draft
November 30, 1979	Amendment No. 16
December 13, 1979	Meeting with applicant to discuss Regulatory Guide 1.97--Summary dated January 11, 1980
December 20, 1979	Letter from applicant transmitting Revision 4 to Fire Protection-Fire Hazards Analysis Report
December 21, 1979	Letter to applicant concerning environmental monitoring for direct radiation
December 21, 1979	Letter to applicant concerning proposed regulation on emergency response plans
December 26, 1979	Letter to applicant transmitting request for information regarding evacuation times
January 14-15, 1980	Site visit to review industrial security and safeguards for the facility and to discuss applicant's response to security questions--Summary dated February 14, 1980
January 17, 1980	Notice of Reconstitution of Board; Herbert Grossman is appointed new chairman

February 5, 1980	Letter to applicant concerning "Interim Staff Position on Equipment Qualification of Safety-Related Electrical Equipment," NUREG-0588
February 21, 1980	Site visit to orient NRC consultant with geology and seismology of site and to discuss geology and seismology of site with applicant--Summary dated March 21, 1980
February 21, 1980	Letter to applicant concerning qualification of safety-related electrical equipment
February 22, 1980	Meeting with applicant to discuss draft Technical Specifications--Summary dated March 2, 1980
February 29, 1980	Letter from applicant transmitting Amendment No. 17 to the Final Safety Analysis Report
March 10, 1980	Letter to applicant transmitting "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants (For Interim Use and Comment)," NUREG-0654/FEMA-REP-1
March 10, 1980	Letter to applicant concerning actions required from operating license applicants of nuclear steam supply systems designed by Westinghouse and Combustion Engineering resulting from Bulletins and Orders Task Force Review regarding TMI-2 accident
March 11, 1980	Letter to applicant advising of change to submittal date for evacuation time estimates
March 11, 1980	Letter from applicant transmitting "Seismic Activity Near the V. C. Summer Nuclear Station - for ther Period October-December 1979," Technical Report No. 79-4
March 28, 1980	Letter to applicant concerning qualifications of reactor operators
April 1, 1980	Letter from applicant transmitting revised and updated wiring schematic package drawings
April 1, 1980	Amendment No. 18
April 4, 1980	Letter to applicant transmitting requests for additional information
April 4, 1980	Letter to applicant advising of change in NRC staff project manager
April 15, 1980	Letter from applicant transmitting preliminary preservice inspection program for weld inspections and hydrostatic tests

April 21, 1980	Letter to applicant requesting information on seismic Category I masonry walls
April 25, 1980	Letter to applicant providing clarification of NRC requirements for emergency response facilities
May 5, 1980	Letter to applicant concerning separation of electrical equipment and systems at nuclear power plants
May 5, 1980	Letter to applicant transmitting requests for additional information
May 5, 1980	Letter to applicant transmitting requests for additional information on physical security plan
May 6, 1980	Letter to applicant transmitting requests for additional information
May 6, 1980	Letter from applicant transmitting "Final Report - Significance of the Monticello Reservoir Earthquake of August 27, 1978, to the Virgil C. Summer Nuclear Station"
May 8, 1980	Letter from applicant advising that there are no seismic Category I concrete masonry walls used in the facility design
May 8, 1980	Letter from applicant transmitting description of program to monitor and control secondary water chemistry
May 14, 1980	Letter to applicant concerning potential design deficiencies in bypass, override, and reset circuits of engineered safety features
May 15, 1980	Letter from applicant transmitting Revision 6 to Fire Protection Evaluation-Fire Hazard Analysis Report
May 19, 1980	Letter from applicant transmitting Preliminary Preservice Inspection Plan for Pumps, Valves and Component Supports
May 20, 1980	Letter to applicant providing additional guidance on "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," NUREG-0577
May 20, 1980	Letter from applicant transmitting "Seismic Activity Near the V. C. Summer Nuclear Station for the Period January-March 1980"
May 21, 1980	Letter from applicant expressing appreciation for April 28, 1980 meeting with H. R. Denton and advising that the facility will be "construction ready" by December 1980

May 28, 1980	Letter from applicant concerning request for detailed field audit of installed electrical equipment and systems to verify conformance to separation criteria
May 30, 1980	Letter to applicant transmitting requests for additional information
June 4, 1980	Letter from applicant transmitting Radiation Emergency Plan
June 11, 1980	Letter from Gilbert/Commonwealth transmitting reactor building purge exhaust isolation valve elementary drawing
June 11, 1980	Letter from Gilbert/Commonwealth transmitting drawings for miscellaneous steel for fuel transfer canal shielding
June 11, 1980	Letter to applicant transmitting requests for additional information
June 13, 1980	Meeting with applicant to discuss structural and geotechnical engineering review--Summary dated June 23, 1980
June 13, 1980	Letter to applicant regarding reorganization of the Office of Nuclear Reactor Regulation
June 18, 1980	Letter from applicant transmitting Amendment No. 19 to the Final Safety Analysis Report
June 18, 1980	Letter to applicant concerning underclad cracking in reactor vessel nozzles
June 25, 1980	Letter to applicant transmitting Commission Memorandum and Order
June 26, 1980	Letter to applicant providing further Commission guidance for power reactor operating licenses regarding NUREG-0694
June 30, 1980	Letter to applicant transmitting request for additional information
June 30, 1980	Letter to applicant transmitting Federal Register notice
July 3, 1980	Letter from applicant forwarding additional information on service water intake structure
July 8-10, 1980	Meeting with applicant to discuss technical qualifications--Summary dated July 28, 1980
July 11, 1980	Letter to applicant requesting construction completion date
July 14, 1980	Letter from applicant forwarding additional information on secondary water chemistry control program

July 18, 1980	Letter from applicant forwarding revision 1 to radiation emergency plan
July 18, 1980	Letter from applicant forwarding station report on NUREG-0660 and NUREG-0694
July 21, 1980	Letter from applicant forwarding response to staff request 31.67
July 21, 1980	Letter from applicant forwarding report on diesel generator prelube oil pump
July 21, 1980	Letter from applicant forwarding response to staff request 31.71 through 31.73
July 22-23, 1980	Meeting with applicant to review emergency planning
July 24, 1980	Letter from applicant forwarding evacuation time estimates
July 29, 1980	Letter from applicant regarding qualification of instructors
July 30, 1980	Meeting to discuss seismology matters--Summary issued October 1, 1980
August 1, 1980	Letter to applicant regarding emergency response facilities
August 6, 1980	Letter from applicant forwarding 1978 and 1979 financial reports
August 6, 1980	Letter from applicant regarding design deficiencies of engineered safety features
August 6, 1980	Letter from applicant advising the construction completion date in December 1980
August 8, 1980	Letter from applicant forwarding results of human factors emergency review of control room design
August 12, 1980	Letter from applicant regarding secondary water chemistry control program
August 12, 1980	Letter to applicant requesting additional information
August 13, 1980	Letter from applicant transmitting Amendment 20 to the Final Safety Analysis Report
August 15, 1980	Letter from applicant forwarding emergency feedwater system reliability analysis
August 15, 1980	Letter from applicant forwarding seismic report

August 15, 1980	Letter to applicant regarding TMI matters
August 20, 1980	Letter from applicant forwarding seismic qualification information
August 25, 1980	Letter from applicant forwarding Amendment 20 to the Final Safety Analysis Report
August 26, 1980	Letter to applicant regarding loss-of-coolant accident analyses
August 27, 1980	Letter from applicant responding to staff requests on shielding
August 28, 1980	Letter from applicant forwarding interim deficiency report
September 2, 1980	Letter to applicant requesting additional information
September 2, 1980	Letter to applicant requesting additional information on emergency planning
September 5, 1980	Letter from applicant forwarding response to seismology requests
September 5, 1980	Letter to applicant regarding clarification of TMI requirements
September 9, 1980	Letter from applicant responding to management review concerns
September 9, 1980	Letter from applicant regarding TMI matters
September 10, 1980	Letter from Federal Emergency Management Agency regarding emergency response plans
September 10, 1980	Letter to applicant requesting additional information
September 12, 1980	Letter from applicant forwarding additional information for Section 2.5 of the Final Safety Analysis Report
September 12, 1980	Letter from Colt Industries concerning possible improper operation prelube pump
September 12, 1980	Letter from applicant forwarding responses to staff requests regarding quality assurance
September 16, 1980	Letter to applicant requesting additional information
September 17, 1980	Letter from applicants forwarding Amendment 2 to Physical Security Plan
September 17, 1980	Letter from applicant responding to staff requests regarding preoperational test program

September 17, 1980	Letter from applicant forwarding preservice inspection plan
September 19, 1980	Letter to applicant regarding clarification to TMI Action Plan
September 23, 1980	Letter from applicant regarding underclad cracking on reactor vessel
September 23-24, 1980	Meeting with applicant to discuss emergency operating procedures--Summary dated October 9, 1980
September 24, 1980	Letter from applicant forwarding response to NUREG-0588
September 25, 1980	Letter from applicant forwarding radwaste solidification system instruction manual
September 25, 1980	Letter from applicant forwarding response to staff request 31.76
September 26, 1980	Meeting with applicant to discuss Technical Specification--Summary dated October 15, 1980
September 27, 1980	Letter from Colt Industries regarding possible field insulation damage
September 30, 1980	Letter from applicant forwarding response to staff request 362.16 thru 362.49
October 2, 1980	Letter from applicant regarding fire protection
October 3, 1980	Letter from applicant responding to NRC audit report
October 3, 1980	Letter from applicant forwarding Amendment 21 to the Final Safety Analysis Report
October 7, 1980	Letter from applicant responding to staff requests on postaccident monitoring
October 7, 1980	Letter to applicant requesting additional information
October 8, 1980	Meeting with applicant to discuss reactor systems--Summary dated October 17, 1980
October 8, 1980	Letter from applicant responding to staff request 3S1.17
October 9, 1980	Letter from applicant forwarding deficiency report
Octobber 13, 1980	Letter from applicant forwarding corrected pages for Chapter 17 of Final Safety Analysis Report
October 13, 1980	Letter from applicant forwarding Revision 2 of Emergency Plan

October 15, 1980	Letter from applicant forwarding revised Figure 17.1-1 of the Final Safety Analysis Report
October 20, 1980	Letter from applicant regarding emergency planning
October 27, 1980	Letter to applicant concerning compliance with NRC regulations
October 27, 1980	Letter to applicant concerning inservice inspection
October 28, 1980	Letter from applicant regarding diesel generator qualification
October 28, 1980	Letter to applicant requesting additional information
October 29, 1980	Letter from applicant forwarding Amendment 3 to the Physical Security Plan
October 29, 1980	Letter from applicant forwarding additional loss-of-coolant accident calculations
October 30, 1980	Letter from applicant responding to reactor systems requests
October 31, 1980	Letter from applicant regarding piping penetration assemblies

APPENDIX B
BIBLIOGRAPHY

Accident Analysis Branch

1. "State of South Carolina Peacetime Radiological Emergency Response Plan," Revision 4, December 30, 1978.

2. ANSI-N101.4 - 1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities," American National Standards Institute, 1972.

3. Letter NS-CE-1352 from C. Eicheldinger (Westinghouse) to C. Heltemes (NRC), February 1, 1977.

Analysis Branch

1. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute.

2. "Critical Heat Flux Testing of 17 x 17 Fuel Assembly Geometry with 22 inch Grid Spacing," Westinghouse Topical Report WCAP-8536, December 10, 1976.

3. "An Experimental Investigation of the Effect of Open Channel Flow on Thermal Hydrodynamic Instabilities," Westinghouse Topical Report WCAP-7240.

4. "THINC-IV - An Improved Program for Thermal and Hudraulic Analysis of Rod Bundle Cores," Westinghouse Topical Report WCAP-7956.

5. "Application of the THINC-IV Program to PWR Design," Westinghouse Topical Report WCAP-8054.

6. Letter from J. Stolz to C. Eicheldinger, "Staff Evaluation to WCAP-7956, WCAP-8054, WCAP-8507, and WCAP-8762," April 12, 1978.

7. Westinghouse Mass and Energy Release Data for Containment Design," Westinghouse Topical Report WCAP-8312A.

8. "Mass and Energy Release Following a Steam Line Break," Westinghouse Topical Report WCAP-8860.

9. "TRANFlØ Steam Generator Code Description," Westinghouse Topical Report WCAP-8859.

Auxiliary Systems Branch

1. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1970).

2. ASME Boiler and Pressure Vessel Code, 1971 and 1974 Editions, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

3. ASME Boiler and Pressure Vessel Code, 1971 and 1974 Editions, Section VIII, Division 1, "Pressure Vessels," American Society of Mechanical Engineers.

4. ANSI B31.1.0-1967, "Power Piping," American National Standards Institute.

5. API Standard 620, Fifth Edition, "Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks," American Petroleum Institute. (1973).

6. API Standard 650, Fifth Edition, "Welded Steel Tanks for Oil Storage," American Petroleum Institute. (1973).

7. AWWA D100-73, "AWW Standard for Steel Tanks, Standpipes, Reservoirs, and Elevated Tanks for Water Storage," American Water Works Association. (1973)

Containment Systems Branch

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers.

2. "Electrical Hydrogen Recombiner for PWR Containments," Westinghouse Topical Report WCAP-7820, Supplements 1, 2, 3, 4, and 6.

Core Performance Branch

1. J. M. Hellman et al., "Fuel Densification Experimental Results and Model for Reactor Application," Westinghouse Report WCAP-8219, October 1973 (Proprietary Version WCAP-8218).

2. "Technical Report on Densification of Westinghouse PWR Fuel," United States Atomic Energy Commission Regulatory Staff Report, May 14, 1976.

3. R. O. Meyer, "The Analysis of Fuel Densification," Nuclear Regulatory Commission Report NUREG-0085, July 1976.

4. J. V. Miller et al., "Improved Analytical Methods Used in Westinghouse Fuel Rod Design Computations," Westinghouse Report WCAP-8785, October 1976 (Proprietary Version WCAP-8720).

5. Letter from J. Stolz (NRC) to T. Anderson (Westinghouse), February 9, 1979.

6. W. J. Leech, "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations - Application for Transient Analysis," WCAP-8720, Addendum 1, September 1979.

7. D. H. Risher et al., "Safety Analysis For The Revised Fuel Rod Internal Pressure Design Basis," Westinghouse Report WCAP-8963, January 1979.

8. R. A. George et al., "Revised Clad Flattening Model," Westinghouse Report WCAP-8377, July 1974.

9. G. H. Eng et al., "Fuel Densification Penalty," Westinghouse Report WCAP-7982, October 1972.

10. V. Stello, Jr., (NRC) memorandum to R. C. DeYoung, January 14, 1975, "Evaluation of Westinghouse Report WCAP-8377."

11. E. E. Demario, "Hydraulic Flow Test of the 17x17 Fuel Assembly," Westinghouse Report WCAP-8279, February 1974 (Proprietary Version WCAP-8278).

12. L. Gesinski and D. Chiang, "Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident," Westinghouse Report WCAP-8288, December 1973, and Addendum #1, March 1974 (Proprietary Version WCAP-8236).

13. K. D. Sheppard, S. Cerni, and J. R. Reavis, "An Evaluation of Fuel Rod Bowing," Westinghouse Report WCAP-8346, May 1974.

14. J. R. Reavis et al., "Fuel Rod Bowing," Westinghouse Report WCAP-8692, December 1975 (Proprietary Version WCAP-8691).

15. "Interim Safety Evaluation Report in Westinghouse Fuel Rod Bowing," Division of Systems Safety, USNRC, April 1976.

16. Memorandum from D. F. Ross, Assistant Director for Reactor Safety, DSS and D. G. Eisenhut, Assistant Director for Operational Technology, DOR to D. B. Vassallo, Assistant Director for Light Water Reactors and K. R. Goller, Assistant Director for Operating Reactors, USNRC, "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," February 16, 1977.

17. J. Skaritka et al., "Fuel Rod Bow Evaluation," Westinghouse Report WCAP-8692, Revision 1, July 1979 (Proprietary Version WCAP-8691, Revision 1).

18. V. Stello (NRC) memorandum to R. C. DeYoung," Safety Evaluation of Westinghouse Report WCAP-8288," May 16, 1974.

19. Letter, NS-CE-972, C. Eicheldinger, Westinghouse, to D. Vassallo, NRC, March 1, 1976.

20. B. L. Siegel, "Evaluation of the Behavior of Waterlogged Fuel in LWRs," NRC Report NUREG-0303, March 1978.

21. Letter, NS-CE-640, C. Eicheldinger, Westinghouse, to V. Stello, NRC, May 15, 1975.

22. Letter from L. M. Mills, Tennessee Valley Authority, to L. S. Rubenstein, NRC, Dockets 50-327 and 50-328, dated November 27, 1979.

23. Letter from T. M. Anderson, Westinghouse, to H. R. Denton, NRC, NS-TMA-2238, dated April 29, 1980.

24. P. S. Check (NRC) memorandum to L. S. Rubenstein, "W Baffle-jetting," April 17, 1980.

25. R. O. Meyer (NRC) memorandum to R. A. Clark, "Trojan License Change Application and Cycle 3 Reload," June 20, 1980.

26. Letter form E. H. Crews, Jr. (South Carolina Electric & Gas) to H. R. Denton (NRC) on reactor vessel internal modification to Virgil C. Summer Nuclear Station.

27. Amendment No. 20, Salem, Unit 1 Operating License, October 1979, Docket No. 50-272.

28. D. G. Eisenhut (NRC), "Information Memorandum No. 19 - Westinghouse 17x17 Grid Damage," October 25, 1979.

29. D. G. Eisenhut (NRC), "Information Memorandum No. 18 - Westinghouse 17x17 Rodlet Drops," October 25, 1979.

30. Letter NS-TMA-2214, T. M. Anderson (Westinghouse) to V. Stello (NRC), March 14, 1980.

31. Letter from D. J. Broehl, Portland General Electric Company, to A. Schwencer, NRC, May 25, 1978.

32. Letter from C. F. Yundt, Portland General Electric Company, to R. R. Engelken, NRC, May 8, 1980.

33. "An Evaluation of the Rod Ejection Accident in Westinghouse Reactors Using Special Kinetic Methods," Westinghouse Topical Report WCAP-7588, Revision 1.

Effluent Treatment Systems Branch

1. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Ractors (PWR-GALE Code)," April 1976.

2. Opinion of the Commission, Docket No. Rm-50-2, "Numerical Guidance for Design Objectives and Limiting Conditions for Operation to Meet the Criterion, 'As Low As Practicable,' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," February 1974.

Geosciences Branch

1. Algermissen, S. T. and Perkins, D. M., 1976, A Probabilistic Estimate of Maximum Acceleration in Rock in the Contiguous U.S., U.S. Geological Survey Open File Report - 74-416, 45 pp.

2. Behrendt, J. C.; Hamilton, R. M.; Ackerman, H. D.; Henry, J. V.; and Bayer, K. C., 1980, Seismic Reflection Evidence for Cenozoic Reactivation of Older Faults on Land and Offshore in the Area of the Charleston, S.C. 1886 Earthquake, GSA Vol. 12, No. 7, Abstract, 93rd Annual Meeting, Atlanta, GA. Geological Society of America

3. Bollinger, G. A., 1973, Seismicity of the Southeast United States, Bull. Seismological Society of America, Vol. 63, No. 5, pp 1785-1808.

4. Bollinger, G. A., 1977, Reinterpretation of the Intensity Data for the 1886 Charleston, South Carolina Earthquake, in Studies Related to the Charleston, South Carolina Earthquake of 1886 - A Preliminary Report, USGS Prof. Paper 1028B.

5. Brune, J. N., 1970, Tectonic Stress and the Spectra of Seismic Shear Waves from Earthquakes, J. Geophys. Res., v. 75, pp. 4997-5009.

6. Brune, J. N., 1971, Correction, J. Geophys. Res., v. 76, 5002.

7. Cook, F. A.; Albaugh, S; Brown, L. D.; Kaufman, S.; Oliver, J. E.; Hatcher, R. D., Jr., 1979, Thin-Skinned Tectonics in the Crystalline Southern Appalachians; COCORP Seismic-Reflection Profiling of the Blue Ridge and Piedmont, Geology, V. 7, No. 12, p. 563-567.

8. Dames and Moore, 1979, Comparison of the August 27, 1978, Monticello Reservoir Earthquake Response Spectra to the Operating Basis Earthquake Response Spectra for the Virgil C. Summer Nuclear Station.

9. Dames and Moore, 1980, Final Report, Significance of the Monticello Reservoir Earthquake of August 27, 1978 to the Virgil C. Summer Nuclear Station.

10. Fletcher, J. B.; Sbar, M. L.; and Sykes, L. R., 1978, Seismic Trends and Travel Time Residuals in Eastern North America and Their Tectonic Implications, Geological Society of America Bull. Vo. 89, pp. 1656-1676.

11. Fletcher, J., 1980, A Comparison between Source Parameters Determined from Body-wave Spectra and In-situ Stress Measurements at Monticello, South Carolina, EOS Trans. AGU, v. 61, No. 46, pp. 1027-1028.

12. Gottfried, D.; Annel, C. S.; and Schwarz, L. J., 1979, Geochemistry of Subsurface Basalt from Deep Corehole (Clubhouse Crossroads Corehole 1) near Charleston, S.C. Magma Type and Tectonic Implications, U.S. Geological Survey Prof. Paper 1028-G, pp. 91-114.

13. Harris, L. D. and Bayer, K. C., 1979, Sequential Development of the Appalachian Orogen Above a Master Decollement - A Hypothesis, Geology, v. 7, No. 12, p. 568-572.

14. Heller, P. L.; Wentworth, C. M.; Poag, C. W.; and Mergner-Keefer, M., 1980, Episodic Post-Rift Subsidence of the Eastern U.S. Continental Margin. Geological Society of America, Vol. 12, No. 7, Abstract 93, Ann. Meeting Atlanta, GA, 1980.

15. Holdahl, S. R. and N. L. Morrison, 1974, Regional Investigations of Vertical Crustal Movements in the U.S. Using Precise Relevelings and Mareograph Data, Tectonophysics, Vol. 23, p. 373-390.

16. Johnson, J. A. and Traubenik, N. L., 1978, Magnitude-dependent Near-source Ground Motion Spectra, Proc. A.S.C.E. Conf., Earthquake Engineering and Soil Dynamics, Pasadena, pp. 530-539.

17. Kane, M. F., 1977, Correlation of Major Eastern Earthquake Centers and Mafic/Ultramafic Basement Masses. U.S. Geological Survey Prof. Paper 1028-0, pp. 119-204.

18. Lyttle, P. T.; Gohn, G. S.; Higgins, B. B.; and Wright, D. S., 1979, Vertical Crustal Movements in the Charleston, S.C.-Savannah, Ga. Area, Tectonophysics, Vol. 52, pp. 183-189.

19. McGuire, R. K. and Hanks, T. C., 1980, RMS Accelerations and Spectral Amplitudes of Strong Ground Motion during the San Fernando, California Earthquake, Bull. Seism. Soc. Am., v. 70, No. 5, pp. 1907-1919.

20. McKeown, F. A., 1975, Mafic Intrusives and Their Contact Zones are Source Zones of Many Earthquakes in Central and Southeastern U.S., Earthquake Notes, Vol. 46, No. 53, Abstract.

21. Meade, B. K., 1971, Report of the Sub-Commission on Recent Crustal Movements in North America. Int'l Assoc. of Geodesy, 15th General Assembly, Moscow, USSR.

22. Newmark, N. M., and Hall, W. J., Seismic Response Spectra for Trans Alaska Pipeline, 5th World Conf. Earthquake Eng., Sess. 2B, Rome, Italy, 1973, Proc., no. 60.

23. Nishenko, S. P., and Sykes, L. R., 1979, Fracture Zones, Mesozoic Rifts and the Tectonic Setting of the Charleston, S.C. Earthquake of 1886. Transactions, American Geophysical Union, Vol. 60, No. 18.

24. Nuttli, O. and Herrmann, R., 1978, State-of-the-Art for Assessing Earthquake Hazard[illegible] the United States, Credible Earthquakes for the Central United State[illegible]ort 12, Waterways Experiment Station.

25. Owens, [illegible]. P., 1970, Post-Triassic Tectonic Movements in the Central and Southern Appalachians as Recorded by Sediments of the Atlantic Coastal Plain: in Fisher, and others, eds., Studies of Appalachian Geology - Central and Southern; Interscience Publishers, John Wiley and Sons, 1970.

26. Packer, D., Cluff, L., Knuepfer, P. and Withers. R., 1979, Study of Reservoir Induced Seismicity, Woodward-Clyde Consultants, Final Technical Report to the U.S. Geological Survey.

27. Popenoe, P. and Zietz, I., 1977, The Nature of the Geophysical Basement Beneath the Coastal Plain of South Carolina and Northeastern Georgia, in Studies Related to the Charleston, S.C. Earthquake of 1886 - A Preliminary Report, U.S. Geological Survey Prof. Paper 1028 I.

28. Rankin, D. W., 1976, Appalachian Salients and Recesses: Late Precambrian Continental Breakup and the Opening of the Iapetus Ocean. Journal Geophysical Research, Vol. 81, No. 32, pp. 5605-5619.

29. ________________ , 1977, Studies Related to the Charleston, S.C. Earthquake of 1886 - Introduction and Discussion, U.S. Geological Survey Prof. Paper 1028A.

30. ________________ , 1978, The Charleston, S.C. Earthquake of 1886 and the Blake Spur Fracture Zone. Geological Society of America, Vol. 10, No. 4, 27th Ann. Mtng. Southeastern Section - Abstracts of Programs.

31. Sbar, M. L. and Sykes, L. R., 1973, Contemporary Compressive Stress and Seismicity in Eastern North America: An Example of Intra-Plate Tectonics, GSA Bull., Vol. 84, No. 6, pp. 1861-1881.

32. Secor, D. T., 1980, Geological Studies in an Area of Induced Seismicity at Monticello Reservoir, South Carolina, First Technical Report to U.S. Geological Survey.

33. Seeber, L. and Armbruster, J. G., 1980, The Charleston, S.C. Earthquake and the Appalachian Detachment, in review.

34. Sheridan, R. E., 1976, Sedimentary Basins of the Atlantic Margin of North America, Tectonophysics, Vol. 36, pp. 113-132.

35. South Carolina Electric and Gas Co., 1971, Preliminary Safety Analysis Report, Virgil C. Summer Nuclear Station, Unit 1.

36. ________________ , 1974, Supplemental Geologic Investigation Report, 14 June 1974, and Addendum, 10 December 1975.

37. ________________ , 1976, Final Safety Analysis Report, Virgil C. Summer Nuclear Station, Unit 1.

38. ________________ , 1980, Supplemental Seismologic Investigation, Virgil C. Summer Nuclear Station, Unit 1, 10 December 1980.

39. Talwani, P., 1979, Seismic Activity Near the V.C. Summer Nuclear Station for the Period October-December 1979, Technical Report 79-4, U.S. Geological Survey Contract No. N230519.

40. Thatcher, W. and Hanks, T.C., 1973, Source Parameters of Southern California Earthquakes, J. Geophys. Res., v. 78, pp. 8547-8576.

41. US Atomic Energy Commission, 1972, Safety Evaluation Report in the Matter of Virgil C. Summer Nuclear Station, Unit 1, 29 August 1972.

42. ________________ , 1974, Safety Evaluation of the Geologic Faults at the Site of the Virgil C. Summer Nuclear Station, Unit 1, 12 February 1974.

43. Wentworth, C. M. and Mergner-Keefer, M., 1980, Atlantic Coast Reverse-Fault Domain: Probable Source of East-Coast Seismicity, Geological Society of America, Vol. 12, No. 7, Abstract 93rd Ann. Mtng., Atlanta, Ga.

44. Winker, C. D. and Howard, J. D., 1977, Correlation of Tectonically Deformed Shorelines on the Southern Atlantic Coastal Plain, Geology, Vol. 5, pp. 123-127.

45. Winker, C. D., 1980, Rates of Regional Quaternary Deformation, Atlantic and Gulf Coastal Plains, Geological Society of America, Vol. 12, No. 7, Abstract 93rd Ann. Mtng., Atlanta, Ga.

46. Zimmerman, R. A., 1980, Patterns of Post-Triassic Uplift and Inferred Fall Zone Faulting in the Eastern U.S., Geological Society of America, Vol. 12, No. 7, Abstract 93rd Ann. Mtng., Atlanta, Ga.

47. Zoback, M. D. and Zoback, M. L., 1980, in: Evernden, J. F., 1980, Magnitude of Deviatoric Stress in the Earth's Crust and Upper Mantle, U.S. Geological Survey, Proceedings of Conf. IX, OFR 80-625, pp. 353-433.

Hydrologic and Geotechnical Engineering Branch

1. U.S. Army Corps of Engineers, "Wave Runup on Rough Slopes," ETL 110-2-227, 1976 and Draft Guide CETA 79, U.S. Army Corps of Engineers Coastal Engineering Research Center, 1979.

2. ANSI-N170-1976, "Standards for Determining Design Basis Flooding at Power Reactor Sites," American National Standards Institute, 1976.

Instrumentation and Control Systems Branch

1. Westinghouse Topical Report WCAP-8892A.

2. IEEE Standard 279-1971, Institute of Electrical and Electronic Engineers.

3. "Failure Mode and Effects Analysis (FMEA) of the Engineered Safety Features Actuation System," Westinghouse Topical Report WCAP-8584.

4. IEEE Standard 308-1971, Institute of Electrical and Electronic Engineers.

5. ANSI N42.7-1972, American National Standards Institute.

6. IEEE Standard 317-1972, Institute of Electrical and Electronic Engineers.

7. IEEE Standard 323-1974, Institute of Electrical and Electronic Engineers.

8. IEEE Standard 336-1971, Institute of Electrical and Electronic Engineers.

9. IEEE Standard 338-1971, Institute of Electrical and Electronic Engineers.

10. IEEE Standard 344-1975, Institute of Electrical and Electronic Engineers.

11. IEEE Standard 379-1972, Institute of Electrical and Electronic Engineers.

12. IEEE Standard 384-1974, Institute of Electrical and Electronic Engineers.

Materials Engineering Branch

1. ASME Boiler and Pressure Vessel Code, 1974 Edition, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" American Society of Mechanical Engineers.

2. ASME Boiler and Pressure Vessel Code, 1971 Edition and 1972 Summer Addenda, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers.

3. "Dynamic Fracture Toughness of ASME SA-505 Class 2a, ASME SA-533 Grade A Class 2 Base and Heat Affected Zone Material and Applicable Weld Metals"; Westinghouse Topical Report WCAP-9292.

4. ASTM Standard E 187-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels."

5. Westinghouse Topical Report WCAP-8082-P-A, January 1975.

6. Westinghouse Topical Report BN-TOP-2, Revision 2, May 1974.

7. ASME Boiler and Pressure Vessel Code, Section I , Parts A, B, and C, American Society of Mechanical Engineers.

<u>Operator Licensing Branch</u>

1. ANSI N18.1, American National Standard Institute, 1971.

2. ANSI 18.7-1976, American National Standard Institute, 1976.

3. "Fire Protection Evaluation Report," Gilbert Associates, Inc., July 1977.

<u>Structural Engineering Branch</u>

1. Paper No. 3269, Volume 126, Part 2, "Wind Forces on Structures, "American Society of Civil Engineering.

2. Kennedy, R. P., "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," Nuclear and Systems Sciences Group, Holmes and Nasves, Inc., September 1975.

3. Gwaltney, R. G., "Missile Generation and Protection in Light-water-Cooled Power Reactor Plants," ORNL-NSIC-22 Oak Ridge National Laboratory September 1968.

4. Stephenson, A. E., "Tornado Vulnerability - Nuclear Production Facilities," Sandia Laboratories, April 1975.

5. Stephenson, A. E., "Addendum to Tornado Vulnerability - Nuclear Production Facilities," Sandia Laboratories, June 1975

6. EPRI NP-148, "Full Scale Tornado - Missile Impact Test," Electric Power Research Institute, April 1976.

7. "Missile Impact Testing of Reinforced Concrete Panels," Calspan Corporation, January 1975.

8. ACI 318-71, "Building Code Requirements for Reinforced Concrete."

9. "AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Building."

Power Systems Branch

1. IEEE Standard 308-1974, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.

2. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.

3. IEEE Standard 384-1974, "Criteria for Independence of Electrical Circuits," Institute of Electrical and Electronic Engineers.

4. IEEE Standard 317-1972, "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations, "Institute of Electrical and Electronic Engineers.

5. IEEE Standard 387-1977, "Standard Criteria Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.

6. ANSI-N-197 "Fuel Oil Systems for Diesel Generators," American National Standards Institute.

7. ASME Boiler and Pressure Vessel Code, Section VIII, American Society of Mechanical Engineers.

8. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components" American Society of Mechanical Engineers.

Reactor Systems Branch

1. ASME Boiler and Pressure Vessel Code Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

2. Letter from C. Eicheldinger (Westinghouse) to J. F. Stolz (NRC), January 1978.

3. Westinghouse Topical Report WCAP-7306

4. Westinghouse Topical Report WCAP-7486.

5. Westinghouse Topical Report WCAP-7706.

6. Westinghouse Topical Report WCAP-7769, Revision 1.

7. Westinghouse Topical Report WCAP-7898

8. "LOFTRAN Code Description," Westinghouse Topical Report WCAP-7907.

9. "A FACTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," Westinghouse Topical Report WCAP-7908.

10. "MARVEL Code Description," Westinghouse Topical Report WCAP-7909.

11. Westinghouse Topical Report WCAP-7956.

12. Westinghouse Topical Report WCAP-7973.

13. Westinghouse Topical Report WCAP-7980.

14. "BLKOUT Code Description, "Westinghouse Topical Report WCAP-7998.

15. Westinghouse Topical Report WCAP-8853.

16. Westinghouse Topical Report WCAP-8963.

17. Westinghouse Topical Report WCAP-8971.

18. "Main Steamline Break Sensitivity Studies," Westinghouse Topical Report WCAP-9227.

19. "Report on the Consequences of a Postulated Main Feedline Repture," Westinghouse Topical Report WCAP-9230.

APPENDIX C

NUCLEAR REGULATORY COMMISSION UNRESOLVED SAFETY ISSUES

C.1 Unresolved Safety Issues

The NRC staff continuously evaluates the safety requirements used in its reviews against new information as it becomes available. Information related to the safety of nuclear power plants comes from a variety of sources including experience from operating reactors, research results, NRC staff and Advisory Committee on Reactor Safeguards safety reviews, and vendor, architect/engineer and utility design reviews. Each time a new concern or safety issue is identified from one or more of these sources, the need for immediate action to assure safe operation is assessed. This assessment includes consideration of the generic implications of the issue.

In some cases, immediate action is taken to assure safety, e.g., the derating of boiling water reactors as a result of the channel box wear problems in 1975. In other cases, interim measures, such as modifications to operating procedures, may be sufficient to allow further study of the issue prior to making licensing decisions. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether existing NRC staff requirements should be modified to address the issue for new plants or if backfitting is appropriate for the long-term operation of plants already under construction or in operation.

These issues are sometimes called "generic safety issues" because they are related to a particular class or type of nuclear facility rather than a specific plant. These issues have also been referred to as "unresolved safety issues." However, as discussed above, such issues are considered on a generic basis only after the staff has made an initial determination that the safety significance of the issue does not prohibit continued operation or require licensing actions while the longer-term generic review is underway.

C.2 ALAB-444 Requirements

These longer-term generic studies were the subject of a Decision by the Atomic Safety and Licensing Appeal Board of the NRC. The Decision was issued on November 23, 1977 (ALAB-444) in connection with the Appeal Board's consideration of the Gulf States Utility Company application for the River Bend Station, Unit Nos. 1 and 2.

In the view of the Appeal Board (pp. 25-29):

> "The responsibilities of a licensing board in the radiological health and safety sphere are not confined to the consideration and disposition of those issues which may have been presented to it by a party or an "Interested State" with the required degree of specificity. To the contrary, irrespective of what matters may or may not have been properly placed in controversy, prior to authorizing the issuance of a construction permit the board must make the finding, inter alia, that there is "reasonable assurance" that "the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public." 10 CFR 50.35(a)...Of necessity, this determination will entail an inquiry into whether the staff review satisfactorily has come to grips with any unresolved generic safety problems which might have an impact upon operation of the nuclear facility under consideration."
>
> "The SER is, of course, the principal document before the licensing board which reflects the content and outcome of the staff's safety review. The board should therefore be able to look to that document to ascertain the extent to which generic unresolved safety problems which have been previously identified in a FSAR item, a Task Action Plan, an ACRS report or elsewhere have been factored into the staff's analysis for the particular reactor -- and with what result. To this end, in our view, each SER should contain a summary description of those generic problems under continuing study which have both relevance to facilities of the type under review and potentially significant public safety implications."
>
> "This summary description should include information of the kind now contained in most Task Action Plans. More specifically, there should be an indication of the investigative program which has been or will be undertaken with regard to the problem, the program's anticipated time span, whether (and if so, what) interim measures have been devised for dealing with the problem pending the completion of the investigation, and what alternative courses of action might be available should the program not produce the envisaged result."
>
> "In short, the board (and the public as well) should be in a position to ascertain from the SER itself -- without the need to resort to extrinsic documents -- the staff's perception of the nature and extent of the relationship between each significant unresolved generic safety question and the eventual operation of the reactor under scrutiny. Once again, this assessment might well have a direct bearing upon the ability of the licensing board to make the safety findings required of it on the construction permit level even though the generic answer to the question remains in the offing. Among other things, the furnished information would likely shed light on such alternatively important considerations as whether: (1) the problem has already

been resolved for the reactor under study; (2) there is a reasonable basis for concluding that a satisfactory solution will be obtained before the reactor is put in operation; or (3) the problem would have no safety implications until after several years of reactor operation and, should it not be resolved by then, alternative means will be available to insure that continued operation (if permitted at all) would not pose an undue risk to the public."

This appendix is specifically included to respond to the decision of the Atomic Safety and Licensing Appeal Board as enunciated in ALAB-444 and as applied to an operating license proceeding Virginia Electric and Power Company (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, 8 NRC 245 (1978).

C.3 Unresolved Safety Issues

In a related matter, as a result of Congressional action on the Nuclear Regulatory Commission budget for Fiscal Year 1978, the Energy Reorganization Act of 1974 was amended (PL 95-209) on December 13, 1977 to include, among other things, a new Section 210 as follows:

"UNRESOLVED SAFETY ISSUES PLAN"

"SEC. 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report of the Commission thereafter."

The Joint Explanatory Statement of the House-Senate Conference Committee for the FY 1978 Appropriations Bill (Bill S.1131) provided the following additional information regarding the Committee's deliberations on this portion of the bill:

"SECTION 3 - UNRESOLVED SAFETY ISSUES"

"The House amendment required development of a plan to resolve generic safety issues. The conferees agreed to a requirement that the plan be submitted to the Congress on or before January 1, 1978. The conferees also expressed the intent that this plan should identify and describe those safety issues, relating to nuclear power reactors, which are unresolved on the date of enactment. It should set forth: (1) Commission actions taken directly or indirectly to develop and implement corrective measures; (2) further actions planned concerning such measures; and (3) timetables and cost estimates of such actions. The Commission should indicate the priority it has assigned to each issue, and the basis on which priorities have been assigned."

In response to the reporting requirements of the new Section 210, the NRC staff submitted to Congress on January 1, 1978, a report describing the NRC generic issues program (NUREG-0410).[1] The NRC program was already in place when PL 95-209 was enacted and is of considerably broader scope than the "Unresolved Safety Issues Plan" required by Section 210. In the letter transmitting NUREG-0410 to the Congress on December 30, 1977, the Commission indicated that "the progress reports, which are required by Section 210 to be included in future NRC annual reports, may be more useful to Congress if they focus on the specific Section 210 safety items."

It is the NRC's view that the intent of Section 210 was to assure that plans were developed and implemented on issues with potentially significant public safety implications. In 1978, the NRC undertook a review of over 130 generic issues addressed in the NRC program to determine which issues fit this description and qualify as "Unresolved Safety Issues" for reporting to the Congress. The NRC review included the development of proposals by the NRC Staff and review and final approval by the NRC Commissioners.

This review is described in a report, NUREG-0510, entitled "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants - A Report to Congress" dated January 1979. The report provides the following definition of an "Unresolved Safety Issue:"

> "An Unresolved Safety Issue is a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants it affects."

Further the report indicates that in applying this definition, matters that pose "important questions concerning the adequacy of existing safety requirements" were judged to be those for which resolution is necessary to (1) compensate for a possible major reduction in the degree of protection of the public health and safety, or (2) provide a potentially significant decrease in the risk to the public health and safety. Quite simply, an "Unresolved Safety Issue" is potentially significant from a public safety standpoint and its resolution is likely to result in NRC action on the affected plants.

All of the issues addressed in the NRC program were systematically evaluated against this definition as described in NUREG-0510. As a result, 17 "Unresolved Safety Issues" addressed by 22 tasks in the NRC program were identified. The issues are listed below. Progress on these issues was discussed in the 1978 NRC Annual Report. The number(s) of the generic task(s) (e.g., A-1) in the NRC program addressing each issue is indicated in parentheses following the title.

[1] NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," issued on January 1, 1978.

"UNRESOLVED SAFETY ISSUES" (APPLICABLE TASK NOS.)

1. Water Hammer - (A-1)
2. Asymmetric Blowdown Loads on the Reactor Coolant System - (A-2)
3. Pressurized Water Reactor Steam Generator Tube Integrity - (A-3, A-4, A-5)[2]
4. BWR Mark I and Mark II Pressure Suppression Containments - (A-6, A-7, A-8, A-39)
5. Anticipated Transients Without Scram - (A-9)
6. BWR Nozzle Cracking - (A-10)
7. Reactor Vessel Materials Toughness - (A-11)
8. Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports - (A-12)
9. Systems Interaction in Nuclear Power Plants - (A-17)
10. Environmental Qualification of Safety-Related Electrical Equipment - (A-24)
11. Reactor Vessel Pressure Transient Protection - (A-26)
12. Residual Heat Removal Requirements - (A-31)
13. Control of Heavy Loads Near Spent Fuel - (A-36)
14. Seismic Design Criteria - (A-40)
15. Pipe Cracks at Boiling Water Reactors - (A-42)
16. Containment Emergency Sump Reliability - (A-43)
17. Station Blackout - (A-44)

In the view of the staff, the "Unresolved Safety Issues" listed above are the substantive safety issues referred to by the Appeal Board in ALAB-444 when it spoke of ". . . those generic problems under continuing study which have . . . potentially significant public safety implications" (page 27). Eight of the 22 tasks identified with the "Unresolved Safety Issues" are not applicable to Virgil C. Summer Nuclear Station, Unit 1 and six of these tasks (A-6,[3] A-7, A-8, A-39, A-10 and A-42) are peculiar to boiling water reactors. With regard to the remaining 14 tasks that are applicable to this facility, the NRC staff has issued NUREG reports providing its proposed resolution of five of the issues. Each of these have been addressed in this Safety Evaluation Report or will be addressed in a future supplement. The table below lists those issues and the section of this Safety Evaluation Report in which they are discussed.

---

[2]Even though Tasks A-4 and A-5 address steam generator tube problems experienced in Combustion Engineering and Babcock & Wilcox plants, there are many common task elements between these tasks and Task A-3 which address Westinghouse steam generator tube problems. For this reason, the Task Action Plans for all three asks have been combined into a single Task Action Plan.

[3]Task A-6 was completed in December 1977.

Task Number	NUREG Report and Title	Safety Evaluation Report Section
A-12	NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports"	Discussed below
A-24	NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"	7.7.2
A-26	NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors" and RSB BTP 5-2	5.4.2
A-31	Regulatory Guide 1.139, "Guidance for in Residual Heat Removal" and RSB BTP 5-1	Will be addressed in a futu supplement
A-36	NUREG-0612, Control of Heavy Loads at Nuclear Power Plants	9.2.4

NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," was issued for comment in November 1979. This report summarizes work performed by the NRC staff and its contractor, Sandia Laboratories, in the resolution of this generic activity. The report describes the technical issues, the technical studies performed by Sandia Laboratories, the NRC staff's technical positions based on these studies, and the NRC staff's plan for implementing its technical positions. As a part of initiating the implementation of the findings in this report, letters were sent to all applicants and licensees on May 19 and 20, 1980. In these letters a revised proposed implementation plan was presented and specific criteria for material qualifications were defined.

Many comments on both the draft of NUREG-0577 and the letters of May 19 and 20 have been received by the NRC staff and detailed consideration is presently being given to these comments. After completing our review and analysis of the comments provided, we will issue the final revision of NUREG-0577 which will include a full discussion and resolution of the comments and a final plan for implementation.

We estimate that our implementation review will require approximately two years. Since many factors (initiating event, low fracture toughness in a critical support member in tension, low operating temperature, large flaw) must be simultaneously present for failure of the support system to ensue the we have determined that licensing for pressurized water reactors should continue during the implementation phase. Our conclusions regarding licensing and subsequent operation are not sensitive to the estimated length of time required for this work.

With regard to the lamellar tearing issue, the results of an extensive literature survey by Sandia revealed that, although lamellar tearing is a common occurrence in structural steel construction, virtually no documentation exists describing inservice failures due to lamellar tearing. Nonetheless, additional research is recommended to provide a more definitive and complete evaluation of the importance of lamellar tearing to the structural integrity of nuclear power plant support systems.

The remaining issues applicable to this facility are listed in the following table:

GENERIC TASKS ADDRESSING UNRESOLVED SAFETY ISSUES
THAT ARE APPLICABLE TO THE VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

1. A-1 Water Hammer
2. A-2 Asymmetric Blowdown Loads on the Reactor Coolant System
3. A-3 Westinghouse Steam Generator Tube Integrity
4. A-9 Anticipated Transients Without Scram
5. A-11 Reactor Vessel Materials Toughness
6. A-17 Systems Interactions in Nuclear Power Plants
7. A-40 Seismic Design Criteria
8. A-43 Containment Emergency Sump Reliability
9. A-44 Station Blackout

With the exception of Tasks A-9, A-43 and A-44, Task Action Plans for the generic tasks above are included in NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants." The technical resolution for Task A-9 is completed and a proposal for rulemaking is in preparation. A Task Action Plan for Task A-44 has been approved and issued. A Task Action Plan for Task A-43 is near completion. The information provided in NUREG-0649 meets most of the informational requirements of ALAB-444. Each Task Action Plan provides a description of the problem; our approach to its resolution; a general discussion of the bases upon which continued plant licensing or operation can proceed pending completion of the task; the technical organizations involved in the task and estimates of the manpower required; a description of the interactions with other NRC offices, the Advisory Committee on Reactor Safeguards and outside organizations; estimates of funding required for contractor supplied technical assistance; prospective dates for completing the task; and a description of potential problems that could alter the planned approach or schedule

We have reviewed the nine "Unresolved Safety Issues" listed above as they relate to Virgil C. Summer Nuclear Station, Unit 1. Discussion of each of these issues including references to related discussions in the Safety Evaluation Report are provided below in Section C.5. Based on our review of these items, we have concluded, for the reasons set forth in Section C.5, that there is reasonable assurance that this facility can be operated prior to the ultimate resolution of these generic issues without endangering the health and safety of the public.

C.4 New "Unresolved Safety Issues"

No new issues have been identified in 1979 for reporting as "Unresolved Safety Issues." The NRC staff has not performed an in-depth review to identify new issues however NRC efforts have been concentrated on implementing new TMI-related requirements on operating plants and on identifying, defining and scoping additional TMI-related issues and tasks. Several broad program areas where issues and tasks are being scoped will likely result in designation of new "Unresolved Safety Issues." These program areas include the following:

1. Man-machine interface and control-room design.

2. Qualification and training of operation, maintenance, and supervisory personnel.

3. Offsite emergency response, emergency planning, and action guidelines.

4. Siting policy, including compensatory design and operating provisions for plants in areas where evacuation would be difficult.

5. Systems reliability and interactions.

6. Consideration in licensing requirements of accidents involving degraded or melted fuel.

Nonetheless, the specific TMI-related requirements for licensing this facility have been identified and are discussed in this Safety Evaluation Report. Many of these are related to the program areas listed above. Long-term "Unresolved Safety Issue" tasks that may be undertaken in the same program areas could provide a basis for further improvements that may or may not be applicable to the Farley plant.

The NRC staff also performed a cursory review of a number of candidate issues from sources other than Three Mile Island accident investigations, including a review of events reported as abnormal occurrences in 1979. Based on this cursory review, none were judged to be of such safety importance to require reporting to the Congress in the 1979 Annual Report as "Unresolved Safety Issues." An in-depth and systematic review of all candidate issues is being performed by the staff and the Commission. A special report will be provided to the Congress after completion of this review, describing the review and new issues designated as "Unresolved Safety Issues." Their applicability to all plants will be determined at that time.

C.5 Discussion of Tasks as they Relate to Virgil C. Summer Nuclear Station, Unit 1

A-1 Water Hammer

Water hammer events are intense pressure pulses in fluid systems caused by any one of a number of mechanisms and system conditions.

Since 1971 there have been over 100 incidents involving water hammer in pressurized water reactors and boiling water reactors. The water hammers have involved steam generator feedrings and piping, decay heat removal systems, emergency core cooling systems, containment spray lines, service water lines, feedwater lines and steam lines. However, the systems most frequently affected by water hammer effects are the feedwater systems. The most serious water hammer events have occurred in the steam generator feedrings of pressurized water reactors. These types of water hammer events are addressed in Section 10.4.3 of this Safety Evaluation Report.

With regard to protection against other potential water hammer events currently provided in plants, piping design codes require consideration of impact loads. Approaches used at the design stage include: (1) increasing valve closure times, (2) piping layout to preclude water slugs in steam lines and vapor formation in water lines, (3) use of snubbers and pipe hangers, and (4) use of vents and drains. In addition, as described in Section 3.9.2 of this Safety Evaluation Report, we require that the applicant conduct a preoperational vibration dynamic effects test program in accordance with Section III of the ASME Code for all ASME Class 1 and Class 2 piping systems and piping restraints during startup and initial operation. These tests will provide adequate assurance that the piping and piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips and other operating modes associated with the design operational transients.

Nonetheless, in the unlikely event that a large pipe break did result from a severe water hammer event, core cooling is assured by the emergency core cooling systems described in Section 6.3 of this Safety Evaluation Report and protection against the dynamic effects of such pipe breaks inside and outside of containment is provided as described in Section 3.6 of this Safety Evaluation Report.

Task A-1 may identify some potentially significant water hammer scenarios that have not explicitly been accounted for in the design and operation of nuclear power plants. The task has not as yet identified the need for requiring any additional measures beyond those already required in the short term.

Based on the foregoing, we have concluded that the facility can be operated prior to ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-2 Asymmetric Blowdown Loads on Primary Coolant Systems

In the very unlikely event of a rupture of the primary coolant piping in a light water reactor, large nonuniformly distributed loads would be imposed upon the reactor vessel, reactor vessel internals, and other components in the reactor coolant system. The potential for such asymmetric loads, which result from the rapid depressurization of the reactor coolant system, was identified in May 1975 and was not considered in the original design of some facilities. The forces associated with a postulated break in the reactor coolant piping near the reactor vessel, for example, could affect the integrity of the reactor vessel supports and reactor pressure vessel internals. A significant failure of the reactor vessel support system, besides impacting the reactor internals, has a potential for (1) damaging systems designed to cool the core following the postulated piping break, (2) affecting the capability of the control rods to function properly, (3) damaging other reactor coolant system components, and (4) causing other ruptures in the initially unbroken reactor coolant system piping loops and attached systems.

As indicated in Section 3 of the Task Action Plan for Task A-2 in NUREG 0649, we currently require that this issue be resolved prior to issuing an operating license. This issue has been acceptably resolved for this facility. Our evaluation and conclusions are provided in Section 3.9.3 of this Safety Evaluation Report. Accordingly, we have concluded that this facility can be operated prior to ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-3 Westinghouse Steam Generator Tube Integrity

The primary concern is the capability of steam generator tubes to maintain their integrity during normal operation and postulated accident conditions. In addition, the requirements for increased steam generator tube inspections and repairs have resulted in significant increases in occupational exposures to workers. Corrosion resulting in steam generator tube wall thinning (wastage) has been observed in several Westinghouse plants for a number of years. Plants operating exclusively with an all volatile secondary water treatment process have not experienced this form of degradation to date. Another major corrosion-related phenomenon has also been observed in a number of plants in recent years, resulting from a buildup of support plate corrosion products in the annulus between the tubes and the support plates. This buildup eventually causes a diametral reduction of the tubes, called "denting," and deformation of the tube support plates. This phenomenon has led to other problems, including stress corrosion cracking, leaks at the tube/support plate intersections, and U-bend section cracking of tubes which were highly stressed because of support plate deformation.

Specific measures such as steam generator design features and a secondary water chemistry control and monitoring program, that the applicant has employed to minimize the onset of steam generator tube problems are described in Section of this Safety Evaluation Report. In addition, Section of this Safety Evaluation Report discusses the inservice inspection requirements. As described in Section, the applicant has met all current requirements regarding steam generator tube integrity. The Technical Specifications will include requirements for actions to be taken in the event that steam generator tube leakage occurs during plant operation.

Task A-3 is expected to result in improvements in our current requirements for inservice inspection of steam generator tubes. These improvements will include a better statistical basis for inservice inspection program requirements and consideration of the cost/benefit of increased inspection. Pending completion of Task A-3, the measures taken at this facility should minimize the steam generator tube problems encountered. Further the inservice inspection and Technical Specification requirements will assure that the applicant and the NRC staff are alerted to tube degradation should it occur. Appropriate actions such as tube plugging, increased and more frequent inspections and power derating could be taken if necessary. Since the improvements that will result from Task A-3 will be procedural, i.e., an improved inservice inspection program, they can be implemented by the applicant at after operation of this facility begins, if necessary.

Based on the foregoing, we have concluded that this facility can be operated prior to ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-9 Anticipated Transients Without Scram

Nuclear plants have safety and control systems to limit the consequences of temporary abnormal operating conditions or "anticipated transients." Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. In some anticipated transients, rapidly shutting down the nuclear reaction (initiating a "scram"), and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. If there were a potentially severe "anticipated transient" and the reactor shutdown system did not "scram" as desired, then an "anticipated transient without scram," or ATWS, would have occurred.

The anticipated transients without scram issue and the requirements that must be met by the applicant prior to operation of the facility are discussed in Section 15.3.5 of this Safety Evaluation Report.

Based on our review, we have concluded that there is reasonable assurance that this facility can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

A-11 Reactor Vessel Materials Toughness

Resistance to brittle fracture, a rapidly propagating catastrophic failure mode for a component containing flaws, is described quantitatively by a material property generally denoted as "fracture toughness." Fracture toughness has different values and characteristics depending upon the material being considered. For steels used in nuclear reactor pressure vessels, three considerations are important. First, fracture toughness increases with increasing temperature. Second, fracture toughness decreases with increasing load rates. Third, fracture toughness decreases with neutron irradiation.

In recognition of these considerations, power reactors are operated within restrictions imposed by the Technical Specifications on the pressure during heatup and cooldown operations. These restrictions assure that the reactor vessel will not be subjected to that combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel material. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these Technical Specification limitations over the life of the plant.

For the service times and operating conditions typical of current operating plants, reactor vessel fracture toughness for most plants provides adequate margins of safety against vessel failure under operating testing, maintenance, and anticipated transient conditions, and accident conditions over the life of the plant. However, results from a reactor vessel surveillance program and analyses performed using currently available methods indicate that the reactor vessels for up to 20 older operating pressurized water reactors and those for some more recent vintage plants will have marginal toughness, relative to required margins at normal full power after comparatively short periods of operation. In addition, results from analyses performed by pressurized water reactor manufacturers indicate that the integrity of some reactor vessels may not be maintained in the event that a main steam line break or a loss-of-coolant accident occurs after approximately 20 years of operation. The principal objective of Task A-11 is to develop an improved engineering method and safety criteria to allow a more precise assessment of the safety margins that are available during normal operation and transients in older reactor vessels with marginal fracture toughness and of the safety margins available during accident conditions for all plants.

Based upon our evaluation of this facility's reactor vessel materials toughness, we have concluded that this unit will have adequate safety margins against brittle failure during operating, testing, maintenance and anticipated transient conditions over the life of the units. However, some pressurized water reactors in the later stages of licensing have the potential, after many years of operation, to have marginal fracture toughness for the postulated accident conditions. When Task Action Plan A-11 is completed and explicit fracture evaluation criteria for accident conditions are defined, all vessels will be reevaluated for acceptability over their design lives. Since Task A-11 is projected to be completed well in advance of this facility's reactor vessel reaching a level of marginal fracture resistance, acceptable vessel integrity for the postulated accident conditions will be assured at least until the reactor vessel is reevaluated for long-term acceptability.

Therefore, based upon the foregoing, we have concluded that this facility can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

A-17 Systems Interactions In Nuclear Power Plants

The licensing requirements and procedures used in our safety review address many different types of systems interactions. Current licensing requirements are founded on the principle of defense-in-depth. Adherence to this principle results in requirements such as physical separation and independence of redundant safety systems, and protection against events such as high energy line ruptures, missiles, high winds, flooding, seismic events, fires, operator errors, and sabotage. These design provisions supplemented by the current review procedures of the Standard Review Plan (NUREG-75/087) which require interdisciplinary reviews and which account, to a large extent, for review of potential systems interactions, provide for an adequately safe situation with respect to such interactions. The quality assurance program which is followed during the design, construction, and operational phases for each plant is expected to provide added assurance against the potential for adverse systems interactions.

In November 1974, the Advisory Committee on Reactor Safeguards requested that the NRC staff give attention to the evaluation of safety systems from a multi-disciplinary point of view, in order to identify potentially undesirable interactions between plant systems. The concern arises because the design and analysis of systems is frequently assigned to teams with functional engineering specialties--such as civil, electrical, mechanical, or nuclear. The question is whether the work of these functional specialists is sufficiently integrated in their design and analysis activities to enable them to identify adverse interactions between and among systems. Such adverse events might occur, for example, because designers did not assure that redundancy and independence of safety systems were provided under all conditions of operation required, which might happen if the functional teams were not adequately coordinated.

In mid-1977, Task A-17 was initiated to confirm that present review procedures and safety criteria provide an acceptable level of redundancy and independence for systems required for safety by evaluating the potential for undesirable interactions between and among systems.

The NRC staff's current review procedures assign primary responsibility for review of various technical areas and safety systems to specific organizational units and assign secondary responsibility to other units where there is a functional or interdisciplinary relationship. Designers follow somewhat similar procedures and provide for interdisciplinary reviews and analyses of systems. Task A-17 will provide an independent investigation of safety functions--and systems required to perform these functions--in order to assess the adequacy of current review procedures. This investigation is being conducted by Sandia Laboratories under contract assistance to the NRC staff.

The contract effort, Phase I of the task, began in May 1978 and is nearing completion. The Phase I investigation is structured to identify areas where interactions are possible between and among systems and have the potential of negating or seriously degrading the performance of safety functions. The investigation will then identify where NRC review procedures may not have properly accounted for these interactions. Preliminary results of the Phase I contracted effort indicate that, within the limitations of the study, there are only a few areas where the review procedures are weak from a systems interaction standpoint. These results are being finalized by the contractor and the staff is considering whether, and if so, what changes in the Standard Review Plan are needed. Finally, a follow-on Phase II of the task will be scoped based on the results of Phase I and the status and scope of other related NRC activities.

The NRC staff believes that its review procedures and acceptance criteria currently provide reasonable assurance that an acceptable level of system redundancy and independence is provided in plant designs. Although some changes to the review procedures will likely result, the preliminary results of the Phase I effort appear to confirm this belief. Therefore, we conclude that there is reasonable assurance that this facility can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-40 Seismic Design Criteria - Short-Term Program

NRC regulations require that nuclear power plant structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants are provided in the NRC regulations and Regulatory Guides. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, rereviews of the seismic design of various plants are being undertaken to assure that these plants do not present an undue risk to the public. Task A-40 is, in effect, a compendium of short-term efforts to support such reevaluation efforts of the NRC staff, especially those related to older operating plants. In addition, some revisions to the Standard Review Plan sections and Regulatory Guides to bring them more in line with the state-of-the-art will result.

As discussed in Section 3.7 of this Safety Evaluation Report the seismic design basis and seismic design of the facility have been evaluated at the operating license stage and have been found acceptable. We do not expect the results of Task A-40 to affect these conclusions because the techniques under consideration are essentially these utilized in the review of this facility. Accordingly, we have concluded that this facility can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-43 Containment Emergency Sump Reliability

Following a postulated loss-of-coolant accident, i.e, a break in the reactor coolant system piping, the water flowing from the break would be collected in the emergency sump at the low point in the containment. This water would be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water would also be circulated through the containment spray system to remove heat and fission products from the containment. Loss of the ability to draw water from the emergency sump could disable the emergency core cooling and containment spray systems. The consequences of the resulting inability to cool the reactor core or the containment atmosphere could be melting of the core and/or loss of containment integrity.

One postulated means of losing the ability to draw water from the emergency sump could be blockage by debris. A principal source of such debris could be the thermal insulation on the reactor coolant system piping. In the event of a piping break, the subsequent violent release to the high pressure water in the reactor coolant system could rip off the insulation in the area of the break. This debris could then be swept into the sump, potentially causing blockage.

Currently, regulatory positions regarding sump design are presented in Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," which address debris (insulation). Regulatory Guide 1.82 recommends, in addition to providing redundant separated sumps, that two protective screens be provided. A low approach velocity in the vicinity of the sump is required to allow insulation to settle out before reaching the sump screening; and it is required that the sump remain functional assuming that one-half of the screen surface area is blocked.

A second postulated means of losing the ability to draw water from the emergency sump could be abnormal conditions in the sump or at the pump inlet such as air entrainment, vortices, or excessive pressure drops. These conditions could result in pump cavitation, reduced flow and possible damage to the pumps.

Currently, regulatory positions regarding sump testing are contained in Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors," which addresses the testing of the recirculation function. Both in-plant and scale model tests have been performed by applicants to demonstrate that circulation through the sump can be reliably accomplished.

As indicated in Section 6.3.3 of this Safety Evaluation Report, the applicant will perform out-of-plant scale model tests of the containment sump design. The applicant will be required to demonstrate that there is reasonable assurance that the sump design will perform as expected following a loss-of-coolant accident.

The near term implementation of Task-A-43 for this facility is expected to be procedural in nature and assure adequate housekeeping and emergency procedures to supplement the sump tests discussed above. Accordingly, we have concluded that this facility can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

A-44 Station Blackout

Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes an offsite alternating current power connection, a standby emergency diesel generator alternating current power supply, and direct current sources.

Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all alternating current power, i.e., a loss of both the offsite and the emergency diesel generator alternating current power supplies. A loss of all alternating current for an extended period of time in pressurized water reactors accompanied by loss of the auxiliary feedwater pumps (usually one of two redundant pumps is a steam turbine driven pump that is not dependent on alternating current power for actuation or operation) could result in an inability to cool the reactor core, with potentially serious consequences. This particular accident sequence was a significant contributor to the overall risk associated with the pressurized water reactor analyzed in the Reactor Safety Study (WASH-1400). The steam turbine driven auxiliary feedwater pump for the pressurized water reactor analyzed in WASH-1400 had no alternating current power dependencies. If the auxiliary feedwater pumps are dependent on alternating current power to function, then a loss of all alternating current power could of itself result in an inability to cool the reactor core and accordingly, this event sequence would be expected to be more important to the overall risk posed by the facility.

A loss of all alternating current power was not a design basis event for this facility. Nonetheless, the combination of design, operation, and testing requirements that have been imposed on the applicant will assure that these units will have substantial resistance to a loss of all alternating current and that even if a loss of all alternating current power should occur there is reasonable assurance that the core will be cooled. These are discussed below.

A loss of offsite alternating current power involves a loss of both the preferred and backup sources of offsite power. Our review and basis for acceptance of the design, inspection, and testing provisions for the offsite power system are described in Section 8.2 of this Safety Evaluation Reprot.

If offsite alternating current power is lost, two diesel generators and their associated distribution systems will deliver emergency power to safety-related equipment. Our review of the design, testing, surveillance, and maintenance provisions for the onsite emergency diesel generators isdescribed in Section 8.3 of the Safety Evaluation Reprot. Our requirements include preoperational testing to assure the reliability of the installed diesel generators in accordance with our requirements discussed in the Safety Evaluation Report.

Even if both offsite and onsite alternating current power are lost, cooling water can still be provided to the steam generators by the auxiliary feedwater system by employing a steam turbine driven pump that does not rely on alternative current power for operation. Our review of the auxiliary feedwater system design and operation is described in Section of the Safety Evaluation Report.

Based on our review, we have concluded that there is reasonable assurance that this facility can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

University of California

LOS ALAMOS SCIENTIFIC LABORATORY

Post Office Box 1663 Los Alamos, New Mexico 87545

In reply refer to: G-2-740-80
Mail stop: 978

December 24, 1980

Dr. Robert E. Jackson, Chief
Geosciences Branch
Division Site Safety and Environmental Analysis
U.S. Nuclear Regulatory Commission, MS P-314
Washington, DC 20555

Dear Dr. Jackson:

This letter reviews the seismic hazard assessment made by the South Carolina Electric & Gas Company (SCE&G) for the engineering design of the Virgil C. Summer Nuclear Station near Columbia, SC. This review is based on material specified by the NRC Standard Review Plans sections 2.5.1, 2.5.2, and 2.5.3 (Basic Geologic and Seismic Information, Vibratory Ground Motion, and Surface Faulting). The LASL Staff preparing this review has only reviewed docketed material and compared those with open literature. No attempt has been made to provide independent research on any of the review topics.

The primary source material for assessing the site seismic hazards has been the Final Safety Analysis Report submitted by SCE&G to the NRC, including amendments through 21. We have also evaluated the SCE&G responses to NRC questions and the Supplemental Seismological Investigation submitted December, 1980. Material supporting the conclusions of this review have been published papers, open file reports, and contract reports (especially, Study of Reservoir Induced Seismicity by Duane R. Packer, Lloyd S. Cluff, Peter L. Knuepfer, and Robert J. Withers; Woodward Clyde Consultants final technical report to the U.S. Geological Survey, August 1979). We believe that SCE&G has obtained and docketed sufficient data to make a seismic hazard assessment for which we can be highly confident. To whatever degree our assessments may differ, the causes lie in the interpretations applied to those data.

Our review has been especially focused on the reservoir induced seismicity (RIS) hazard assessment. The RIS problem appeared shortly after filling began in Lake Monticello in December 1977, many months after the construction permit for V. C. Summer was issued.

We recognize the importance of the 1886 Charleston, SC, earthquake to the earthquake hazard assessment for the V. C. Summer site. Because the U.S. Geological Survey has been studying the causes of seismicity in the Charleston seismic zone from several kinds of data, their conclusions should be preferred to the ones expressed herein if there are differences.

Previous meetings between the LASL staff and the SCE&G staff have been a 21 February 1980 FSAR review and site tour and three meetings in the NRC Bethesda offices, 30 July 1980, 9 October 1980, and 18 November 1980. The last three meetings were convened to discuss the NRC questions and the SCE&G responses to them.

The following abbreviations and terms will be used to shorten this letter:

SCE&G - also referred to as the utility or the applicant
SSE - safe shutdown earthquake
MMI - modified Mercalli intensity
DRS - design response spectrum
USGS - United States Geological Survey
SRP - U.S. NRC Standard Review Plan, NUREG-75/087
OBE - operating basis earthquake

The SCE&G has made the following seismic hazard assessment in arriving at the SSE design response spectrum:

1. The greatest MMI that will occur at the site during the planned plant lifetime of 40-50 years is VII.

2. The largest Lake Monticello RIS event will be $M_L = 4.0$.

3. Peak horizontal accelerations of 0.15 g in rock and 0.25 g in soil are conservative SSE values.

This review will consider the following topics, with emphasis given to information obtained since the writing of the construction permit (CP) Safety Evaluation Report (SER):

1. Geologic investigation of regional and local seismic hazards.

2. Geophysical surveys of buried structures.

3. Historic seismicity catalog for MMI $\geq$ IV.

4. Non-RIS instrumental seismicity catalog for $M_L > 3.0$.

5. Hazards posed by Fairfield Dam and Lake Monticello.

6. Potential for maximum earthquake.

   a. Geologic structure

   b. Tectonic province

   c. Lake Monticello RIS

7. Selection of Design Response Spectrum.

1. Geologic Investigation of Regional and Local Seismic Hazards

The FSAR position is that no capable faults are found within 250 miles of the V.C. Summer site. We find no suggestions in the literature we have read on this subject that would lead to any suspicion of this observation. Indeed, the youngest fault movements dated in the region occurred tens of millions of years ago. Recently proposed is the plate tectonic theory that describes the crystalline Paleozoic rocks in the region as being a part of a 6 to 15 km thick sheet that was thrust up to 260 km over younger Paleozoic sediments. The thrusting apparently took place in several episodes between 200 and 500 million years ago. It thus may be that nearly all of the seismic motions in the region occurred along deeply buried horizontal surfaces and that the surface faults reflect minor adjustments in an otherwise coherent plate motion. Given the present-day in-situ stress regime, it seems highly unlikely that any of the deep faulting that is symptomatic of this model can occur in many lifetimes of the V.C. Summer plant.

Geologic mapping studies being done by the University of South Carolina within the four quadrangles surrounding the site have found evidence of a previously unrecognized fault striking in the direction of the V. C. Summer plant. Unless future findings demonstrate otherwise, this Wateree Creek fault should not be considered capable or to exist in a region of RIS.

2. Geophysical Surveys of Buried Structure

The magnetic and gravity surveys reported on in the FSAR lack the resolution to reveal anything but very gross features. The COCORP seismic-reflection profiling in Georgia near the South Carolina border has detected many structural discontinuities from the Brevard Zone to the Modoc fault, many of which are within 100 miles of the Summer site. A reasonable inference may be drawn from the regional epicenter maps in the FSAR and elsewhere that nearly all of these features are aseismic. Possibly the RIS is associated with small scale anomalous features, but the evidence is too weak to draw such a conclusion. However, it is our opinion that the best seismic hazard assessment for most sites in the Southeastern U.S. will be inferred from seismicity observations, and so the poor resolution of the geophysical surveys presented in the FSAR does not negatively impact the quality of the assessment made.

3. Historic Seismicity Catalog for MMI $\geq$ IV

Historic earthquakes have been felt at the site with greater intensity than any that have o[illegible]red in the last 15 years when instrumental magnitude dete[illegible]ions have been possible. None has been reported to date that [illegible]ater significance for design accelerations than the Union Count[illegible] earthquake and the Charleston 1886 earthquake. The maximum intensity for the 1913 event seems to be correctly stated as VII in the FSAR. From the damage described, intensity VIII would be a great exaggeration. (The intensity VIII for this event appearing in articles by Bollinger is a Rossi-Forel intensity which is equivalent to MMI = VII.) The maximum intensity reported for the 1886 event of MMI = X also matches the observations well and is the commonly accepted value.

4. Non-RIS Instrumental Seismicity Catalog for $M_L > 3.0$

Several instrumentally recorded earthquakes have possible significance to assessing the seismic hazard to the site. Some of these events were not discussed in the FSAR. There were several earthquakes in the center of South Carolina in 1971 and 1972; the largest had $M_L = 4.5$. This area had no record of earthquakes prior to this time, and it has raised the question of the existence of a buried weakness linked to an $M_L = 4.7$ earthquake at Summerville, SC, in 1974 and to the 1886 Charleston earthquake (Talwani, Amick and Logan, 1979). At this point, the existence of such a linear feature cannot be highly regarded when one considers the separation of epicenters and a calculated source dimension of 1250 m for an $M_L = 4.5$ earthquake. Additional evidence for rejecting this hypothesis comes from geophysical observations in the coastal plain.

Another earthquake which seems more significant is the $M_L = 4.9$ (?) event in August 1974 near Clark Hill reservoir. (The National Earthquake Information Service, USGS, gave the magnitude as $m_b = 4.3$, which is equivalent to an $M_L \approx 3.4$ using Gutenberg's quadratic formula.) Clark Hill Reservoir is in the Piedmont, has similar site geology, has a dam height 5 m greater and reservoir volume 5-7 times greater than Fairfield Dam. Stress regimes based on focal mechanisms are extensional at both Clark Hill and Monticello according to the Woodward-Clyde RIS report. More recent data and analysis by Talwani shows that thrust mechanisms are most common at Monticello. The microseismicity at Clark Hill from 1952, when the dam was completed, until the 1974 macroseismicity, is unknown because the distance to the nearest seismograph was too great, and so there is some doubt about calling the macroseismicity "reservoir induced". Epicenters of Clark Hill microearthquakes have a diffuse pattern which is centered on an old fault of uncertain dip and unknown offset and extent. It is possible that some of the microearthquakes are occurring on the old fault, but most appear to occur along contacts between metamorphic rock units. The significance of the Clark Hill Reservoir seismicity to the seismic hazard assessment at the V.C. Summer site will be discussed.

5. Hazards Posed by Fairfield Dam and Lake Monticello

That the microseismicity around Lake Monticello is reservoir-induced is reasonably certain; however, correlations between microearthquake time history (after the initial filling) and reservoir water levels and fluctuations are quite low. Although the largest events, $2.5 < M_L < 2.9$, have triggered a strong motion accelerograph (SMA) located on a shared abutment of Fairfield Dam, the SMA instrumentation operating within the Summer station has never triggered. (All of the SMA instruments within the plant should have triggered for accelerations as great as 0.01 g.) At least one of the accelerograms has high frequency acceleration components which exceeded the DRS zero-period acceleration.

At present, the most commonly used model for RIS is the one which has the reservoir raising pore fluid pressures in the underlying rock, which results in reducing effective normal stresses across faults subject to prestresses that are near failure criteria, causing seismic slip to occur. Hypocenters computed from seismograms recorded on magnetic tape from the local seismic network have yet to indicate the presence of any lineations that might be faults capable of $M_L > 4.0$ earthquakes.

6. Potential for Maximum Earthquake

The SRP guidelines provide the procedure for obtaining the SSE and OBE design response spectra for sites, such as the one being reviewed here, where the historic earthquakes have been located where no faulting is observable, and where the largest events have not been recorded by seismographs. There are no guidelines established for consideration of reservoir-induced seismicity, and so we are giving it special consideration.

a. Geologic Structure

We have not found any reports, nor have we seen any evidence, that would cast doubt upon the FSAR conclusion that there are no capable faults found within 200 miles of the site.

b. Tectonic Province

We find no justification for moving the 1886 Charleston earthquake epicentral intensity outside of the "Charleston seismic zone" where more recent seismicity is concentrated. Furthermore, we believe that the probability analysis of the tectonic hypotheses for the Charleston seismicity, presented in the Supplemental Seismologic Investigation (December 1980), strongly supports that position.

Therefore, in accordance with 10 CFR Part 100, Appendix A, the Union County earthquake of 1913, which resulted in epicentral area damage of intensity VI to VII categories, should be migrated near the plant site to obtain the SSE and OBE design ground motions.

c. Lake Monticello Reservoir-Induced Seismicity

Nearly all cases of reservoir-induced macroseismicity have been associated with preexisting faults. The only cases of RIS that I know of where no causative fault has been identified have been reservoirs like Lake Monticello where only microseismicity occurs. If we were to migrate the maximum RIS Piedmont province earthquake to the site as is done for the maximum tectonic earthquake for establishing conservative peak design accelerations, then, assuming the 1974 Clark Hill earthquake was induced, the maximum possible sustained vibratory ground motions so obtained are smaller than the SSE design levels. The expected spikes of greater acceleration levels than the DRS will be discussed under review topic 7.

Another approach to estimating the largest induced earthquake to occur near Lake Monticello during the plant life of 40-50 years is to extrapolate the recorded RIS. The utility has estimated future RIS for Monticello Reservoir based on the magnitude-frequency relationship of approximately 2 years of data. Although an $M_L = 3.4$ earthquake is calculated to have a return period of 2 years, the largest event recorded in nearly 3 years is an $M_L = 2.9$ earthquake. Statistically, the largest earthquake forecasted to occur every 50 years is $M_L = 4.45$. In spite of their argument that local geologic factors may limit the maximum earthquake to the order of $M_L = 3$, we believe that uncertainties in assessing those factors require a conservative estimate of an $M_L = 4.5$ earthquake occurring during the life of the plant. We note that this approach gives the same maximum magnitude induced earthquake as the estimate based on the maximum suspected Piedmont RIS event.

7. Selection of Design Response Spectrum

Since the utility received a construction permit, there have been several studies that have found formal relationships among magnitude, maximum intensity, and peak acceleration. The utility has applied the Brune model and formulas from McGuire and Hanks (1980) to calculate a peak acceleration for the maximum tectonic earthquake. Although the latter was done in response to a question submitted by the NRC, the results are overly conservative.

Nuttli, Bollinger and Griffiths (1979) have found more accurate ways of determining magnitudes of historic seismicity where isoseismal maps exist, but instrumental recordings do not. Based on their study, I find that the largest historical earthquakes in the Piedmont and Coastal Plain provinces south of Chesapeake Bay (excluding the Charleston seismic zone) have been $m_b = 4.5$ with an uncertainty of up to a half magnitude. This includes the Union County, SC, earthquake of 1913.

Applying recent formulas of Nuttli and others to find peak accelerations for all tectonic earthquakes of concern to the V.C. Summer site, I find that the utility's DRS is appropriately conservative.

Concerning the reservoir induced seismicity, the maximum event is expected to be no greater than the largest tectonic event for which the DRS was chosen. It is expected, however, that small, near-field earthquakes will generate acceleration spikes that may be twice the SSE design acceleration. The utility has shown in Appendix X of the Supplemental Seismologic Investigation that these acceleration spikes have practically no damageability.

Sincerely,

Carl A. Newton

Attach. References

kc: CRMO (2)
C. A. Newton
D. J. Cash
G-2 File

REFERENCES

McGuire, R. K. and T. C. Hanks, "RMS Accelerations and Spectral Amplitudes of Strong Ground Motion During the San Fernando, California Earthquake", Bull. Seis. Soc. Am. 70, 1907-1919 (October 1980).

Nattli, O. W., G. A. Bollinger and D. W. Griffiths, "On the Relation Between Modified Mercalli Intensity and Body-Wave Magnitude", Bull. Seis. Soc. Am. 69, 893-909 (June 1979).

South Carolina Electric & Gas, "Supplemental Seismological Investigation, Virgil C. Summer Nuclear Station Unit 1", USNRC Docket No. 50/395 (December 1980).

Talwani, P., D. C. Amick and R. Logan, "A Model to Explain the Intraplate Seismicity in the South Carolina Coastal Plain", abstract, EOS, Trans., Am. Geophys. Union 60, 311 (May 1979).

United States Department of the Interior

GEOLOGICAL SURVEY
RESTON, VA. 22092

OFFICE OF THE DIRECTOR

In Reply Refer To:
EGS-Mail Stop 106

December 30, 1980

Dr. Robert E. Jackson
Chief, Geosciences Branch
Division of Engineering
U.S. Nuclear Regulatory Commission
M.S. P-314
Washington, D.C. 20555

Dear Bob:

This is in response to your request for an update on our information concerning the occurrence of earthquakes similar to the Charleston, S.C., event of August 31, 1886. That earthquake is rated as a Modified Mercalli intensity IX-X and, as such, is the principal controlling event for the determination of engineering design of nuclear power plants and processing facilities in the southeastern United States. In the past, the seismic engineering input has been based upon an earthquake of similar intensity occurring in the vicinity of Charleston, S.C., but not occurring randomly throughout the entire Coastal Plain province.

During the past decade, there has been considerable research by the geologic community directed toward a better understanding of seismicity in the southeast and, in particular, toward identification of the structure that generated the 1886 earthquake. The spring 1978 regional meeting of the Geological Society of America in Chattanooga, Tennessee, devoted a half-day symposium to the discussion of recent investigations in the Charleston area. The USGS has published (1978) Professional Paper 1028, and is planning a second one, which describes the progress of its research at Charleston. In addition, new ideas about southeastern U.S. seismicity are being presented at meetings of geologic and seismologic societies and are being discussed in the various geological journals. As a result, several new working hypotheses have been presented. However, although geologic mapping, stratigraphic drilling, seismic reflection profiling, and gravity and magnetic surveys have been underway for several years, no direct correlation between structures and earthquakes has been possible. The only significant structure recognized in the 1886 earthquake meisoseismal area is a northeast-trending reverse fault called the Cooke Fault. This feature has been interpreted primarily from seismic reflection surveys as there is no surface expression at all. To date, however, no evidence has been presented that associates the Cooke Fault directly with the 1886 earthquake. In fact, the length of the fault, as presently known (15 km), does not appear sufficient for generating an earthquake of intensity MM-IX-X, if standard fault length-earthquake size relationships are used. However, until further research provides more definitive concepts of southeastern U.S. seismicity and of its fault length and history of movement, the Cooke Fault by virtue of its coincidence of location with the Charleston earthquake should remain as a candidate structure to associate with that earthquake. Consequently, it should be considered as having a potential for generating similar events in the future.

Recently, several other faults have been recognized in the Charleston-Summerville areas, but none appear to have the potential of generating the higher intensity earthquakes. Individual faults recognized in the area appear to have responded to stress fields of different orientation (compressional or extensional) rather than reverse movement, and have inferred slip movements either in a normal or reverse sense depending upon the direction of stress of a given time. Other faults mapped throughout the eastern Coastal Plain and Piedmont provinces have had similar histories but none have been proven to be active at the present time, particularly in context of capability according to NRC criteria.

During the course of recent investigations, various additional hypotheses on the probable causative mechanism of the 1886 earthquake have included models of association with mafic plutons, stress concentration along block interface, and projection of the Blake Spur Fracture Zone at depth (basement structure).

Currently, there is much discussion of an interpretation of structure that hypothesizes a large east dipping decollement extending from Georgia northeastward to New England (Harris, L. D., et al, 1979) (Cook, F. A., et al, 1979). The decollement has been interpreted primarily from deep seismic reflection surveys done by COCORP (Schilt, F. S., et al, in prep.). Whether the decollement extends under the Charleston region is controversial (Hamilton, R. M., et al, in prep.). Present day seismicity ranges in depth from 3 to 13 km, with two-thirds of the events in the 5-8 km range. Composite focal mechanism solutions indicate sub-horizontal nodal planes as one of the possible orientations. However, there is no evidence which suggests that the generation of the present low-magnitude seismicity is related to the larger magnitude earthquakes such as the 1886 event. A structure such as the hypothesized decollement, however, should it exist and still be undergoing sufficient stresses to cause continued movement, may be capable of generating a large earthquake.

However, where the hypothesized decollement projects to the surface along the western margin of the valley and ridge province, there has been no recognition of quaternary surface rupture. In addition, this zone is characterized by a relatively low level of seismicity.

The problem regarding identification of specific tectonic structures capable of generating large earthquakes in the east is far from resolution. Local structures near Charleston are incompletely known at present and the larger structural element, the decollement, is as yet hypothetical. However, the concentration of seismicity in the Charleston earthquake epicenter both before and after the August 31, 1886, event and the lack of post Miocene faulting in the Coastal Plain or any evidence for localizing large earthquakes indicate that the likelihood of a Charleston sized event in other parts of the Coastal

Plain and Piedmont is very low. Consequently, earthquakes similar to the 1886 event should be considered as having the potential to occur in the vicinity of Charleston and seismic engineering parameters should be determined on that basis.

The research on the causative mechanism of the Charleston and other east coast earthquakes must continue if a more definitive resolution of this problem is to be obtained.

Sincerely yours,

James F. Devine

James F. Devine
Assistant Director for
Engineering Geology

Enclosure

Selected References

U.S. Geological Survey Professional Paper 1028. Studies related to the Charleston, South Carolina Earthquake of 1886 - a preliminary report.

Hamilton, Robert M., J. C. Behrendt, and H. D. Ackerman, Land Multichannel Seismic-Reflection Evidence for Tectonic Features near Charleston, South Carolina (in preparation).

Seeber, L., and J. G. Armbruster, Lamont Geological Observatory, The 1886 Charleston, South Carolina Earthquake and the Appalachian Detachment (preprint).

Harris, L.D., and K. C. Bayer, Sequential development of the Appalachian orogen above a master decollement - a hypothesis, Geology 7, pp. 568-572, 1979.

Wentworth, Carl M., and Marcia Mergner-Keefer, Regenerate faults of small Cenozoic offset: probable earthquake sources in the southeastern U.S. (in preparation).

Cook, F. A., Albaugh, D. S., Brown, L. D., Kaufman, S., Oliver, J. E., and Hatcher, R. D., Jr., 1979, Thin-skinned tectonics in the crystalline southern Appalachians; COCORP seismic-reflection profiling of the Blue Ridge and Piedmont, Geology, v. 7, p. 563-567.

Schilt, F. S., Brown, L. D., Oliver, J. E., and Kaufman, S., 19__, New evidence of recent faulting near Charleston, South Carolina: results of COCORP reflection profiling in the Atlantic Coastal Plain: this volume.

APPENDIX F

EMERGENCY PREPAREDNESS EVALUATION REPORT
FOR
VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

1.0 Introduction

Evaluation by the Nuclear Regulatory Commission (NRC) of the state of emergency preparedness associated with the Virgil C. Summer Nuclear Station, Unit 1 involves review of the applicant's emergency preparedness and the Federal Emergency Management Agency's findings on State and local radiological emergency preparedness. This evaluation addresses the applicant's emergency preparedness. In a supplement to this Safety Evaluation Report, we will address the findings and determinations of the Federal Emergency Management Agency on the adequacy of the State and local emergency response plans and the staff's overall conclusions on the status of emergency preparedness associated with the facility.

The applicant, South Carolina Electric & Gas Company, by letter dated October 13, 1980, filed with the NRC a comprehensive revision to the Virgil C. Summer Nuclear Station, Unit 1 Radiation Emergency Plan (hereinafter referred to as the emergency plan). We have reviewed this revised emergency plan. Previously, we reviewed preliminary versions of the emergency plan, visited the facility, and held a public meeting on emergency preparedness.

The emergency plan was reviewed against the 16 planning standards in Section 50.47 of 10 CFR Part 50, the requirements of Appendix E to 10 CFR Part 50, and the specific corresponding criteria of NUREG-0654/FEMA-REP-1 entitled "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, November 1980.

This evaluation follows the format of Part II of NUREG-0654. Each of the planning standards is listed and followed by a summary of applicable portions of the emergency plan that relate principally to that specific standard. The conclusions of our review are provided in Section 3.0 of this Appendix to the Safety Evaluation Report.

2.0 Evaluation of Applicant's Emergency Plan

2.1 Assignment of Responsibility (Organization Control)

2.1.1 Standard

Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the emergency planning zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

2.1.2 Evaluation

When an emergency condition arises, the shift supervisor will be designated as the Interim Emergency Director and it will be the shift supervisor's responsibility to evaluate the situation. If, in the shift supervisor's judgment, conditions meet or exceed any of the emergency classification action levels, it will be the shift supervisor's responsibility to implement the emergency plan. There will be a 24-hour per day communication linkage capability between the facility and Federal, State, and local response agencies and organizations to assure rapid transmittal of accurate notification information and emergency assessment data.

The authority, responsibility, and duties of the plant staff personnel for coping with emergencies are clearly defined for both the normal operating staff and the augmented staff. The operational relationships between the onsite emergency centers and offsite agencies are identified. The Emergency Control Officer will be responsible for assuring continuity of the applicant's resources and overall management of the emergency and recovery operation.

The emergency plan describes the functions and responsibilities of each State and local organization with response roles. The South Carolina Emergency Preparedness Division - Adjutant General's office will be responsible for coordinating emergency response action decisions affecting the general public with the State and involved local governments. The South Carolina Department of Health and Environmental Control, Bureau of Radiological Health, will be responsible for initiating the State emergency plan and for offsite radiological monitoring and assessments. The Fairfield County Disaster Preparedness Agency, Newberry County Public Safety Department, Richland County Civil Defense Agency, and Lexington County Public Safety Department will be responsible for implementing protective actions in their respective counties.

In the event of an emergency, the applicant will contact the State Emergency Operations Center and the Fairfield County, Newberry County, Richland County, and Lexington County Emergency Operation Centers by dedicated phone linkup. These facilities will be manned on a 24-hour per day basis. Arrangements have been made for the counties to accomplish protective actions based upon the applicant's protective action recommendations.

Updated written agreements have been executed with Federal and local agencies and organizations to provide for radiological support, medical assistance, medical transportation, and fire protection during an emergency.

The applicant has committed to provide the emergency plans for South Carolina, Fairfield County, Newberry County, Richland County, and Lexington County and a written agreement with Lexington County in a forthcoming revision to the emergency plan.

2.2 Onsite Emergency Organization

2.2.1 Standard

On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and the interfaces among various onsite response activities and offsite support and response activities are specified.

2.2.2 Evaluation

The Shift Supervisor, designated as the Interim Emergency Director, has the responsiblity and authority to implement the emergency plan and initiate any necessary emergency actions. The shift supervisor will be relieved by the Station Manager, designated the Emergency Director, after that individual arrives onsite and becomes thoroughly cognizant of the situation. The emergency director will operate from the technical support center. The Emergency Director will not delegate the responsibility to notify and make protective action recommendations to offsite authorities.

Plant staff emergency assignments have been made and the relationship between the emergency organization and normal staff complement are specified and illustrated in the emergency plan. Positons and/or titles of shift and plant personnel (both onsite and offsite) assigned emergency functional duties are listed. The shift and augmented staffing specified in the emergency plan approximate the specific staffing requirements expressed in Table B-1 of NUREG-0654. However, the emergency plan does not meet the staffing requirements pertaining to augmentation capability within 30 minutes, the on-shift rad/chem technician, and the on-shift mechanical maintenance capability.

The applicant's corporate personnel, under the direction of the Emergency Control Officer, will provide assistance to the onsite organization from the emergency operations facility. Interfaces between and among the applicant's onsite and offsite organizations and governmental and private sector organizations have been specified.

The following items require resolution:

1. The staffing requirements in Table B-1 of NUREG-0654 must be satisfied.

2. The emergency plan must clearly specify that the Interim Emergency Director has the authority and responsibility to immediately and unilaterally provide protective action recommendations to offsite authorities.

2.3 Emergency Response Support and Resources

2.3.1 Standard

Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's near-site emergency operations facility have been made, and other organizations capable of augmenting the planned response have been identified.

2.3.2 Evaluation

The emergency operations facility will be placed in a standby status for the emergency "Alert" emergency classification and will be actuated for a "Site Emergency" or "General Emergency". Provisions have been made to accommodate representatives from Federal, State, and local government organizations and from contractor and other support groups. It will be the central location for collecting and providing information and making recommendations for offsite protective actions.

Request for support under the Federal Radiological Monitoring and Assessment Plan will be coordinated through the South Carolina Department of Health and Environmental Control, Bureau of Radiological Health. An updated written agreement with the Department of Energy pertaining to the Federal response has been completed.

In addition to the plant's laboratory facilities, a backup laboratory will be available at the applicant's Parr Steam and Hydroelectric facility, two miles from the plant, which can be operational within one hour of an accident. The South Carolina Bureau of Radiological Health maintains a laboratory facility at Columbia, South Carolina and has a mobile laboratory which can be actuated within three hours for analyzing environmental samples.

The applicant has not committed to dispatch a representative to the principal offsite governmental emergency operations centers and has not provided sufficient information concerning support of the Federal response. These items must be addressed in a revision to the emergency plan.

2.4 Emergency Classification System

2.4.1 Standard

A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

2.4.2 Evaluation

The four standard emergency classes (i.e., unusual event, alert, site emergency, and general emergency) have been established by the applicant. Observable and measurable emergency action levels have been established which, if exceeded, will initiate each emergency class, consistent with the criteria of Appendix 1 to NUREG-0654. Emergency action levels are provided using specific instrumentation, parameters, and equipment status. Emergency plan procedures contain specific information and guidance for determining the appropriate emergency action level and properly classifying the emergency condition, as well as the appropriate actions to be taken.

The applicant must make several clarifications to the classifications and emergency action level section of the emergency plan and provide the aforementioned procedures for staff review.

2.5 Notification Methods and Procedures

2.5.1 Standard

Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all response organizations; the content of initial and followup messages to response organizations and the public have been established; and means to provide early notification and clear instructions to the populace within the plume exposure pathway emergency planning zone have been established.

2.5.2 Evaluation

The emergency plan and associated procedures establish and describe a notification and verification system which is consistent with Appendix 1 to NUREG-0654. The system provides for notification of the South Carolina Bureau of Radiological Health for each class of emergency, and for notification of Fairfield, Newberry, Richland, and Lexington counties for "Alert", "Site Emergency", and "General Emergency".

The following items require resolution:

1. Prompt notification of offsite authorities as set forth in Appendix 1 (i.e., 15 minutes or less) is not satisfied.

2. Emergency messages, both initial and followup, which are to originate from the applicant are not included or addressed.

3. Notification and prompt instructions to the public within the plume exposure pathway emergency planning zone is not addressed.

2.6 Emergency Communications

2.6.1 Standard

Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

2.6.2 Evaluation

Primary and backup communication links will be provided with the Federal, State, and local emergency response organizations. Provisions exist for 24-hour per day notification to and actuation of these organizations. Offsite communications systems include commercial telephone, private telephone lines, dedicated telephone lines, radio systems, and a microwave system.

Communications with the State and local governments will be tested monthly, and communications with Federal response organizations, the State and local emergency operations centers, and field monitoring teams will be tested annually as part of the communication drills.

2.7 Public Information

2.7.1 Standard

Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency; the principal points of contact with the news media for dissemination of information during an emergency (including physical location or locations) are established in advance; and procedures for coordinated dissemination of information to the public are established.

2.7.2 Evaluation

The applicant will institute a public education program of the public within the plume exposure pathway emergency planning zone. The program provides for the annual update of information provided and an annual survey to determine the effectiveness of the program. The program will be designed to assure that the population will be: (1) educated in general radiation health, (2) able to recognize radiological emergency notification (e.g., sirens), and (3) knowledgeable of the proper protective actions.

The applicant will conduct annual training for personnel of the news media which will acquaint these persons with the emergency plan, information concerning radiation, and points of contact for release of public information during an emergency.

The applicant has committed to establish a news media area in close proximity to the emergency operations facility, with equipment and facilities adequate to support media representatives. The Media Coordinator will be responsible for disseminating information to the public via the news media. The Media Coordinator will hold press conferences and release information approved by the offsite emergency coordinator.

The following items require resolution:

1. The public education program must address special needs of the handicapped and the actual means to be utilized in disseminating information to the public must be described in the emergency plan.

2. The public education program must provide for the transient adult population within the plume exposure emergency planning zone.

3. Additional information pertaining to the news media area capability is necessary.

2.8 Emergency Facilities and Equipment

2.8.1 Standard

Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

2.8.2 Evaluation

Emergency facilities to support an emergency response have been established as follows:

1. Technical Support Center

   The technical support center will be located in the control building in close proximity to the control room. The role of the technical support center will be to provide command and control functions, provide dose assessment, and provide technical assistance to the control room and emergency operations facility staffs. The Emergency Director and support staff will be located at the technical support center.

2. Operational Support Center

The operational support center will be located in the control building next to the control room. The purpose of the operational support center is to provide an assembly area for emergency team personnel to muster.

3. Interim Emergency Operations Facility

The interim emergency operations facility will be located at the administrative office annex approximately 1000 feet from the reactor building. From the interim emergency operations facility, the offsite emergency organization will evaluate and coordinate all of the applicant's activities related to the emergency and will interface with Federal, State, and local organizations.

4. Backup Emergency Operations Facility

The backup emergency operations facility will be located at the applicant's Parr facility, approximately two miles from the reactor building.

The technical support center and operational support center will be activated for an "Alert" condition and the emergency operation facility will be brought to a standby status. The technical support center, operational support center, and emergency operation facility will be activated for "Site Emergency" and "General Emergency" conditions.

The emergency plan describes the following means used to initiate and assess emergencies: (1) meteorological instrumentation, (2) radiological monitors to include field survey monitors, (3) process monitors, (4) fire detection devices, (5) an environmental radiological monitoring program, and (6) laboratory facilities. The description of the instrumentation identified in the emergency action levels includes location, type, alarms, setpoints, and range.

Meteorological instrumentation is located on a 61-meter self-supporting tower. Meteorological data is recorded in the control room.

The emergency plan contains a summary of emergency equipment and supplies and a listing of radiological monitoring equipment. Emergency plan procedures specify the calibration and maintenance of emergency equipment.

The following items require resolution:

1. The emergency facilities must meet the criteria of NUREG-0696, Revision 1.
2. The environmental radiological monitoring program must be consistent with the NRC Radiological Assessment Branch Technical Position.
3. The meteorological criteria of Appendix 2 to NUREG-0654 must be satisfied.

2.9 Accident Assessment

2.9.1 Standard

Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

2.9.2 Evaluation

The emergency plan identifies specific instrument readings and other observable and measurable parameters which, if exceeded, will initiate an emergency as discussed in Section 2.4 of this Appendix to the Safety Evaluation Report. The estimation of doses, both onsite and offsite, will be accomplished by a real time system, the dose assessment and measurement system. The system receives automatic inputs of relevant parameter values and has provisions for manual entry of readings. The system provides rapid evaluation of parameter inputs to assist in determining protective action recommendations.

The dose projections and their relationship to the Environmental Protection Agency Protection Action Guides are used as part of the accident classification system.

The emergency plan describes the applicant's field monitoring teams which can be functioning within approximately 15 minutes.

The following items require resolution:

1. The method and technique for determining the source term and the magnitude of the release must be addressed.

2. The relationship between effluent monitor readings and exposures and contamination must be addressed.

3. The capability of acquiring and evaluating meteorological information specified by Appendix 2 to NUREG-0654 must be provided.

4. The methodology to determine release rate/projected doses if instrumentation is offscale/inoperable must be described.

5. The means for relating measured parameters to dose rates for key isotopes and for estimating integrated doses must be described.

2.10 Protective Response

2.10.1 Standard

A range of protective actions have been developed for the plume exposure pathway emergency planning zone for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway emergency planning zone appropriate to the locale have been developed.

2.10.2 Evaluation

The emergency plan establishes guides for determining when protective actions are required onsite to include evacuation, distribution of radio-protective drugs, and the use of respiratory protection. Monitoring and decontamination of onsite evacuees will be conducted at the designated assembly area. The storage locations for emergency equipment and supplies are specified in the emergency plan. Personnel accountability will be performed at the assembly area within 30 minutes of the emergency. Provisions are made for transportation and evacuation routes for onsite personnel.

The emergency plan provides for the prompt notification and recommendation of protective actions to State and local authorities for the population-at-risk in the plume exposure pathway emergency planning zone. Time estimates for evacuation within the plume exposure pathway emergency planning zone are provided.

The following items require resolution:

1. The warning of contractor/construction personnel and other persons within the owner-controlled area must be clarified.

2. The provisions for evacuation of onsite personnel must be clarified.

3. Evacuation time estimates must satisfy the criteria of Appendix 4 to NUREG-0654.

4. Maps pertaining to evacuation and maps illustrating population distribution must be provided.

5. The means for notifying the transient population must be provided.

6. Expected protection in residential units for direct and inhalation exposure must be addressed.

2.11 Radiological Exposure Control

2.11.1 Standard

Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with Environmental Protection Agency Emergency Workers and Lifesaving Activity Protective Action Guides.

2.11.2 Evaluation

The Emergency Director will authorize any potential personnel exposure exceeding the occupational limits of 10 CFR Part 20 and the emergency plan identifies onsite exposure guidelines consistent with Environmental Protection Agency guidelines for emergencies. The emergency plan provides for personnel decontamination facilities and emergency plan procedures identify the action levels requiring decontamination. The emergency plan provides for 24-hour per day determination of doses received by onsite emergency workers and offsite response personnel and for appropriate record keeping.

The following items require resolution:

1. Provisions for the distribution and reading of personnel dosimetry devices must be clarified.

2. The action levels indicating tne need for decontamination and the means for radiological decontamination must be addressed in the emergency plan.

3. The contamination control measures must be more completely described in the emergency plan.

2.12 Medical and Public Health Support

2.12.1 Standard

Arrangements are made for medical services for contaminated and injured individuals.

2.12.2 Evaluation

The applicant has made arrangements with the Pinner Clinic (located approximately three miles from the plant) and the Richland Memorial Hospital (located in Columbia, South Carolina) to provide medical assistance to personnel injured or exposed to radiation and/or radioactive material. Transportation of victims will be provided by the Fairfield County Emergency Medical Service. Augmented transportation capability, including air rescue, can be provided by the U.S. Army Military Assistance for Safety and Traffic Operation. Onsite first aid capability will be provided by the applicant.

2.13 Recovery and Reentry Planning and Postaccident Operations

2.13.1 Standard

General plans for recovery and reentry are developed.

2.13.2 Evaluation

The emergency plan describes the applicant's general plans for recovery and reentry. The Emergency Control Officer will be responsible for determining the need for and aspects of the recovery plan and organization. Emergency plan procedures provide instructions for reentry activities.

The following items require resolution:

1. The composition of the recovery organization must be provided in the emergency plan.

2. The means for informing response organizations of recovery operations must be described in the emergency plan.

3. The method for periodically estimating total population exposure must be described in the emergency plan.

2.14 Exercises and Drills

2.14.1 Standard

Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

2.14.2 Evaluation

An emergency exercise will be conducted annually and will be based on an accident scenario postulating at least a "Site Emergency" condition. The scenario will be varied such that all plans and preparedness organizations are tested within a five-year period. One exercise will start between midnight and 6:00 a.m. and another between 6:00 p.m. and midnight once every six years.

The Emergency Coordinator will be responsible for the planning, scheduling, and coordinating of drills and exercises. The annual exercise will be approved by the General Manager, Nuclear Operations.

Each drill and exercise will be conducted to test the state of emergency preparedness and will be designed to meet a list of specific objectives which are specified in the emergency plan. The Emergency Coordinator will coordinate and implement revisions to the emergency plan and required corrective actions resulting from the drills and exercises.

Drills will be supervised instruction periods aimed at testing, developing, and maintaining skills. Drills will include the following:

1. Communication drills - initial plant contact with State and county governments will be tested monthly; communications with Federal response agencies, offsite emergency centers, and field assessment teams will be tested annually.

2. Fire drills - quarterly

3. Medical emergency drills -annually

4. Radiological monitoring drills - annually

5. Onsite radiation protection drills - semiannually.

The following items require resolution:

1. The emergency plan should address all of the criteria pertaining to exercises.

2. Radiological monitoring drills should include each of the specific criteria.

3. The emergency plan should specifically address the objectives for exercise and drill scenarios that are itemized in the criteria.

4. The emergency plan must specifically address the role of observers and the required critique for exercises.

5. The means for evaluating comments and implementing improvements in the emergency plan and associated procedures must be addressed in the emergency plan.

2.15 Radiological Emergency Response Training

2.15.1 Standard

Radiological emergency response training is provided to those who may be called upon to assist in an emergency.

2.15.2 Evaluation

The emergency plan provides for training and qualifying all personnel on the emergency tasks for which they are responsible as specified in the emergency plan. The Nuclear Training Coordinator will be responsible for coordinating the training of all plant personnel. The training and retraining of personnel comprising the offsite organization is to include details of the emergency plan, procedures relevant to the emergency operations facility, facilities at the emergency operations facility, and the role of offsite agencies and organizations.

The applicant will provide training and annual retraining for those offsite organizations whose services may be required in an emergency, such as fire, police, medical support, and rescue personnel. The training will be consistent with the organizations' emergency functions.

Selected station personnel on each shift will attend the multimedia National Red Cross First Aid Course.

The training program for members of the applicant's emergency organization will include practical drills as discussed in Section 2.14 of this Appendix to the Safety Evaluation Report.

2.16 Responsibility for the Planning Effort: Development, Periodic Review and Distribution of Emergency Plans

2.16.1 Standard

Responsibilities for plan development and review and for distribution of emergency plans are established and planners are properly trained.

2.16.2 Evaluation

The Emergency Planning Coordinator has the authority and responsibility for the applicant's emergency response planning. Changes to the emergency plan will be reviewed by the Emergency Coordinator and submitted to the Plant Manager and Plant Safety Review Committee for review and approval. The emergency plan, and revisions thereto, will be maintained and distributed under strict administrative controls.

The emergency plan and appended letters of agreement will be reviewed and updated on an annual basis and an audit, by the corporate Nuclear Safety Review Committee, will be performed every two years.

The following items require resolution:

1. Training must be provided for persons responsible for the planning effort.

2. The emergency plan audit should include review of training, readiness testing, and equipment, and should be accomplished annually.

3. The telephone numbers in emergency procedures should be updated at least quarterly.

3.0 Conclusions on the Applicant's Emergency Plan

Based on our review against the criteria in "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0654, Revision 1, November 1980, we conclude that the Virgil C. Summer Nuclear Station, Unit 1 Radiation Emergency Plan, upon satisfactory correction of those items requiring resolution as identified in Section 2.0 of this appendix of the Safety Evaluation Report, will provide an adequate planning basis for an acceptable state of emergency preparedness and will meet the requirements of 10 CFR Part 50 and Appendix E thereto.

After receiving the findings and determinations made by the Federal Emergency Management Agency on State and local emergency response plans and after reviewing the revision(s) to the applicant's emergency plan, we will supplement this Safety Evaluation Report to provide our overall conclusions on the status of emergency preparedness for the Virgil C. Summer Nuclear Station, Unit 1 and related emergency planning zones.

The final NRC approval of the state of emergency preparedness for the site will be made following implementation of the emergency plans to include development of procedures, training, and qualifying of personnel, installation of equipment and facilities, and a joint exercise of all the plans (applicant, State, and local).

NRC FORM 335 (7 77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER (Assigned by DDC) NUREG-0717
4. TITLE AND SUBTITLE (Add Volume No., if appropriate) Safety Evaluation Report Related to the Operation of Virgil C. Summer Nuclear Station, Unit No. 1	2. (Leave blank)
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S)	5. DATE REPORT COMPLETED MONTH February YEAR 1981
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Office of Nuclear Reactor Regulation U.S. Nuclear Regulatory Commission Washington, D. C. 20555	DATE REPORT ISSUED MONTH February YEAR 1981
	6. (Leave blank)
	8. (Leave blank)
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Same as 9 above	10. PROJECT/TASK/WORK UNIT NO.
	11. CONTRACT NO.
13. TYPE OF REPORT	PERIOD COVERED (Inclusive dates)
15. SUPPLEMENTARY NOTES Docket No. 50-395	14. (Leave blank)

16. ABSTRACT (200 words or less)

The Safety Evaluation Report for the application filed by South Carolina Electric & Gas Company for a license to operate the Virgil C. Summer Nuclear Station, Unit 1 (Docket No. 50-395), located in Fairfield County, South Carolina, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. Subject to favorable resolution of the items discussed in the Safety Evaluation Report, the staff concludes that the plant can be operated by the South Carolina Electric & Gas Company without endangering the health and safety of the public.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
17b. IDENTIFIERS/OPEN-ENDED TERMS	

18. AVAILABILITY STATEMENT	19. SECURITY CLASS (This report)	21. NO. OF PAGES
Unlimited	Unclassified	
	20. SECURITY CLASS (This page)	22. PRICE $