DRAFT SAFETY EVALUATION REPORT

for

VOGTLE ELECTRIC GENERATING PLANT
Units 1 and 2

November 1984

ABSTRACT

The Safety Evaluation Report for the application filed by Georgia Power Company, Municipal Electric Authority of Georgia, Olgethorpe Power Corporation, and City of Dalton, Georgia, as applicants and owners, for licenses to operate the Vogtle Electric Generating Plant, Units 1 and 2 (Docket Nos. 50-424 and 50-425), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Burke County, Georgia, approximately 41.5 km (26 mi) south-southeast of Augusta and on the Savannah River. Subject to favorable resolution of the items discussed in this report, the staff concludes that the applicant can operate the facility without endangering the health and safety of the public.

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1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

This report is the U.S. Nuclear Regulatory Commission (NRC) staff's Safety Evaluation Report (SER) on the application for an operating license (OL) for the Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle or the facility). On August 1, 1972, the Georgia Power Company filed an application with the Atomic Energy Commission (AEC) for a license to construct and operate the proposed facility consisting of four units. This application was docketed on February 13, 1973. The site is located in Burke County, Georgia, along the Savannah River.

The AEC, now the NRC (or Commission), reported the results of its preconstruction review in an SER dated March 8, 1974. Following a public hearing before an Atomic Safety and Licensing Board, Construction Permit (CP) Nos. CPPR-108, CPPR-109, CPPR-110 and CPPR-111 were issued on June 28, 1974. On September 12, 1974, Georgia Power Company cancelled Units 3 and 4.

The Georgia Power Company (hereinafter referred to as the applicant) acting as agent and representative for the owners (Municipal Electric Authority of Georgia, Oglethorpe Power Corporation, and City of Dalton, Georgia) tendered an application for an operating license for Vogtle, Units 1 and 2, by letter dated June 30, 1983, which included the Final Safety Analysis Report (FSAR) for the facility. When the NRC staff acceptance review was completed, the FSAR for Vogtle, Units 1 and 2, was docketed by a letter dated September 16, 1983. The applicant's Environmental Report (ER) was tendered separately from the OL application by letter dated August 31, 1983. Upon completion of the staff's acceptance review, the ER was docketed by letter dated November 30, 1983. The staff's review of the ER is contained in the Draft and Final Environmental Statements.

Before issuing an OL for a nuclear power plant, the NRC staff is required to conduct a review of the effects of the plant on public health and safety. The staff safety review of Vogtle, Units 1 and 2, has been based on NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Reactors, LWR Edition" (SRP). An audit review of each of the areas listed in the Areas of Review section of the SRP was performed according to the guidelines provided in the Review Procedures portion of the SRP. Exceptions to this practice are noted in the applicable sections of this report.

This SER summarizes the results of the staff's radiological safety review of Vogtle, Units 1 and 2, and delineates the scope of the technical details considered in evaluating the radiological safety aspects of its proposed operation. The design of the facility was reviewed against the federal regulations, CP criteria, and the SRP, except where noted otherwise. The SRP covers a variety of site conditions and plant designs. Each section is written to provide the complete procedure and all acceptance criteria for all of the areas of review pertinent to the section. However, for any given application, the staff may select and emphasize particular aspects of each SRP section as appropriate for the application. In some cases, the major portion of the review of a plant feature may be done on a generic basis, with the designer of that feature rather than in the context of reviews of individual applications from utilities. In other cases, a plant feature may be sufficiently similar to that of a previous plant so that a de novo review of the feature is not needed.

During the course of its review, the staff held a number of meetings with representatives of the applicant to discuss the design, construction, and proposed operation of the plant. The staff requested additional information, which the applicant provided in amendments to the FSAR. This information is available to the public for review at the NRC Public Document Room at 1717 H Street, N.W., Washington, D.C., and at the local Public Document Room at the Burke County Library, 4th Street, Waynesboro, Georgia 30830.

Following the accident at Three Mile Island Unit 2 (TMI-2), the Commission paused in its licensing activities to assess the impact of the accident. During this pause, the recommendations of several groups established to investigate the lessons learned from the TMI-2 accident became available. All available recommendations were correlated and assimilated into a TMI Action Plan, now published as NUREG-0660, entitled "NRC Action Plan Developed as a Result of the TMI-2 Accident." Additional guidance relating to implementation of the Action Plan is in NUREG-0737, "Clarification of TMI Action Plan Requirements," and in Supplement 1 to NUREG-0737. Licensing requirements based on the lessons learned from the TMI-2 accident have been established to provide additional safety margins. These have been incorporated into the design and operation of Vogtle, Units 1 and 2. Table 1.1 provides a cross-reference relating the TMI items to the sections in this report where they are discussed.

Sections 2 through 22 of this report contain the NRC review and evaluation of both the non-TMI- and TMI-related issues. Section 23 presents the staff's conclusions.

Appendix A is a chronology of NRC's principal actions related to the safety (or radiological) review of the application. Appendix B is a bibliography of the references used during the course of the review. Availability of all material cited in this report is described on the inside front cover of this report. Sections of Title 10 of the Code of Federal Regulations (10 CFR) (including the general design criteria (GDC) in Appendix A to Part 50), NRC regulatory guides (RGs), and sections of the SRP, including branch technical positions (BTPs), will be identified as appropriate. They are not included in Appendix B. Appendix C is a discussion of how various unresolved safety issues (USIs) relate to the application. Appendix D is a list of abbreviations and acronyms used in this report. Appendix E is a list of principal contributors.

As part of its review of the application against the NRC regulations, the staff will ask the applicant to certify that Vogtle, Units 1 and 2, meets the applicable requirements of 10 CFR 20, 50, 51, and 100. Following the applicant's response to this request, the staff will address its findings in this area in a supplement to this Safety Evaluation Report.

In accordance with the provisions of the National Environmental Policy Act (NEPA) of 1969, a Draft Environmental Statement (DES) (NUREG-1087) that sets forth the environmental considerations related to the proposed construction and operation of Vogtle, Units 1 and 2, was prepared by the staff and was published in November 1984. The Final Environmental Statement (FES) is scheduled to be published in March 1985 and will include a consideration of public comments received on the DES.

The review and evaluation of Vogtle, Units 1 and 2, for an operating license is only one of many stages at which the staff reviews the design, construction, and operating features of the facility. The facility design was extensively reviewed before the applicant was granted a construction permit for the facility. Construction of the facility has been monitored in accordance with a detailed monitoring and inspection program at the OL stage. The NRC staff has reviewed the final design of the facility to determine that the Commission's regulations 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 Commission's regulations, and the facility will be subject to the staff's continuing inspection program.

In addition to the NRC staff review, the Advisory Committee on Reactor Safeguards (ACRS) will review the application and will meet with both the applicant and the staff to discuss the final design and proposed operation of the plant. The Committee's report to the Chairman of the NRC will be included in a supplement to this SER.

The NRC Project Manager assigned to the OL application for Vogtle, Units 1 and 2, is Melanie A. Miller. Ms. Miller may be contacted by calling (301) 492-4259 or by writing

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1.2 General Plant Description

The major structures of the Vogtle facility include containment structures, equipment buildings, an auxiliary building, fuel-handling buildings, radwaste solidification and transfer buildings, an engineered safety features building, a main steam valve building, a turbine building, a service building, a control building, emergency diesel generator buildings, nuclear service cooling water towers, circulating water cooling towers, and auxiliary feedwater pumphouses.

The containment structure of each unit houses the nuclear steam supply system (NSSS). The NSSS incorporates a pressurized-water reactor and a four-loop reactor coolant system (RCS). Each loop contains a reactor coolant pump and steam generator, two-loop isolation valves, an isolation bypass valve, and a bypass line. The NSSS also contains an electrically heated pressurizer and auxiliary systems. The NSSS is designed for a power output of 3,411 MWt with a gross electrical output of 1,157 MWe.

The reactor is a low-alloy-steel vessel with interior stainless steel cladding. The reactor coolant piping and all of the pressure-containing and heat-transfer surfaces in contact with the reactor water are stainless steel or stainless steel clad except for the steam generator tubes, which are Inconel, and the fuel tubes, which are Zircaloy.

The reactor vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The core is composed of fuel rods made of slightly enriched uranium dioxide pellets contained in Zircaloy tubes pressurized with helium and fitted with welded end plugs. The fuel rods are grouped and supported in fuel assemblies. The fuel assemblies are initially loaded within the core using three different enrichments of U-235. In subsequent refuelings, one-third of the fuel is discharged from the central portion of the core and transferred to fuel storage. New fuel is loaded into the periphery of the core and the remaining fuel is arranged in the central two-thirds of the core in a checkerboard fashion to achieve optimum power distribution.

The reactor is controlled during operation by control rod movement and regulation of the concentration of boric acid, a neutron absorber, in the reactor coolant. Mechanical rod cluster control assemblies consist of stainless-steel-clad hafnium or silver-indium-cadmium neutron-absorber rods that are inserted in Zircaloy guide tubes located in certain fuel assemblies. The rod cluster control assemblies are attached to stainless steel drive shafts, which will be raised and lowered within the core by individual control rod drive mechanisms. The concentration of boron is varied to control reactivity changes that occur relatively slowly.

Water will serve as both the moderator and the coolant and will be circulated through the reactor vessel and core by four vertical, single-speed centrifugal pumps driven by water/air-cooled, three-phase induction motors. One reactor coolant pump is located in the cold leg of each loop. The reactor coolant will be heated by the core and circulated through four steam generators where heat will be transferred to the secondary system to produce steam for the turbine generator. The coolant will then be pumped back to the reactor to complete the cycle.

An electrically heated pressurizer connected to the hot-leg piping of one of the loops will maintain RCS pressure during normal operation, limit pressure variations during plant load transients, and keep system pressures within design limits during abnormal conditions. The pressurizer provides a surge chamber and a water reserve to accommodate changes in reactor coolant volume during operation.

The steam generators are vertical shell and U-tube evaporators, which contain Inconel tubes; they are Westinghouse Model F. The steam produced in the steam generators will be used to drive a tandem-compound, six-flow, 1,800-rpm turbine generator and will be condensed in a two-pass, three-shell, deaerating surface condenser. These components are housed in the turbine building. Condenser circulating water is drawn from the circulating water intake structure and supplied to the condenser by two half-capacity circulating water pumps. Upon exiting the condenser, the water flows to the discharge network in the hyperbolic natural draft cooling tower. Circulating water is pumped through the tubes of the condenser to remove heat from, and thus condense, the steam after it has passed through the turbine. The condenser is equipped with titanium condenser tubes, which resist corrosive action.

NSSS auxiliary components are provided to charge makeup water into the RCS, purify reactor coolant, provide chemistry for corrosion inhibition and reactivity control, cool system components, remove decay heat, provide for emergency safety injection, process wastes and provide containment ventilation and cooling.

An engineered safety features actuation system is provided that automatically initiates appropriate action whenever a condition monitored by the system approaches preestablished limits. This system will act to shut down the reactor, close isolation valves, and initiate operation of engineered safety features should any or all of these actions be required.

Supervision and control of both the NSSS and the steam and power conversion system will be accomplished from the Vogtle control room, located in the control building. The control room contains all instrumentation and control equipment required for startup, operation, and shutdown, including normal and accident conditions.

The emergency core cooling system (ECCS) is designed to cool the reactor core and to provide shutdown capability by injecting borated water following a loss of coolant accident. The ECCS also provides continuous long-term core cooling following an accident by recirculating borated water between the reactor core and the containment sump. The ECCS consists of safety injection accumulators, cha ging pumps, safety injection pumps, residual heat removal pumps and heat exchangers, containment recirculation pumps and coolers, boron injection tank, and the refueling water storage tank along with the associated piping, valves, instrumentation, and other related equipment. The active components of the ECCS are powered from separate safety-related buses, which are energized from offsite power supplies. In addition, emergency diesel generators ensure redundant sources of auxiliary onsite power in the event of a loss of offsite power. The emergency diesel generators are located within separate compartments in the emergency diesel generator building.

The containment structure housing the NSSS is a carbon-steel-lined, prestressed post-tensioned concrete cylinder and hemispherical dome. The containment is designed to withstand the internal pressure and temperature which result from the energy released in the event of a high-energy-line-break accident. The containment spray system consists of two redundant, full-capacity trains designed to reduce postaccident iodine concentrations inside containment so that offsite doses are within allowable limits. The containment cooling system is composed of two redundant, independent, full-capacity trains which contain equipment to facilitate postaccident safe shutdown of the facility.

Each unit is supplied with electrical power from two independent offsite power sources and is provided with independent and redundant onsite emergency power supplies capable of supplying power to engineered safety features.

Table 1.2 compares principal design features of Vogtle with those of similar facilities.

1.3 Unique Plant Features

The design of Vogtle, Units 1 and 2, includes unique features that have been or are being reviewed by the staff. They include

(1) Nuclear Service Cooling Water (NSCW) Towers

Two 100% capacity NSCW towers per unit serve as the ultimate heat sink at Vogtle. The NSCW towers are part of the NSCW system whose purpose is to remove heat from plant auxiliaries to facilitate safe shutdown of the reactor. Each tower includes a basin containing the water of the ultimate heat sink and an upper portion where heat is transferred to the atmosphere. The design of the ultimate heat sink is such that a single failure in combination with loss of offsite power will not cause inadequate core cooling or prevent safe shutdown. No components of the ultimate heat sink are shared between units. The NSCW towers are circular mechanical draft towers of reinforced concrete constructed to seismic Category I. Cooling water from the tower basins is supplied to each unit via two of three NSCW pumps in each of the two trains. After cooling plant components, the water is returned to the cooling towers where the heat is rejected to the atmosphere. Makeup water is provided from NSCW makeup wells and, if necessary, from the Savannah River.

(2) Auxiliary Feedwater Steam Generator Nozzle

The auxiliary feedwater (AFW) system supplies feedwater to the steam generators whenever the temperature of the reactor coolant is above 350°F. Vogtle has a separate 6-in. steam generator nozzle for the auxiliary feedwater flow, rather than having the auxiliary feedwater flow enter the steam generator through the main feedwater nozzle. The auxiliary feedwater nozzle is connected to the main 16-in. feedwater line by a 6-in. bypass feedwater line. The auxiliary feedwater nozzle is used during low-load or hot standby conditions when the feedwater flow and temperature are low. No water is added to the steam generator through the main feedwater line and nozzle at this time thereby reducing the possibility of main feedline and nozzle cracking. An elbow with a short transition piece is connected to the main and bypass feedwater nozzles minimizing that part of the feedwater piping which is able to drain into the steam generator and fill with steam. Additionally, both lines contain no high-point pockets that could trap steam and possibly lead to water hammer.

(3) Safe Shutdown From Outside the Control Room

Instead of one remote shutdown panel per unit, each Vogtle unit has two remote shutdown panels on separate trains. This arrangement provides redundancy so that the plant can be maintained in a hot standby and hot or cold shutdown condition in case the main control room can not be occupied due to an abnormal plant condition. The main purpose of the shutdown panels is to maintain hot shutdown outside the control room. However, the panels may be used to implement cold shutdown. The train A panel is located on level A of the control building next to the train A 4160-V switchgear room. The train B panel is also on level A but adjacent to the lower cable spreading room. Automated control access terminals control access to the panels.

(4) Radwaste Solidification Building

The radwaste solidification building is a separate structure containing the radwaste volume reduction and solidification system. The volume reduction system has the capability to incinerate dry waste. The solidification system solidifies the particulate and ash product from the volume reduction system. The resin transfer system provides the capability to transfer remotely spent radioactive resin from the radwaste transfer building through a transfer tunnel to the radwaste solidification building.

(5) Refueling Water Storage Tank (RWST)

The RWST is the source of water for the safety injection, centrifugal charging, and residual heat removal pumps following an accident. The Vogtle RWST has a large capacity of 715,000 gallons, which is considerably larger than that of most other plants. This capacity is sufficient to supply the water required during the injection phase. The RWST is a cylindrical, reinforced concrete tank containing a stainless steel liner.

(6) Main Steam Isolation Valves (MSIVs)

Vogtle has two main steam isolation valves per main steamline rather than the more usual number of a single MSIV per steamline. The valves are situated as close to the containment as practical but outside the containment building. The redundancy of the MSIVs in each line will provide positive shutoff with minimum leakage during possible line break situations either upstream or downstream of the valves.

1.4 Significant Issues

During the course of the staff review, certain significant issues were identified that involved one or more of the following:

(1) novel features of the plant or site resulting in special safety concerns

(2) unique technical approaches by the applicant in dealing with safety

(3) recently developed staff safety concerns for which a solution has not been standardized by the staff or nuclear industry

(4) a major disagreement between the staff and applicant

(5) a major modification to the facility during the course of the staff review

(6) a high level of effort, either by the applicant or the staff, to resolve

For Vogtle, the following is a list of such issues:

Later

1.5 Open Items

The staff has identified certain open items in its review that had not been resolved with the applicant at the time this report was issued. The staff will complete its review of these items before the operating license is issued. The staff will discuss the resolution of each of these items in a supplement to this report. These items are listed in Table 1.3 and are discussed further in the sections of this report as indicated.

1.6 Confirmatory Items

At this point in the review there are some items that have essentially been resolved to the staff's satisfaction, but for which certain confirmatory information has not yet been provided by the applicant (see Table 1.4). In these instances, the applicant has committed to provide the confirmatory information in the near future. If staff review of the information provided for an item does not confirm preliminary conclusions, that item will be treated as open and the staff will report on its resolution in a supplement to this report.

1.7 License Condition Items

There are certain issues for which a license condition may be desirable to ensure that staff requirements are met during plant operation (see Table 1.5). The license condition may be in the form of a condition in the body of the operating licenses or a limiting condition for operation in the Technical Specifications appended to the licenses.

1.8 Unresolved Safety Issues

Section 210 of the Energy Reorganization Act of 1974, as amended, reads as follows:

Unresolved Safety Issues Plan

> Section 210. The Commission shall develop a plan for 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 to the Congress thereafter.

In response to this reporting requirement, the NRC provided a report to the Congress, NUREG-0410, in January 1978, which describes the generic issues program of the Office of Nuclear Reactor Regulation (NRR) that had been implemented early in 1977. The NRR program described in NUREG-0410 provides for the identification of generic issues, the assignment of priorities, the development of detailed task action plans to resolve the issues, the projections of dollar and personnel costs, continuing high-level management oversight of task progress, and public dissemination of information related to the tasks as they progress. Since the issuance of NUREG-0410, each annual report has described NRC progress in resolving these issues.

The staff continually evaluates the safety requirements used in its review against new information as it becomes available. In some cases, the staff takes immediate action or interim measures to ensure safety. In most cases, however, the initial staff 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 staff 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 these matters and the NRC program for the resolution of these generic issues is provided in Appendix C to this report, which includes references to sections of this report for specific discussions concerning Vogtle, Units 1 and 2.

1.9 Identification of Agents and Contractors

Georgia Power Company (GPC) acts as the agent for the applicant and is responsible for the design, construction, operation, maintenance, and testing of Vogtle, Units 1 and 2. GPC uses the technical support services of Southern Company Services, Inc. (SCS) in licensing, engineering, design, and quality assurance activities.

The applicant has retained Bechtel Engineering Corporation in Norwalk, California, to perform architectural-engineering and construction management services for Vogtle, Units 1 and 2. The Westinghouse Electric Corporation designed, manufactured, and delivered to the site the NSSS and initial core for Vogtle, Units 1 and 2.

The turbine generator is manufactured by General Electric Company.

The applicant utilizes consultants, as required in specialized areas; for example, Pickert, Lowe and Garrick assists in areas of meteorological analyses; General Physics provided input to the preliminary control room design review; Law Engineering assists in geotechnical testing and evaluation; and Dames and Moore provides services in meteorological tower calibration.

1.10 Summary of Principal Review Matters

The staff technical review and evaluation of the information submitted by the applicant considered, or will consider, the principal matters summarized below.

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

(2) The design, fabrication, construction, and testing criteria and the expected performance characteristics of the plant structures, systems, and components important to safety to determine that (a) they are in accord with the general design criteria, quality assurance criteria, regulatory guides and other appropriate rules, codes, and standards, and (b) any departures from these criteria, codes, and standards have been identified and justified.

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

(4) The applicant's engineering and construction organization, plans for the conduct of plant operations (including the organizational structure and the general qualifications of operating and technical support personnel),

the plans for industrial security, and the plans for emergency actions to be taken in the unlikely event of an accident that might affect the general public to determine that the applicant is technically qualified to operate the facility safely.

(5) The design of the systems provided for control of radiological effluents from the facility to determine that (a) these systems are capable of controlling the release of radioactive wastes from the facility within the limits of the Commission regulations in 10 CFR 20 and (b) the applicant is capable of operating the equipment provided so that radioactive releases are reduced to levels that are as low as is reasonably achievable within the context of the Commission regulations in 10 CFR 50 and to meet the dose-design objectives of Appendix I to 10 CFR 50.

(6) The applicant's quality assurance program for the operation of the facilities to ensure that (a) the program complies with the Commission regulations in 10 CFR 50 and (b) the applicant will have proper controls over the facility operations so that there is reasonable assurance that the facility can be operated safely and reliably.

Table 1.6 lists completed and estimated licensing, construction, and operation milestones. The future milestones listed are projections based on experience and as such, are subject to significant change depending on the progress of the project.

Table 1.1 Cross-reference table for TMI-2 Task Action Plan items

TMI item	Shortened title	SER section
I.A.1.1	Shift technical advisor	13.1.2
I.A.1.2	Shift Supervisor responsibilities	13.5
I.A.1.3	Shift staffing	13.1
I.A.2.1	Immediate upgrade of RO and SRO training and qualification	13.2.1.3
I.A.2.3	Administration of training program	13.2.1.3
I.A.3.1	Revised scope and criteria for licensing exams	13.2
I.B.1.2	Independent Safety Engineering Group	13.4
I.C.1	Short-term accident and procedure review	13.5
I.C.2	Shift and relief turnover procedures	13.5.1
I.C.3	Shift Supervisor responsibilities	13.5
I.C.4	Control room access	13.5.1
I.C.5	Feedback of operating experience	13.5.1
I.C.6	Verification of correct performance of operating activities	13.5.1
I.C.7	NSSS vendor review of procedures	13.5.2
I.C.8	Pilot monitoring of selected emergency procedures for NTOLs	13.5.2
I.D.1	Control room design review	18
I.D.2	Safety parameter display system	18
I.G.1	Training during low-power testing	14
II.B.1	Reactor coolant system vents	15.9.1
II.B.2	Plant shielding	12.3.2
II.B.3	Postaccident sampling	9.3.2
II.B.4	Training for mitigating core damage	13.2.1.3
II.D.1	Relief and safety valve test requirements	3.9.3.2, 5.4.7

Table 1.1 (Continued)

TMI item	Shortened title	SER section
II.D.3	Relief and safety valve position indication	5.2.2, 7.5.2.3
II.E.1.1	Auxiliary feedwater system evaluation	10.4.9
II.E.1.2	Auxiliary feedwater system initiation and flow indication	7.3.3.1
II.E.3.1	Emergency power for pressurizer heaters	8.3.3.4
II.E.4.1	Dedicated hydrogen penetrations	6.2.5
II.E.4.2	Containment isolation dependability	6.2.4
II.F.1.1	Noble gas monitor	11.5
II.F.1.2	Iodine/particulate sampling	11.5
II.F.1.3	Containment high-range monitor	12.3.4
II.F.1.4	Containment pressure	7.5.2.4
II.F.1.5	Containment water level	7.5.2.4
II.F.1.6	Containment hydrogen	7.5.2.4, 6.2.5
II.F.2	Instrumentation for detection of inadequate core cooling	4.4.8, 7.5.2.5
II.F.3	Instrumentation for monitoring accident conditions	7.5.2.6
II.G.1	Power supplies for pressurizer relief valves and level indicators	5.2.2, 8.3.3.4
II.K.1.5	Review of ESF valves	15.9
II.K.1.10	Operability status	15.9
II.K.2.13	Effect of HPI for small-break LOCA with no auxiliary feed	15.9
II.K.2.17	Voiding in RCS	15.9
II.K.2.19	Benchmark analysis sequential AFW flow	15.9
II.K.3.1	Auto PORV isolation system	15.9
II.K.3.2	Report on PORV failures	15.9

Table 1.1 (Continued)

TMI item	Shortened title	SER section
II.K.3.3	Reporting SRV failures	15.9
II.K.3.5	Auto trip of RCPs	15.9
II.K.3.9	PID controller modification	7.7.2.4
II.K.3.10	Applicant's proposed anticipatory trip at high power	15.9
II.K.3.12	Confirm anticipatory trip upon turbine trip	7.2.2.5
II.K.3.17	Report of ECCS outage	15.9.4
II.K.3.25	Loss of power to pump seal coolers	15.9.4
II.K.3.30	Small-break LOCA methods	15.9.4
II.K.3.31	Plant-specific calculations	15.9.4
III.A.1.2	Upgrade emergency support facilities	13.3
III.A.2	Emergency preparedness	13.3
III.D.1.1	Primary coolant outside containment	15.9
III.D.3.3	Inplant radioiodine monitoring	12.3.4.2
III.D.3.4	Control room habitability	6.4

Table 1.2 Comparison of principal design features of Vogtle, Units 1 and 2, and other facilities

Design feature	Vogtle	Comanche Peak	SNUPPS
Containment type*	A	A	A
Rated thermal power, MWt	3411	3411	3411
Gross electrical output, MWe	1157	1159	1188
Total steam flow, 10^6 lb/hr	15.12	15.14	15.14
Total core flow rate, 10^6 lb/hr	142.1	140.3	142.1
Nominal system pressure	2250	2250	2250
Fuel lattice	17 x 17	17 x 17	17 x 17
Number of fuel assemblies	193	193	193
Number of fuel rods per fuel assembly	264	264	264
Number of cluster control assemblies full/part length	53/0	53/0	53/0
Reactor vessel inside diameter, in.	173	173	173
Overall reactor vessel height, ft-in.	43-10	43-10	43-8
Reactor vessel design pressure, psig	2485	2485	2485
Reactor vessel minimum cladding thickness, in.	0.125	0.125	0.125
Number of loops	4	4	4

Table 1.2 (Continued)

Design feature	Vogtle	Comanche Peak	SNUPPS
Number of high-pressure safety injection pumps	2	2	2
Number of intermediate safety injection pumps	2	2	2
Number of low-pressure safety injection pumps	2	2	2
Maximum heat flux, Btu/ft^2-hr	436,500	440,300	440,300
Peak linear power for normal operation, kW/ft	12.5	12.6	12.6
Minimum DNBR	>1.30	>1.30	>1.30
Total peaking factor	2.30	2.32	2.32

*A = atmospheric

Table 1.3 Listing of open items

Item		SER section
(1)	Design basis temperatures for auxiliary systems and components	2.3.1
(2)	Upgrade of operational meteorological measurements program	2.3.3
(3)	Foundation competency of clay marl stratum	2.5.4.1.3, 2.5.4.2, 2.5.4.4.3, 2.5.4.4.6
(4)	Verification FSAR commitments on compaction of Category 1 backfill	2.5.4.3
(5)	Submittal and evaluation of settlement records	2.5.4.1.3, 2.5.4.4, 2.5.4.4.3
(6)	Foundation design and construction information on radwaste buildings and tunnels	2.5.4.4
(7)	Bearing capacity stability	2.5.4.4.2
(8)	Long-term groundwater and settlement monitoring requirements	2.5.4.4.3, 2.5.4.5
(9)	Acceptability of variations in soil dynamic properties	2.5.4.4.6
(10)	Final pipe whip and jet impingement evaluation for high-energy piping.	3.6.2
(11)	Clarification of pipe break criteria and pipe whip restraints	3.6.2
(12)	Methods to relate measured vibration values to stress levels	3.9.2.1
(13)	Safety-related instrument lines in vibration monitoring program	3.9.2.1
(14)	Design of seismic interface anchors	3.9.2.2
(15)	Use of damping values and equivalent static factors	3.9.2.2
(16)	Piping analysis procedures for main steam and feedwater piping outside containment	3.9.2.2

Table 1.3 (Continued)

Item		SER section
(17)	Methodology for load combinations	3.9.3.1
(18)	Piping for service level C and D loadings	3.9.3.1
(19)	Compliance with NUREG-0737, Item II.D.1	3.9.3.2
(20)	Design of safety and relief valves	3.9.3.2
(21)	Design and construction of component supports	3.9.3.3
(22)	Snubber pre-service examination and preoperational testing program	3.9.3.3
(23)	Preservice and inservice testing of pumps and valves	3.9.6
(24)	Acceptable leak rates	3.9.6
(25)	Seismic and dynamic equipment qualification	3.10.1
(26)	Pump and valve operability assurance	3.10.2
	(a) Extent to which standards are used	3.10.2
	(b) Compliance with RG 1.148	3.10.2
	(c) Methods and standards for qualification	3.10.2
	(d) Qualification of pump and motor	3.10.2
	(e) Aging and sequence of environmental conditions in maintenance program	3.10.2
	(f) Pumps affected by static shaft analysis	3.10.2
	(g) Generic testing criteria for qualifying check valves	3.10.2
	(h) Administrative control of component qualification	3.10.2
	(i) Onsite audit	3.10.2
	(j) Dependability of containment isolation (purge valves)	3.10.2
	(k) Long-term operability of deep draft pumps (IE Bulletin 79-15)	3.10.2, 6.3.1

Table 1.3 (Continued)

Item		SER section
(27)	Sensitivity of CVCS letdown monitor for detecting fuel rod failures	4.2.4.2
(28)	Postirradiation fuel surveillance program additional surveillance	4.2.4.3
(29)	Postirradiation fuel surveillance program disposition of failed fuel	4.2.4.3
(30)	Flow measurement capability with crud buildup	4.4.4.2
(31)	Thermal-hydraulic design comparison	4.4.5
(32)	Loose parts monitoring system	4.4.7
(33)	Compliance with NUREG-0737, Item II.F.2	4.4.8
(34)	Steamline break DNBR	4.4.9
(35)	Overpressure protection during low temperature operation	5.2.2.2
(36)	Preservice inspection program	5.2.4.3, 6.6
(37)	Impact test data and C_v curve for vessel beltline materials	5.3.1
(38)	Steam generator tube preservice inspection	5.4.2.2.2
(39)	Effect of neutron irradiation damage on limiting weld metal	5.3.1
(40)	Withdrawal schedules for surveillance specimens	5.3.1
(41)	Pressure-temperature curves to include closure flange regions	5.3.2
(42)	Natural circulation boration and cooldown tests	5.4.7.5
(43)	RHRS operation above 450 psig	5.4.7.5
(44)	Target Rock valves in RVHVS	5.4.12
(45)	Compliance with NUREG-0737, Item II.K.1.5	6.3.1
(46)	Compliance with NUREG-0737, Item II.K.3.10	6.3.1

Table 1.3 (Continued)

Item		SER section
(47)	Operator errors during switchover to recirculation	6.3.2
(48)	Analysis for large break LOCA with $C_D = 1.0$	6.3.5.1
(49)	Air leakage discrepancy in control room leak rate	6.4
(50)	Data used to estimate control room dose following a LOCA	6.4
(51)	Toxic gas evaluation of chemicals	6 4
(52)	Compliance with RG 1.52 and ANSI 509	6.5.1
	(a) No high alarm and trip-alarm signals provided in control room for a temperature sensor located between the heater and first HEPA filter	
	(b) No recorded indication is provided in the control room for the pressure drop across the first HEPA filter	
(53)	Justification for not providing a cooling mechanism for the ESF filtration units	6.5.1
(54)	Design modification for automatic reactor trip using shunt coil trip attachment	7.2.2.3
(55)	Level measurement errors resulting from environmental temperature effects on level instrument reference legs	7.3.3.4
(56)	Auxiliary feedwater system	7.3.3.7
(57)	Override of isolation signals	7.3.3.8
(58)	Isolators used in the BOP design	7.3.3.9
(59)	Auxiliary relays used with no-go tested slave relays	7.3.3.10
(60)	Electrical tunnel ventilation system	7.3.3.11
(61)	Control room ventilation isolation	7.3.3.12
(62)	Emergency response capability - RG. 1.97, Rev. 2	7.5.2.1

Table 1.3 (Continued)

Item		SER section
(63)	Compliance with NUREG-0737, Item II.D.3	7.5.2.3
(64)	Bypass and inoperable status panel- conformance to Position C.2 of RG 1.47	7.5.2.4
(65)	IE Bulletin 79-27, Loss of non-class 1E instrumentation and control power system bus during operation	7.5.2.5
(66)	Freeze protection for instrumentation sensing and sampling lines	7.5.2.6
(67)	RCS overpressure protection during low temperature operation	7.6.2.1
(68)	Compliance with NUREG-0737, Item II.K.3.1	7.6.2.2
(69)	Instrumentation for process measurements used for safety functions	7.6.2.3
(70)	High-energy-line breaks and consequential control system failures	7.7.2.2
(71)	Control system failure caused by malfunctions of common power source or instrument line	7.7.2.3
(72)	Compliance with NUREG-0737, Item II.B.3	9.3.2
	(a) Criterion 2	
	(b) Criterion 10	
(73)	Fire hazards analysis	9.5.1, C.1.b
(74)	Fire doors	9.5.1, C.5.a
(75)	Fire dampers	9.5.1, C.5.a
(76)	Soundproofing materials	9.5.1, C.5.a
(77)	Safe shutdown	9.5.1, C.5.b
(78)	Alternate shutdown	9.5.1, C.5.c
(79)	Power supplies for ventilation	9.5.1, C.5.f

Table 1.3 (Continued)

Item		SER section
(80)	Fire detection	9.5.1, C.6.a
(81)	Valve supervision	9.5.1, C.6.c
(82)	Automatic sprinkler systems	9.5.1, C.6.c
(83)	Standpipes	9.5.1, C.6.c
(84)	Halon 1301 systems	9.5.1, C.6.d
(85)	Control room complex	9.5.1, C.7.b
(86)	Secondary water chemistry monitoring and control program	10.3.5
	(a) Sampling schedule and control limits	
	(b) Procedures	
	(c) Sampling points	
	(d) Data management	
	(e) Responsible authority	
(87)	Quality assurance for main condenser evacuation system	10.4.2
(88)	Quality assurance for turbine gland sealing system	10.4.3
(89)	Volume reduction system	11.4.3, 11.5
(90)	Initial training program	13.2.1.1
	(a) Simulator training	
	(b) Walkthrough training	
	(c) Review and audit	
	(d) Description of SRO training	
	(e) Description of training program for heat transfer, fluid flow and thermodynamics	
	(f) Training complete before preop tests begin	
	(g) Numbers of personnel for whom training and licensing is planned to meet Tech Specs	

Table 1.3 (Continued)

Item	SER section
(91) Licensed operator requalification training program	13.2.1.2
(a) Implementation schedule	
(b) Procedures for record retention	
(c) Lack of heat transfer, fluid flow and thermodynamic training	
(d) Training for mitigating core damage	
(e) Review of abnormal and emergency procedures	
(92) Compliance with NUREG-0737, Item I.A.2.1	13.2.1.3
(a) Lack of heat transfer, fluid flow, and thermodynamic training	
(b) Retesting of simulator response	
(93) Compliance with NUREG-0737, Item I.A.2.3	13.2.1.3
(94) Compliance with NUREG-0737, Item II.B.4	13.2.1.3
(95) Training for nonlicensed plant staff	13.2.2
(a) Organization teaching the course	
(b) Distribution of training	
(c) Health physics training for mechanical and electrical maintenance personnel	
(d) Refresher instruction	
(e) Schedule	
(f) Number of personnel for whom training is planned to meet Tech Specs	
(96) Fire protection training	13.2.2.1, 9.5.1
(a) Fire fighting plan	
(b) Content of instruction as per BTP CMEB 9.5-1 (e.g., meetings every 3 months, drills, refresher training)	

Table 1.3 (Continued)

Item		SER Section
(96)	Fire protection training (continued)	13.2.2.1, 9.5.1
	(c) Qualified individuals as instructors	
	(d) Training for other plant employees	
(97)	Shift technical advisor training	13.2.2.2
	(a) Mitigating core damage	
	(b) INPO recommendations	
(98)	Procedures generation package	13.5.2
	(a) Plant-specific technical guideline	
	(b) Writer's guide	
	(c) Validation and verification programs	
	(d) Training program description	
(99)	Initial test program	14
(100)	Technical Specifications to require four valves to be closed during refueling	15.4.6
(101)	Inadvertent boron dilution during modes 3, 4, and 5	15.4.6
(102)	Compliance with NUREG-0737, Items II.K.3.1/II.K.3.2	15.6.1
(103)	Radiological consequences of SGTR	15.6.3
(104)	Operator action in event of steam generator tube rupture	15.6.3
(105)	Compliance with regulatory guides	17.5
(106)	Operational QA program	17.5
(107)	Detailed control room design review	18
(108)	Safety parameter display system	18

Table 1.4 Listing of confirmatory items

Item		SER section
(1)	Parameters measured at meteorological tower	2.3 3
(2)	Correlation and analysis of data from old and new meteorological towers	2.3.3
(3)	Locations and description of observed cavities	2.5.4.4.1
(4)	Method used to establish dynamic passive pressures	2.5.4.4.4
(5)	Rod bowing analysis	4.2.3.1(6)
(6)	Correct references for the cladding rupture and cladding ballooning and flow blockage models for large-break LOCA	4.2.3.2(6), 4.2.3.3(3)
(7)	Testing and inspection of new fuel	4.2.4.1
(8)	On-line fuel failure detection methods	4.2.4.2
(9)	N-1 loop operation	4.4.6
(10)	Yield strength of cold-worked austenitic stainless steels	4.5.1
(11)	Discrepancy between WCAP 10529 and FSAR	5.2.2.2
(12)	Tech Spec for maximum permissible temperature mismatch	5.2.2.2
(13)	Operability requirements for vent system in Technical Specifications	5.4.12
(14)	GDC 51	6.2.7
(15)	SBLOCA below P-11 interlock	6.3.5.2
(16)	Lead, lag, and rate time constant setpoints used in safety system channels	7.2.2.1
(17)	Turbine trip following a reactor trip	7.2.2.2
(18)	Trip setpoint and margins	7.2.2.4
(19)	Compliance with NUREG-0737, Item II.K.3.10	7.2.2.5
(20)	Test of engineered safeguards P-4 interlock	7.3.3.2
(21)	Undetectable failure in online circuitry for engineered safeguards relays	7.3.3.3

Table 1.4 (Continued)

Item		SER section
(22)	Steam generator level instrumentation	7.3.3.5
(23)	IE Bulletin 80-06 concerns	7.3.3.6
(24)	Compliance with NUREG-0737, Item II.D.3	7.5.2.3
(25)	Bypass and inoperable status panel	7.5.2.4
(26)	Process control program	11.4.3
(27)	Exemption from 10 CFR 70.24	12.3.4.1
(28)	Program to minimize post-LOCA leakage from ESF system outside containment	15.4.1.2
(29)	Compliance with NUREG-0737, Items II.K.3.30 and II.K.3.31	15.6.5
(30)	Compliance with NUREG-0737, Item II.F.1	15.9.2
(31)	Compliance with NUREG-0737, Item III.D.1.1	15.9.5

Table 1.5 Listing of license conditions

License condition	SER section
Later	

Table 1.6 Major licensing, construction, and operation milestones

Milestone	Date
Limited work authorization (LWA) issued	May 28, 1974
Construction permit issued	June 28, 1974
Suspension of Units 1 and 2, cancellation of Units 3 and 4	September 12, 1974
Full reactivation of Units 1 and 2	July 1, 1978
Amendment extending construction completion dates from April 1981 and April 1982 to April 1983 and April 1984 for Units 1 and 2, respectively, and adding additional owners*	January 1977
Amendment extending construction completion dates from April 1983 and April 1984 to March 1988 and September 1989 for Units 1 and 2, respectively	June 1982
Final Safety Analysis Report docketed	September 16, 1983
Environmental Report docketed	November 28, 1983
Safety Evaluation Report issued	June 1985
ACRS full committee meeting	July 1985**
Safety hearings	February 1986
Ready for fuel loading (applicant)	September 1986

*Oglethorpe Electric Membership Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, Georgia.

**Estimated.

2 SITE CHARACTERISTICS

2.1 Geography and Demography

The geography and demography of Vogtle Electric Generating Plant were reviewed in accordance with Sections 2.1.1, 2.1.2, and 2.1.3 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981 (SRP). The results of this review are presented below.

2.1.1 Site Location and Description

The site for the Vogtle plant, which is a proposed two-unit plant, consists of 3,169 acres of land located on the west bank of the Savannah River, at River Mile 151, in the eastern part of Burke County in the state of Georgia. The exclusion area, which is designated by the plant's property line, and the plant layout are shown in Figure 2.1. The Vogtle low population zone and the area within 5 mi (8 km) of the site are shown in Figure 2.2. A map of the area in the general vicinity (25 mi) of the site is shown in Figure 2.3. This map shows some of the towns and transportation routes in the area as well as the location of the Vogtle site in relation to the Savannah River Plant on the opposite side of the river. The 1970 Bureau of the Census statistics for the most populated areas within 100 mi of the site are shown in Figure 2.4.

As can be seen in Figure 2.1, the Unit 1 and Unit 2 reactors are located about 1,097 m (3,600 ft) and 1,189 m (3,900 ft), respectively, west of the river. The site is about 24 km (15 mi) east-northeast of Waynesboro, Georgia, and about 41.5 km (26 mi) south-southeast of Augusta, Georgia. Augusta is the largest populated area near Vogtle. Except for some limited activity on the river and the restricted activity associated with the Savannah River Plant located on the east side of the river in Barnwell County, South Carolina, there is very little other activity within the 5-mi area surrounding the site. About 30% of this area is gently rolling farmland. Most of the remaining land within 5 mi, particularly west of the river, is wooded. The coordinates of the Vogtle site are 33° 08' 29" north latitude and 81° 45' 45" west longitude. The Universal Transverse Mercator (UTM) coordinates are 3,667,028.6 m north and 428,881.8 m east, in zone 17.

2.1.2 Exclusion Area Authority and Control

The exclusion area for the Vogtle site is defined as an irregularly shaped area which conforms to the site's boundary lines. The minimum distance to the exclusion area boundary is 1,097 m (3,600 ft) measured from the center of the Unit 1 containment building. The applicant owns all of the surface and mineral rights in the designated exclusion area, and thereby has the authority to determine all activities within the area as required by the Commission's regulation, "Reactor Site Criteria," 10 CFR Part 100. The shortest distance from the plant's gaseous effluent release point to the exclusion area boundary is 1,128 m (3,700 ft). No one resides within the exclusion area and there are no highways, railways, or waterways crossing the area. The applicant owns and operates Plant Wilson, a combustion turbine plant, that is located in the east-southeast portion of the site property. Aside from Plant Wilson, the only other activities that may occur within the exclusion area that are unrelated to plant operations are those associated with persons in and around the Visitors Center, and those connected with the operation of the Vogtle simulator. It should be noted that during Unit 1 operation there will be onsite construction activities associated with Unit 2 until its completion. Approximately 15 of the applicant's employees work at Plant Wilson. Also, 8 staff members and approximately 40 trainees are at the simulator. Because of the remoteness of the site, only a few persons are expected at the Visitors Center occasionally. The applicant has made arrangements to control and, if necessary, evacuate the exclusion area in the event of an emergency. Section 13.3 of this SER has more details about these arrangements.

The staff concludes, by virtue of ownership of the land and control of the mineral rights within the exclusion area, and on the basis that suitable arrangements have been made to control all activity unrelated to plant operations, that the applicant has the authority to determine all activity within the exclusion area as required by 10 CFR Part 100. The staff further concludes that the activities unrelated to plant operation within the exclusion area will not interfere with normal plant operations.

2.1.3 Population Distribution

The resident population in the vicinity of the Vogtle site is shown as a function of distance in the table below. The year 2030 is the nearest census year to the end of plant life.

The closest resident lives about 1.9 km (1.2 mi) from the reactor building. The nearest communities in the vicinity of the site with a population of more than 1,000 persons are Sardis, Georgia, 20.8 km(13 mi) south, and Waynesboro, Georgia, 24 km (15 mi) west-southwest, with populations of 1,179 and 5,760, respectively, in 1980. Girard, Georgia, 12 km (7.5 mi) south-southeast, is the closest residential area and it had a population of only 225 in 1980. There are no large communities with populations of 5,000 or more persons within 8 km (5 mi) of the site. The population within 5 mi in 1980 was 1,085 and within 10 mi it was 2,560. As indicated in Table 2.1, the population within 5 mi of the site by 1990 is expected to drop to about 262 (after Units 1 and 2 become operational) and is expected to almost double during the life of the plant. The applicant reported that there were 509,222 people living within 80 km (50 mi) of the site in 1980, and they expect this number to increase to 589,111 by 1990. By the year 2028, the population within 50 mi of Vogtle is projected to reach 903,493. Augusta, Georgia, located about 41.5 km (26 mi) north-northwest, is the largest populated area around the site and it had a 1980 population of 47,532. There are no cities larger than Augusta within 50 mi of Vogtle. Columbia, South Carolina, with a 1980 population of 101,208 is the closest large city, and it is about 120 km (75 mi) away. The applicant conservatively projects a population growth rate of about 48% during the life of the plant for the area within 50 mi of the site. This represents a growth of about 12% per decade for this period. The staff, using the Bureau of Economic Analysis (BEA) projections, calculated the population within 50 mi of the site and determined that the population will increase about 44% for the same period.

The applicant has designated a low-population zone (LPZ) for the site which is a circular area with a 3.2 km (2 mi) radius measured from the midpoint of the centerline between Units 1 and 2. Except for the Savannah River, the LPZ consists mostly of wooded areas and a small amount of agricultural land much the same as the rest of the area in the general vicinity around the Vogtle plant.

There is very little transient population within the LPZ because of the remoteness of the area. A limited amount of recreational activity takes place on the river, including fishing and boating. During the winter season there is a minimal amount of hunting on the Georgia side of the river, but there are no hunting lodges or camps in the area. According to the applicant, about 500 people reside within the LPZ and will remain there until construction work at Unit 2 is completed. By 1990, however, only about 30 people will live in the LPZ; this number is expected to increase to about 65 during the life of the plant. There is one church, the Ebenezer Baptist Church, located within the LPZ about 2.7 m (1.7 mi) from the site which has a congregation of about 100 persons.

The transient population within a 10-mi radius of the Vogtle site is quite low because of the lack of recreational facilities or industry in the vicinity, and because of the restricted nature of the Savannah River Plant which occupies almost one-half of the 10-mi area. The Savannah River Plant employs about 6,675 persons. There are no migrant workers in this area. Five private boat-landing facilities are located within a 10-mi stretch of the river near the site. The applicant estimates that about 231 fishermen use these facilities during the span of a year. Surveys conducted by the Georgia Department of Natural Resources indicate that Burke County has the lowest hunting yields in the state, thus most of the hunting is done outside the 10-mi area, specifically in the Georgia counties northwest of the site.

There are no prisons, hospitals, nursing or convalescent homes, day-care centers, federal/state parks or forests, beaches, amusement parks, or federal highways within 10-mi of the plant. The Girard Elementary School is the only school within 16 km (10 mi) of the site. It has a staff of 23 and an enrollment of 200 students, but is expected to close before Unit 1 goes into operation.

There are about 24 churches within 10 mi of Vogtle. The frequency of services at these churches varies considerably, from twice a week to once a month, and services usually occur on Sunday. The total attendance at all church services combined is about 4,800 per month. In the 10 mi surrounding the Vogtle site there are about 26 cemeteries or burial grounds. Some are private family gravesites or church affiliated cemeteries, and some are open to the public. About one-third of them are inactive, while the others may be used from once every two years to once a month. Section 13.3 of this SER provides a discussion of the emergency preparedness plans for protecting the public in this area.

The nearest densely populated center of about 25,000 or more persons, as defined by 10 CFR Part 100, is Augusta, Georgia, which is about 26 mi north-northwest of the site. This distance is at least 1-1/3 times the distance to the LPZ outer radius, as required by 10 CFR Part 100.

2.1.4 Conclusion

This review is based on the 10 CFR Part 100 definitions of the exclusion area, the LPZ, and the population center distance, as well as the staff's analysis of the onsite meteorological data, from which the relative concentration factors (x/Q) were calculated (see Section 2.3 of this SER), and the calculated potential radiological dose consequences of design-basis accidents (see Section 15 of this SER). The staff has concluded that the exclusion area, LPZ, and population center distance satisfy the criteria of 10 CFR Part 100 and are acceptable.

2.2 Nearby Industrial, Transportation, and Military Facilities

Insofar as activities at nearby industrial, transportation, and military facilities are concerned, Vogtle was reviewed in accordance with Sections 2.2.1, 2.2.2, 2.2.3, 3.5.1.5, and 3.5.1.6 of the SRP. The results of the review are contained in this section.

2.2.1 Transportation Routes

There are no highways, railways, or waterways traversing the Vogtle exclusion area. River Road, a secondary road which formerly ran in a southeasterly direction through the center of the exclusion area, has been relocated and now runs just inside the western and southern edge of the site. This road is used principally for local traffic. There are several side roads exiting off River Road that provide access to the plant area, the Visitors Center, the simulator building and Plant Wilson. The nearest primary road in the vicinity of the site is Georgia Highway 23 which is located about 4.5 mi south-southwest. Highway 23 handles commercial traffic and serves as a link for hauling timber and wood products between Augusta and Savannah, Georgia. The only major roads in the area are U.S. Highways 25 and 301 which are located about 15 mi west and 20 mi southeast of the plant, respectively. South Carolina Highway 125 runs through the Savannah River Plant, on the opposite side of the river, about 8.8 km (5.5 mi) northeast of Vogtle. There is essentially no hazardous material transported on River Road, except for that used at the Vogtle site. Because of the separation distances between the major highways and the site, accidents which may occur on these roads do not pose a threat to the safe operation of the plant.

The closest railway to Vogtle is the Seaboard Coast Line Railroad which runs through and provides service to the Savannah River Plant. It is located about 7.2 km (4.5 mi) northeast of the site. Hazardous materials transported on this railroad include: ammonia, carbon dioxide, and sulfuric acid. The applicant's analysis takes into consideration the frequency of shipments and the fact that there have been no accidents on this line in the vicinity of the plant in more than 10 years. This provides assurance that the likelihood of an accident affecting the safe operation of the plant is sufficiently low so as to pose little risk. In addition, because of the separation distance involved, an accident on this railroad would not pose a significant hazard to the plant. A rail spur connected to the Central of Georgia Railway, about 12 mi west of the site, will service the Vogtle site.

The Vogtle plant is located about 3,600 ft west of the Savannah River. The river separates the site from the Savannah River Plant. In addition to some minor recreational activity, the river is also used for relatively infrequent commercial traffic between Augusta and Savannah, Georgia. There are no locks or dams on the river in the vicinity of the site, and there have been no accidents within a 30-mi stretch of the river near Vogtle in the past 15 years. Approximately 100 barge tows carry less than 100 tons of cargo past the site annually. Gasoline and fuel oil comprise about 3% and 90% of the shipments, respectively. About 7% of the shipments are solid chemicals and less than 0.1% are steel products. The applicant analyzed the potential hazards to the plant resulting from a transportation accident on the river involving gasoline and fuel oil. They determined that the risk associated with a potential explosion, considering the frequency of shipment on the river, quantity of material shipped, the previous accident rate, and the potential concentrations necessary to cause ignition and subsequent explosion is lower than 10-7 per year. The staff evaluated the analysis and agrees with the applicant's conclusion that the risks associated with the postulated explosions do not pose a hazard to the Vogtle plant. With respect to thermal hazards, the thermal flux from a gasoline or oil fire on the river would be less than the peak solar flux (1,000 - 1,200 watts/m^2) measured in the United States, which would not affect the safe operation of the plant.

Accidents involving river traffic, with the potential of striking the intake structure or spilling corrosive or other material into the river, will not prevent the safe shutdown of the plant because this system only provides makeup water to the nonsafety-related circulating water system.

2.2.2 Nearby Facilities

The closest industrial activity, other than that associated with the construction of Unit 2, is in connection with the operation of Plant Wilson. This is an oil-fired combustion plant operated by the applicant that is located within the exclusion area about 5,000 ft east-southeast of the nuclear plant. The storage capacity of the three fuel storage tanks at Plant Wilson totals 9 x 106 gallons. These tanks are surrounded by an earthen dike so that, in the event of a tank rupture, the tank contents will be contained within the dike. Because of the distance involved, the potential hazard to the nuclear plant is less from the fuel oil stored at Plant Wilson than from that transported on the river. To protect control room personnel from potential smoke inhalation, the applicant has installed redundant smoke detectors in both the control room and the outside air intakes. There is some industrially related activity at the Department of Energy's Savannah River Plant (Savannah) located on the east bank of the Savannah River. Savannah is a closed government reservation with controlled access. The closest activity at Savannah, in relation to the Vogtle site, is the heavy-water extraction and recovery facility located about 4.5 mi from the Vogtle site. There are several nuclear production and test reactors, and associated support facilities at the Savannah complex. The functions of these facilities, in addition to extracting heavy water from natural water, include: fuel fabrication, dissolution of irradiated material, and the separation of nuclear products from radioactive products. Because of the distances involved, these facilities do not present a problem to the safe operation of the Vogtle plant. There is a 34-ton chlorine storage facility at Savannah that is located about 3-mi from the Vogtle plant. There are also 48 one-ton cylinders and two 150-lb cylinders of chlorine on the Savannah site. These cylinders are located at four different storage areas and are used for the various water treatment systems. Redundant chlorine detectors have been installed in the outside air intakes to the control room which will automatically isolate the control room in the event of an accidental release of chlorine. Section 6.4 of this SER provides more details on the control room habitability systems. There are no other industrial facilities nearby and no significant industrial expansion program is planned for the area around the Vogtle plant in the foreseesable future.

There are no pipelines located within 10-mi of the Vogtle plant. The three closest pipelines, all carrying natural gas and varying in size from 8 in. to 16 in. in diameter, are located about 20 mi from the plant. There are no plans to move these pipelines or transport other products in them. On the basis of past reviews of natural-gas pipelines, these pipelines are sufficiently far from the Vogtle plant so that they do not pose a significant hazard to the operation of the plant. No mining or quarry operations and no other hazardous gas or liquid storage facilities are located within 5 mi of the plant.

There are no military bases, bombing ranges, munitions plants, missile installations, or major airports within the general vicinity of the Vogtle plant. The closest airport, Burke County Airport, is located about 16 mi west-southwest of the site. This airport has a 3,200-ft runway and is used by single-engine aircraft for private use and for crop dusting operations. About 15,000 flights per year leave from or arrive at this airport and no increase in traffic is expected in the near future. The closest commercial airport is Bush Field which is about 17 mi north-northwest of the site. It has one 6,000-ft and one 8,000-ft runway, and considering all types of traffic (commercial, general, and military), the airport handled about 72,000 flights in 1980. Some expansion is proposed at Bush Field including a new 10,000-ft runway and a 500-ft extension to the current 8,000-ft runway. Assuming this expansion is completed as planned, traffic at Bush Field is expected to increase about 60% by 1990. There are no low-level military aircraft training routes or federal airways in the airspace around the Vogtle site. On the basis of separation distances and the nature of these facilities, as well as previous staff reviews of aircraft hazards, the staff concludes that the aircraft activities in the vicinity of the site will not affect the safe operation of the Vogtle plant.

2.2.3 Conclusions

On the basis of (1) the information provided by the applicant, and (2) the staff's review based upon criteria in 10 CFR Part 50 (Appendix A, GDC 4) and in SRP Section 2.2.3, the staff has determined that the plant is adequately protected and can be operated with an acceptable degree of safety - activities at nearby industrial, transportation, and military facilities notwithstanding.

2.3 Meteorology

Evaluation of regional and local climatological information, including extremes of climate and severe weather occurrences which may affect the design and siting of a nuclear plant, is required to ensure that the plant can be designed and operated within the requirements of Commission regulations. Information concerning atmospheric diffusion characteristics of a nuclear power plant site is required for a determination that radioactive effluents from postulated accidental releases, as well as routine operational releases, are within Commission guidelines. Sections 2.3.1 through 2.3.5 have been prepared in accordance with the review procedures described in the Standard Review Plan (NUREG-0800), utilizing information presented in FSAR Section 2.3, responses to requests for additional information, and generally available reference materials as described in the appropriate sections of the Standard Review Plan (SRP).

2.3.1 Regional Climatology

The plant is located in eastern Georgia along the Savannah River, about 26 miles south-southeast of Augusta.

Maritime tropical air masses dominate the region in summer and alternate with continental air masses in winter. The mean annual temperature in the area is about 17.4°C (63°F), ranging from about 7.8°C (46°F) in December and January to about 26.7°C (80°F) in July. Annual precipitation in the area is about 1090 mm (43 in.).

The Vogtle plant is located near a principal track of cyclonic storms that originate along the Gulf Coast and move northeastward along the East Coast, resulting in a variety of severe weather phenomena. About 77 thunderstorms can be expected on about 56 days each year, being most frequent in June, July, and August. Considering the frequency of thunderstorms in the region, the applicant has estimated about 10 lightning strikes per year in the square kilometer area containing the Vogtle plant. Hail often accompanies severe thunderstorms. In the period 1955-1967, hail with diameters 19 mm (3/4 in.) or greater was reported six times in the one-degree latitude-longitude square containing the site.

Tornadoes also occur in the area. About 30 tornadoes have occurred within the one-degree latitude-longitude square containing the site in the period 1954-1983, resulting in an annual tornado occurrence frequency of 1.1. The applicant has conservatively computed a recurrence interval for a tornado at the plant site to be about 500 years. The staff has performed an independent assessment of tornado occurrences in the Vogtle region and computed a recurrence interval for a tornado at the plant site to be about 4800 years. Waterspouts are not considered likely on the Savannah River in the vicinity of the Vogtle plant.

The design-basis tornado characteristics selected by the applicant conform to the recommendations of RG 1.76, "Design Basis Tornado for Nuclear Power Plants," for this region of the country. These characteristics are: rotational speed - 290 mph; translational speed - 70 mph; and a total pressure drop of 3 psi occurring at a rate of 2 psi/sec.

Hurricanes or remnants of hurricanes pass through the region occasionally. During the period 1871-1982, 40 tropical cyclones (tropical depressions, tropical storms and hurricanes) passed within 100 nautical miles of the site.

Occurrences of high windspeeds in the area are associated with severe thunderstorms, extratropical cyclones, tropical storms, and hurricanes. The highest "fastest mile" wind speed reported at Augusta was 62 mph in June 1965. The applicant has identified the "fastest mile" wind speed at a height of 30 ft with a return period of 100 years of 105 mph. However, for design of seismic Category I structures, the applicant has used a "design wind velocity" (operating basis windspeed) of 110 mph at 30 ft above the ground, including gust factors and vertical velocity profiles developed in accordance with the criteria of American National Standards Institute (ANSI) A58.1, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures" (1972).

The applicant has identified the basins of the nuclear service cooling water towers (circular mechanical draft) as the ultimate heat sink for the Vogtle plant. The applicant examined meteorological data from Augusta, Georgia, for the period 1947-1981 to determine the meteorological design conditions for the ultimate heat sink. The conditions to maximize water temperature and water usage were selected as the 1-hour combination of dry bulb and wet bulb temperatures in the period of record resulting in the highest water usage and in the maximum temperature in the cooling tower basins followed by the 24-hr period resulting in the maximum average temperature in the basins and highest 1-day water usage. These conditions were repeated to synthesize a 30-day period. The synthesized meteorological conditions selected by the applicant for the design of the ultimate heat sink appear appropriately conservative.

Heavy snowfall is not common in the region, but roof loads may accumulate from a wintertime mixture of snow, ice and rain. Average annual snowfall at Augusta is only about 25 mm (1 in.). The applicant has reported the maximum snowfall in a 24-hr period in the area to be 350 mm (13.7 in.) in February 1973. The applicant has estimated the weight of the 100-year return period snowpack at ground level to be 8 psi. Ice storms, which can plug drains and scuppers as well as disrupt offsite power, occur in the area. The applicant has indicated that freezing rain occurs on about 2 days each year. The applicant has also indicated that freezing rain has been reported to last as long as 17 hr. The accumulation of water on the roofs is the most likely cause of severe and extreme environmental loads for consideration in the design of the roofs of safety-related structures at Vogtle. The adequacy of such roof loadings is discussed in Section 2.4.3 of the SER.

The applicant has considered the following meteorological conditions in the design of the HVAC systems for all safety-related buildings: 98°F dry bulb/80°F wet bulb temperatures for summer air conditioning; 93°F dry bulb/78°F wet bulb temperatures for summer ventilation; 17°F dry bulb temperature for winter heating; and average windspeeds of 7.5 mph and 15 mph for summer and winter, respectively. The bases for the selection of the temperatures for air conditioning and heating were the 1% probability of occurrences (summer) and 99% (winter) probability of occurrence values from the distributions presented by American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE). The bases for the summer ventilation conditions was the 5% ASHRAE values for temperature. Windspeeds are supposed to be characteristic seasonal averages. The applicant has indicated in the response to Question 451.5 that safety-related auxiliary systems and components (including the diesel generator air intake, service water valves, main steam isolation valves, and impulse lines) are enclosed in seismic Category I structures maintained within acceptable environmental conditions by safety-related HVAC systems. Extreme temperatures of 41.7°C (107°F) and -16.1°C (3°F) have been reported at Augusta. Temperatures in excess of 32.2°C (90°F) are expected at Augusta about 60 days each year. The applicant has analyzed onsite data for the periods December 1972-December 1973 and April 1977-December 1983 and determined that "the maximum consecutive duration of the dry bulb and wet bulb temperatures exceeding the design values is 8 hours and 5 hours, respectively," and that "the maximum consecutive duration of the dry bulb temperature lower than the design value is 18 hours." Conditions with return periods of 100 years are normally considered for design of safety-related auxiliary systems and components. The extreme temperatures for the 100-year return period in the Vogtle area are approximately 43.9°C (111°F) and -20°C (-4°F). Although temperature excursions beyond the design values are infrequent, further justification of the adequacy of the extreme temperatures considered by the applicant for the design of safety-related auxiliary systems and components is required. Also, the applicant has not directly addressed the diesel generator air intake. The design-basis temperatures for auxiliary systems and components is an open item.

Large-scale episodes of atmospheric stagnation occur frequently in the region. About 90 atmospheric stagnation cases totaling about 360 days were reported in the area in the period 1936-1970. Ten of these cases lasted 7 days or more.

As discussed above, the staff has reviewed available information relative to the regional meteorological conditions of importance to the safety design and siting of this plant in accordance with the criteria contained in SRP Section 2.3.1. On the basis of this review, the staff concludes that, with the exception of design-basis temperatures for auxiliary systems and components, the applicant has identified and considered appropriate regional meteorological conditions in the design and siting of this plant, and, therefore, meets requirements of 10 CFR 100.10 and GDC 2 (10 CFR 50, Appendix A). The applicant selected design-basis tornado characteristics that conform to the position set forth in RG 1.76, and, therefore, meet the requirement of GDC 4 (10 CFR 50, Appendix A), to determine an acceptable design-basis tornado for missile generation.

2.3.2 Local Meteorology

Climatological data from Augusta, Georgia, and available onsite data have been used to assess local meteorological characteristics of the plant site.

Precipitation is well distributed throughout the year, ranging from about 55 mm (2.2 in.) in October to about 130 mm (5.1 in.) in July. Maximum and minimum monthly amounts of precipitation observed at Augusta have been 355 mm (14.0 in.) in July 1906 and a trace (less than 0.01 in.) in October 1953. The maximum amount of precipitation in a 24-hr period at Augusta was 250 mm (9.8 in.) in October 1929. Snowfall is not common at Augusta, although snow has occurred in each month from November through March. The maximum monthly snowfall at Augusta was 355 mm (14 in.) in February 1973, and the maximum amount of snowfall in a 24-hr period was 350 mm (13.7 in.) also in February 1973. The annual average total precipitation measured at the site for the composite period December 1972-November 1973 and April 1977-March 1979 is about 675 mm (26.6 in.) compared with the annual average at Augusta of 1035 mm (40.7 in.) for the same period of record. Annual average precipitation recorded at Augusta for the same composite 3-year period as the onsite data showed reasonable agreement with the long-term climatological averages. Although spatial variability in precipitation occurrences is expected and may contribute to these anomalies, the most likely source of the differences in annual average amounts is the applicant's measurement of precipitation and data reduction techniques.

Wind data taken from the 10-m level of the onsite meteorological tower for a 3-year composite period of record (April 4, 1977-April 4, 1979 and April 1, 1980-March 31, 1981) indicate that winds are well distributed, and that wind direction frequencies vary~~ing~~ from about 4% to about 8%.

The average windspeed at the 10-m level is about 4 m/sec. Calm conditions (defined as windspeeds less than the starting threshold of the anemometer) occur infrequently, at less than 0.5% of the time.

Slightly stable (Pasquill type E) conditions predominate at the Vogtle site, occurring about 34% of the time as defined by the vertical temperature gradient between the 45.7-m and 10-m levels. Moderately stable (Pasquill type F) and extremely stable (Pasqill type G) conditions occur about 16% and 9% of the time, respectively, for the same stability indicator. Moderately stable and extremely stable conditions were observed with relatively the same frequency during the preoperational program (December 4, 1972-December 4, 1973) for the Vogtle plant.

As discussed above, the staff has reviewed available information relative to local meteorological conditions of importance to the safe design and siting of this plant in accordance with the criteria contained in SRP Section 2.3.2. Although the staff is concerned about apparently anomalous precipitation measurements at the Vogtle site, the staff concludes that the applicant has identified and considered appropriate local meteorological conditions in the design and siting of this plant and, therefore, meets the requirements of 10 CFR 100.10 and GDC 2 (10 CFR 50, Appendix A).

2.3.3 Onsite Meteorological Measurements Program

Meteorological measurements at the Vogtle site were initiated in April 1972. The meteorological tower used to provide data to support both the construction permit and operating license applications is located about 1500 m (5000 ft) south-southwest of the Unit 1 containment building. Windspeed and wind direction are measured at the 10-m (33-ft) and 45.7-m (150-ft) levels, and vertical temperature gradient is measured between the 10-m and 45.7-m levels. Ambient dry bulb and dew point temperatures are measured at the 10-m level, and precipitation and solar radiation are measured near the ground. The applicant has analyzed the overall measurement system accuracies for each parameter, and concluded that the system accuracies for analog recording are not within the specifications presented in RG 1.23. System accuracies for digital recording appear to comply with the specifications presented in RG 1.23. The meteorological data provided with the operating license application have been checked for reasonableness. The preliminary results indicate that the data collected by the meteorological measurements program are reasonable compared with other data collected in the area. However, the staff's review is not yet complete.

The applicant provided three years (April 4, 1977-April 4, 1979 and April 1, 1980-March 31, 1981) of meteorological data with the operating license application. Meteorological data from all the collection periods (including data for the period December 4, 1972-December 4, 1973) have been compared, and no significant differences have been identified. The three most recent years of onsite data have been combined into joint frequency distribution of windspeed and wind direction by atmospheric stability for use in the atmospheric dispersion assessments presented in Sections 2.3.4 and 2.3.5. Windspeed and wind-direction data for these assessments were based on measurements at the 10-m level, and atmospheric stablity was defined by the measurement of vertical temperature gradient between the 10-m and 45.7-m levels.

Analog strip charts have been used to record meteorological data provided with the operating license application. Since 1977, the applicant has calibrated the system twice per year. Joint data recovery of windspeed and wind direction at the 10-m level by atmospheric stability (defined by the vertical temperature gradient between the 10-m and 45.7-m levels) was 92% for the 3-year composite period described above. Because the periods of missing data were sufficiently random during the 3 years of record, the composite data set is expected to reasonably reflect expected diurnal, seasonal, and annual airflow and stability patterns at the Vogtle site. The 3-year period of record is also expected to reasonably represent occurrences of extreme atmospheric conditions of importance for assessments of local transport and diffusion characteristics. The frequencies of occurrence of moderately stable and extremely stable conditions at Vogtle agrees reasonably well with other sites in the southeastern United States. Dose consequence assessments based on available onsite meteorological data are expected to be reasonably conservative. Extreme meteorological conditions for design of safety-related structures, systems, and components (discussed in Section 2.3.1) were based on long-term (30 years or more) climatological data from nearby National Weather Service stations, and not directly on the 3 years of onsite data. However, the representativeness of long-term offsite data was determined by comparisons of concurrent offsite data with available onsite data.

For the postoperational meteorological measurements program, the applicant has installed a new meteorological tower located in the vicinity of the old tower. The tower will be instrumented at the 10-m and 60-m levels. Although the applicant has not specified the parameters to be measured, most likely windspeed and direction will be measured at the 10-m and 60-m levels and vertical temperature difference will be measured between the 60-m and 10-m levels. The applicant should clarify this information. This is a confirmatory item. The new tower was installed in January 1984, and the applicant has indicated that one full year of data from this tower will be available in February 1985. The applicant will correlate and analyze data from both the old and new towers and will provide data to the staff in time for the staff to prepare the SER.

To address meteorological requirements for emergency preparedness planning outlined in 10 CFR 50.47 and Appendix E to 10 CFR 50, the applicant will be required to upgrade the operational meteorological measurements program to meet the criteria in NUREG-0654, Appendix 2, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." The upgrade~~s~~ must be in accordance with the schedule of NUREG-0737, "Clarification of TMI Action Plan Requirements," Item III.A.2, or its supplement. The incorporation of current meteorological data into a real-time atmospheric dispersion model for dose assessments will also be considered as part of the upgraded capability. This is an open item.

The staff has reviewed the onsite meteorological measurements system in accordance with the criteria contained in SRP Section 2.3.3. The applicant has indicated that instrumentation and data reduction procedures for analog recording do not conform to the recommendations of RG 1.23, "Onsite Meteorological Programs." The staff is continuing to check the reasonableness of the data collected to date, and the staff will ensure that the new meteorological measurements program conforms to the specifications of RG 1.23. The current meteorological measurements program appears to have provided data to represent onsite meteorological conditions as required in 10 CFR 100.10; however, the staff is continuing its evaluation of the adequacy of these data. Nevertheless, the staff concludes that the site data provide a reasonable basis for making preliminary conservative estimates of atmospheric dispersion conditions for estimating consequences of design-basis accident and routine releases from the plant because the resulting windspeed, wind direction, and atmospheric stability distributions appear reasonable for the location of the Vogtle site. Additional analyses will be performed to confirm this conclusion.

2.3.4 Short-Term (Accident) Diffusion Estimates

To audit the applicant's estimates, the staff has performed an independent assessment of short-term (less than 30 days) accidental releases from buildings and vents using the direction-dependent atmospheric dispersion model described in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," with consideration of increased lateral dispersion during stable conditions accompanied by low windspeeds. Three years (as described in Section 2.3.3) of onsite data were used for this evaluation. Windspeed and wind-direction data were based on measurements at the 10-m level and atmospheric stability was defined by the vertical temperature gradient measured between the 10-m and 45.7-m levels. A ground-level release with a building wake factor, cA, of 1134 m^2 was assumed. The relative concentration (χ/Q) value for the 0-2-hr time period was determined to be 1.8×10^{-4} sec/m^3 for the 5% overall site limit at an exclusion boundary distance of 1097 m. Virtually identical χ/Q values were calculated at the exclusion area boundary in the east-northeast, south, south-southwest, and southwest sectors. The 5% overall site limit χ/Q at the outer boundary of the low-population zone (LPZ) was also slightly higher than the χ/Q values in individual sectors. The χ/Q values for appropriate time periods at the LPZ distance of 3218 m are:

Time period	χ/Q (sec/m^3)
0-8 hr	3.1×10^{-5}
8-24 hr	2.2×10^{-5}
1-4 days	1.0×10^{-5}
4-30 days	3.4×10^{-6}

The applicant has calculated an identical χ/Q value at the exclusion area boundary. The χ/Q values calculated by the applicant for various time periods at the LPZ distance are very similar to those calculated by the staff, with the largest difference (about 25%) occurring for the value for the 4-30-day time period.

On the basis of the above evaluation performed in accordance with the criteria contained in SRP Section 2.3.4, the staff concludes that the applicant has considered appropriate atmospheric dispersion estimates for assessments of the consequences of radioactive releases in accordance with the requirements of 10 CFR 100.11. The staff used the atmospheric dispersion estimates provided in this section in an independent assessment of the consequences of radioactive releases for design-basis accidents. The results of this assessment are discussed in Section 15 of this SER.

2.3.5 Long-Term (Routine) Diffusion Estimates

To audit the applicant's estimates, the staff has performed an independent calculation of annual average relative concentration (χ/Q) and relative deposition (D/Q) values using the straight-line Gaussian atmospheric dispersion model described in RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactors." The results of this model were adjusted to reflect spatial and temporal variations in airflow using the correction factors contained in NUREG/CR-2919.

Releases from the plant vents (atop the containment building) were considered as a mixture of elevated and ground level, except for the transport directions (affected sectors) of east-northeast and east, where the natural draft cooling towers could significantly affect atmospheric dispersion. For the transport directions of east-northeast and east, releases from plant vents were considered as ground level. Releases from the turbine building (including the air ejector exhausts) were considered as ground level, with mixing in the turbulent wake of the major plant structures. Releases from the radwaste building were also considered as ground level, with mixing in the turbulent wake of that building. The same 3-year period of record described in Section 2.3.4 was used for this evaluation.

On the basis of ~~on~~ the above evaluation performed in accordance with the criteria contained in SRP Section 2.3.5, the staff concludes that site-specific atmospheric dispersion conditions have been considered in demonstrating compliance with the numerical guides for doses contained in 10 CFR 50, Appendix I. The atmospheric dispersion estimates developed by the staff are included in the assessment of the radiological impact to persons resulting from routine releases to the atmosphere contained in the staff's environmental statement.

2.4 Hydrologic Engineering

Later

2.5 Geology and Seismology

For this DSER, the staff has reviewed all available relevant geologic and seismologic information obtained since the issuance of the Safety Evaluation Report - Construction Permit stage (SER-CP) and supplements to the SER-CP in 1974 (NRC, 1974) in accordance with the SRP.

In the SER-CP the staff and its consultant, the U.S. Geological Survey (USGS), concluded that:

(1) Geologic and seismologic investigations and information provided by the applicant offer an adequate basis for determining that no faults exist at, or in the immediate vicinity of, the plant site that could localize seismicity.

(2) Ground motion values of 0.20 g and 0.12 g for the safe shutdown earthquake (SSE) and the operating basis earthquake (OBE), respectively, are adequately conservative.

Since the issuance of the CP-SER, the applicant has performed further detailed geologic and geophysical investigations of the site and site region. This includes geologic mapping of the excavation for the main power block area, and a fault investigation prompted by a USGS open-file report postulating the existence of the Millett Fault 7 mi south of the plant site. A staff review of this investigation is discussed in Section 2.5.3 of this DSER.

During the current review, the NRC staff identified the following issues for evaluation:

(1) new geological and seismological information discovered since the CP review

(2) the postulated Millett Fault south of the Vogtle site

(3) significance of clastic dikes and associated structures at the site and in the site region

(4) the adequacy of the seismic design response spectrum

Much of the new geologic and seismic data have been developed from research in the southeastern United States, particularly in the Charleston, South Carolina, area. During past licensing decisions the NRC (and the former AEC) have held to the position that the relatively high seismic activity within the Coastal Plain Province in the vicinity of Charleston, S.C., including the 1886 Modified Mercalli Intensity (MMI) X earthquake, was, in the context of Appendix A to 10 CFR Part 100, related to a unique tectonic structure there. Therefore, in the context of the tectonic province approach, an MMI X earthquake should not be assumed to occur anywhere else. This conclusion was based primarily on the persistent historical seismicity that has characterized the meizoseismal zone of the 1886 Charleston, S.C., earthquake. It was also based on evidence, though not strong, of unique geologic structure. Lacking definitive information, both the NRC and the AEC based their conclusions in part on advice from the USGS.

In 1973, with AEC funding, the USGS began extensive geologic and seismic investigations in the Charleston, S.C., region. These studies are still under way. As a result of these investigations, a great deal of information has been obtained, but the source mechanism of the seismicity still is not known. Many working hypotheses have been developed based on the research data. These hypotheses are described in the Virgil C. Summer Safety Evaluation Report (NRC, 1981), and will not be discussed here, except to say that some of these theories postulate that an earthquake the size of the Charleston, S.C., earthquake of 1886 could recur in other areas of the Piedmont, Atlantic Coastal Plain, and continental shelf in addition to the epicentral area.

Because of the wide range of opinions within the scientific community concerning the tectonic mechanism for the Charleston, S.C., seismicity, the USGS clarified its position regarding the localization of the seismicity in the vicinity of Charleston, S.C., including the 1886 MMI X earthquake (November 18, 1982 letter from James F. Devine, USGS, to Robert E. Jackson, NRC). The NRC staff has formulated an interim position concerning eastern seismicity in general and Charleston, S.C., seismicity in particular (see Section 2.6 of this DSER). As part of future research efforts described in that position, the NRC staff is addressing the uncertainties about eastern seismicity by probabilistic studies funded by NRC and conducted by Lawrence Livermore National Laboratory (LLNL). At the conclusion of these studies, the NRC staff will assess the need for a modified position with respect to specific sites. Considering the speculative nature of most of the eastern seismicity hypotheses, the low probability of large earthquakes in the eastern U.S. and present knowledge of the geology and seismology of the region, the NRC staff considers the Vogtle design basis appropriate. The staff does not consider this issue an open item.

After careful review of the new information provided and evaluated by the applicant, the staff concludes that there is no basis for altering its conclusions stated in the CP-SER concerning the safety of the Vogtle site.

The staff has evaluated the FSAR and subsequent documents and information including excavation and trench mapping, and the fault investigation report, "Studies of Postulated Millett Fault" (Georgia Power Co., 1983). The staff has concluded that the applicant has (1) performed satisfactory site and regional geologic and geophysical investigations, (2) reviewed all available pertinent literature, and (3) provided the staff with all information necessary to evaluate, assess, and support the applicant's conclusions concerning the safety of the Vogtle site from the geologic and seismologic standpoint. In addition, the staff finds the applicant has satisfied the requirements of and is in compliance with applicable portions of the following:

(1) Appendix A to 10 CFR 50

(2) Appendix A to 10 CFR 100

(3) SRP Sections 2.5.1, 2.5.2, and 2.5.3

(4) RG 1.70 "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Rev. 2

(5) those portions of RG 1.132, "Site Investigations for Foundations of Nuclear Power Plants," applicable to the development of geologic and seismologic information relevant to the stratigraphy, lithology, geologic history, and structural geology of the site

(6) RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations"

(7) RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants"

In Sections 2.5.1 through 2.5.3, the staff reviews the geologic and seismologic information and bases.

In Section 2.5.4 the staff reviews the stability of subsurface materials and foundations.

2.5.1 Basic Geology

The three fundamental geologic concerns addressed in this review, in order to confirm the geologic safety of the site, were:

(1) the large body of new information rapidly accumulating in the southeastern U.S., in the Piedmont and Coastal Plain, partly because of NRC-funded research, that has resulted in greater knowledge of the subsurface and modifications of interpretations of the tectonic history of eastern North America

(2) the problematic, and almost ubiquitous, occurrence of clastic dikes in the upper Eocene and lower Miocene strata of Georgia, South Carolina, and North Carolina, the origin of which had not been investigated in depth

(3) the possibility of a fault of unknown age exists 7 mi south of the Vogtle site (postulated by USGS Open-File Report 82-156)

The conclusions reached:

(1) No new information detrimental to the safety of the Vogtle site has been uncovered.

(2) Although the origin of the clastic dikes is still not demonstrated with certainty, their apparent great age, in the hundreds of thousands to millions of years, ensures that they are not a safety concern to the plant site.

(3) Geological and geophysical investigations confirm that no fault is present at or near the site that has offset any strata younger than 40 million years old.

The sections that follow provide the background and justification for these conclusions.

2.5.1.1 Regional Geology

The Vogtle site is located within the Atlantic Coastal Plain Province about 25 mi southeast of Augusta, Georgia, and on an upland surface about 100 ft above and adjacent to the Savannah River. At the plant site Coastal Plain sediments range in thickness from 800 to 1,000 ft; consist predominantly of sandstone, shale, limestone, claystone, and marl; and range in age from Upper Cretaceous (138 million years before present to 63 mybp) to Holocene (10,000 years before present to the present).

2.5.1.1.1 Stratigraphy

Except for river alluvium and gravels of Quaternary age (2 mybp to present), the youngest, most extensively exposed formation in the region is the Hawthorn (or Altamaha) Formation, a red and yellow, thick-bedded sandy clay, of Miocene age (25 mybp-5 mybp).

Underlying the Hawthorn Formation is the Upper Eocene (55 mybp-38 mybp) Barnwell Group, with a basal soluble limestone, and a variety of rapidly changing assorted sandy and sandy limestone formations and facies that change rapidly both laterally and vertically.

The oldest exposed rock unit, the Blue Buff marl of the Lisbon Formation which underlies the Barnwell Group, is a calcareous, silty, grayish fossiliferous unit, distinctive in lithology and fauna, and of middle Eocene age (50 mybp-40 mybp). The marl is limited in exposure in the banks of the Savannah River which has begun to cut down through it. Because of its clay content, its density and compaction, the marl is the bearing stratum upon which the applicant chose to build the main power block.

The Eocene formations rest unconformably on Paleocene (63 mybp-55 mybp) clays and black lignitic sands which are also distinctive. These overlie the late Cretaceous (96 mybp-63 mybp) Tuscaloosa Formation unconformably. The Cretaceous sediments are the oldest of the late Mesozoic (240 mybp-60 mybp) marine transgression deposits that constitute the Coastal Plain covermass. These consist of layers of coarse-grained sand and gravel with lenses of clay.

Adjacent to and northwest of the Atlantic Coastal Plain Province is the Piedmont Province. The boundary between the two provinces is known as the Fall Line which is approximately 25 mi northwest of the site. Rocks that characterize the Piedmont disappear beneath the Coastal Plain sediments at the Fall Line, but no structurally significant boundary exists.

2.5.1.1.2 Structure and Tectonics

In several places in the Piedmont, west and north of the Coastal Plain, Triassic down-faulted basins filled with distinctive Triassic sedimentary rocks and some Triassic-Jurassic basaltic igneous rocks are exposed. Similar basins have been recognized beneath the Coastal Plain. One such basin, the Dunbarton Basin, trending northeastward, has been identified on the basis of aeromagnetic anomalies and drilling for the Savannah River Plant (Marine and Siple, 1974). The Vogtle plant overlies the basin, close to the northern boundary. As this basin is downfaulted, presumably into Piedmont rocks, it is assumed that the rocks below and surrounding the basis are Precambrian (older than 570 mybp) and early Paleozoic (570 mybp-240 mybp) metamorphic rock.

At the surface, the Coastal Plain strata overlying the Piedmont and Triassic Basins gently dips toward the southeast, and, on the large scale, is relatively undeformed, reflecting the relative tectonic quiescence of a passive continental margin at great distances from a lithospheric plate boundary, where most seismic, volcanic, and tectonic activity occur according to the plate tectonics paradigm.

Recent deep seismic reflection profiling (Cook et al., 1979) has identified a large-scale detachment surface under the Appalachians, from the Allegheny Plateau to at least the central part of the Southern Piedmont, indicating a large allochthonous mass above the continental basement. There is, however, no certainty that the detachment continues under the Coastal Plain (Iverson and Smithson, 1983). Although it has been suggested that this detachment surface may be localizing seismicity (Seeber and Armbruster, 1982), this has not been demonstrated. As this is also one of the hypotheses regarding the source of the Charleston 1886 earthquake, it is being addressed in the Interim Charleston Position of the NRC (see Appendix A of the DSER).

Geophysical evidence has suggested the possibility that the Piedmont and its extension under the Coastal Plain is constructed of several discrete, lithologically and geophysically distinct masses (or terrains). These are postulated to have coalesced by accretion during middle and late Paleozoic tectonic events as the pre-Atlantic Iapetus Ocean closed, resulting in the continental collision of North America with Africa (Williams and Hatcher, 1982). Although attempts to correlate modern seismicity with these terrain boundaries have been made (Wheeler and Bollinger, 1984) the correlation is not convincing and, therefore, this hypothesis has not gained much favor. Moreover, the Vogtle site is not near the postulated terrain boundaries and thus is not affected by this information.

A recent USGS report (Prowell, 1983) catalogues faults of Cretaceous and Cenozoic age (63 mybp to 2 mybp), primarily in the Coastal Plain and the Piedmont. The faults of those ages closest to the Vogtle site are the Belair en echelon faults at least 25 mi north of the site, and an unnamed pair of small faults forming a 10-ft wide graben in a kaolin mine about 27 mi north of the site, in South Carolina. These faults offset upper Cretaceous sediments. No seismicity has ever been associated with these faults. There are no other faults listed that are near the site, or within 50 mi.

It is significant to note that the Millett Fault, postulated by the same author in an earlier report (Faye and Prowell, 1982), is not included in the more recent catalogue of documented Cretaceous and younger faults.

2.5.1.1.3 Clastic Dikes

Dikelike structures are common and widespread in the younger Tertiary sedimentary strata of the Miocene (24 mybp-5 mybp). They are found in the upper Coastal Plain from Georgia to North Carolina in two upper Miocene formations, the Barnwell Formation and the overlying Hawthorn Formation.

The conclusions drawn by the applicant from a limited reference and full investigation required by the staff is that the dikes do not represent a safety issue because they appear to be very old, between 10,000 yr and 100,000 yr.

The staff concurs with this assessment and suggests the possibility of the dikes having formed close to 20 million years ago.

The following material provides some of the background and information that were the bases for the above-stated conclusions.

The origin of the structures, primarily "clastic" dikes, some with associated faults and folds, is still not understood. Because of the lack of detailed information, they have been proposed, among other possible causes, to have had a seismic origin, possibly related to the Charleston 1886 earthquake (Seeber and Armbruster, 1983), the result of subsidence, differential compaction, weathering and soil formation, or infilled extension cracks in soil. Several authors have studied the dikes (Heron et al., 1971; Zupan and Abbott, 1975; Secor, 1979; McDowell and Houser, 1983). None of these, however, was a long-term systematic study to attempt to relate all the features, nor to determine the age, extent, and geometry.

At the request of the staff, the applicant did more reference work and reconnaissance field observations to provide further information, in an effort to determine if the dikes represent a safety-related concern.

During the course of study, the applicant dug a large waste disposal trench within the Vogtle exclusion area, 2 mi south of the plant, for burying construction-related wastes. The trench, about 900 ft x 75 ft and 35 ft high, exposed several interesting features, including subsidence sags, faults, dikes, and small diapiric structures.

Along with other features seen in the vicinity and within 30 mi of the plant, the trench exposure provided more information on some aspects of dike genesis in the area. The applicant logged the trench and submitted a detailed report and analysis. A staff geoscientist toured the area with the applicant's consultants and examined the trench before a map was made.

The report describes several lithic units exposed in the trench, interpreted to be the upper sands of the Barnwell Formation--the lower massive sandy clay of the Hawthorn Formation which is truncated by erosion and overlain unconformably by a layer of fine gray aeolian sands. The strata are warped into several folds; the downwarps are accompanied commonly by normal faults and grabenlike structures, indicative of differential subsidence of the strata into voids below.

Narrow, relatively planar dikes of clay emerge from the Barnwell sands, commonly along the faults and, often, unrelated to faults. The dikes appear to flare out upward in the dense, compact sandy clay, into a myriad of branches or distributaries, irregular and nonplanar in shape, sometimes somewhat vertical but curving to an arch over downwarps. The gray clay dikes are distinguished only by their light color that stands out against the red or yellow host strata. In some areas along the trench, blobs and clumps of the same material as the dikes can be seen near the contact with the overlying sands.

The applicant reports channeling and weathering of the middle thick sandy clay dike-bearing unit to a soil below the overlying gray sand. The dikes appear to be truncated at the contact with the aeolian sands.

Elsewhere in the site area, dikes often are fairly planar, much thicker than in the trench, to about 3-4 in. as opposed to 1-3 in. in the trench. The most striking feature of the thicker dikes outside the trench is the grading from a thin clay center outward to a yellow sandy border and a deep red rind or crust bounding the dike. The red crust is resistant to the characteristic weathering of the host strata, and thus stands out in relief within the sandy units in which they are found.

A few exposures along the local roads show narrow dikelike projections in coarse sand units in which there appears to be no distinction except in color between the dikes and host stratum. The applicant reports distinctions in the clay content of the "dikes" and the host stratum.

These observations have led the applicant to conclude in the report that the dikes are primarily a weathering phenomenon in which groundwater has made its way along pre-existing fractures, bearing and depositing transported clays or leaching out chemicals to alter the character of the fracture zones. The altered fractures then appear to have been intruded by clastic or clay material. In addition, the surface weathering, soil development, channeling, and truncation of dikes at the aeolean sand-dike bed contact suggest to the applicant a great length of time from the formation of the dikes and the weathering and erosion of the upper part of the unit. The estimate based on weathering alone is 10,000 to 100,000 years ago.

Although the staff does not agree with all aspects of the applicant's report, particularly the mode of formation of the dikes, it concurs that the dikes are very old, probably having formed early in the development of the strata in which they are found, which is between 24 mybp and 20 mybp.

As the dikes intrude sandy strata in narrow channels and flare outward into very dense clay layers, which is the reverse of what would be expected if they were liquefaction structures, it is suggested that the strata must have been very loosely consolidated or still saturated in order for the dike material to penetrate what is now almost impermeable clay layers. Such a condition would exist in the early stages of sedimentation almost 20 mybp-25 mybp.

Therefore, on the basis of current knowledge, and considering the likely great age of the dikes, there is no evidence that these features represent a safety issue for the plant, whatever their origin.

2.5.1.2 Site Geology

It is the applicant's view that extensive core drilling and mapping of the main power block excavation has provided evidence that the bearing stratum is sound, lacking faults, solutioning, or any other geologic feature that may represent a safety concern.

Although small depressions can be seen in several localities in and around the plant site, drilling has shown they are the result of solutioning of the Utley Limestone, a thin, fossiliferous, basal unit of the Barnwell Group. Overlying strata, such as that seen in the trench, have subsided into solution cavities

The applicant reports that no evidence for solutioning in the Lisbon clay-marl bearing stratum below the Utley Limestone has been found in drilling or in the excavations.

The staff's evaluation of the characteristics of the bearing stratum are addressed in Section 2.5.4 of this SER.

2.5.2 Vibratory Ground Motion

The conclusion reached during the construction permit (CP) review by the staff and the staff's consultants, the U.S. Geological Survey, and the U.S. Army Corps of Engineers was that 0.20 g (SSE) and 0.12 g (OBE) accelerations were adequate.

During the operating license (OL) review, the staff's seismological review was based on geologic and seismologic information in the Vogtle PSAR and FSAR and other available literature. The review concentrated on the following:

(1) seismicity of the region and the association of earthquakes with geologic and tectonic features

(2) vibratory ground motion at the site determined from the maximum historical earthquake within the tectonic province, and from recurrence of the 1886 Charleston, S.C., earthquake

(3) ground motion estimated in (2) above compared with the SSE proposed for the site

The staff's review indicates that those conclusions reached during the CP review regarding the adequacy of the SSE and OBE at Vogtle are still appropriate.

2.5.2.1 Seismicity

Bulletins of the southeastern U.S. Seismic Network describe the seismicity since 1977 in the vicinity of Vogtle. Before that, most of what is known about seismicity in the region of the site was based mainly on intensity data.

In general, the seismicity within 50 mi of the site is very low. The maximum historical event within that radius is Modified Mercalli Intensity (MMI) IV. Within 200 mi of the site, the only earthquake of epicentral intensity greater than or equal to VII was the Union County, S.C., earthquake of January 1, 1913. This earthquake, which occurred in the southern Piedmont had an epicentral MMI of VII (Stover et al., 1984) and was not felt at the Vogtle site. In addition to this earthquake, larger earthquakes at distances greater than 200 mi were examined. The New Madrid, Missouri, earthquake sequence of 1811-1812 occurred in the New Madrid Seismic Zone about 530 mi northwest of Vogtle and included a maximum epicentral MMI of XI (Stover et al., 1984). Nuttli (1973) indicated that this earthquake was felt in Georgia with a maximum MMI of VI. The Giles County, Virginia, earthquake of May 31, 1897, occurred about 280 mi north of the site in the Valley and Ridge Province and had an epicentral MMI VIII (Bollinger, 1973). Bollinger indicated that intensity III may have been felt at the site. Another earthquake was the New Brunswick earthquake of January 9, 1982, which was about 1250 mi from the Vogtle site and had a magnitude of 5.75 and an epicentral intensity of VI. This earthquake occurred in the Piedmont-New England Tectonic Province.

An earthquake of significance to Vogtle is one of MMI X (Stover et al., 1984) at Charleston-Summerville, S.C. This is the largest historic event along the eastern seaboard of the United States and occurred in a concentration of seismic activity in the Atlantic Coastal Plain Province about 78 mi from the site. The intensity of this earthquake at the site was VI. Other earthquakes in the Atlantic Coastal Plain are discussed in Section 2.5.2.2 of this report.

2.5.2.2 Tectonic Provinces and Maximum Historical Earthquakes

The Vogtle site is located in the Atlantic Coastal Plain Province. This province extends from the Fall Line (the southern boundary of the Piedmont Province) 25 mi to the northwest, to the edge of the continental shelf to the east and southeast. Other tectonic provinces within 200 mi of the site include the Valley and Ridge Province and the New England-Piedmont Tectonic Province. Other than the Coastal Plain Province, the above provinces and all other provinces outside the 200-mi radius are at sufficient distance so as not to have any impact on the Vogtle seismic design. In the Southern Appalachian area, the staff has, for the purpose of licensing, treated the southern Piedmont as a separate area within the assumed New England-Piedmont Tectonic Province (i.e., McGuire, Summer, Catawba SERs). On the basis of available information, it was not possible to relate past earthquakes to geological structures in the southern Piedmont or the Coastal Plain Province. Except for the Charleston, S.C., area where the high seismicity cannot be considered typical of the rest of the Atlantic Coastal Plain, this province is characterized by low to moderate seismicity. The largest reported earthquakes in the Atlantic Coastal Plain Province are the Asbury Park, New Jersey, earthquake of 1927 (MMI VII), and the Wilmington, Delaware, earthquake of 1871 (MMI VII). Therefore, an event similar to the MMI VII event mentioned above should be considered as the maximum historical earthquake likely to affect the site. On the basis of the estimated felt area, the Asbury Park earthquake and the Wilmington earthquake (Kafka, 1980) had an estimated magnitude less than 5.0. Nuttli and Herrmann (1978) indicated that an appropriate equivalent magnitude to an epicentral intensity of VII is a magnitude 5.3. The staff concludes that the maximum random earthquake in the Coastal Plain Province can conservatively be defined as having an estimated magnitude of 5.3.

The August 31, 1886, Charleston, S.C., earthquake is listed with epicentral intensity X (Stover et al., 1984). The center of the area of maximum intensity was located near Middleton, S.C. The USGS and other investigators are presently investigating the Charleston-Summerville, S.C., region. Interpretations that have been published so far regarding the cause of the Charleston earthquake differ considerably as far as the possible mechanisms are concerned.

The current staff position, as in the past (V.C. Summer Nuclear Station, NUREG-0717) is that in accordance with the tectonic province approach (Appendix A to 10 CFR 100), the effects of a recurrence of an event the size of the Charleston earthquake in the Charleston-Summerville area shall be postulated so as to assess its influence on Vogtle. For discussion refer to Section 2.5 of this report. Additional discussion of the Charleston earthquake is found in Sections ?.5 and 2.6 of this report.

2.5.2.3 Safe Shutdown Earthquake

At the CP stage (SER-CP, Suppl. 1) the staff concluded that the maximum site intensity will be no greater than VII and the SSE acceleration of 0.2 g used for the Vogtle units is adequately conservative.

The staff's position regarding the Vogtle site is that the following seismic issues should be considered for the SSE design.

(1) the maximum random event in the Coastal Plain Tectonic Province, an event of MMI VII equivalent to $m_b = 5.3$, where m_b equals body wave magnitude (Nuttli and Herrmann, 1978) in the vicinity of the site

(2) an event of the size of the 1886 Charleston earthquake (MMI X) occurring in the vicinity of the Charleston Summerville area about 78 mi from the site.

On the basis of the tectonic province approach, the staff finds that the maximum random event in the Coastal Plain was of MMI VII. The resulting mean value of peak horizontal acceleration at the site was estimated to be 0.13 g (Trifunac and Brady, 1975). In recent safety evaluation reports (for example, Millstone, NUREG-1031; Limerick, NUREG-0991) the staff has indicated that site-specific spectra obtained from appropriate suites of strong motion records of earthquakes are more in accord with the controlling earthquake size, frequency content of response spectra, and local site conditions than are standard RG 1.60 spectra. In this method the use of the peak acceleration and RG 1.60 spectrum shapes are replaced by spectra obtained from earthquakes within half a magnitude of the SSE earthquake recorded at a distance less than 25 km from the earthquake source with geologic conditions similar to those at the site. It is the staff's position that spectra obtained by this method more realistically estimate the seismic ground motion for Vogtle. The spectra should be based an appropriate ensemble of records with M_{blg} = 5.3 ± 0.5 obtained at a soil site within 25 km of the source where M_{blg} equals the body wave magnitude based on the 1-g seismic wave. The staff's position has been that the 84th percentile spectrum is appropriate for describing ground motion to be used in evaluating the design spectra of nuclear power plants.

Previous staff reviews of site-specific spectra for soil sites (Wolf Creek, NUREG-0881; Palo Verde, NUREG-0857) indicate that the RG 1.60 spectrum anchored at 0.2 g is adequate for describing ground motion for a magnitude 5.3 event.

With respect to the Charleston earthquake of 1886, Nuttli et al., (1979) estimated the magnitude (m_b) to be 6.6. The distance between the Vogtle site and the meisoseismal area of the 1886 earthquake is 78 mi. Using the equations derived by Nuttli (1983) and Campbell (1981), the staff estimated the 84th percentile accelerations at Vogtle from the reoccurrence of such an event to be less than 0.2 g.

On the basis of consideration of both the local magnitude 5.3 and the reoccurrence of an earthquake the size of the 1886 earthquake in the Charleston area, the staff considers the RG 1.60 response spectrum anchored at 0.2 g used for the design of Vogtle to be acceptable.

2.5.2.4 Operating Basis Earthquake

The applicant has proposed 0.12 g for the acceleration level corresponding to the OBE. This represents more than half of the SSE acceleration 0.20 g, consistent with Appendix A to 10 CFR 100 which indicates that the OBE is at least half of the SSE. Therefore, the staff finds the acceleration level of the operating basis earthquake acceptable.

2.5.3 Surface Faulting

2.5.3.1 Postulated Millett Fault

For the construction permit, the applicant's geological investigations conclu
that there was no surface faulting in the vicinity of the Vogtle site.

Before the FSAR was submitted for the OL review, a document released by the USGS, Open-File Report (OFR) 82-156 (Faye and Prowell, 1982), postulated the existence of a fault, the Millett Fault, 7 mi south of the site. Although the report was a water-resources study about the hydrology and geology of the Coastal Plain in the vicinity of the Savannah River, indirect evidence from secondary sources suggested to the authors the possibility of a fault across the Savannah River.

The report further postulated a second fault, the Statesboro Fault, 32 mi south of the plant.

As interpreted by Faye and Prowell, the Millett Fault trends northeastward across the Savannah River, is approximately 40 mi long, and has vertically uplifted the buried Triassic/Cretaceous contact more than 600 ft on the southeast side of the fault.

The main evidence for the inferred fault came from a comparison of well cuttings taken several years before the USGS study from two water wells, P5R and AL66. Interpretation of the lithic fragments suggested that Triassic rocks were present below -1100 ft in ~~5~~ P5R but above 600 ft in AL66 four miles to the south. Further, an examination of surface and groundwater flow records over a period of 40 years indicated some anomalous characteristics which Faye and Prowell interpreted as resulting from a subsurface barrier. By extrapolation from the Belair Fault 35 mi to the north, the USGS study inferred that an impermeable gouge zone above the postulated Millett Fault forced the south-flowing groundwater from a lower aquifer to a higher one on the south side of the fault. As the trace of the postulated fault traverses a segment of the Savannah River where a straight stretch of the river changes to a more characteristic meandering flow pattern, the USGS study considered this observation additional support for the inferred fault.

Evidence for the trend and length of the postulated faults was by extrapolation from other post-Cretaceous Coastal Plain faults in the southeastern United States.

Although no age of faulting was suggested, the USGS study indicated that rocks at least through the Eocene epoch (55 mybp-38 mybp) were involved.

2.5.3.2 Fault Investigation

2.5.3.2.1 Introduction

At the request of the NRC staff, the applicant undertook a detailed investigative program, because the age and, therefore, capability of the inferred faults were undetermined.

In October 1982, the applicant submitted to the NRC a report of a fault-specific investigation entitled, "Studies of Postulated Millett Fault" (Georgia Power Company, 1983). Two postulated faults, the Millett and Statesboro Faults, were investigated; the primary focus was on the Millett Fault, which was closer to Vogtle.

Techniques Used in the Investigation

The applicant utilized a wide range of techniques to explore the surface and subsurface for both geologic and hydrologic information in order to locate and date the fault. Included in the study was (1) field geologic mapping, (2) aerial and Landsat imagery for remote sensing analysis, (3) core drilling on both sides of the Savannah River and straddling the interval of the two wells described in the open file report, (4) petrographic, x-ray, and heavy mineral analyses of core samples, (5) downhole geophysical studies including gamma, neutron, and electric logging, (6) seismic reflection profiling, (7) regional geophysical and seismicity studies, (7) well-water-level monitoring, (8) groundwater modeling, and (9) analysis of surface water flow.

Staff Review

Because of the wide range of techniques used, several NRC staff reviewers have contributed to this evaluation of the utility's report: a geologist covered the varied geological investigations, a seismologist covered the historic and presentday seismicity of the area, a geophysicist covered the seismic reflection profiling, and a hydrologist reviewed the surface and groundwater study. In addition, the geologist and one of the geophysicists have visited the site region with the NRC project manager and the applicant's staff to examine the cores drilled for this study, the stratigraphy, and various aspects of the surface features, in order to have first-hand experience in evaluating the fault investigation report.

Conclusions:

On the basis of the results of the study, the applicant concluded that there is no evidence a capable fault exists; and that if such a fault does exist, it is more than 40 mybp million years old. The staff agrees that this conclusion is consistent with the reported information and results of the various investigative techniques.

2.5.3.2.2 Summary of Fault Investigation and Results

A brief summary of these results, the applicant's views, and the bases for the applicant's and staff's conclusions follow.

Geologic Investigation

The geologic investigation included (1) field mapping and remote sensing to identify evidence for surface faulting; (2) core drilling with petrographic, x-ray, and heavy-mineral analysis of the strata in the cores in order to correlate layers from core to core to determine any offset of the strata; (3) downhole geophysical logging, which identifies individual strata by characteristic signatures that are dependent upon the physical properties of the rock units, also with a view to correlating the strata from core to core; and (4) review of other geophysical studies.

(1) Field mapping and remote sensing techniques failed to uncover any evidence of surface faulting, or linear features indicative of surface or near-surface rupture. The staff reviewed ~~of~~ some of the original imagery used for the study and checked the site area to verify this conclusion.

(2) Twelve drill cores were taken along two parallel north-south lines, crossing the inferred trace of the Millett Fault, one in South Carolina close to the two wells studied by Faye and Prowell, and one in Georgia. Along both lines, one core was taken from the location closest to the trace of the fault and the others were equally spaced north and south of the fault. Eight holes were cored in Georgia, about 1 mi apart (north-south), and four were cored in South Carolina. Visual examination and petrographic and mineralogic analyses identified distinctive marker beds. One of these, the Blue Bluff marl, appeared consistently between 100 ft above sea level to 100 ft below, showing no change in elevation (other than that due to the gentle regional dip to the south) on either side of the postulated fault in Georgia and South Carolina. This indicates that no vertical offset, and therefore no fault of the type postulated in the USGS open file report, is present down to strata at least 40 million year old at the inferred location of the postulated fault. Furthermore, core VSC-4, on strike with, and 200 ft from, well AL66 in which the USGS interpreted Triassic rocks at an elevation of -600 ft, was the deepest hole in the study. The core at -1000 ft was still in distinctive Cretaceous Tuscaloosa sands. The staff examined this core and agrees that it has none of the characteristics of Triassic rocks and looks much like other Cretaceous samples. This result is in direct disagreement with the USGS interpretation upon which the fault is postulated.

(3) Downhole geophysical logs, especially the gamma log, provided distinctive signatures that verified the petrographic identification of the strata, and in particular the Blue Bluff marl, confirming the continuity of the unit across the inferred location of the Millett Fault.

Seismic Reflection Study

A 19-mi acoustical seismic reflection survey was conducted in the Savannah River in the vicinity of the postulated Millett Fault. The survey used three different energy sources (Uniboom, a 10-in.^{3} air gun, and a 20-in.^{3} air gun), to obtain high resolution and deep penetration of the subsurface formations. The survey was carried out in water between 10-ft and 25-ft deep.

The reflection survey identified several key horizons, A through I, at different depths ranging between +70 and -1150 ft. Some of the horizons, such as reflector E, which correlates with the top of the limestone that is the lowest unit in the Barnwell Group, of the late Eocene age (40 mybp) and a reflector G, which represents the unconformable contact between the unnamed sands of the Middle Eocene Lisbon Formation and the top of the Paleocene kaolinitic clay (60 mybp) are well correlated with adjacent core holes. Some of the reflectors are well defined; others are weakly defined. Some of the horizons showed small features which may indicate past channelization or buried karstic surfaces. The continuity of the reflectors above the Triassic/Cretaceous contact and the absence of noticeable displacement in the higher horizons above -500 ft elevation indicate the absence of faulting within the last 60 million years in the vicinity of the postulated Millett Fault. This conclusion agrees with that of the applicant's report, that no capable fault has been identified by the seismic reflection data obtained for this study.

In addition, seismic reflection data obtained by the applicant from the Savannah River Plant investigation of the deeper horizons in the vicinity of both plants suggest the possibility of a small normal offset of the Triassic/Cretaceous contact of 50-100 ft in the vicinity of the postulated fault. No evidence, however, for offset in younger horizons can be detected.

Seismology Study

The available seismicity information includes (1) felt earthquakes, (2) recent instrumentally located events, and (3) data from the Savannah River Plant array, just across the Savannah River from the Vogtle site. The applicant concludes that historic seismicity reveals no evidence of active faulting in the area. This conclusion is consistent with the data. The seismicity within

50 mi of the site has been scattered and low level (maximum MMI of IV). No clustering of earthquakes is occurring near the postulated Millett or Statesboro Faults.

Hydrology Study

Faye and Prowell used, as part of their case, several hydrologic arguments to support the existence of the postulated fault. The applicant investigated these arguments thoroughly in this study. The applicant addressed issues of different unit base flows in river reaches (generally above and below the postulated fault), levels of well water across the postulated fault, and groundwater piezometric surface contours. After carefully reviewing all the contractor's hydrologic evaluations, the staff concurs in the conclusion that the hydrologic data provide "no basis to support or preclude the existence of a fault."

Conclusions

On the basis of (1) the continuity of strata across the inferred location of the postulated Millett Fault as determined by (a) drill cores, (b) downhole geophysical logging, and (c) seismic reflection profiles; (2) the absence of Triassic rocks at levels above -1000 ft, as shown by drill core VSC-4 on the south side of the postulated fault; (3) the scattered and low-level seismicity; and (4) the hydrologic information which neither indicates nor disproves the presence of a fault, the applicant's report concludes that there is no evidence that a capable fault exists in the vicinity of the Vogtle plant.

The staff has carefully reviewed the report, has visited the site and examined the cores, the logs, and remote sensing imagery, and checked the surface features. The ~~reviewers~~ staff considers the applicant's conclusion to be consistent with the data as reported, and conclude, therefore, that no capable fault as defined in Appendix A to 10 CFR Part 100 is present in the vicinity of the Vogtle plant, based on all presently available data.

Further support for this conclusion comes from Prowell's later (1983) report on Cretaceous and younger faults of eastern North America. The Millett and Statesboro Faults are not included as documented faults.

It is concluded, therefore, that no surface faulting capable of localizing earthquakes is present at the plant site on in the vicinity of the site.

2.5.4 Stability of Subsurface Materials and Foundations

Sections 2.5.4.1 through 2.5.4.7 summarize the staff's geotechnical engineering review of the Vogtle Electric Generating Plant, Units 1 and 2, as presented in the Final Safety Analysis Report (FSAR) through Amendment 9, dated August 1984, and the applicant's response to staff questions Q241.1 through Q241.24. The stability of subsurface materials and foundations (FSAR Section 2.5.4) has been evaluated in accordance with the applicable criteria outline in 10 CFR 50; 10 CFR 100; Appendix A of 10 CFR 100; RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (Rev. 3); RG 1.132, "Site Investigations for Foundations of Nuclear Power Plants;" RG 1.138, "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants"; and the Standard Review Plan (SRP), NUREG-0800, July 1981.

2.5.4.1 Site Conditions

The site conditions which exist at the Vogtle site do not involve stability of slopes nor embankment and dams, and therefore, FSAR Sections 2.5.5 and 2.5.6 are not addressed in this report. X

2.5.4.1.1 General

The Vogtle Electric Generating Plant Units 1 and 2 is located on the southwest side of the Savannah River in Burke County, Georgia, approximately 26 mi southeast of Augusta, Georgia. The topography of the site is one of rolling hills with original ground surface elevations in the immediate plant area (excluding the river intake canal and structure) generally ranging from el 255 ft above mean sea level to el 280 ft. Final plant grade at el 219.5 ft required the removal of the upper natural soils. The Savannah River at its closest point to the site is approximately 3000 ft northeast from the main plant area and has a normal water elevation of 80 ft. The maximum water level in the Savannah River has been estimated at el 165 ft under assumed probable maximum flood (PMF) conditions that include allowance for wave runup. As described in Section 2.4 of this SER, groundwater movement has been observed in an upper water table aquifer system and a confined aquifer system that is located below approximately el 70 ft. The foundation designs of seismic Category I structures have been based on a maximum groundwater elevation of 165 ft and this maximum level would be located in the upper water table aquifer system.

2.5.4.1.2 Site Foundation Description

The subsurface conditions as revealed by explorations and foundation excavations in the plant site area may be divided into three principal strata. The top stratum consists of sands (SP), silty sands (SM), and clayey sands (SC) and in the FSAR this top layer is identified as the upper sand stratum. The upper sand stratum is about 85 ft in thickness below plant grade and has a bottom elevation at approximately el 135 ft. A shelly limestone (Utley Limestone), which subsurface explorations showed to be subjected to extensive leaching and to solution cavities, is located at the base of the upper sand stratum and ranged up to 12 ft in thickness. The stratum below the Utley Limestone is the major foundation-supporting layer and is identified as the clay marl bearing stratum. The clay marl stratum is approximately 65 ft in thickness in the main plant area and ranges in elevation between 135 ft and 70 ft. The clay marl stratum is a gray to greenish-gray, calcareous silty clay with shell fragments and interbedded with limestone and sand lenses. Drilling recoveries show the marl to be predominantly a hard to very hard, weakly cemented material with some zones of softer marl. Seismic explorations indicated a velocity interface about 15 ft below the top of the clay marl stratum which is a reflection of weathering in the upper 15 ft of the marl zone. A thick, dense, coarse-to-fine sand zone with minor interbedding of silty clay and clayey silt layers is located beneath the clay marl stratum. This lower sand stratum is estimated to be in excess of 750 ft in thickness;

recorded blow counts per foot of penetration in the standard penetration test (SPT) are generally in excess of 100 blows.

The applicant decided to excavate the upper natural soils and extend this excavation into the clay marl stratum in order to avoid foundation difficulties with the shelly limestone layer and to eliminate any potential for liquefaction in the upper sand stratum. Liquefaction had been indicated to be a possibility in the upper sands when evaluated, allowing for a seismic event equivalent to the safe shutdown earthquake (SSE). This extensive foundation excavation operation required the removal of approximately 5 million cubic yards of soil to el 130 ft and measured approximately 1000 ft along each side at the bottom of the excavation which was roughly square in shape. A deeper excavation to el 108.5 ft in the clay marl stratum was made over a rectangular area measuring 120 ft x 440 ft to accommodate the basemat for the deeper portion of the auxiliary building. Description of the geologic mapping, dewatering activities, rebound monitoring, surface cleanup and protection measures, and foundation inspection and approval procedures are provided in Appendix 2B of the FSAR.

The foundations of seismic Category I structures that are evaluated in this report include the reactor containment buildings, nuclear service cooling water (NSCW) towers and pumphouses, auxiliary building, fuel-handling building, control building, diesel generator buildings, diesel fuel oil storage tanks and buildings, condensate storage tanks, auxiliary feedwater pumphouses, refueling water storage tanks, reactor makeup water storage tanks, and Category I piping, conduits and tunnels. Reinforced concrete mat foundations were used for Category I structures with the exception of wall footings for certain tanks, and box culverts for piping and tunnels. FSAR Figure 241.[illegible]-1 provides a plan view of the main plant layout and identifies the outline of seismic Category I structures.

Excavating the natural soils to the clay marl stratum placed the foundations of most seismic Category I structures on compacted backfill. Only the more deeply founded auxiliary building, NSCW towers, and instrumentation cavity of the containment building are founded on the clay marl. All other foundations of the power block structures are supported on Category 1 backfill and have foundation elevations ranging from el 158 ft (reactor building) to el 218 ft (reactor makeup water storage tanks).

Category 1 backfill was selectively excavated from nearby borrow sources and consisted of medium to fine sands (SP) and sands with some silt (SP-SM). Although permitted by PSAR and FSAR documentation to contain up to 25% by weight passing the No. 200 sieve, the percent of fines actually contained in the Category 1 backfill that was placed and compacted was limited in the field to about 12%. All Category 1 backfill in the power block area was to be compacted to an average of 97% of the maximum dry density determined by American Society of Testing Materials (ASTM) D1557, with no tests below 93% and not more than 10% of the tests between 95 and 93%. On the basis of the results of test fill studies, the applicant indicated his intent was to control the placement moisture content of the Category 1 backfill to within ± 2% of the optimum moisture content determined by ASTM D1557 (FSAR Section 2.5.4.5.2.7). The staff's evaluation on the adequacy of the compacted backfill is subsequently discussed in this DSER in Section 2.5.4.3.

2.5.4.1.3 Site Investigations

Field investigations at the site were initially started in January 1971 and were continued during construction; 38 borings were drilled from the bottom of the foundation excavation on the top of the clay marl bearing stratum in 1977 in the power block area. The field investigations have included drilling, geophysical seismic surveys, groundwater studies, and geologic mapping of foundation excavations. A total of 474 holes have been drilled, of which 111 holes were completed subsequent to the PSAR investigations. Table 2B-1 of the FSAR provides a list of borings with summary information for foundation investigations completed for the PSAR.

The site investigations were completed to define the various subsurface materials and stratification, to obtain soil samples for laboratory testing and the establishment of engineering properties, to identify sources of suitable borrow, to permit measurement of shear and compression wave velocities, to determine in situ foundation material permeabilities and groundwater movement and for geologic mapping and inspection (e.g. for faulting, cavities, soft zones) of foundation excavations for approval before concrete was placed. The investigations completed at the Vogtle site did not extend to firm bedrock which is estimated to be approximately 750 ft below the bottom of the clay marl stratum.

On the basis of its review of the information presented in the FSAR, the staff concludes that the site investigations completed by the applicant are acceptable and adequate to identify the important subsurface features and foundation conditions, with the exception of the "CS" series holes drilled in 1977 from on top of the clay marl stratum.

In Question 241.3 of its review of the FSAR, the staff attempted to understand the reasons for the poor core recovery in the clay marl stratum that was indicated in 9 of the 36 borings which were drilled in 1977. Because the staff cannot accept portions of the applicant's response to Q241.3 which were provided in Amendment 6 (May 1984), this concern remains an open review item. The basis for this staff's position includes the following:

(1) The Applicant's response to Q241.3 indicates that the 1977 marl-sampling program was not an exploration program and was not designed to obtain 100% core recovery but, rather, was intended to obtain selected samples of the clay marl for laboratory testing. The staff has great difficulty in understanding this response. The staff finds that the borings of the "CS" series, some of which were drilled within the foundation limits of the NSCW towers, auxiliary building, containment buildings, control building, and fuel-handling building, were important to assessing the foundation competency of the clay marl stratum and should have been drilled in accordance with good engineering practice and the guidelines of RG 1.132 "Site Investigations for Foundations of Nuclear Power Plants." Good engineering practice would require a full and complete description of the materials encountered for the entire depth and an explanation on the boring logs for 0% recoveries in order to properly assess this condition on the adequacy of foundation design and future building peformance. Supplementary explorations specifically intended to determine the features of the zones of the poor core recovery would normally be completed. The applicant's response

that this program was intended to obtain selected representative samples of the marl stratum needs to be further explained, if it implies only good intact rock core specimens were to be laboratory tested.

(2) The staff is unclear as to the significance in the applicant's response of indicating that six of the nine borings with zero recovery are located outside of the limits of seismic Category I structures. Certainly the borings are close enough to safety-related structures to reasonably permit extrapolation of the subsurface information to these structures. The staff recognizes that the applicant has made similar extrapolation of subsurface data in its assessment of the clay marl stratum where it has relied on information from caisson excavations and outcrop locations that exist at considerably greater distances from seismic Category I structures. The staff also recognizes that widely spaced borings may not, in many instances, allow detection of adverse anomalies, discontinuities or lenses or pockets of unsuitable material and that there is an important need to respond to these indications, such as 0% recovery when it does occur, particularly at locations where leaching and solution cavities have been observed.

(3) Although the staff agrees with the applicant that the preponderance of subsurface information indicates that no open cavities exist below the top of the clay marl stratum, the staff is less certain that zones of softer material do not exist in the clay marl. Such softer zones could be a factor as engineering properties there would be significantly lower than values used in foundation design. The staff attempted to gain confidence in the foundation adequacy of the marl layer, in spite of the difficulties with the "CS" borings, by reviewing the recorded settlements of structures founded on or close to the top of the clay marl stratum. As discussed in Section 2.5.4.4.3 of this report, the settlement records provided for the auxiliary building and reactor containment building are indicating total settlements larger than anticipated for the years of plant operation with approximately 87% of the total static loading already placed. Settlement records for the NSCW towers have not yet been provided.

2.5.4.2 Engineering Properties of Foundation Materials

The types of foundation materials have been described in Section 2.5.4.1.2 of this report. The engineering properties of these materials were established by laboratory testing and field testing and are summarized in the following FSAR tables and figures:

- Range of Engineering Static Properties for Site (Natural) soils (FSAR Table 2.5.4-1)

- Engineering Static Properties Adopted in Design-Site (Natural) Soils (FSAR Table 2.5.4.2)

- Engineering Dynamic Properties Adopted in Design-Site Soils (FSAR Table 214.12-1 and FSAR Figures 3.7.B.1-9, 3.7.B.1-10, 3.7.B.2-6, 3.7.B.2-7)

On the basis of its review of the information provided in the FSAR, the staff concludes that the engineering properties determined for foundation materials are acceptable and conform with the applicable portions of the Commission's regulations, the Standard Review Plan, and RG 1.138, "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants." The staff also notes, however, the extremely variable properties of undrained shear strength and soil modulus of elasticity for the clay marl that have been established in laboratory testing. It is the extreme variability that gives the staff concern for the appropriateness of the adopted design values for undrained shear strength and soil modulus of elasticity (FSAR Table 2.5.4-2) and for the strain-dependent dynamic soil shear moduli and damping curves (Figures 3.7.B.2-6 and 3.7.B.1-9) for the clay marl stratum. The staff's concerns are further discussed in this report in Sections 2.5.4.1.3, 2.5.4.4.3, and 2.5.4.4.6.

2.5.4.3 Engineering Properties of Backfill Materials

A description of the materials placed and compacted as Category 1 backfill soils has been provided in Section 2.5.4.1.2. In localized areas that restricted compaction because of space limitations, lean concrete was used in place of backfill. The engineering properties of Category 1 backfill were established by laboratory testing and are summarized in the following FSAR tables and figures:

- Engineering Static Properties Adopted in Design (FSAR Table 2.5.4-8)

- Engineering Dynamic Properties Adopted in Design (FSAR Figures 241.12-1, 3.7.B.1-8, 3.7.B.2-5)

Test fills on Category 1 backfill were constructed to determine the appropriate lift thickness and the number of passes, and to evaluate the performance of different compactors in order to achieve the required maximum densities.

The FSAR, as originally submitted, did not provide information on the actual results from field control testing on compacted Category 1 backfill. In response to staff Q241.4 and to discussions at a March 1984 site visit, the applicant provided compaction control records for backfill material placed during the first 6 months of 1983. Following its review and evaluation of the field records, the staff expressed the following difficultes with the submitted information.

(1) Many of the laboratory-determined maximum dry densities (ASTM 1557) appeared unusually low. These low densities, when used to establish the percent compaction, would result in the reporting of values in excess of 100%.

(2) The field procedures used by the applicant to demonstrate that fill placement moisture contents met FSAR commitments also gave a problem to the staff. The field procedures followed would consist of running a fill moisture content immediately before compaction to verify that the fill moisture was within the specified range of an average optimum moisture content that had been predetermined on stockpiled fill material. The staff's problem resulted from the moisture testing of the fill before compaction and using this result to decide on moisture acceptability rather than the more normal practice of testing the fill after compaction. The normal practice of testing after compaction has the advantage

of verifying the uniform mixing of water throughout the entire lift thickness, which is required by the compaction control specification. Also it is the compacted condition of the fill (density and molding water content) which will govern the resulting engineering properties. The applicant's procedure of using an average optimum moisture content also presented a problem to the staff because it differs from normal procedures where the optimum moisture is directly established in the lab on the same type of material that is field tested for density.

In order to address the staff's concern with compaction of Category 1 backfill, a confirmatory laboratory testing program was agreed upon with the applicant and testing was initiated in June 1984. The major objectives of the confirmatory testing program consisted of the following:

(1) Evaluate the acceptability of the quality control test procedures and test results for the compacted Category 1 backfill by determining whether FSAR commitments (Section 2.5.4.5.2) had been met in obtaining the required maximum dry densities. The check on acceptability was to be made by requiring both the field laboratory and an independent testing laboratory to perform control tests (gradation, moisture-density relationships, relative density and permeability) on the same Category 1 backfill material. Identical samples of fill material were selected from existing stockpiles.

(2) Reexamine the FSAR commitments on compaction control (maximum dry density and placement moisture contents) after evaluation of the results from the confirmatory testing program and determine if modifications of FSAR commitments are warranted for the future control of Category 1 backfill that remains to be placed.

The laboratory results of the confirmatory testing program were provided to the NRC in an August 10, 1984, submittal. The applicant has also submitted a report to the NRC dated September 27, 1984, which evaluates the testing program results. The staff has not yet evaluated this report. The confirmatory testing program is an open item. Preliminary observations of the staff based on the results provided in the August 10, 1984, submittal indicate the following:

(1) A comparison of the maximum dry densities determined by the field laboratory and the independent testing laboratory indicates that the independent laboratory results show higher value of maximum densities in all of the 12 tests performed using ASTM D1557. The increase in densities ranged from 0.8 lb/ft^3 up to 3.5 lb/ft^3. The maximum difference in dry density from the loosest state to the densest state for the medium to fine sand (SP) is about 20 lb/ft^3. The differences in results between the testing laboratories for optimum moisture content determinations were more widely scattered--differences ranged from 7.6% moisture below optimum to 2.5% above for the tests on the same type of material.

(2) The test results also indicate that the backfill soils which have a small amount of fines (less than 6% passing the No. 200 sieve) attained their highest densities when tested in the relative density test (ASTM D4253) in six of the seven tests performed. The increase in maximum dry densities between modified Proctor (ASTM D1557) by the field laboratory and the relative density (ASTM D4253) testing ranged from 2.4 lb/ft^3 up to 4.5 lb/ft^3. Recognition of these results would encourage a modification to current control procedures that requires the running of both the relative density test and the modified Proctor test in order to establish the maximum dry densities and percent compaction for this type backfill which has the small amount of fines.

The opportunity for the staff to observe a portion of the actual testing by the independent laboratory has helped the staff understand why higher densities are not more consistently obtained in the ASTM D1557 test. During this laboratory test a large part of the heavy compaction effort that is specified is actually lost during compaction, because of the large shear displacements which repeatedly occur in the test sample mold under the impact of the hammer weight. These displacements and resulting loss in compactive effort appear to be greatest for soils being compacted at moistures on the wet side of optimum moisture content. The staff believes the large displacements and resulting loss in compactive effort are a major reason for the differences in test results between the field laboratory which was indicated to use a mechanical tamper, and the independent testing laboratory, which manually compacted the test specimens. Under manual compaction conditions there is a natural tendency to locate the next hammer blow where the displacements are occurring, whereas in mechanical tamping, a set pattern and sequence in compaction effort is followed. The differences in results between the two testing laboratories, therefore, are more the result of the particular soil behavior under the specified compactor and allowable test procedures of ASTM D1557, rather than the result of errors or unacceptable test procedures between the laboratories. The staff also believes the densities obtained in the relative density tests are higher because the displacements do not occur and that the relative density test is better suited for Vogtle backfill materials with less than 6% fines.

The staff anticipates that in the applicant's future report which addresses the objectives of the confirmatory test program, the higher maximum dry densities obtained, for the three types of backfill materials tested, will be used to establish the percent compaction for all Category 1 backfill compacted to date. Preliminary observations, when using the higher densities for the field records from the first six months of 1983, indicate that FSAR requirements have essentially been met but at lower percent compaction values than originally reported.

2.5.4.4 Foundation Stability

With the exception of the NSCW towers, the instrumentation cavity of the containment building and the auxiliary building, all seismic Category I structures are founded on compacted Category 1 backfill. The applicant's response to staff question Q241.17 indicates that little or no settlement data are presently available for the auxiliary feedwater pumphouses, diesel generator buildings, diesel fuel oil storage pumphouses and Category I tanks since these structures are either in the initial stages of construction or construction has not begun. Also in response to question Q241.17, the applicant indicates that settlement records for the NSCW towers and Category I tunnels are to be submitted to the NRC. Until this construction is completed and described and the settlement data are provided to the NRC for evaluation, the staff would not be in a position to complete its final report on foundation stability. Therefore, this is an open item.

The applicant's response to Q241.1 indicates that the radwaste transfer building and radwaste transfer tunnel, although not seismic Category I structures, could potentially adversely affect seismic Category I structures. On the basis of guidance in RG 1.29, the foundation design and construction of these structures would also then be required to meet the equivalent of seismic Category I requirements. The foundation design and construction information for these structures should, therefore, be provided for staff review. The applicant is also asked to provide the reasons why the radwaste solidification building, which has been founded on drilled caissons (refer to staff question Q241.20), would not fall into this same category. This item is open.

2.5.4.4.1 Construction Notes

During the years of plant construction, several conditions became evident that could affect the long-term stability of the foundation. These conditions included (1) detecting the solution cavities in the Utley Limestone above the clay marl stratum and then treating the foundation and (2) detecting the erosion of already placed Category 1 backfill (November 1979) that resulted from the heavy rainfall and surface runoff that entered the foundation excavation area.

Several cavities of varying size were exposed on the slopes of the foundation excavation where the slopes intersected the limestone shell bed. The largest cavity was located on the northwest corner of the power block area and measured 10 ft x 10 ft at the opening and extended approximately 30 ft back into the slope where it narrowed to a small size. Other small cavities were encountered at varying intervals all along the north side of the power block foundation excavation. The cavities were cleaned of loose debris and then backfilled with crushed rock. In the larger cavity the crushed rock was forced into the opening for at least a 25-ft length beyond the opening by a ram attached to the blade of a bulldozer. The cavities were filled to provide a buttress on the foundation excavation slope against which structural backfill could be placed and compacted.

The areas affected by the soil erosion problem that became evident in November 1979 included zones (1) between the electrical shafts of the control building for Units 1 and 2 and the turbine building, (2) between the containment buildings for Units 1 and 2 and the electrical tunnels, (3) along the perimeter of the Unit 1 containment building, and (4) under the mud slab of the Unit 2 tendon gallery. The applicant provided detailed description of the areas affected and the remedial measures completed in the August 15, 1980 report, "Final Report on Dewatering and Repair of Erosion in Category 1 Backfill in Power Block Area." The limits of the areas disturbed by erosion were determined by inspection, field explorations, and testing, using proving ring and dynamic core penetrometers and sand cone density tests. The remedial measures completed between January and August, 1980, included (1) reshaping foundation excavation slopes and protecting them with gunite, (2) improving surface water controls, (3) installing additional dewatering measures and piezometers to ensure that the level of the water table was deep enough in the Category 1 backfill to allow mud slabs and disturbed soils to be replaced under dry conditions, and (4) pumping grout into voids in the backfill in space-restricted areas.

The staff concludes that the applicant's investigations and remedial measures are acceptable, but also recognizes that the success of these actions in addressing the erosion problem and filling the cavities can best be judged by continued visual inspections of the structures' performance and long-term settlement behavior. The staff asks that the applicant locate all observed cavities on appropriate FSAR figures and describe them in terms of their extent. This information could prove useful when future settlement records are reviewed. This is a confirmatory item.

2.5.4.4.2 Bearing Capacity

The applicant's responses to staff questions Q241.5 and 241.15 indicate that the results of bearing capacity analysis under static and dynamic loading will be submitted to the staff by December 1, 1984. The staff will complete its safety evaluation on the acceptability of the resulting margins of safety against bearing-capacity-type failure after it reviews the December 1984 submittal. This item is open.

2.5.4.4.3 Settlement

The applicant has responded to staff questions Q241.17 and 241.18 by providing a portion of the settlement records for seismic Category I structures. The settlement records that remain to be submitted are identified in Section 2.5.4.4 of this report. The settlement records provided in reponse to Q241.17 are in a form that makes review and evaluation difficult. The applicant needs to use the same time scale for plotting data on settlement versus time and for plotting data on application of loading versus time. Only then can settlement behavior under structure loading be reasonably evaluated. Such improvement is important for future data submittals and essential before the staff can decide on long-term settlement-monitoring requirements for the Technical Specifications.

The applicant's reply to staff question Q241.18 is not acceptable for the following reasons:

(1) Contrary to the staff's request there is no discussion or comparison between total and differential settlements allowed for in design and actual settlement records at specific locations of structures. The applicant's statement that, because all major seismic Category I structures are separated from each other by seismic gaps they are unaffected by differential settlements, fails to recognize that excessive settlements can cause (a) unacceptable cracking in structures and (b) high stresses and unacceptable tipping of the structure. The staff's review of the limited settlement records that the applicant has provided indicates that values of total settlement larger than the upper predicted values have been recorded for certain settlement markers (Nos. 128, 133, 134, 234, 235, 323, 324, 325) at the auxiliary building and Unit 1 containment building. These larger settlements have been recorded with approximately 87% of the total static load applied in comparison to the predicted values which were estimated for the 40-yr plant life. The discussion requested in staff question Q241.18 asked that the specific maximum recorded settlements be identified for each structure and be compared to design estimates. The recorded settlements may have potentially significant and adverse impact to future structural performance (e.g., cracking, high stresses). The larger settlements being observed may be the result of the clay marl stratum being

less competent than orginally anticipated or may possibly be related to the soil erosion problem. Foundation design modifications may be warranted. The engineering basis for the response to the above considerations should be clearly described in the applicant's response.

(2) The applicant's response to Q241.18 provides general information on differential settlements which are typically addressed in the design of seismic Category I piping but the response needs to be completed by providing information specific for Vogtle (locations, sectional views where appropriate) showing where total and differential settlements have been recorded and discussing the significance of these settlements on the piping system's capability to safely withstand them.

2.5.4.4.4 Lateral Pressures

The walls of seismic Category I structures below plant grade el 219.5 ft were designed for static loading to resist at-rest lateral earth pressures using the equivalent fluid pressure concept. The adopted design pressure diagrams are presented in response to Q241.21 and are discussed in FSAR Section 2.5.4.10.5. An at-rest lateral earth pressure coefficient of 0.7 was used in design for the backfill materials. A water level at el 165 ft was conservatively used to establish the hydrostatic pressure contribution to lateral pressures.

For dynamic loading conditions, the Seed simplified version of the Mononobe-Okabe method was used for active earth pressures. Dynamic passive pressures were calculated using a method by Kapila that is based on the Mononobe-Okabe method. A peak horizontal ground surface acceleration of 0.20 g was used for SSE condition to estimate inertial forces.

With the exception of the approach for dynamic passive pressures, the staff concludes that the methods used to estimate lateral earth pressures are conservative and acceptable and are in accordance with current state-of-the-art engineering practice. The staff needs to complete its evaluation of the method used to establish dynamic passive pressures. This is a confirmatory item.

2.5.4.4.5 Liquefaction Potential

The applicant's decision to remove all the upper sand stratum materials and excavate the power block area to the top of the clay marl stratum has the significant advantage of eliminating the potential for liquefaction that was indicated for the upper sand material. The staff agrees with the applicant that neither the clay marl stratum nor the deeper, dense lower sand stratum is susceptible to liquefaction under SSE conditons assuming a peak horizontal ground surface acceleration of 0.20 g.

To demonstrate that an acceptable margin of safety against liquefaction is available for structures and piping founded in the Category 1 backfill, the applicant conducted cyclic shear strength tests on a representative range of backfill materials which were compacted to 97% of maximum dry density determined by ASTM D1557. The lowest factors of safety against liquefaction type failure were on the order of 1.9 to 2.0 using the Seed and Idriss (1971) simplified method and the cyclic test results.

The staff concurs with the applicant's findings that an acceptable margin of safety against liquefaction potential does exist for Category 1 backfill that is compacted to 97% of maxium dry density. The staff plans to reexamine this conclusion on liquefaction potential following resolution of the concern discussed in SER Section 2.5.4.3 on compaction control procedures. This is a portion of that open item.

2.5.4.4.6 Dynamic Loading

In staff questions Q241.11 and 241.12 the applicant was asked to provide the soil properties (shear modulus and damping values) for the soil springs used in the finite-element and lumped-parameter dynamic studies and to compare these properties with the results from field geophysical surveys and laboratory cyclic triaxial testing completed for Vogtle. The staff has reviewed the applicant's responses and concludes that:

(1) The strain-dependent soil damping curves for the compacted sand backfill (FSAR Figure 3.7.B.1-8) and the lower sand stratum (FSAR Figure 3.7.B.1-10) are reasonable best estimates and are acceptable. The staff does not understand the basis for the change made by the applicant for the damping curve (FSAR Figure 3.7.B.1-9) for the clay marl stratum between the time of FSAR docketing and Amendment 6 (May 1984). The staff requires that the applicant provide the basis for this change and include a comparison that permits evaluation of the effects on structure behavior (e.g., reponse spectra) when both soil damping curves are used in design. This is a part of the open item discussed previously in Sections 2.5.4.1.3, 2.5.4.2, and 2.5.4.4.3 (item 1 in Table 2.2).

(2) The staff finds the strain-dependent shear moduli curves (FSAR Figures 3.7.B.2-5 through 3.7.B.2-7) to be reasonable best estimates. The staff requires, however, that the results of the study which varied the soil shear moduli values (discussed in FSAR Section 3.7.B.2.4.1) by a factor of ± 1.5 be provided and discussed, and that the results permit a comparison of the resulting response spectra with final design spectra for the range of shear moduli values considered. The acceptability of variations in soil dynamic properties is an open item.

2.5.4.5 Instrumentation and Monitoring

Because of the primary importance of the groundwater regime in the solution process and the resulting potential for ground subsidence, the staff will require adequate monitoring of both groundwater levels and settlement during the life of the Vogtle project.

The observation wells which will be active as indicated in the applicant's response to Q241.10 need to be supplemented with additional wells closer to the main plant complex and be located in both the upper water table aquifer and in the clay marl stratum at representative depths. The staff requires that the applicant provide a plan which locates, as requested, the additional wells and the pertinent information on well installation and monitoring that is requested in Q241.10.

The staff requires clarification of the applicant's response to Q241.19 as to whether it is intended that all settlement markers shown on FSAR Figure 241.19-1 are to be monitored for the entire life of the Vogtle. The staff feels it is initially important to resolve the issues identified in Section 2.5.4.4.3 of this report, particularly to have a better understanding of the significance of the settlements which have already occurred. Following resolution of the concerns expressed in Section 2.5.4.4.3, the staff would be in a better position to evaluate the applicant's proposal for long-term settlement monitoring. This is an open item.

2.5.4.6 Remaining Issues

The remaining operating license safety review items which have been identified and discussed in the preceding SER sections are listed in Table 2.2.

2.5.4.7 Conclusions

On the basis of the staff's review of the information provided by the applicant in the FSAR, the staff has concluded that the following features of foundation stability are acceptable, except as impacted by items in Table 2.2.

(1) site investigations
(2) engineering static properties of foundation materials
(3) foundation preparation measures including treatment of cavities
(4) methods for estimating lateral earth pressures
(5) margin of safety against liquefaction potential
(6) engineering dynamic soil properties

Final conclusions on plant foundation stability requires resolution of the remaining issues identified in Table 2.2.

2.5.5/2.5.6 The site conditions which exist at Vogtle do not involve stability of slopes nor embankment and dams; therefore, these are not addressed.

2.6 Interim Position on Charleston Earthquake for Licensing Proceedings

The staff position with respect to the MMI X 1886 Charleston earthquake has been that, in the context of the tectonic province approach used for licensing nuclear power plants, this earthquake should be restricted to the Charleston vicinity. This position was based, in part, on information provided by the USGS in a letter dated December 30, 1980 from J. E. Devine to R. E. Jackson (see Summer Safety Evaluation Report). The USGS has been reassessing its position and issued a clarification on November 18, 1982, in another letter from J. E. Devine to R. E. Jackson. As a result of this letter, a preliminary evaluation and outline for NRC action was forwarded to the Commission in a memorandum from W. J. Dircks on November 19, 1982.

The USGS letter states that:

> Because the geologic and tectonic features of the Charleston region are similar to those in other regions of the eastern seaboard, we conclude that although there is no recent or historical evidence that other regions have experienced strong earthquakes, the historical record is not, of itself, sufficient grounds for ruling out the occurrence in these other regions of strong seismic ground motions similar to those experienced near Charleston in 1886. Although the probability of strong ground motion due to an earthquake in any given year at a particular location in the eastern seaboard may be very low, deterministic and probabilistic evaluations of the seismic hazard should be made for individual sites in the eastern seaboard to establish the seismic engineering parameters for critical facilities.

The USGS clarification represents not so much a new understanding but rather a more explicit recognition of existing uncertainties with respect to the causative structure and mechanism of the 1886 Charleston earthquake. Many hypotheses have been proposed as to the locale in the eastern seaboard of future Charleston-size earthquakes. Some of these could be very restrictive in location while others would allow this earthquake to recur over very large areas. Presently none of these hypotheses are definitive and all contain a strong element of speculation.

The staff is addressing this uncertainty in both longer term deterministic and shorter term probabilistic programs. The deterministic studies, funded primarily by the Office of Research of the NRC should reduce the uncertainty by better identifying (1) the causal mechanism of the Charleston earthquake and (2) the potential for the occurrence of large earthquakes throughout the eastern seaboard. The probabilistic studies, primarily that being conducted for NRC by Lawrence Livermore National Laboratory (LLNL), will take into account existing uncertainties. They will have as their aim to determine differences, if any, between the probabilities of seismic ground motion exceeding design levels in the eastern seaboard (i.e., as affected by the USGS clarified position on the Charleston earthquake) and the probabilities of seismic ground motion exceeding design levels elsewhere in the central and eastern United States. Any plants for which the probabilities of exceeding design level ground motions are significantly higher than those calculated for other plants in the central and eastern United States will be identified and evaluated for possible further engineering analysis.

Given the speculative nature of the hypotheses with respect to the recurrence of large Charleston-type earthquakes as a result of present limited scientific knowledge and the generalized low probability associated with such events, the staff does not see a need for any action for specific sites at this time. It is the staff's position, as it has been in the past, that facilities should be designed to withstand the recurrence of an earthquake the size of the 1886 earthquake in the vicinity of Charleston, S.C. At the conclusion of the shorter term probabilistic program and during the longer term deterministic studies, the staff will be assessing the need for a modified position with respect to specific sites.

Figure 2.1 Exclusion area/property line and plant layout of Vogtle

Figure 2.2 Low-population zone and area within 5 miles of Vogtle site

Figure 2.3 Transportation routes and area within 25 miles of Vogtle site

Figure 2.4 Heavily populated areas within 100 miles of Vogtle site

Table 2.1 Resident population vs. distance from reactor building

Year	0-1 mi	0-2 mi	0-3 mi	0-4 mi	0-5 mi	0-10 mi
1980	-	495	773	885	1085	2560
1987	-	517	806	923	1133	2669
1990	-	27	74	121	262	1830
2030	-	66	153	235	499	2551

Table 2.2 Remaining safety review items in Section 2.5.4 *

Review item	SER Sections	Status
(1) Foundation competency of clay marl stratum	2.5.4.1.3, 2.5.4.2, 2.5.4.4.3, 2.5.4.4.6	Open
(2) Verification of FSAR commitments on compaction of Category 1 backfill	2.5.4.3	Open
(3) Submittal and evaluation of settlement records	2.5.4.1.3, 2.5.4.4, 2.5.4.4.3	Open
(4) Foundation design and construction information on radwaste buildings and tunnels	2.5.4.4	Open
(5) Locations and description of observed cavities	2.5.4.4.1	Confirmatory
(6) Bearing capacity stability	2.5.4.4.2	Open
(7) Long-term groundwater and settlement monitoring requirements	2.5.4.4.3, 2.5.4.5	Open
(8) Acceptability of variations in soil dynamic properties	2.5.4.4.6	Open
(9) Method used to establish dynamic passive pressures	2.5.4.4.4	Confirmatory

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 Conformance with General Design Criteria and NRC Regulations

In FSAR Section 3.1 the applicant presents a discussion of conformance of the NRC general design criteria (GDC) for nuclear power plants specified in Appendix A to 10 CFR 50. The staff has reviewed a final design and the design criteria using this information to verify that the plant has been designed to satisfy the requirements of the GDC.

In its review of structures, systems, and components, the staff relied extensively on the application of industry codes and standards that have been used as accepted industry practice. These codes and standards are cited in this report and have been previously reviewed by the staff, found acceptable, and incorporated into the Standard Review Plan (NUREG-0800).

3.2 Later

3.3 Later

3.4 Later

3.5 Later

~~3.6 Later~~ Insert

~~3.7 Later~~

~~3.8 Later~~

~~3.9 Later~~

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

3.6.1 Later

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping

General Design Criterion 4, "Environmental and Missile Design Bases", of 10 CFR Part 50, Appendix A, requires that structures, systems, and components important to safety shall be designed to be compatible with and to accommodate the effects of the environmental conditions as a result of normal operations, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be adequately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power plant.

The staff's review, conducted in accordance with Standard Review Plan (NUREG-0800), Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping", pertains to the methodology used for protecting safety-related structures, systems, and components against the effects of postulated pipe breaks both inside and outside containment. The staff has used the review procedures identified in SRP 3.6.2 to evaluate the effect that breaks in high energy fluid systems would have on adjacent safety-related structures, systems, or components with respect to jet impingement and pipe whip. The staff also reviewed the location, size, and orientation of postulated failures and the methodology used to calculate the resultant pipe whip and jet impingement loads that might affect nearby safety-related structures, systems, or components. The details of the staff's review follow.

Pipe whip need only be considered in those high-energy piping systems having fluid reservoirs with sufficient capacity to develop a jet stream. The criteria for determining high- and moderate-energy lines is found in Branch Technical Position ASB 3-1 of Standard Review Plan 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

Th~~is~~ese criteria ha~~s~~ve been used correctly by the applicant. A list of all high energy systems is included in the FSAR.

For high energy piping within the containment penetration area where breaks are not postulated, SRP Section 3.6.2 sets forth certain criteria for the analysis and subsequent augmented inservice inspection requirements. Breaks need not be postulated in those portions of piping within the containment penetration region that meet the requirements of the ASME Code, Section III, Subarticle NE-1120 and the additional requirements outlined in Branch Technical Position MEB 3-1 of SRP Section 3.6.2. Augmented inservice inspection is required for those portions of piping within the break exclusion region.

For ASME Section III Class 1 high energy fluid system piping not in the containment penetration area, SRP Section 3.6.2 states that breaks are to be postulated at every location where the fatigue cumulative usage factor, as determined by the ASME Code, is greater than 0.1. Additionally, breaks are also to be postulated at those ASME Class 1 piping locations where the primary or secondary stress intensity range (including the zero load set) as calculated by equation (10) and either equation (12) or (13) in Paragraph NB-3653 of ASME Section III exceeds 2.4 S_m for normal and upset conditions including the Operating Basis Earthquake (OBE).

The applicant has provided drawings of break locations showing types of breaks, structural barriers, restraint locations and constrained directions for each restraint for the primary coolant loop and all breaks inside containment.

The following are considered to be open items.

- The applicant has not yet completed the final pipe whip and jet impingement evaluation for all high energy piping systems. The staff's review cannot be completed until this information is available for review. (Q240.30)

- Clarification is required on some aspects of the applicant's pipe break criteria and pipe whip restraints. (Q210.25-Q210.29, Q210.31, Q210.32)

Based on the staff's review of FSAR Section 3.6.2 and resolution of the above open items, the staff's findings are as follows.

In its evaluation, the staff concludes that the pipe rupture postulation and the associated effects are adequately considered in the plant design, and, therefore, are acceptable and meet the requirements of General Design Criterion 4. This conclusion is based on the following.

(1) The proposed pipe rupture locations have been adequately assumed and the design of piping restraints and measures to deal with the subsequent dynamic effects of pipe whip and jet impingement provide adequate protection to the structural integrity of safety-related structures, systems and components.

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

(3) The proposed piping and restraint arrangement and applicable design considerations for high- and moderate-energy fluid systems inside and outside of containment, including the reactor coolant pressure boundary, will provide adequate assurance that the structures, systems, and components important to safety that are in close proximity to the postulated pipe rupture will be protected. The design will be of a nature to mitigate the consequences of pipe ruptures so that the reactor can be safely shut down and maintained in a safe shutdown condition in the event of a postulated rupture of a high or moderate energy piping system inside or outside of containment.

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3.9 MECHANICAL SYSTEMS AND COMPONENTS

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The review performed under SRP Sections 3.9.1 through 3.9.6 of NUREG-0800 pertains to the structural integrity and functional capability of various safety-related mechanical components in the plant. The staff's review is not limited to ASME Code components and supports, but is extended to other components such as control rod drive mechanisms, certain reactor internals, and any safety-related piping designed to industry standards other than the ASME Code.

The staff reviews such issues as load combinations, allowable stresses, methods of analysis, summary of results, and preoperational testing. The staff's review must arrive at the conclusion that there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

The review of this section was performed following SRP Section 3.9.1, "Special Topics for Mechanical Components." All areas of review and review procedures identified in SRP Section 3.9.1 were followed. The staff has reviewed the design transients and methods of analysis used for all seismic Category I components, component supports, core support structures and reactor internals designated as Class 1 and CS under the ASME Code, Section III, and those not covered by the Code. The assumptions and procedures used for the inclusion of transients in the fatigue evaluation of ASME Code Class 1 and CS components have been reviewed. The staff's review also covered the computer programs used in the design and analysis of seismic Category I components and their supports and experimental and inelastic analytical techniques.

The applicant has provided a list of the design transients and the number of cycles for each design transient used for design. Five OBEs of ten cycles each and one Safe Shutdown Earthquake (SSE) of ten cycles have been included. This is in conformance with the requirements of SRP 3.9.1.

Analysis of mechanical components by the use of computer programs was performed by the applicant. A list showing all computer programs used by the applicant for static and dynamic analyses to determine the structural integrity and functional integrity of seismic Category I Code and non-Code items, and the analyses to determine stresses along with a description of the program is included in the FSAR. Design control measures to verify the adequacy of the design of safety-related components is required by 10 CFR Part 50, Appendix B.

Based upon the staff's review of FSAR Section 3.9.1 its findings are as follows.

The staff concludes that the design transients and resulting loads and load combinations with appropriate specified design and service limits for mechanical components and supports are acceptable and meets the relevant requirements of General Design Criteria 1, 2, 14, 15; and 10 CFR Part 50, Appendix B; and 10 CFR Appendix A. This is based on the following.

(1) The applicant has met the relevant requirements of General Design Criteria 14 and 15 by demonstrating that the design transients and resulting loads and load combinations with appropriate specified design and service limits which the applicant has used for designing Code Class 1 and CS components and supports, and reactor internals provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant.

(2) The applicant has met the relevant requirements of General Design Criteria 2 and 10 CFR Part 100, Appendix A by including seismic events in design transients which serve as design bases to withstand the effects of natural phenomena.

(3) The applicant has met the relevant requirements of 10 CFR Part 50, Appendix B, and General Design Criteria 1 by having submitted information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I Code Class 1, 2, 3 and CS structures, and non-Code structures within the present state-of-the-art limits and by having design control measures which are acceptable to assure the quality of the computer programs.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

The staff has reviewed the methodology, testing procedures, and dynamic analyses employed by the applicant to ensure the structural integrity and functionality of piping systems, mechanical equipment, and their supports under vibratory loadings. The principal document used in this review is SRP (NUREG-0800) Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment." All areas of review and review procedures identified in SRP Section 3.9.2 were followed. The staff's review included (1) the piping vibration, thermal expansion, and dynamic effect testing, (2) the seismic system analysis methods, (3) the dynamic responses of structural components within the reactor caused by steady-state and operational flow transient conditions for non-prototype reactors, (4) flow-induced vibration testing of reactor internals to be conducted during the preoperational and start-up test program, and (5) the dynamic analysis methods used to confirm the structural design adequacy and functional capability of the reactor internals and piping attached to the reactor vessel when subjected to loads from a loss-of-coolant accident (LOCA) in combination with an SSE.

3.9.2.1 Piping Preoperational Vibration and Dynamic Effects Testing

Piping vibration, thermal expansion, and dynamic effects testing will be conducted during a preoperational testing program. The purpose of these tests is to assure that the piping vibrations are within acceptable limits and that the piping system can expand thermally in a manner consistent with the design intent. During the Vogtle plant's preoperational and start-up testing program, the applicant will test various piping systems for abnormal, steady-state or transient vibration and for restraint of thermal growth. Systems to be monitored will include 1) ASME Code Class 1, 2 and 3 piping systems, 2) high energy piping systems inside seismic Category I structures, 3) high energy portions of systems whose failure could reduce the functioning of seismic Category I plant features to an unacceptable safety level, and 4) seismic Category I portions of moderate energy piping systems located outside containment. Steady-state vibration, whether flow-induced or caused by nearby vibrating machinery, could cause 10^8 or 10^9 cycles of stress in the pipe during its 40-year life. For this reason, the staff requires that the stresses associated with steady-state vibration be minimized and limited to acceptable levels. The test program will consist of a mixture of instrumented measurements and visual observations by qualified personnel.

The following are considered to be open items.

* A description is needed for the methods to be used to relate measured vibration values to stress levels. (Q210.40)

* Assurance is needed that all essential safety-related instrument lines will be included in the vibration monitoring program during preoperational or start-up testing. (Q210.41)

Based upon the staff's review of FSAR Section 3.9.2.1 and resolution of the open items, the staff concludes that the applicant has met the relevant requirements of General Design Criteria 14 and 15 with respect to the design and testing of the reactor coolant pressure boundary. This provides reasonable assurance that rapidly propagating failure and gross rupture will not occur as a result of vibratory loadings. In addition, the testing assures that design conditions are not exceeded during normal operation including anticipated operational occurrences by having an acceptable vibration, thermal expansion, and dynamic effects test program which will be conducted during start-up and initial operation of specified high and moderate energy piping, including all associated restraints and supports. The tests provide adequate assurance that the piping and piping supports have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design basis flow conditions. In addition, the tests provide assurance that adequate clearances and free movement of snubbers exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations. The planned test will develop loads similar to those experienced during reactor operations.

3.9.2.2 Seismic Subsystem Analysis

The staff's review performed according to Standard Review Plan Section 3.9.2 included Section 3.7.3 of the applicant's FSAR, "Seismic Subsystem Analysis."

Areas reviewed were seismic analyses methods, determination of the number of earthquake cycles, basis for selection of frequencies, the combination of modal responses and spatial components of an earthquake, criteria used for damping, torsional effects of eccentric masses, interaction of other piping with Category I piping, and Category I buried piping systems.

The scope of the review of Vogtle's seismic system and subsystem analysis includes the seismic analysis methods for all seismic Category I piping systems and components. The staff has reviewed the manner in which the dynamic system analysis is performed, the method of selection of significant modes, whether the number of masses or degrees of freedom is adequate, and how consideration is given to maximum relative displacements. The review included design methodologies and procedures used for the evaluation of the interaction of non-seismic Category I piping with seismic Category I piping, and the seismic methods which consider the effect of settlement and movement at support points, penetration, and anchors for seismic Category I buried piping systems. In addition, the staff reviewed seismic analysis procedures for reactor internals. The system and subsystem analyses are performed by the applicant on an elastic basis. Modal response spectrum, multi-degree of freedom and time history methods form the basis for the analyses of all major seismic Category I systems and components. When the response spectrum method is used, modal responses are combined by the square-root-sum-of-the-squares (SRSS) rule.

For the dynamic analysis of seismic Category I piping, each piping system was idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system was determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as change in stiffness due to curved members. Next, the mode shapes and the undamped natural frequencies were obtained. The dynamic response of the system was calculated by using the response spectrum method of analysis. For a piping system which was supported at points with different dynamic excitations, the response analysis was performed using an enveloped response spectrum.

The following are considered to be open items.

* The staff has requested further information on the design of seismic interface anchors. (210.39)

* Clarification is required on the use of damping values and equivalent static factors other than those discussed in the SRP. (Q210.33, Q210.34, Q210.37)

* The staff has requested detailed information regarding the piping analysis procedures used for the main steam and feedwater piping outside containment. (Q210.35, Q210.36)

Based upon the staff's review of FSAR Section 3.7.3, and contingent upon resolution of the open items, the staff concludes that the applicant has met the relevant requirements of General Design Criteria 2 with respect to demonstrating the design adequacy of all Category I piping systems, components, and their supports to withstand earthquakes by meeting the regulatory positions of ~~Regulatory Guides~~ RGs 1.61 and 1.92 and by providing acceptable seismic analysis procedures and criteria. The scope of review of the seismic system analysis included the seismic analysis methods of all Category I piping systems, components, and their supports. It included review of procedures for modeling, and inclusion of torsional effects, seismic analysis of multiply-supported equipment and components with distinct inputs, and determination of composite damping. The review has included design criteria and procedures for evaluation of the interaction of non-Category I piping with Category I piping. The review has also included criteria and seismic analysis procedures for reactor internals.

3.9.2.3 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Flow-induced vibration testing of reactor internals will be conducted during the preoperational and startup test program. The purpose of this test is to demonstrate that flow-induced vibrations similar to those expected during operation will not cause unanticipated flow-induced vibrations of significant magnitude, or structural damage.

The Indian Point Unit 2 reactor has been established as the prototype for the Westinghouse four-loop plant internals verification program. The only significant differences between Vogtle's internals and Indian Point Unit 2's internals are the replacement of the annular thermal shield with neutron shield panels and the substitution of 17x17 fuel assemblies for 15x15 assemblies, and the change to the UHI-style inverted top hat support structure configuration.

The change to the neutron shield panels and 17x17 fuel assemblies has been tested at the Trojan plant. The change to the UHI-style inverted top hat support structure configuration has been tested at the Sequoyah Unit 1 plant. The Four Loop Internals Assurance Program conducted on Indian Point Unit 2 supplemented by the Trojan and Sequoyah Unit 1 data jointly satisfy Regulatory Guide 1.20.

The applicant has committed to test the reactor internals in accordance with the provisions of Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Start-Up Testing," Revision 2, for non-prototype Category I plants. The applicant will conduct a visual inspection prior to hot functional testing and after hot functional testing the applicant has committed to inspecting all major load-bearing surfaces, torsional, lateral, and vertical restraints, locking and bolting devices whose failure could adversely affect the structural integrity of the internals, and all other locations examined on the prototype design. The inside of the vessel will be inspected with all the internals removed both prior to and subsequent to hot functional testing to verify that no loose parts or foreign material are present.

The applicant will subject the internals to an operating time of sufficient duration to assure that a minimum of 10^6 cycles of vibration will be experienced by the critical components. At completion of the flow test, the vessel head will be removed and the internals will be inspected for evidence of wear and loose parts. The inspection will cover all components which were examined on the prototype design. Important welds, bearing surfaces, and alignment and locking devices in the internals will be inspected with the aid of 5x or 10x magnifying glass.

The staff finds the inspection program to be sufficient and the hot functional test to be of adequate length. Based upon the staff's review of FSAR Section 3.9.2.4, findings are as follows.

The staff concludes that the applicant has met the relevant requirements of General Design Criteria 1 and 4 with respect to the reactor internals being designed and tested to quality standards commensurate with the importance of the safety functions being performed and being appropriately protected against dynamic effects by meeting the regulatory positions of RG ~~Regulatory Guide~~ 1.20 for the conduct of preoperational vibration tests and by having a preoperational vibration program planned for the reactor internals which 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 the reactor 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 reactor operation and shutdown.

3.9.2.4 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

The applicant has analyzed its reactor internals and unbroken loops of the reactor coolant pressure boundary, including the supports, for the combined loads due to a simultaneous loss-of-coolant-accident and safe shutdown earthquake. The applicant has described the methodology used in developing the dynamic loads resulting from an asymmetric load from a postulated pipe break at the RPV nozzle safe-end in FSAR Section 3.9.N.2.5.

Based on the staff's review of FSAR Section 3.9.N.2.5 and the load combinations and stress limits as presented in tables contained in FSAR Section 3.9.3, the staff concludes that the applicant has met the relevant requirements of General Design Criteria 2 and 4 with respect to the design of systems and components important to safety to withstand the effects of earthquakes and the appropriate combinations of the effects of normal and postulated accident conditions with the effects of the safe shutdown earthquake (SSE) by performing a dynamic system analysis which provides an acceptable basis for confirming the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic loads of a postulated loss of coolant accident (LOCA) and the SSE. The analysis provides adequate assurance 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 at 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 system analysis. The proposed combination of component and system analyses are, therefore, acceptable. The assurance of structural integrity of the reactor internals under LOCA conditions for the most adverse postulated loading event provides added confidence that the design will withstand a spectrum of lesser pipe breaks and seismic loading events.

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

The staff's review under Standard Review Plan Section 3.9.3 is concerned with the structural integrity and functional capability of pressure-retaining components, their supports, and core support structures which are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, or earlier industrial standards. All areas of review and review procedures identified in SRP Section 3.9.3 were followed. The staff has reviewed loading combinations and their respective stress limits, the design and installation of pressure relief devices, and the design and structural integrity of ASME Code Class 1, 2, and 3 components and component supports. Details of the staff's review are included in the following sections.

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

The first area of review is the methodology used for load combinations and allowable stress limits in FSAR Section 3.9.3.

The following is considered an open item.

- The applicant's methodology used for load combinations does not appear to conform to the acceptance criteria in SRP 3.9.3. Specifically, the applicant does not appear to have included the LOCA loads in evaluation of the faulted condition limits for ASME Class 2 and 3 components and their supports where such loads are appropriate. (Q210.43)

- The applicant is to provide the basis for assuring that ASME Code Class 1, 2, and 3 piping can perform its intended function for service levels C and D loadings. (Q210.42)

Based upon the staff's review of FSAR Sections 3.9.B.3.1 and 3.9.N.3.1 and contingent upon the satisfactory resolution of the open items, the staff's findings will be as follows.

The applicant has met the requirements of 10 CFR 50.55a and General Design Criteria 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 components by ensuring that systems and components important to safety are designed to quality standards commensurate with their importance to safety and that these systems can accommodate the effects of normal operation as well as postulated events such as loss-of-coolant accidents and the dynamic effects resulting from earthquakes. The specified design and service combinations of loading as applied to ASME Code Class 1, 2, and 3 pressure retaining components 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 for other service loading caused by postulated events or system operating transients, the resulting combined stresses imposed on system components 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.

3.9.3.2 Design and Installation of Pressure Relief Devices

The staff has reviewed Section 3.9.3.3 of the applicant's FSAR with respect to the design and installation, and testing criteria applicable to the mounting of pressure relief devices used for the overpressure protection of ASME Class 1, 2, and 3 components. This review, conducted in accordance with SRP Section 3.9.3 (NUREG-0800), includes evaluation of the applicable loading combinations and stress criteria. The design review extends to consideration of the means provided to accommodate the rapidly applied reaction force when a safety valve or relief valve opens, and the transient fluid-induced loads applied to the piping downstream of a safety or relief valve in a closed discharge piping system. The staff also reviewed the applicant's relief and safety valve test results as required in Item II.D.1 of NUREG-0737.

In accordance with Item II.D.1 of NUREG-0737, pressurized water reactor and boiling water reactor licensees and applicants are required to conduct testing to qualify the reactor coolant system relief and safety valves, block valves, and associated piping and supports under expected operating conditions for design-basis transients and accidents.

The Electric Power Research Institute (EPRI) was contracted by the PWR Owners Group to develop and carry out a generic test program and to provide the generic test data to be used by the PWR utilities to satisfy the NUREG-0737, Item II.D.1, requirements.

Testing of valves in the EPRI program was completed by December 21, 1981.

By letter dated April 1, 1982, from D. P. Hoffman, Chairman of the PWR Safety and Relief Valve Test Program Subcommittee, the EPRI/PWR Owners Group transmitted the following reports to NRC:

(1) Valves Selection/Justification Report

(2) Valve Inlet Fluid Condition for Pressurizer Safety and Relief Valves in Westinghouse-Designed Plants (note: two other NSSS vendor reports were also received)

(3) Test Condition Justification Report

(4) Safety and Relief Valve Test Report

(5) Application of RELAP5/MOD 1 for Calculation of Safety and Relief Valve Discharge Piping Hydrodynamic Loads.

Additionally, by letter dated June 1, 1982, from R. C. Youngdahl to H. Denton, reports documenting block valve testing performed by EPRI were transmitted to NRC. These generic reports are currently being reviewed by the staff. On the basis of a preliminary review of the EPRI generic reports, the staff has concluded that they contain data that can be used by the applicant to prepare an Item II.D.1 plant-specific response for the valves and associated piping for Vogtle.

The staff requires that these plant-specific submittals be made before fuel load in accordance with the schedule of NUREG-0737 and the September 29, 1981, clarification letter on this matter. Once the staff has received this information, it will report its findings in a supplement to this SER. This is an open item.

The staff requires additional information on the design of safety and relief valves. (Q210.44 and Q210.46) This is an open item.

The applicant has met the requirements of 10 CFR 50.55a and General Design Criteria 1, 2, and 3 with respect to the criteria used for design and installation of ASME Code Class 1, 2, and 3 overpressure relief devices by ensuring that safety and relief valves and their installations are designed to standards which are commensurate with their safety functions, and that they can accommodate the effects of discharge caused by normal operation as well as postulated events such as loss-of-coolant accidents and the dynamic effects resulting from the safe shutdown earthquake. The relevant requirements of General Design Criteria 14 and 15 are also met with respect to assuring that the reactor coolant pressure boundary design limits for normal operation including anticipated operational occurrences are not exceeded. The criteria used by the applicant in the design and installation of ASME Class 1, 2, and 3 safety and relief valves provide adequate assurances 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 and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function.

3.9.3.3 Component Supports

The staff's review of Section 3.9.3.4 of the applicant's FSAR relates to the methodology used by the applicant in the design of ASME Class 1, 2, and 3 component supports. The review includes assessment of design and structural integrity of the supports. The review addresses three types of supports: plate and shell, linear, and component standard types. Additional information to ensure a complete basis and consistent approach for the design and construction of component supports is required. The specific concern has been transmitted to the applicant in Q210.45. Additionally, a commitment is required from the applicant regarding the snubber pre-service examination and pre-operational testing program discussed in Q210.47. This is an open item.

Based upon the staff's review of FSAR Section 3.9.3.4 and contingent upon the resolution of the open item, the staff findings will be as follows.

The applicant has met the requirements of 10 CFR 50.55a and General Design Criteria 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports by ensuring that component supports important to safety are designed to quality standards commensurate with their importance to safety, and that these supports can accommodate the effects of normal operation as well as postulated events such as loss-of-coolant accidents and the dynamic effects resulting from the safe shutdown earthquake. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination, has met the positions and criteria of Regulatory Guides 1.124 and 1.130 and are in accordance with NUREG-0484, Revision 1. The specified design and service loading combinations used for the design of ASME Code Class 1 2, and 3 component supports in systems classified as seismic Category I provide assurance that in the event of an earthquake or other service loadings caused by postulated events or system operating transients, the resulting combined stresses imposed on system components 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 support components to withstand the most adverse combination of loading events without loss of structural integrity.

Class CS component evaluation findings are covered in SER Section 3.9.5 in connection with reactor internals.

3.9.4 Control Rod Drive Systems

The staff's review under Standard Review Plan Section 3.9.4 covers the design of the control rod drive system up to its interface with the control rods. The rods and drive mechanism shall be capable of reliably controlling reactivity changes either under conditions of anticipated normal plant operational occurrences, or under postulated accident conditions. The staff reviewed the information in FSAR Section 3.9.4 relative to the analyses and tests performed to assure the structural integrity and functionality of this system during normal operation and under accident conditions. The staff also reviewed the life-cycle testing performed to demonstrate the reliability of the control rod drive system over its 40-year life.

A detailed review of the design of the control rod drive system with respect to its capability of controlling reactivity and cooling the reactor core with appropriate margin in conjunction with either the emergency core cooling system or the reactor protection system was not performed because of the system similarity with other Westinghouse plants which were found to be acceptable. The staff is not aware of any significant design changes in the control rod drive system for the Vogtle plant.

Based on the staff's review of the above information, it was concluded that the design of the control rod drive system is acceptable and meets the requirements of General Design Criteria 1, 2, 14, 26, 27, and 29, and 10 CFR 50.55a. This conclusion is based on the following.

(1) The applicant has met the requirements of GDC 1 and 10 CFR 50.55a, with respect to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for control rod drive systems are in conformance with the requirements of appropriate ANSI and ASME codes.

(2) The applicant has met the requirements of GDC 2, 14, and 26 with respect to designing the control rod drive system to withstand effects of earthquakes and anticipated normal operational occurrences with adequate margins to assure its structural integrity and functional capability and with extremely low probability of leakage or gross rupture of reactor coolant pressure boundary. The specified design transients, design and service loadings, combination of loads, and limiting the stresses and deformations under such loading combinations are in conformance with the requirements of appropriate ANSI and ASME codes and acceptable regulatory positions specified in SRP Section 3.9.3.

(3) The applicant has met the requirements of GDC 27 and 29 with respect to designing the control rod drive system to assure its capability of controlling reactivity and cooling the reactor core with appropriate margin, in conjunction with either the emergency core cooling system or the reactor protection system. The operability assurance program is acceptable with respect to meeting system design requirements in observed performance as to wear, functioning times, latching, and overcoming a stuck rod.

3.9.5 Reactor Pressure Vessel Internals

The staff's review under Standard Review Plan 3.9.5 is concerned with the load combinations, allowable stress limits and other criteria used in the design of the Vogtle reactor internals. The staff has limited their review of SRP Section 3.9.N.5 to include the design and analysis of the reactor internals and the deformation limits specified for those components. A detailed review of the configuration and general arrangement of the mechanical and structural internal elements was not performed because of the similarity with other Westinghouse plants which were found acceptable. The staff is not aware of any significant design changes in the reactor internals for the Vogtle plant.

Based on the staff's review of FSAR Section 3.9.5, the staff concludes that the design of reactor internals is acceptable and meets the requirements of General Design Criteria 1, 2, 4, and 10 and 10 CFR 50.55a. This conclusion is based on the following.

(1) The applicant has met the requirements of GDC 1 and 10 CFR 50.55a with respect to designing the reactor internals to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the reactor internals are in conformance with the requirements of Subsection NG of the ASME Code, Section III.

(2) The applicant has met the requirements of GDC 2, 4, and 10 with respect to designing components important to safety to withstand the effects of earthquake and the effects of normal operation, maintenance, testing, and postulated loss-of-coolant accidents with sufficient margin to ensure that capability to perform its safety functions is maintained and the specified acceptance fuel design limits are not exceeded.

The specified design transients, design and service loadings, and combination of loadings as applied to the design of the reactor internals structures and components provided reasonable assurance that in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components would not exceed allowable stresses and deformations under such loading combinations. This 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.

3.9.6 Inservice Testing of Pumps and Valves

The review under Standard Review Plan 3.9.6 is concerned with the inservice testing of certain safety-related pumps and valves typically designated as ASME Class 1, 2, or 3. Other pumps and valves not categorized as Code Class 1, 2, or 3 may be included if they are considered to be safety-related by the staff.

In Sections 3.9.2 and 3.9.3 of the Safety Evaluation Report, the staff discusses the design of safety-related pumps and valves in the Vogtle plant. The load combinations and stress limits used in the design of pumps and valves assure that the component pressure boundary integrity is maintained. In addition, the applicant will periodically test and perform periodic measurements of all its safety-related pumps and valves. These tests and measurements are performed in accordance with the rules of Section XI of the ASME Code. The tests verify that these pumps and valves operate successfully when called upon. The periodic measurements are made of various parameters and compared to baseline measurements in order to detect long-term degradation of the pump or valve performance. The staff reviews the applicant's program for preservice and inservice testing of pumps and valves using the guidance of SRP Section 3.9.6, and gives particular attention to the completeness of the program and to those areas of the test program for which the applicant requests relief from the requirements of Section XI of the ASME Code. The applicant must provide a commitment that the inservice testing of ASME Class 1, 2, and 3 components will be in accordance with the rules of 10 CFR Section 50.55a, paragraph (g).

The applicant has submitted its program for the preservice testing of pumps and valves by letter dated May 1, 1984. The applicant has not yet submitted its program for the inservice testing of pumps and valves. The staff has not yet completed its review. The staff will report the resolution of these issues in a supplement to this Safety Evaluation Report. The preservice and inservice testing of pumps and valves is an open item. (Q210.49)

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure system. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems.

Pressure isolation valves are required to be Category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code, except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also, surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated as less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. Leak testing should also be performed after all disturbances to the valves are complete such as prior to reaching power operation following a refueling outage, and maintenance.

The staff's position on leak rate limiting conditions for operation is that leak rates must be equal to or less than 1 gallon per minute (gpm) for each valve to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves. In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

The applicant has provided a list of Vogtle pressure isolation valves to be included in the leak rate testing program. However, the applicant has not committed to the staff's position on acceptable leak rates. This is an open item and will be addressed in a supplement to this SER. (Q210.48)

3.10 Seismic and Dynamic Qualification of Safety-Related Mechanical and Electrical Equipment

3.10.1 Seismic and Dynamic Qualification

The staff evaluated the adequacy of the applicant's program for qualification of safety-related mechanical and electrical equipment for seismic and dynamic loads. The staff determined the acceptability of the procedures used, standards followed, and the completeness of the program in general, and audited (on site) selected equipment items to develop the basis for the staff to judge the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program.

The Seismic Qualification Review Team (SQRT), consisting of engineers from the Equipment Qualification Branch (EQB) and the Idaho National Engineering Laboratory (INEL, EG&G), has reviewed the methodology and procedures of the equipment seismic and dynamic qualification program contained in the pertinent FSAR Sections 3.9.2, 3.9.3, and 3.10. The SQRT has concluded that the information contained in the FSAR sections mentioned above does meet the intent of the current licensing criteria as described in IEEE Std. 344-1975, RGs 1.92 and 1.100, and SRP Section 3.10. In FSAR Sections 3.9.2, 3.9.3, and 3.10, a surveillance and maintenance program was not addressed which would ensure that all safety-related Class 1E and age-sensitive mechanical components in both harsh and mild environments will be functional throughout the entire life of the plant. In addition, the qualified life of these equipment items was not discussed. The staff also found that in the above FSAR sections there is a lack of applicant commitment in establishing a central filing system that is capable of retrieving qualification documentation, in an auditable manner, before the plant operates.

The staff has informed the applicant that a substantial portion (85% to 90%) of the equipment should be qualified, documented in an auditable manner, and installed on site before the SQRT can perform an onsite audit. The staff also indicated to the applicant the type of information necessary for the SQRT to select the equipment items for a detailed onsite review. The applicant has not indicated when its work will be substantially complete. The staff is currently waiting for information it needs in order to determine a target audit date and to select the equipment to be audited. The staff's review of this area will be completed after the applicant has demonstrated the adequacy of its qualification program through a satisfactory audit. The staff shall report the results of its audit in a future supplement to the SER.

3.10.2 Pump and Valve Operability Assurance

The staff's evaluation of the adequacy of the applicant's pump and valve operability assurance program consists of two parts. First a determination is made of the completeness of the program with regard to the standards and guides used and the procedures used for program implementation. This determination is based on the sufficiency of information in the FSAR and its supporting documents which gives positive evidence of the applicant following a disciplined and thorough program for operability assurance and equipment qualification.

The staff's pump and valve operability review team (PVORT) has reviewed the scope, methodology, and procedures of the pump and valve operability assurance program described in FSAR Sections 3.9 and 3.10.

The information in the FSAR suggests compliance with the general intent of the staff's acceptance criteria as specified in SRP Section 3.10. On the basis of the commitments in the FSAR, the applicant's qualification program for nuclear steam supply system and balance of plant equipment satisfies the requirements and recommendations of IEEE Std. 323-1974 and IEEE Std. 344-1975. However, the staff requires that the FSAR describe further what use is made of applicable references in SRP Section 3.10 and other guidelines to ensure that the equipment is qualified and will operate properly under all imposed design and service conditions, including the loadings imposed by the safe shutdown earthquake (SSE), postulated accidents, and loss-of-coolant accidents. The following areas require clarification or resolution:

(1) The extent to which the complete draft standards ANSI/ASME QNPE-1 (N551.1), QNPE-2 (N551.2), QNPE-3 (N551.3), QNPE-4 (N551.4), and N41.6, and issued standard ANSI/ASME B.16.41 are used needs to be clearly stated in the FSAR.

(2) The applicant has stated compliance with RG 1.148. However, the discussion of testing in the operational condition is limited. Assessment of degraded conditions and how testing was tailored to meet the requirements of SRP Section 3.10, paragraph II.1a(2) should be addressed in the FSAR. Also, a commitment to RG 1.148 for replacement components should be clearly stated in the FSAR.

(3) The applicant should amend the existing tables of pumps and valves in the FSAR to include the methods and standards used for qualification. As an alternate, a separate table may be provided which includes the above information correlated to Tables 3.9.B.3-8, 3.9.B.3-9, and 3.11.N.1-1.

(4) For those components for which qualification and/or operability assurance was performed by analysis alone, some question remains as to the confidence level assured by this methodology. The need for additional component testing is being considered, and such need cannot be established without an inspection at the plant site.

(5) In many cases the motor of an assembly was independently qualified and the pump separately qualified for operation, using the inputs at the mounting. Further justification is needed in the FSAR to describe how an acceptable qualification of the assembly was determined, considering simultaneous dynamic interactions between the pump, motor, and pedestal/mounting structure.

(6) Aging and the sequence of environmental conditions on the qualification process are only briefly addressed in the FSAR. The applicant should clarify how these findings will be reflected in the maintenance and surveillance program. The FSAR should include the criteria for the maintenance program as it relates to equipment qualification test and analysis results.

(7) FSAR Section 3.10.N.2.2.1, indicates that static shaft analysis of the rotor is performed with SSE accelerations and compared with allowable rotor clearances. FSAR Section 3.10.N.2.2.1 further states, "if rubbing or impact is predicted, it is required that it be shown by prototype or

existing documented data the pump will not be damaged. . . ." The applicant should identify all pumps affected and the test and/or documentation used to ensure operability.

(8) The FSAR should be amended to include the generic testing criteria for qualifying check valves for service conditions. The applicant should address considerations of load conditions (end loads, vibrations, seismic and reverse flow) and environmental conditions (thermal and radiation aging of sensitive materials) and their impact on valve function and valve leakage.

(9) The FSAR should be amended to include information regarding administrative control of component qualification. The information should describe component qualification, the equipment qualification file, the handling of documentation, the applicant's internal acceptance review procedures, etc.

The applicant should submit FSAR amendments to resolve the identified FSAR deficiencies. In addition, the PVORT will follow the applicant's effort closely, and will confirm its implementation during the onsite audit. During the plant site audit the staff will review in detail: the applicant's implementation of the qualification program to confirm that all applicable loads and combinations of loads have been defined and utilized; operability has been verified through appropriate tests and analyses; assemblies rather than individual components have been verified operable; and that, for all safety-related equipment, operability can be ensured throughout the plant life. At least 85% of the safety-related equipment must be qualified, documented in an auditable manner, and installed on site before the PVORT can perform an onsite audit. When the applicant indicates that its work is substantially complete, the PVORT will schedule an onsite audit. The staff will report the results of the audit and the followup and resolution of the concerns described above in a future supplement to the SER.

In addition, the staff has not received information in the following areas and these areas remain open:

(1) dependability of containment isolation (purge valves)
(2) long-term operability of deep draft pumps (IE Bulletin 79-15)

3.11 Environmental Design of Mechanical and Electrical Equipment

Later

4 REACTOR

4.1 Summary Description

The Vogtle, Units 1 and 2 (Vogtle), nuclear steam supply system (NSSS) is supplied by Westinghouse Electric Corporation and is designed to operate at a core thermal power of 3411 MWt. Sufficient margin exists to ensure that fuel damage will not occur during steady-state operation or anticipated operational occurrences.

The NSSS, a four-loop design, has a primary coolant flow rate of 138.7×10^6 lb/hr. The reactor coolant and moderator is light water at a nominal system pressure of 2250 psia. The reactor core consists of 193 fuel assemblies of similar mechanical design, but different fuel pellet enrichments. Each assembly is a 17 x 17 array containing 264 fuel rods. Each fuel assembly has 24 positions for guide thimbles for the rod cluster control assemblies which consist of stainless-steel-clad hafnium or silver-indium-cadmium neutron absorber rods. There are 24 absorber rods per cluster. The center position in each assembly is used for incore instrumentation.

The design of the Vogtle reactors is similar to that of the Millstone Unit 3 and Standardized Nuclear Unit Power Plant System (SNUPPS) reactors.

The review addressed in this section was performed in accordance with the applicable portions of the Standard Review Plan (SRP) (NUREG-0800).

4.2 Fuel Design

The fuel assembly described in the FSAR for Vogtle is a 17 x 17 array of fuel rods having a diameter of 0.374 in. This design will be referred to as the standard fuel assembly (SFA) in the following paragraphs.

Section 4.2 of the FSAR presents the design bases for the SFA. For the Westinghouse (W) analysis, plant design conditions are divided into four categories of operation that are consistent with traditional industry classification (ANSI Stds. N18.2-1973 and N-212-1974): condition 1 is normal operation, condition 2 is incidents of moderate frequency, condition 3 is infrequent incidents, and condition 4 is limiting faults. Fuel damage is then related to these conditions of operation, which are coupled to the fuel design bases and design limits. The subsections of the design bases section address topics such as (a) cladding, (b) fuel material, (c) fuel rod performance, (d) spacer grids, (e) fuel assembly structural design, (f) in-core control components, and (g) surveillance program. Thus, as part of the discussion of the cladding design bases, cladding mechanical properties, stress-strain limits, vibration and fatigue, and cladding chemical properties are also pre-sented. A similar approach is taken for the other major subtopics.

The review and safety evaluation will be performed according to ~~follow~~ SRP Section 4.2 (NUREG-0800, ~~Rev. 2~~ July 1981). The objectives of this fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged" is defined as meaning that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements GDC 10 of 10 CFR Part 50, Appendix A ("General Design Criteria for Nuclear Power Plants"), and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR 100 ("Reactor Site Criteria") for postulated accidents. "Coolability," which is sometimes termed "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the GDC 27 and 35. Specific coolability requirements for the loss-of-coolant accidents are given in 10 CFR 50.46 ("Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors").

To meet the above-stated objectives of the fuel system review, the following specific areas are critically examined: (a) design bases, (b) description and design drawings, (c) design evaluation, and (d) testing, inspection, and surveillance plans. In assessing the adequacy of the design, several items involving operating experience, prototype testing, and analytical predictions are weighed in terms of specific acceptance criteria for fuel system damage, fuel rod failure, and fuel coolability. Recently, Westinghouse developed the optimized fuel assembly (OFA), which is described in WCAP-9500. WCAP-9500 is mentioned in the last paragraph (p. 4.2-2) of Section 4.2 of the FSAR for Vogtle but is not included in the reference list for that section. WCAP-9500 was approved by the NRC (Tedesco, May 22, 1981). The OFA design also consists of a 17 x 17 array of fuel rods but the rods have a diameter of 0.360 in., which is somewhat smaller than the rod diameter in the SFA. Because the format of WCAP-9500 followed RG 1.70, some of the fuel design bases and design limits for the OFA were not presented in WCAP-9500 in a form that permitted NRC to cross-check these with the acceptable criteria provided in SRP Section 4.2. Therefore, several questions were issued to clarify the design bases and limits. Responses to those questions are contained in letters from Westinghouse (Anderson, January 12, 1981, and April 21, 1981). These responses are applicable to the standard fuel assembly to be used in Vogtle as well (Petrick, September 9, 1981). References to these questions and answers will be made at several places in the review that follows.

4.2.1 Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and suggest limiting values for important parameters such that damage will be limited to acceptable levels. For convenience, acceptance criteria for these design limits are grouped into three categories in the Standard Review Plan: (a) fuel system damage criteria, which are most applicable to normal operation (W plant condition 1), including anticipated operational occurrences (W plant condition 2), (b) fuel rod failure criteria, which apply to normal operation (W plant condition 1), anticipated operational occurrences (W plant condition 2), and postulated accidents (W plant conditions 3 and 4), and (c) fuel coolability criteria, which apply to postulated accidents (W plant conditions 3 and 4).

4.2.1.1 Fuel System Damage Criteria

The following paragraphs discuss the evaluation of the design bases and corresponding design limits for the damage mechanisms listed in the SRP. These design limits along with certain criteria that define failure (see Section 4.2.1.2 of this report) constitute the SAFDLs required by GDC 10. The design limits in this section should not be exceeded during normal operation including anticipated operational occurrences.

(1) Cladding Design Stress

In Section 4.2.1.1 of the FSAR, it is indicated that the cladding stresses under conditions 1 and 2 are less than the Zircaloy yield stress, with due consideration of temperature and irradiation effects. The design basis for fuel rod cladding stress as given in the response (Anderson, January 12, 1981, and April 21, 1981) to Q231.2* is that the fuel system will not be damaged because of excessive fuel rod cladding stresses. The design limit for fuel rod cladding stress under condition 1 and 2 modes of operation is that the volume-averaged effective stress calculated with the von Mises equation, considering interference due to uniform cylindrical pellet-to-cladding contact (caused by pellet thermal expansion and swelling, uniform cladding creep, and fuel rod/coolant system pressure differences), is less than the Zircaloy 0.2% offset yield stress as affected by temperature and irradiation. This is a traditional limit consistent with previous Westinghouse design practice, but with credit being taken by Westinghouse for irradiation-induced strengthening. The staff has approved (Thomas, July 21, 1983) WCAP-9179, Revision 1, which includes approval for taking such credit.

*All questions and responses referred to in this manner were part of the review of WCAP-9500, and the first application of the SFA, on the Shearon Harris docket.

(2) Cladding Design Strain

With regard to cladding strain, a design limit for fuel rod cladding plastic tensile creep (due to uniform cladding creep and uniform cylindrical fuel pellet swelling and thermal expansion) of less than 1% from the unirradiated condition is given in the response (Anderson, January 12, 1981, and April 21, 1981) to Q231.2 and in Section 4.2.1.1 of the FSAR. Furthermore, the total tensile strain transient limit (due to uniform cylindrical pellet thermal expansion during the transient) is stated to be less than 1% from the pre-transient value. This value of less than 1% is consistent with past practice (no numerical value for normal operation cladding strain is provided as an acceptance criterion in the Standard Review Plan), and thus there is reasonable assurance that 1% total plastic creep strain is an acceptable design limit for normal operation, including condition 1 power changes (load following). For transient-induced deformation, the Standard Review Plan indicates that 1% uniform cladding strain is an acceptable damage limit that should preclude some types of pellet/cladding interaction (PCI) failures. Such a limit, however, while consistent with past practice, should not be construed to be a broadly applicable PCI damage limit because there is evidence that PCI failures can occur at less than 1% uniform cladding strain. Westinghouse has indicated in its response (Anderson, January 12, 1981, and April 21, 1981) to Q231.24 that 1% plastic strain from the pretransient value is not meant to serve as a broadly applicable PCI criterion. Nevertheless, the 1% cladding transient plastic strain criterion appears to be an acceptable design limit for the type of application indicated in SRP Section 4.2. For fuel assembly structural design, Westinghouse set design limits on stresses and deformations due to various non-operational, operational, and accident loads. As indicated in Section 4.2.1.5 of the FSAR, the stress categories and strength theory presented in Section III of the ASME Code are used as a general guide. This is consistent with acceptance criterion II.A.1(a) of SRP Section 4.2 and is acceptable.

(3) Strain Fatigue

According to Section 4.2.1.1 of the FSAR, the cumulative strain fatigue cycles are less than the design strain fatigue life, which is consistent with proven practice (WCAP-8183). The strain fatigue criteria given in the response

(Anderson, January 12, 1981, and April 21, 1981) to Q231.2 and in Section 4.2.3.3.1 of the FSAR are the same as those described in SRP Section 4.2, viz., a safety factor of 2 on stress amplitude or of 20 on the number of cycles and are, therefore, acceptable.

(4) Fretting Wear

Although the Standard Review Plan does not provide numerical bounding-value acceptance criteria for fretting wear, it does stipulate that the allowable fretting wear should be stated in the safety analysis report and that the stress and fatigue limits should presume the existence of this wear.

In Sections 4.2.1.1 and 4.4.4.7 of the FSAR, it is indicated that potential fretting wear owing to vibration is prevented, assuring that the stress-strain limits are not exceeded during the design life. From the response (Anderson, January 12, 1981, and April 21, 1981) to Q231.5 it can also be seen that the Westinghouse design basis for fretting wear is that fuel rods shall not fail during condition 1 and 2 events. Furthermore, Westinghouse does not use an explicit fretting wear limit in its stress and fatigue analysis for fuel rods. However, Westinghouse does use a value of wall thickness as a general guide in evaluating cladding imperfections, including fretting wear. Cladding imperfections including fretting wear are thus considered in the stress and fatigue analysis, albeit in a qualitative manner. In view of the apparently small effects of these defects and large stress and fatigue margins (see Section 4.2.3.1(4) of this report), this design method is acceptable.

The design basis for guide thimble tubes [see response (Anderson, January 12, 1981, and April 21, 1981) to Q231.41] is that the thinning of the guide thimble tube walls should not result in the failure of the fuel assembly structural integrity or functionability of the guide thimble tubes. The staff finds this to be an acceptable design basis.

With regard to a design limit for guide thimble tube wear, Westinghouse has determined from stress analyses that the most limiting load on the fuel assembly structure is that which might occur during a fuel-handling accident. For the analysis of this accident, Westinghouse uses a design criterion of 6 g, as noted in Section 4.2.1.5 of the FSAR. This design limit is, therefore, used for degraded guide thimble tubes and has been previously accepted for Westinghouse fuels.

(5) Oxidation and Crud Buildup

The SFA design basis for cladding oxidation and crud buildup is that the increase in cladding temperature due to cladding oxidation and crud buildup is not excessive (see Section 4.2.1.2(3), below).

SRP Section 4.2 identifies cladding oxidation, hydriding, and crud buildup as potential fuel system damage mechanisms. Hydriding is discussed in Section 4.2.1.2(1), below. Because of the increased thermal resistance of these layers, there is an increased potential for elevated temperature within the fuel as well as the cladding. Because the effect of oxidation and crud layers on fuel and cladding temperature is a function of several different parameters (such as heat flux and thermal-hydraulic boundary conditions), a design limit on oxide or crud layer thickness does not preclude fuel damage as a result of these layers. Rather, it is necessary that these layers be appropriately considered in other temperature-related fuel system damage and failure analyses. This approach (e.g., see Sections 4.4.2.9.1, 4.4.2.11, and 4.4.2.11.5 of the FSAR) taken by Westinghouse in the design of the standard fuel assembly is found by the NRC staff to be acceptable.

(6) Rod Bowing

Fuel rod bowing is a phenomenon that alters the pitch dimensions between adjacent fuel rods. Bowing affects local nuclear power peaking and the local heat transfer to the coolant. Rather than placing design limits on the amount of bowing that is permitted, the effects of bowing are included in the safety analysis (see Sections 4.2.3.1 and 4.2.3.3.5 of the FSAR). This is consistent with the Standard Review Plan and is acceptable. The methods used for predicting the degree of rod bowing are evaluated in Section 4.2.3.1(6) of this report, and the impact of the resulting bow magnitude is evaluated in Sections 4.3 and 4.4 of this report.

(7) Axial Growth

In the SFA design, the core components requiring axial-dimensional analyses are the control rods, neutron source rods, burnable poison rods, fuel rods, and fuel assemblies (thimble plugging rods are omitted because they are short and not axial-growth limited). The axial growth of the first three of these components is primarily dependent upon the behavior of poison, source, or spacer pellets and their Type 304 stainless-steel cladding. The growth of the latter two is mainly governed by the behavior of fuel pellets, Zircaloy-4 cladding, and Zircaloy-4 guide thimble tubes.

The Westinghouse design bases for core component rods are that (a) dimensional stability and cladding integrity are maintained during condition 1 and 2 events and (b) these components do not interfere with shutdown during condition 3 and 4 events.

Westinghouse does not have specific design limits on the axial growth of its control, source, and burnable poison rods. However, allowances are made to accommodate (a) pellet swelling due to gas production and (b) relative thermal expansion between the stainless-steel cladding and the encapsulated material. Westinghouse does not account for irradiation growth of the stainless steel cladding and has cited experiments (Foster and Strain, October 1974) as justification for the insignificance of irradiation growth of stainless steel at PWR operating conditions.

For the Zircaloy cladding and fuel assembly components, the axial-dimensional behavior is governed by creep (due to mechanical or hydraulic loading) and irradiation growth. The critical tolerances that require controlling are (a) the spacing between the fuel rods and the fuel assembly (shoulder gap) and (b) the spacing between the fuel assemblies and the core internals. Failure to adequately design for the former may result in fuel rod bowing, and for the latter may result in collapse of the holddown springs. With regard to inadequately designed shoulder gaps, problems have been reported (Schenk, October 1973; Kuffer and Lutz, 1973; FSAR of R. E. Ginna Unit 1, 1972; Clark, July 24, 1983; and Nerses, April 28, 1983) in foreign (Obrigheim and Beznau) and domestic (Arkansas-2, Ginna, and St. Lucie-2) plants that have necessitated predischarge modifications to fuel assemblies.

With regard to a design basis for shoulder gap spacing, it is indicated in Sections 4.4.2 and 4.2.3.5.1 of the FSAR and it is stated by Westinghouse in the responses (Anderson, January 12, 1981, and April 21, 1981) to Q231.2, Q231.8, Q231.25, and Q231.40 that interference is precluded by having clearance between the fuel rod end and the top and bottom nozzles. The design clearance accommodates the differences in growth, fabrication tolerances, and the differences in thermal expansion between the fuel cladding and the thimble tubes. Westinghouse does not have specific limits on growth, but does provide a gap spacing that is equal to or greater than a percentage of the fuel rod length.

With regard to fuel assembly growth, Westinghouse has a design basis that there shall be no axial interference between the fuel assembly and upper and lower core plates caused by temperature or irradiation. As a design limit, Westinghouse provides a minimum gap, which is a fraction of the fuel assembly length, between the fuel assembly and the reactor internals.

The above design bases and limits dealing with axial growth are acceptable.

(8) <u>Fuel Rod and Nonfuel Rod Pressures</u>

For condition 1 and 2 events, the mechanical design basis for core component rods described in the FSAR is that dimensional stability and cladding integrity are maintained. A necessary corollary of this design basis is that the driving force, rod internal pressure, is never so great as to result in loss of dimensional stability and cladding integrity.

SRP Section 4.2 identifies rod internal pressure as a potential fuel system damage mechanism. In this sense, damage is defined as an increased potential for elevated temperatures within the rod as well as an increased potential for cladding failure. Although the Standard Review Plan mentions only fuel and burnable poison rods, the mechanism also applies to control rods, neutron source rods, and other core component rods. Because rod internal pressure is a driving force for, rather than a direct mechanism of, fuel system damage, it is not necessary that a damage limit be specified. It is only necessary that the phenomenon be appropriately considered in other fuel system damage and fuel failure analyses. In other words, rod internal pressure must be considered in calculating the temperature of the rod internals, cladding deformation, and cladding bursting.

In order to simplify the analysis of fuel system damage due to excessive rod internal pressure, the Standard Review Plan states that rod internal gas pressure should remain below the nominal system pressure during normal operation unless otherwise justified. Westinghouse has elected to justify limits other than ~~that~~ those provided in the Standard Review Plan.

For the fuel rods, revised internal rod pressure criteria as described in WCAP-8963, an approved (Stolz, May 19, 1978) topical report, were used in the FSAR. Briefly stated, these criteria (Section 4.2.1.3 of the FSAR) allow the fuel rod internal pressure to exceed the system pressure under certain conditions: (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 (DNB) propagation does not occur during normal operation and any accident event. These criteria have been previously approved and are therefore acceptable.

It is stated in Section 4.2.3.1 of the FSAR that the burnup-dependent fission gas release model in WCAP-8720, which has been approved by the staff was used in the FSAR. Addendum No. 1 to WCAP-8720 has also been approved by the NRC (Bernard, July 20, 1982).

For the nonfuel rods, the rod internal pressure is limited such that the mechanical design limits, discussed in Section 4.2.1.5 of the FSAR, are not exceeded for condition 1 and 2 events. This implies a stress limit of 2/3 of the material yield stress and a strain limit of 1%. These limits are unchanged from previously approved Westinghouse fuel designs and remain acceptable for this FSAR.

(9) Assembly Liftoff

The Standard Review Plan calls for the fuel assembly holddown capability (gravity and springs) to exceed worst-case hydraulic loads for normal operation, which includes anticipated operational occurrences. The SFA design basis provides for positive holddown for condition 1, but allows momentary liftoff during one condition 2 event (see Section 4.4.2.6.2 of the FSAR). This design basis is acceptable provided that it can be shown that the affected fuel assemblies will reseat properly without damage and without other adverse effects during the event. The ability of the affected fuel assemblies to satisfy this provision will be discussed in Section 4.2.3.1, below.

(10) Control Material Leaching

The Standard Review Plan and General Design Criteria require that control rod reactivity be maintained. Control rod reactivity can sometimes be lost by leaching of certain poison materials if the control rod cladding has been breached. The mechanical design basis for the control rods is stated in Section 4.2.1.6 of the FSAR to be consistent with the loading conditions of Section III of the ASME Code. Thus, the design basis for the SFA control rods is to maintain cladding integrity; because cladding integrity would ensure that reactivity is maintained, this design basis might appear to be acceptable. However, under some circumstances, unexpected breaches might go undetected, so the staff does not normally accept control rod cladding integrity as a sufficient design basis. A discussion will be presented under Section 4.2.3.1, below, that shows that adequate surveillance will be provided by the applicant to ensure maintenance of reactivity.

4.2.1.2 Fuel Rod Failure Criteria

The evaluation of fuel rod failure thresholds for the failure mechanisms listed in the SRP is presented in the following paragraphs. When these failure thresholds are applied to normal or transient operation, they are used as limits (the specified acceptable fuel design limits of GDC 10), since fuel failures under those conditions should not occur (according to the traditional conservative interpretation of GDC 10). When these thresholds are applied to accident analyses, the number of fuel failures must be determined for input to the radiological dose calculations required by 10 CFR 100. The basis or reason for establishing these failure thresholds is thus predetermined, and only the threshold values are reviewed below.

(1) Internal Hydriding

Hydriding as a cladding failure mechanism is precluded by controlling the level of moisture and other hydrogenous impurities during fabrication. As described in the revised response (Anderson, January 12, 1981, and April 21, 1981) to Q231.6, the moisture levels in the uranium dioxide fuel are limited by Westinghouse to less than or equal to 20 ppm. This specification is compatible with the ASTM specification for sintered uranium dioxide pellets, which allows two micrograms of hydrogen per gram of uranium (2 ppm). These are the same limits provided in the Standard Review Plan and are therefore acceptable.

(2) Cladding Collapse

If axial gaps in the fuel pellet column were to occur due to densification, the cladding would have the potential of collapsing into a gap (flattening). Because of the large local strains that would result from collapse, the cladding is assumed to fail. As indicated in Section 4.2.1.3 of the FSAR and in responses (Anderson, January 12, 1981, and April 21, 1981) to Q231.2, Q231.9, and Q231.34 it is a Westinghouse design basis that cladding collapse is precluded during the fuel rod design lifetime. This design basis is the same as that in the Standard Review Plan and is therefore acceptable.

(3) Overheating of Cladding

The design basis as given in Section 4.4.1.1 of the FSAR for the prevention of fuel failures due to overheating is that there will be at least 95% probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation or any transient conditions arising from faults of moderate frequency (condition 1 and 2 events) at a 95% confidence level. This design basis is consistent with the thermal margin criterion of SRP Section 4.2 and is, thus, acceptable. The specific DNBR limits and methods of analysis are reviewed in Section 4.4 of this report.

(4) Overheating of Fuel Pellets

As a second method of avoiding cladding failure due to overheating, Westinghouse avoids centerline fuel pellet melting as a design basis. This design basis is the same as given in the Standard Review Plan and is thus acceptable.

The design limit (Sections 4.2.1.2 and 4.4.1.2.1 of the FSAR) corresponding to the design basis given above is that, during modes of operation associated with condition 1 and condition 2 events, there is at least a 95% probability that the peak kW/ft fuel rod will not exceed the UO_2 melting temperature. This design limit is an acceptable representation of the design basis given above.

(5) Pellet/Cladding Interaction

As indicated in SRP Section 4.2, there are no generally applicable criteria for PCI failure. However, two acceptance criteria of limited application are presented in the SRP for PCI: (a) less than 1% transient-induced cladding strain and (b) no centerline fuel melting. The response (Anderson, January 12, 1981, and April 21, 1981) to Q231.2 indicates that the 1% cladding plastic strain limit is met for the SFA design, and as stated in Section 4.2.1.2 of the FSAR, the SFA design ensures that UO_2 centerline melting will not occur through selection of a calculated fuel centerline temperature of 4700°F as an overpower limit. Thus the SFA design basis and limits agree with the existing licensing criteria for PCI.

(6) Cladding Rupture

In the LOCA analysis for SFA-designed plants, an empirical model is used to predict the occurrence of cladding rupture. The failure temperature is expressed as a function of differential pressure across the cladding wall. There are no specific design limits associated with cladding rupture, and the rupture model is a portion of the ECCS evaluation model, which is documented in Revision 1 of WCAP-9220-P-A and WCAP-9221-A.

4.2.1.3 Fuel Coolability Criteria

For major accidents in which severe fuel damage might occur, core coolability must be maintained as required by several GDCs (e.g., GDC 27 and 35). The following paragraphs discuss the evaluation of limits that will assure that coolability is maintained for the severe damage mechanisms listed in SRP Section 4.2.

(1) Fragmentation of Embrittled Cladding

For LOCA analysis (Section 15.6.5.1 in the FSAR), Westinghouse uses the acceptance criteria of 2200°F on peak cladding temperature and 17% on maximum cladding oxidation as prescribed by 10 CFR 50.46. For events other than the LOCA, the staff does not have separately established temperature or oxidation criteria. Yet it is clear that for short-term events such as locked rotor, the 2200°F peak cladding temperature and 17% oxidation LOCA criteria are not really meaningful, because the temperature history for such an event is much shorter than that of a LOCA. For events such as locked rotor, therefore, Westinghouse uses a unique peak-cladding-temperature (PCT) criterion of 2700°F (e.g., see Section 15.3.3.2 and Section 15.4.8.1.2 of the FSAR).

The Westinghouse 2700°F PCT limit was selected taking into consideration the short time (a few seconds) that the fuel is calculated to be in DNB for a locked-rotor type event and the fact that the PCT and total metal-water reaction at the fuel hot spot would not be expected to impact fuel coolable geometry. Although this limit has been used by Westinghouse for several years, the basis for the limit has only recently been reviewed. However, an assessment by the staff of the available experimental information indicates that fuel rod cladding will, indeed, retain its rodlike geometry after exposure to short-term (a few seconds) peak cladding temperature of 2700°F. That conclusion is based on four Japanese reports (Shiozawa, March 1979; Hoshi, May 1980; JAERI-M-9011, September 1980; and Fukishiro, October 1980) that describe experimental results for reactor test programs reported since 1979. The staff, therefore, concludes that there is reasonable assurance that the 2700°F PCT limit for short-term events such as locked rotor is an acceptable coolability limit for the Westinghouse SFA design.

It should be noted that NRC acceptance of the 2700°F PCT limit for fuel rod coolability is currently restricted to undercooling events such as locked rotor. For overpower events such as control rod ejection, which involve a pellet-to-cladding mechanical interaction, the staff has not determined the applicability of a PCT limit and currently uses a fuel rod enthalpy criterion of 280 cal/gm for coolability of a rod-ejection accident.

(2) Violent Expulsion of Fuel Material

The design bases that there should be little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves are given in Section 15.4.8.1.2 of the FSAR and are equivalent to those in the Standard Review Plan.

The design limits given in the FSAR are:

(a) Average fuel pellet enthalpy at the hot spot will be below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.

(b) Average cladding temperature at the hot spot will be below the temperature at which cladding embrittlement may be expected (2700°F).

(c) Peak reactor coolant pressure will be less than that which could cause pressures to exceed the faulted condition stress limits.

(d) Fuel melting will be limited to less than 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits in (a), above.

These limits are more conservative than the single 280-cal/gm limit given in RG 1.77. They have been previously approved in the review of WCAP-7588, and are therefore acceptable.

(3) Cladding Ballooning and Flow Blockage

In the LOCA analyses for SFA-designed plants, empirical models are used to predict the degree of cladding circumferential strain and assembly flow blockage at the time of hot-rod and hot-assembly burst. These models are each expressed as functions of differential pressure across the cladding wall. There are no specific design limits associated with ballooning and blockage, and the ballooning and blockage models are portions of the ECCS evaluation model, which is documented in Revision 1 of WCAP-9220-P-A and WCAP-9221-A.

(4) Structural Damage from External Forces

Section 4.2.3.5 of the FSAR states that the fuel assembly will maintain a geometry that is capable of being cooled under the worst-case accident condition 4 event and that no interference between control rods and thimble tubes will occur during a safe shutdown earthquake. This is equivalent to the design basis as presented in the Standard Review Plan and is therefore acceptable.

4.2.2 Description and Design Drawings

The description of fuel system components, including fuel rods, bottom and top nozzles, guide and instrument thimbles, grid assemblies, rod cluster control assemblies, burnable poison assemblies, neutron source assemblies, and thimble plug assemblies, is contained in Section 4.2.2 of the FSAR. In addition, Tables 4.1-1 and 4.3-1 of the FSAR provide numerical values for various core component parameters. While each parameter listed in SRP Subsection 4.2.2 is not provided in the FSAR, enough information is provided in sufficient detail to provide a reasonably accurate representation of the SFA design, and this information is thus acceptable. However, the number of fuel rods in Table 4.3-1 of the FSAR should be 50,952 and not 50,592 (see Table 4.1-1 in the FSAR). The applicant should change this value in an amendment to the FSAR.

4.2.3 Design Evaluation

Design bases and limits were presented and discussed in Section 4.2.1, above. In this section, the staff reviews Westinghouse methods of demonstrating that the SFA fuel design conforms to the design criteria that have been established.

This section will, therefore, correspond point by point to Section 4.2.1, above. The methods of demonstrating that the design criteria have been satisfied include operating experience, prototype testing, and analytical predictions.

4.2.3.1 Fuel System Damage Evaluation

The following paragraphs discuss the evaluation of the ability of the SFA fuel to satisfy the fuel system damage criteria described in Section 4.2.1.1, above. Those criteria apply only to normal operations and anticipated transients.

(1) Cladding Design Stress

As indicated in the response (Anderson, January 12, 1981, and April 21, 1981) to Q231.2, Westinghouse used its performance-analysis and design (PAD) code, WCAP-8720, to analyze cladding stress. That code has been reviewed and found acceptable (Bernard, July 20, 1982). Typical calculated design values for cladding effective stress provided in the response (Anderson, January 12, 1981, and April 21, 1981) to Q231.2 are stated to be considerably below the 0.2% offset yield stress design limit.

(2) Cladding Design Strain

The NRC-approved Westinghouse fuel performance code (PAD) was used in the strain analysis, as indicated in the response (Anderson, January 12, 1981, and April 21, 1981) to Q231.2. Typical design values of steady-state and transient creep strain, as calculated by that code, are found to be below the 1% strain criterion. Hence, the staff concludes that the SFA cladding strain design limits have been met.

(3) Strain Fatigue

As indicated in the response (Anderson, January 12, 1981, and April 21, 1981) to Q231.2, Westinghouse used its approved PAD code for the strain range and strain fatigue life usage analysis. Experimental data obtained from Westinghouse testing programs (see Section 4.2.3.3.1 of the FSAR) were used by Westinghouse to derive the Zircaloy fatigue design curve, according to the response (Anderson, January 12, 1981, and April 21, 1981) to Q231.4 For a given strain range, the number of fatigue cycles is less than that required for failure, considering a minimum safety factor of 2 on stress amplitude or a minimum safety factor of 20 on the number of cycles, (the fatigue usage factor is less than 1.0). The computations were performed with an approved code. It is concluded that the SFA fatigue design basis has been met.

(4) Fretting Wear

With regard to the Westinghouse fretting analysis of the fuel cladding, the conclusions of the review are the following:

(a) Cladding fretting and fuel vibration have been experimentally investigated, as shown in WCAP-8278 (and nonproprietary version WCAP-8279) and noted in Section 4.2.3.1 of the FSAR. WCAP-8278 (and WCAP-8279) has been approved by the staff (Tedesco, April 2, 1981).

(b) The out-of-pile flow tests and analyses (WCAP-9401) to determine the magnitude of fretting wear that is anticipated for the OFA design have been previously reviewed and found acceptable (Tedesco, May 7, 1981). These analyses are also acceptably conservative for SFA applications. as compared to the criterion discussed in Section 4.2.1.1 (4) of this report

(c) LWR operating experience demonstrates that the number of fretting-induced fuel failures is insignificant.

(d) There should be only a small dependence of cladding stresses on fretting wear because this type of wear is local at grid-contact locations and relatively shallow in depth.

(e) The built-in conservatisms (that is, safety factors of 2 on the stress amplitudes and 20 on the number of cycles) in the strain fatigue analysis as well as the calculated margin to fatigue life limit adequately offset the effect of fretting wear degradation.

Therefore, it is concluded that the SFA fuel rods will perform adequately with respect to fretting wear.

Fretting wear has also been observed on the inner surfaces of guide thimble tubes where the fully withdrawn control rods reside. Significant wear is limited to the relatively soft Zircaloy-4 guide thimble tubes because the Inconel or stainless steel control rod claddings are relatively wear resistant. The extent of the wear is both time-dependent and plant-dependent and has, in some non-Westinghouse cases, extended completely through the guide thimble tube wall.

Westinghouse has predicted that an SFA can operate under a rod cluster control assembly (RCCA) for a period of time that exceeds the amount of rodded time (the amount of time that the control rod is inserted in the guide thimble tube) expected with current 3-cycle fuel schemes before fretting wear degradation would result in exceeding the present margin to the 6-g load criterion for the fuel-handling accident. However, the staff ~~requires~~ required several applicants to perform a surveillance program because of the uncertainties in predicting wear rates for the standard 17 x 17 fuel assembly design. The objective of this program was to demonstrate that no holes formed in rodded guide thimble tubes, thus providing some confidence that ~~scrammability~~ the ability to scram is ensured. These applicants formed an owners' group, which has submitted a generic report (Leasburg, March 1, 1982) that provides postirradiation examination results on guide thimble tube wear in the Westinghouse 17 x 17 fuel assembly design. On the basis of this report, the staff has concluded (Rubenstein, April 19, 1982) that the Westinghouse 17 x 17 fuel assembly design is resistant to guide thimble tube wear.

(5) Oxidation and Crud Buildup

In the FSAR, there is no explicit discussion of cladding oxidation, hydriding, and crud buildup. Crud and oxide are mentioned in Sections 4.4.2.9.1, 4.4.2.11, 4.4.2.11.5, and 4.4.4.5.2 of the FSAR. The applicable models for cladding oxidation and crud buildup are discussed in the supporting documentation (Salvatori, January 4, 1973) for the Westinghouse fuel performance code PAD-3.1. These models were previously approved by the staff. A new temperature-dependent cladding oxidation model is also presented in WCAP-9179. Because the temperature-independent model in PAD-3.1 is conservative with respect to the approved model in WCAP-9179, the staff continues to find the older models applicable. These models affect the cladding-to-coolant heat transfer coefficient and the temperature drop across the cladding wall. Mechanical properties and analyses of the cladding are not significantly impacted by oxide and crud buildup. On the basis of the Westinghouse discussion (Anderson, January 12, 1981) of the impact of cladding hydriding on fuel performance, and on our review of the oxidation and crud buildup models, the staff concludes that these effects have been adequately accounted for in the standard fuel design.

(6) Rod Bowing

In Section 4.2.3.1 of the FSAR, the applicant has indicated that the model used for evaluation of fuel rod bowing is in WCAP-8691 (nonproprietary version is WCAP-8692), which was withdrawn by Westinghouse. Revision 1 to WCAP-8691 was subsequently submitted by Westinghouse and has been approved by the staff (Thomas, December 29, 1982). The applicant needs to use Revision 1 to WCAP-8691 as the reference for the fuel rod bowing model and to confirm that the rod bowing analysis for Vogtle fuel has been performed with this model. This is a confirmatory item.

(7) Axial Growth

Relative to the discussion in Section 4.2.1.1(7), above, on stainless steel growth, the staff is aware of supporting information (Bloom, April 1972, and Appleby, April 1972) that was not cited by Westinghouse, but which also implies that irradiation growth of stainless steel should not be significant at the temperatures and fluences that are associated with PWR operation. Furthermore, because the staff is unaware of any operating experience that indicates axial-growth-related problems in Westinghouse NSSS plants, the staff concludes that

Westinghouse has made sufficient accommodations for control, source, and burnable poison rod axial rod growth in its NSSS designs.

The Westinghouse analysis of shoulder gap spacing for the SFA has found that interference will not occur until achieving burnups beyond traditional values. The staff, therefore, finds that the required shoulder gap spacing has been reasonably accommodated. However, for extended burnup applications, the adequacy of the spacing should be reverified. Furthermore, because stress-free irradiation growth of zirconium-bearing alloys is sensitive to texture (preferred cystallographic orientation) and retained cold work, which, in turn, are strongly dependent on the specific fabrication techniques that are employed during component production, the design shoulder gap should be reverified if current Westinghouse fabrication specifications are significantly altered.

Finally, the staff finds the Westinghouse analysis of fuel assembly growth to be acceptable. However, as stated in the above discussion on shoulder gap spacing, the fuel assembly growth should be reverified if significant changes are made in the current Westinghouse fabrication techniques.

(8) Fuel Rod and Nonfuel Rod Pressures

As noted in Section 4.2.1.3 of the FSAR, the analysis of fuel rod internal pressure for the standard fuel design is described in an approved (Stolz, May 19, 1978) topical report, WCAP-8963. The evaluation relies on the Westinghouse PAD-3.3 fuel performance code, which has also been approved (Stolz, February 9, 1979).

The analysis of the internal pressure of nonfueled rods for the SFA is generally based on Section III, Subsection NG-3000, of the ASME Code (see Section 4.2.1.6 of the FSAR). Absorber rod, burnable poison rod, and neutron source rod cladding is cold-worked Type 304 stainless steel, which is not covered by the ASME Code. Westinghouse therefore defines as the stress limit an intensity value S_m equal to 2/3 of the material yield stress. The yield stress for this material is approximately 62,000 psi. A strain limit of

1% also applies to the cladding. Predicted maximum values of rod internal pressure have been provided in the response (Anderson, January 12, 1981, and April 21, 1981) to Q231.2 and they are well below those imposed by the cladding stress and strain limits.

The staff concludes that there is adequate assurance that nonfueled core component rods can operate safely during conditions 1 and 2 because appropriate stress and strain limits are met even though the maximum internal rod pressure may exceed system pressure.

(9) Assembly Liftoff

In response to the staff's question on this topic, Westinghouse has confirmed that momentary liftoff will occur only during a turbine overspeed transient (this is also stated in Section 4.4.2.6.2 of the FSAR). Westinghouse has further found that (a) proper reseating will occur after momentary liftoff, (b) damage to adjacent assemblies will not occur even if one assembly is fully lifted and the adjacent ones remain seated, and (c) no ill consequences of momentary liftoff are expected. The staff concludes, therefore, that fuel assembly liftoff has been adequately addressed for the SFA design.

(10) Control Material Leaching

Although the design basis for the SFA control rods is to maintain cladding integrity, and although the probability of control rod cladding failures appears to be quite low, the staff has considered the corrosion behavior of the Vogtle control material and burnable poison and concludes that a breach in the cladding should not result in serious consequences because the Ag-In-Cd or hafnium absorber material and the poison material (borosilicate glass) are relatively inert.

4.2.3.2 Fuel Rod Failure Evaluation

The following paragraphs discuss the evaluation of (a) the ability of the SFA fuel to operate without failure during normal operation and anticipated transients, and (b) the accounting for fuel rod failures in the applicant's accident analysis. The fuel rod failure criteria described in Section 4.2.1.2, above, were used for this evaluation.

(1) Internal Hydriding

Westinghouse has used moisture and hydrogen control limits in the manufacture of earlier fuel types and has found that typical end-of-life cladding hydrogen levels are less than 100 ppm--a level below which hydride blister formation is not anticipated in fuel cladding. The staff therefore concludes that reasonable evidence has been provided that hydriding as a fuel failure mechanism will not be significant in the SFA.

(2) Cladding Collapse

In calculating the time at which cladding collapse will occur, Westinghouse uses the generic methods described in WCAP-8377, which the staff has approved (letter dated February 15, 1975, from D. B. Vassallo, NRC to C. Eicheldinger, Westinghouse). Inputs to the analysis include cladding ovality, helium prepressurization, free volume of the fuel rod, and limiting power histories.

The applicant has confirmed (FSAR Amendment No. 9, August 1984) that the calculated cladding collapse time for Vogtle using WCAP-8377 methods is more than the expected lifetime of the fuel. Consequently, it is concluded that the criterion for cladding collapse is satisfied.

(3) Overheating of Cladding

As stated in SRP Section 4.2, adequate cooling is assumed to exist when the thermal margin criterion to limit the departure from nuclear boiling (DNB) or boiling transition in the core is satisfied. The method employed to meet the DNB design basis is reviewed in Section 4.4 of this report.

(4) Overheating of Fuel Pellets

The design evaluation of the fuel centerline melt limit is performed with the Westinghouse fuel performance code, PAD-3.3 (WCAP-8720). This code, which has been approved by the staff (Bernard, July 20, 1982), is also used to calculate initial conditions for transients and accidents described in SRP Section 15 (see Section 4.2.3.3(1), below, for further comments on PAD-3.3).

In applying the PAD-3.3 code to the centerline melting analysis, the melting temperature of the UO_2 is assumed to be 5080°F unirradiated and is decreased by 58°F per 10,000 MWd/t. This relation has been almost universally adopted by the industry and has been accepted by the staff in the past. The expressions for thermal conductivity and gap conductance, described in Section 4.4.2.11 of the FSAR, are unchanged from that originally described in the PAD code. Therefore, they are acceptable.

The peak linear heat rate resulting from overpower transients/operator errors (assuming a maximum overpower of 118%) for Vogtle is 18.0 kW/ft. As noted in Section 4.4.2.11.6 of the FSAR, the centerline temperature at this peak linear heat rate is below that required to produce fuel melting.

Consequently, it is concluded that the criterion for the prevention of fuel centerline melting is satisfied.

(5) Pellet/Cladding Interaction

The only two PCI criteria in current use in licensing (1% cladding strain and no fuel melting), while not broadly applicable, are easily satisfied. As noted in the discussion of the cladding stress and strain evaluation, Westinghouse uses an approved code (PAD) to calculate creep strain, and the values calculated by that code are found to be below the 1% strain criterion. And, as indicated in the discussion on overheating failures in Section 4.4.2.11.6 of the FSAR, the no-centerline-melt criterion is satisfied. Therefore, the two existing licensing criteria for PCI have been satisfied.

In addition to the SRP-type treatment of PCI, however, the response (Anderson, January 12, 1981, and April 21, 1981) to Q231.23 and FSAR Section 4.2.3.3(a) address PCI from the standpoint of its effect on fatigue life. PCI produces cyclic stresses and strains that can affect fatigue life of the cladding. Furthermore, gradual compressive creep of the cladding onto the fuel pellet occurs from the differential pressure exerted on the fuel rod by the coolant. Westinghouse contends that by using prepressurized fuel rods, the rate of cladding creep is reduced, thus delaying the time at which fuel-to-cladding contact first occurs. The staff agrees that fuel rod prepressurization should improve PCI resistance, albeit in a presently unquantified amount.

In conclusion, Westinghouse has used approved methods to demonstrate that the present PCI acceptance criteria have been met.

(6) Cladding Rupture

In the LOCA analysis for SFA-designed plants, an empirical model is used to predict the occurrence of cladding rupture. The rupture model utilized for the large-break analysis is stated in Section 15.6.5.3.1.1 of the FSAR to be the 1981 version of the LOCA evaluation model; however, the references (8 and 13) stated for that model in the last paragraph of that section of the FSAR are incorrect. The correct reference (11, which is Revision 1 of WCAP-9220-P-A [and WCAP-9221-A] and has been approved by the staff) is in the reference list but is not used in the text of that section of the FSAR. The applicant should verify these references and make any necessary changes in an amendment to the FSAR. This is a confirmatory item.

The rupture model utilized for the small-break analysis was the approved October 1975 version of the ECCS evaluation model (see Section 15.6.5.3.1.2 of the FSAR). This model has been found acceptable for this analysis.

The appropriate references for the large-break LOCA analysis need to be confirmed. The overall impact of cladding rupture on the response of the SFA design to the loss-of-coolant accident is evaluated in Section 15.6.5 of this report.

4.2.3.3 Fuel Coolability Evaluation

The following paragraphs discuss the evaluation of the ability of the SFA fuel to meet the fuel coolability criteria described in Section 4.2.1.3, above. Those criteria apply to postulated accidents.

(1) Fragmentation of Embrittled Cladding

The primary degrading effect of a significant degree of cladding oxidation is embrittlement of the cladding. Such embrittled cladding will have a reduced ductility and reduced resistance to fragmentation. The most severe occurrence of such embrittlement is during a LOCA. The overall effects of cladding embrittlement on the SFA design for the loss-of-coolant accident are analyzed in Section 15.6.5 of this report and are not reviewed further in this section.

One of the most significant analytical methods that is used to provide input to the analysis in Section 15.6.5 of this report is the steady-state fuel performance code, which is reviewed in Section 4.2. This code provides fuel pellet temperatures (stored energy) and fuel rod gas inventories for the ECCS evaluation model as prescribed by Appendix K to 10 CFR 50. The code accounts for fuel thermal conductivity, fuel densification, gap conductance, fuel swelling, cladding creep, and other phenomena that affect the initial stored energy. For this purpose, Westinghouse uses a relatively new fuel performance code called PAD-3.3 (WCAP-8720). This code was approved by the staff's safety evaluation (Bernard, July 20, 1982).

For non-LOCA events, the locked-rotor accident (one-pump seizure with four loops operating) is the most severe undercooling event that is analyzed. This event is analyzed in Section 15.3.3 of the FSAR, where it is found that the peak cladding temperature is 1835°F, which is well below the 2700°F design limit. The analysis of this event is reviewed in Section 15.3.3/15.3.4 of this report, but it is clear that the SFA meets the non-LOCA peak cladding temperature design limit.

(2) Violent Expulsion of Fuel Material

The analysis that demonstrates that the design limits are met for this event for the SFA is presented in Section 15.4.8 of the FSAR and is discussed in Section 15.4.8 of this report.

(3) Cladding Ballooning and Flow Blockage

The cladding ballooning and flow blockage models for the large break LOCA are integral parts of the Westinghouse ECCS evaluation model. Consequently, the concern expressed in Section 4.2.3.2(6) of this report as to the appropriate references for the Westinghouse ECCS model used for the large break LOCA analysis needs to be addressed before this analysis can be approved. This is confirmatory.

The cladding ballooning and flow blockage analysis for the small break LOCA was performed with correlations from the approved October 1975 ECCS evaluation model (see Section 15.6.5.3.1.2 of the FSAR). This model has been found to be acceptable for this analysis.

The appropriate references for the large-break LOCA analysis need to be confirmed. The overall impact of cladding ballooning and assembly flow blockage models on the response of the SFA design to the loss-of-coolant accident is evaluated in Section 15.6.5 of this report.

(4) Structural Damage From External Forces

The applicant has stated in Section 4.2.3.4 of the FSAR that Westinghouse has performed these analyses utilizing models described in WCAP-8236 (and WCAP-8288) and WCAP-9401 (and WCAP-9402). WCAP-9401 essentially augments the information presented in WCAP-8236 because both WCAP reports apply to similar assemblies. WCAP-9401 has been reviewed and approved (Tedesco, May 7, 1981); therefore, these models are acceptable for these analysis.

The maximum grid and non-grid component loads from the safe shutdown earthquake (SSE) and LOCA events were calculated using the above approved models. The maximum impact loads from these two events were combined using the square-root-of-sum-of-squares (SRSS) method, as per SRP Section 4.2, Appendix A, for each component and found to be less than the allowable stresses for each of these Vogtle components. Consequently, these analyses are found to be acceptable.

4.2.4 Testing, Inspection, and Surveillance Plans

4.2.4.1 Testing and Inspection of New Fuel

As required by SRP Section 4.2, testing and inspection plans for new fuel should include verification of significant fuel design parameters. While details of the manufacturer's testing and inspection programs should be documented in quality control reports, the programs for onsite inspection of new fuel and control assemblies after they have been delivered to the plant should also be described in the FSAR. This is a confirmatory item.

The Westinghouse quality control program that will be applied to Vogtle fuel is discussed in Section 4.2.4 of the FSAR and addresses fuel system components and parts, pellets, rod inspection, assemblies, other inspections, and process control. Fuel system component inspection depends on the component parts and includes dimensions, visual appearance, audits of test reports, material certification, and nondestructive examinations. Pellet inspections, for example, are performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Fuel rod, control rod, burnable poison rod, and source rod inspections reportedly consist of nondestructive examination techniques such as leak testing, weld inspection, and dimensional measurements. Process control procedures are described in detail. In-core control component testing and inspection is described in Section 4.2.4.4 of the FSAR. In addition, the applicant states in Section 4.2.4.5 of the FSAR that if any tests and inspections are to be performed by others on behalf of Westinghouse, Westinghouse will review and approve the quality control procedures, inspection plans, and so forth, to ensure that they are equivalent to the description provided in Sections 4.2.4.1 through 4.2.4.4 of the FSAR and are performed properly to meet all Westinghouse requirements.

On the basis of the information provided in Section 4.2.4 of the FSAR and the commitment by Westinghouse to ensure the acceptability of any tests and inspections performed by others on behalf of Westinghouse, the staff concludes that the fuel testing and inspection program for new fuel is acceptable.

4.2.4.2 On-Line Fuel Failure Monitoring

In Section 11.5.2.3 and Table 11.5.2-1 of the FSAR, the applicant has provided a description of the chemical volume and control system (CVCS) letdown monitor for on-line fuel rod failure detection. A definite commitment to use the fuel failure detection instruments is required to meet the guidelines of paragraph II.D.2 of SRP Section 4.2. The applicant has indicated (see response to Q490.4-1 in Amendment No. 9, dated August 1984, to the FSAR) that this information will be provided in Amendment No. 10 to the FSAR. This is a confirmatory item.

Section 11.5.2.3 and Table 11.5.2-1 of the FSAR do not include information about the sensitivity of the CVCS letdown monitor for detecting fuel rod failures. The sensitivity of the monitor needs to be confirmed, as stipulated in paragraph II.D.2 of SRP Section 4.2. This is an open item.

4.2.4.3 Postirradiation Surveillance

Westinghouse has extensive experience with the use of 17 x 17 standard fuel assemblies in other operating plants. As noted in Section 4.2.3.3.2 of the FSAR, this experience is summarized in WCAP-8183, which is periodically updated to provide the most recent information on operating plants. Additional test assembly and test rod experience is given in Sections 8 and 23 of WCAP-8768, Revision 2.

Section 4.2.4.6 of the FSAR indicates that it is currently anticipated by the applicant that postirradiation poolside surveillance of the Vogtle fuel assemblies will not exceed a qualitative visual examination of some discharged fuel assemblies from each refueling, which satisfies part of paragraph II.D.3 of SRP Section 4.2. To satisfy the remaining part of paragraph II.D.3, the applicant must (a) make a commitment in the surveillance program to perform additional surveillance if unusual behavior is noticed in the visual examination or if plant instrumentation indicates gross fuel failures and (b) address the disposition of failed fuel in the postirradiation fuel surveillance program. These are both open items.

4.2.5 Evaluation Findings

The applicant has not yet provided:

(1) confirmation of the correct references for the cladding rupture and cladding ballooning and flow blockage models for the large-break LOCA (see Sections 4.2.3.2(6) and 4.2.3.3(3) in this report)

(2) a commitment to use the on-line fuel failure detection methods (see Section 4.2.4.2 in this report)

(3) the sensitivity of the chemical volume and control system letdown monitor for detecting fuel rod failures (see Section 4.2.4.2 in this report)

(4) a commitment in the postirradiation fuel surveillance program to perform additional surveillance if unusual behavior is noted in the visual examination or if plant instrumentation indicates gross fuel failures (see Section 4.2.4.3 in this report)

(5) information in the postirradiation fuel surveillance program that addresses the disposition of failed fuel (see Section 4.2.4.3 in this report)

(6) confirmation that the rod bowing analysis has been performed with the latest approved model (see Section 4.2.3.1(6) in this report)

When the above are provided, the staff will be able to conclude that the Vogtle fuel has been designed so that (a) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (b) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (c) core coolability will always be maintained, even after severe postulated accidents, and thereby satisfies the related requirements of 10 CFR 50.46; 10 CFR 50, Appendix A; GDC 10, 27, and 35; 10 CFR 50, Appendix K; and 10 CFR 100. This conclusion is based on the following.

(1) The applicant has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with structural response, control rod ejection, and fuel densification have been performed in accordance with (1) the guidelines of RG 1.77, and methods that the staff has reviewed and found to be acceptable alternatives to RGs 1.60 and 1.126, and (b) the guidelines for "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces" in Appendix A to SRP Section 4.2.

(2) The applicant has provided for testing and inspection of the fuel to ensure that it is within design tolerances at the time of core loadings. The applicant has made a commitment to perform on-line fuel failure monitoring and postirradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

The staff concludes that the applicant has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated and thereby satisfies the related requirements of 10 CFR 100. In satisfying these requirements, the applicant has (a) used the fisrsion product release assumptions of RGs 1.4, 1.25, and 1.77, and (b) performed the analysis for fuel rod failures for the rod ejection accident in accordance with the guidelines of RG 1.77.

On the basis of the review, it is concluded that the applicant's fuel system design has satisfied all the requirements of the applicable regulations, regulatory guides, and current regulatory positions.

4.3 Nuclear Design

The Vogtle power plants each have a reactor core consisting of 193 fuel assemblies of the Westinghouse standard 17 x 17 design. The core has a design heat output of 3411 MWt and is similar in most respects to the Callaway reactor and to other recent Westinghouse 4-loop reactors. The staff has reviewed the nuclear design of the Vogtle reactors in accordance with the guidelines provided by SRP Section 4.3, and was based on information contained in the FSAR, amendments thereto, and the referenced topical reports.

4.3.1 Design Bases

Design bases are presented which comply with the applicable GDC. Acceptable fuel design limits are specified (GDC 10), a negative prompt feedback coefficient is specified (GDC 11), and tendency toward divergent operation (power oscillation) is not permitted (GDC 12). Design bases are presented which require a control and monitoring system (GDC 13) which automatically initiates a rapid reactivity insertion to prevent exceeding fuel design limits in normal operation or anticipated transients (GDC 20). The control system is required to be designed so that a single malfunction or single operator error will cause no violation of fuel design limits (GDC 25). A reactor coolant boration system is provided which is capable of bringing the reactor to cold shutdown conditions (GDC 26) and the control system is required to control reactivity changes during accident conditions when combined with the engineered safety features (GDC 27). Reactivity accident conditions are required to be limited so that no damage to the reactor coolant system boundary occurs (GDC 28).

The staff finds the design bases presented in the FSAR to be acceptable.

4.3.2 Design Description

The FSAR contains the description of the first cycle fuel loading which consists of three different enrichments and has a first cycle length of approximately 1-1/2 years. 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 (0.55 and 0.44) and prompt neutron lifetime (19.4 and 18.1 microseconds) at beginning and end of cycle, respectively, are consistent with those normally used and are acceptable.

Power Distribution

The design bases affecting power distribution are:

- The peaking factor in the core will not be greater than 2.30 during normal operation of full power in order to meet the initial conditions assumed in the loss of coolant accident analysis.

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

- 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 factor).

The 2.30 F_Q peaking factor is determined and maintained via calculations of — Insert ½ C1 extremes of allowed transient power distributions and periodically measured radial power distributions and radial peaking factors F_{xy} and $F_{\Delta H}$. These also — Insert ½ provide maximum initial conditions for events described in Section 15 which ensure that peak full power does not cause centerline fuel melting or result in departure from nucleate boiling during anticipated operational occurrences.

The applicant has described the manner in which the core will be operated and power distribution monitored so as to ensure that these limits are met. The core will be operated in the constant axial offset control (CAOC) mode which has been shown to result in peaking factors less than 2.30 for both constant power and load following operation. The applicant has elected to use an improved load follow package, developed by Westinghouse, in Vogtle.

CAOC is described in WCAP-8385 (proprietary) and WCAP-8403 (non-proprietary), "Power Distribution Control and Load Following Procedures." This report contains methodology for operation with and without part-length control rods. The former mode allows better return to power capability than the latter. Use of part-length rods has been withdrawn from Westinghouse reactors. The improved load follow strategy provides a return to power capability during operation without part-length rods comparable to the level previously obtainable from operation with part-length rods.

The improved load-follow strategy involves a redesignated control rod bank and modified overlap that allows greater reactivity insertion than the former design bank within the constraints of a widened, asymmetric CAOC band. The control bank has been changed from eight to four rods. The four rods removed from the control bank have been reassigned as a shutdown bank, thus maintaining shutdown margins. The CAOC band has been changed from ±5 to +3, -12 delta flux difference. The greater inserted reactivity is available for return to power capability upon control rod withdrawal. Another element in the load follow strategy is the use of moderator temperature reductions to augment return to power capability. The temperature reduction adds reactivity during rapid return to power through the inherently negative moderator temperature coefficient.

The analysis used to calculate the maximum peaking factor which can occur using the improved strategy expands the set in the CAOC topical report to 18 calculational cases. However, with the reassigned control bank, maneuvers resulting in greater control rod insertion for a longer duration become operationally practical but tend to become slightly more limiting in terms of total peaking factors. Therefore, simulated load-follow maneuvers which return the flux difference to the target value (and thereby reduce control rod insertion) have been replaced by load-follow strategies which maintain the deeper rod insertion. As a result of its evaluation, the staff agrees with Westinghouse's conclusion that substitution of these more conservative cases will maintain the limiting nature of the 18-case load-following analysis.

The analysis performed by Westinghouse indicated that the peaking factor limit could not be met at beginning of life (BOL) of cycle 1 because of the wide flux difference band. This resulted in limiting the width of the band for the first 20% of the cycle typically, and until 3,000 MWd/MTU burnup for Vogtle to the value of ±5%. This ±5% is the value previously justified by the CAOC analysis. These features will be incorporated in the Vogtle Technical Specifications.

The staff concludes, for the reasons stated above, that the improved load-follow package will continue to prevent the 2.30 peaking factor limit from being exceeded in normal operation of the power plant, and therefore is acceptable.

The two types of instrumentation systems are normally provided to monitor core power distribution. Excore detectors with two axial sections are used to monitor core power, axial offset and azimuthal tilt for the 2.30 F_Q limit, and Insert movable incore detectors permit detailed power distributions to be measured. These systems are used in operating reactors supplied by Westinghouse and the staff finds their use acceptable for Vogtle when a 2.30 limit is the minimum requirement (or possibly lower when cycle specific 18-case or equivalent analyses so indicate).

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 FSAR and has evaluated the uncertainties of these values. The staff has reviewed the calculated values of reactivity coefficients and has concluded that calculated values adequately represent the full range of expected values. The staff has reviewed the reactivity coefficients used in the transient and accident analyses and concludes 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 ensure 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. The excess reactivity is controlled by a combination of full length control rods and soluble boron. Soluble boron is used to control changes due to:

- moderator density and temperature changes from ambient to operating temperatures

- equilibruim xenon and samarium buildup

- fuel depletion and fission product buildup - that portion not controlled by lumped burnable poison

- transient xenon resulting from load following

Control rods are used to control reactivity change due to:

- moderator reactivity changes from hot zero to full power

- 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 0.987 during initial 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 GDC 26.

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 10%. In addition bank worth measurements are performed as part of the startup test program to ensure that conservative values have been used in safety analyses.

On the basis of these comparisons, the staff concludes that the applicant has made suitably conservative assessments of reactivity control requirements and that adequate control rod worths have been provided to ensure shutdown capability.

Control Rod Patterns and Reactivity Worths

The 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 required in the Technical Specifications to ensure that:

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

- 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.

The staff has reviewed the calculated rod worths and the uncertainties in these worths, and concludes 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 Vogtle cores to xenon-induced spatial oscillations is discussed in the FSAR. The overall negative reactivity (power) coefficient provides assurance that the reactor will be stable against total power oscillation.

The applicant 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. The staff concurs with this conclusion.

This core is predicted to be unstable with respect to axial xenon oscillations after about 12,000 MWd per ton of exposure. The applicant has acceptably shown that axial 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 presents information on calculational techniques and assumptions used to ensure that criticality is avoided. The staff has reviewed this information and the criteria which will be employed and finds 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 (S_n) which results — Insert ½ c
in a neutron flux of 2.1×10^{10} neutrons/cm^2/sec having energy greater than 10^6 electron-volts at the inner vessel boundary. This results in a fluence of 2.2×10^{19} neutrons/cm^2 for a 40-year vessel life with an 80% use factor. The methods used for these calculations are state of the art, and the staff concludes that acceptable analytical procedures have been used to calculate the vessel fluence. The requirements for surveillance programs and the pressure-temperature limits for operation are addressed in Section 5.3.3 of this report.

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 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. The staff, therefore, concludes that these methods are acceptable for use in calculating the nuclear characteristics of Vogtle.

4.3.4 Summary of Evaluation Findings

The Vogtle nuclear design was reviewed according to SRP Section 4.3 (NUREG-0800). All areas of review and review procedures from that section have been followed either for this reactor or for previous similar reactors (e.g., Callaway) or for topical report reviews.

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 Vogtle.

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 0.987 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 its review, the staff concludes 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 ensure shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes available. The staff also concludes that nuclear design bases, features, and limits have been established in conformance with the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28.

This conclusion is based on the following:

(1) The applicant has satisfied the requirements of GDC 11 with respect to prompt inherent nuclear feedback characteristics in the power operating range by:

 (a) calculating a negative Doppler coefficient of reactivity, and

 (b) using calculational methods that have been found acceptable.

 The staff has reviewed the Doppler reactivity coefficients in this case and found this to be suitably conservative.

(2) The applicant has satisfied the requirements of GDC 12 with respect to power oscillations which could result in conditions exceeding specified acceptable fuel design limits by:

 (a) showing that such power oscillations are not possible and/or can be easily detected and thereby remedied, and

 (b) using calculational methods that have been found acceptable.

(3) The applicant has satisfied the requirements of GDC 13 with respect to provisions of instrumentation and controls to monitor variables and systems that can affect the fission process by:

 (a) providing instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other process variables such as temperature and pressure, and

(b) providing suitable alarms and/or control room indications for these monitored variables.

(4) The applicant has satisfied the requirements of GDC 26 with respect to provision for two independent reactivity control systems of different designs by:

(a) having a system that can reliably control anticipated operational occurrences,

(b) having a system that can hold the core subcritical under cold conditions, and

(c) having a system that can control planned, normal power changes.

(5) The applicant has satisfied the requirements of GDC 27 with respect to reactivity control systems that have a combined capability in conjunction with poison addition by the emergency core cooling system of reliably controlling reactivity changes under postulated accident conditions by:

(a) providing a movable control rod system and a liquid poison system, and

(b) performing calculations to demonstrate that the core has sufficient shutdown margin with the highest-worth stuck rod.

(6) The applicant has satisfied the requirements of GDC 28 with respect to postulated reactivity accidents by (reviewed under Section 15.4.8):

(a) meeting the regulatory position in RG 1.77,

(b) meeting the criteria on the capability to cool the core, and

(c) using calculational methods that have been found acceptable for reactivity insertion accidents.

(7) The applicant has satisfied the requirements of GDC 10, 20, and 25 with respect to specified acceptable fuel design limits by providing analyses demonstrating:

(a) that normal operation, including the effects of anticipated operational occurrences, have met fuel design criteria,

(b) that the automatic initiation of the reactivity control system ensures that fuel design criteria are not exceeded as a result of anticipated operational occurrences and ensures the automatic operation of systems and components important to safety under accident conditions, and

(c) that no single malfunction of the reactivity control system causes violation of the fuel design limits.

4.4 Thermal-Hydraulic Design

4.4.1 Performance and Safety Criteria

The performance and safety criteria for the Vogtle core design as stated in Section 4.4.1 of the FSAR are:

(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 rod failures. 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 immediate resumption of operation."

(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."

4.4.2 Design Bases

The performance and safety criteria listed above are implemented through the following design bases.

4.4.2.1 Departure From Nucleate Boiling

The margin to departure from nucleate boiling at any point in the core is expressed in terms of the departure from nucleate boiling ratio (DNBR). The DNBR 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, as stated in Section 4.4.1.1 of the Vogtle FSAR, for the prevention of departure from nucleate boiling is as follows:

"There will be a 95 percent probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation and operational transients and any transient arising from faults of moderate frequency (Condition I and II events) at a 95 percent confidence level."

4.4.2.2 Fuel Temperature

The fuel temperature design basis given in Section 4.4.1.2 is:

"During modes of operation associated with Condition I and Condition II events, there is at least a 95 percent probability that the peak kW/ft fuel rods will not exceed the UO_2 melting temperature at the 95 percent confidence level."

This design basis is evaluated in the Safety Evaluation Report in Section 4.2 "Fuel System Design."

4.4.2.3 Core Flow

Section 4.4.1.3 of the FSAR has the following core flow design basis.

"A minimum of 95.5 percent of the primary coolant flow will pass through the fuel rod region of the core and be effective for fuel rod cooling."

4.4.2.4 Hydrodynamic Stability

The hydrodynamic stability design basis given in Section 4.4.1.4 is as follows.

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

4.4.3 Thermal Hydraulic Design Methodology

4.4.3.1 Departure from Nucleate Boiling

The thermal-hydraulic design analysis was performed using the W-3 critical heat flux (CHF) correlation in conjunction with the THINC-IV analysis. THINC-IV is an open channel computer code which determines the coolant density, mass velocity, enthalpy, vapor void, static pressure, and DNBR distribution along parallel flow channels within a reactor core.

The W-3 correlation was developed from data obtained from experiments conducted with fluid flowing inside single heated tubes. As test procedures progressed to the use of rod bundles instead of tubes, the correlation was modified to include the effects of "R" and "L" mixing vane grids and axially non-uniform power distributions.

A correlation factor is developed to adopt the W-3 correlation to 17 x 17 fuel assemblies with top split mixing vane grids (R grid). This correlation factor, termed the "modified spacer factor," was developed as a multiplier on the W-3 correlation. A description of the 17 x 17 fuel assembly test program and a summary of the results are described in the NRC approved WCAP-8298-P-A and WCAP-8299-A. The test program predicted heat flux includes a 0.88 multiplier which is part of the 17 x 17 modified spacer factor. However, a multiplier of 0.86 has been conservatively applied for all DNB analyses. The test results indicated that a reactor core using this geometry may operate with a minimum DNBR of 1.28 and satisfy the design criterion. However, a minimum DNBR of 1.30 is conservatively used for this plant.

The applicant has proposed this minimum departure from nucleate boiling ratio of 1.30 to ensure that there is a 95% probability at a 95% confidence that critical heat flux will not occur on the limiting fuel rod. The use of the W-3 correlation with a minimum DNBR of 1.30 has been previously approved by the staff.

A description of the THINC-IV computer code is given in WCAP-7956, "THINC-IV, An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores." The design application of the THINC-IV program is given in detail in WCAP-8054, "Application of the THINC-IV Program to PWR Design." Both WCAP-7956 and WCAP-8054 have been reviewed and approved by the staff in a letter dated April 19, 1978, from J. Stole, NRC, to C. Eicheldinger, Westinghouse.

The staff has previously reviewed under a different docket, a November 2, 1977, letter from C. Eicheldinger (Westinghouse) to J. Stolz (NRC) which described THINC-IV analyses using a cosine upper plenum radial pressure gradient with a maximum value of 5 psi at the core center and 0 psi at the periphery. The results of these analyses showed that the effects of a core pressure distribution on the minimum DNBR are negligible. The staff conducted a similar sensitivity study using COBRA-IV. The staff's results also showed that the effects are small (NUREG-0847). On the basis of these analyses, the staff concludes that the use of a nonuniform exit pressure gradient in the Vogtle thermal-hydraulic design is acceptable.

The design calculational procedure, using THINC-IV, is to perform a core-wide analysis followed by a hot assembly and hot subchannel analyses.

For the hot assembly and hot subchannel analyses, a set of hot channel factors are used to account for deviations due to manufacturing tolerances. A reload review of a pressurized water reactor, not of Westinghouse design, showed that the hot channel factors used in the thermal-hydraulic analysis of the initial core did not bound future cycles (i.e., beyond the first cycle). The staff questioned the applicant to determine if their methods appropriately bound future cycles. The applicant responded that the safety analysis is intended to be valid for all plant cycles and the values of the input parameters used in the safety analysis are selected to bound the values expected in all subsequent cycles. The applicant further stated that when all of the reload related parameters for a given accident are bounded, the reference safety analysis is valid; however, if a parameter is not bounded, further evaluation is necessary. This further evaluation is to confirm that the margin of safety defined in the basis for any Technical Specification is not reduced. On the basis of the information given above, the staff concludes that the applicant has adequately addressed the staff's concerns on future cycle considerations.

The staff also asked ~~that~~ the applicant to provide information as to whether there are plans to use the Westinghouse optimized fuel assembly or Westinghouse improved thermal margin procedure as described in WCAP-8567 for Vogtle. The applicant indicated that Vogtle currently has no intention of using the optimized fuel assemblies or WCAP-8567 in the initial core. However, advanced fuels will be covered in subsequent refuelings and the applicant will then amend the FSAR as necessary.

On the basis of the staff's findings that the CHF correlation and the thermal-hydraulic computer code used by the applicant have been previously approved by the staff, that the applicant has appropriately bounded future cycles in its safety analyses, and that the use of a uniform core exit pressure gradient has been adequately justified, the staff concludes that the DNB design methodology used in the design of the Vogtle units is acceptable.

4.4.3.2 Core Flow

The core flow design basis requires that the minimum flow which will pass through the fuel rod region and be effective for fuel rod cooling is 95.5% of the primary coolant flow rate. The remainder of the flow, called bypass flow, will be ineffective for cooling since it will take the following bypass paths:

(1) flow through the spray nozzles into the upper head,
(2) flow into the rod cluster control rod guide thimbles,
(3) leakage from the vessel inlet nozzle directly to the vessel outlet nozzle,
(4) flow between the baffle and barrel, and
(5) flow in the gaps between the fuel assemblies.

The amount of bypass flow (4.5%) is determined by a series of hydraulic resistance calculations on the core and vessel internals and verified by model flow tests. Since the amount of bypass is consistent with approved plants of similar design, the staff concludes that the core flow given in the Vogtle FSAR, 95.5%, is acceptable. It is noted that the nominal value for reactor coolant flow per loop is 95,700 gpm (a total of 382,800 gpm for the four loops) as shown in FSAR Table 15.0.3-3. The minimum required reactor coolant flow ~~should~~ will be put in the Technical Specifications.

4.4.3.3 Hydrodynamic Stability

For steady-state, two-phase heated flow in parallel channels, the potential for hydrodynamic instability exists.

The applicant stated that the core design is stable since Westinghouse reactors will not experience any Ledinegg instability over Condition I and II operational ranges, and open channel configurations, which are a feature of Westinghouse PWRs, are more stable than closed channel configurations. This was shown by flow stability tests which were conducted at pressures up to 2200 psia. The results showed that for flow and power levels typical of power reactor conditions, no flow oscillations could be induced above 1200 psia.

Also, a method developed by Ishii (Saha et al., 1976) for evaluating density wave stability in parallel closed channel systems was used to assess the stability of typical Westinghouse reactor designs. The results indicate that a large margin to density wave instability exists. Finally, data from numerous rod bundle tests which were performed over wide ranges of operational conditions show no evidence of premature DNB or of inconsistent data which might be indicative of flow instabilities in the rod bundles.

The staff is conducting a generic study of the hydrodynamic stability of light water reactors. Limitations to the thermal-hydraulic design resulting from the staff study will be compensated for by appropriate operating restrictions; however, none are anticipated.

In the interim, the staff concludes that past operating experience, flow stability experiments and the inherent thermal-hydraulic characteristics of Westinghouse pressurized water reactors serve as a basis for accepting the Vogtle stability evaluation for issuance of an operating license.

4.4.4 Operating Abnormalities

4.4.4.1 Fuel Rod Bowing

A significant parameter which affects the thermal-hydraulic design of the core is rod-to-rod bowing within fuel assemblies. The Westinghouse methods for predicting the effects of rod bow on DNB, WCAP-8691, Revision 1 "Fuel Rod Bow Evaluation," have been approved by the staff in a letter dated December 29, 1982, from C. Thomas, NRC, to E. P. Rahe, Westinghouse.

In response to a question, the applicant stated that Westinghouse-designed plants do not consider the effects of rod bow for an assembly average burnup greater than 33,000 MWD/MTU because beyond this burnup, the burndown effects preclude the fuel from achieving the limiting value of $F_{\Delta H}$.

For the worst case, at an assembly average burnup of 33,000 MWD/MTU, the calculated penalty is less than 3%. However, sufficient margin (9.1%) is maintained to sustain full and low flow DNBR penalties.

On the basis of its review of the information in the FSAR response to questions, and the fact that the methods used have been previously approved by the staff, the staff has concluded that the proposed rod bow calculations are acceptable. The available thermal margin used to offset the rod bow penalty is required to be put in the bases of the Technical Specifications.

For plants designed by Westinghouse, the staff has approved the following generic margins ("Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," December 1976), which may be used to offset the reduction in DNBR due to rod bowing as shown in Table 4.1.

4.4.4.2 Crud Deposition

Crud deposition in the core and an associated change in core-pressure drop and flow have been observed in some PWRs not of Westinghouse design. The applicant has stated that: (1) operating experience on Westinghouse reactors indicates that a flow resistance allowance for crud deposition is not required; and (2) the effects of crud enter into the calculations by the use of a surface roughness factor three times greater than those obtained from operating Westinghouse PWRs. In response to a staff question the applicant stated that the reactor coolant system flow rate will be measured prior to initial criticality, at 50% power, and at least once per 18 months. A surveillance test procedure will give guidelines for reactor coolant system flow measurement each 18 months and will utilize the same method of measurement used in the startup test procedures. In the event of a lower than design flowrate, the action to perform an engineering analysis and justify continued operation is acceptable. The applicant should also provide a description of the means of detecting reduced flow in between the 18-month flow measurement test periods as it has not been verified that venturi fouling will be adequately accounted for in determining the core flow rate. This issue must be adequately addressed before the staff can approve the applicant's capability to measure core flow. This is an open item.

4.4.5 Thermal-Hydraulic Comparison

The thermal-hydraulic design parameters for Vogtle are listed in Table 4.2 and are compared to the values of these parameters for the SNUPPS plants as presented in the Vogtle FSAR.

Vogtle is designed to operate at the same thermal power as the SNUPPS plants. The W-3 CHF correlation and THINC-IV computer program were used in the design of both the Vogtle and SNUPPS plants.

The differences between the Vogtle and SNUPPS plants as shown in Table 4.2 are slightly lower values for the Vogtle maximum heat flux and maximum linear thermal output. These differences between the thermal-hydraulic designs of the Vogtle and SNUPPS plants are negligible. The SNUPPS plant has been previously reviewed and approved by the staff. Although the Vogtle FSAR shows the thermal-hydraulic values of the Vogtle are very similar to those of SNUPPS, the current SNUPPS FSAR shows some other differences (e.g., coolant flow and temperatures). The staff requested information on these differences and will report on our findings in a supplement to this SER. This is an open item.

4.4.6 N-1 Loop Operation

N-1 loop operation is when one of the reactors coolant loops is out of service. Thus, only three coolant loops are available to supply coolant to the reactor core.

In response to a staff question, the applicant stated that as a limiting condition for operation all reactor coolant loops shall be in operation during modes 1 and 2. The staff will require that the Technical Specification include appropriate provision to ensure that N-1 type of operation is prohibited. The applicant should also state whether or not there is any intention of operating in the N-1 mode in the future. This is a confirmatory item. The staff will report on this in a supplement to the SER.

4.4.7 Loose Parts Monitoring System

The applicant has provided a description of the loose parts monitoring system (LPMS) which will be used by Vogtle. This system is called a metal impact monitoring system (MIMS). The design will consist of 12 active instrumentation channels, each comprising a piezoelectric accelerometer (sensor), signal conditioning equipment and diagnostic equipment. Two redundant sensors are fastened mechanically to the reactor coolant system (RCS) at each of the following potential loose parts collection regions:

(1) ~~R~~eactor pressure vessel-upper head region
(2) ~~R~~eactor pressure vessel-lower head region
(3) ~~E~~ach steam generator-reactor coolant inlet region

The system will be capable of detecting a metallic loose part that weighs from 0.25 to 0.30 lb impacting within 3 ft of a sensor and having a kinetic energy of 0.5 ft-lb on the inside surface of the reactor coolant system (RCS) pressure boundary.

The system is designed to remain functional for a seismic event up to and including the operating basis earthquake.

The applicant's response was incomplete and also took exception to some items in RG 1.133 for the LPMS. Therefore, the staff requested more information (Q492.1). A response has not been provided. The staff will require the licensee to provide an LPMS consistent with the provisions of RG 1.133 and to commit to provide, before power operation, a final design report which contains the following: (a) an evaluation of the LPMS for conform-ance to RG 1.133; (b) a description of the system hardware, operation, and implementation of the loose parts detection program, including plans for startup testing, acquisition of baseline data, and alarm settings; (c) a description and evaluation of diagnostic procedures used to confirm the presence of a loose part; and (d) a description of the operator training program. This is an open item.

The staff will report its findings in a future SER supplement.

4.4.8 Instrumentation for Detection of Inadequate Core Cooling

The applicant's response to Q492.3 with respect to the design require-ments stated in NUREG-0737, Item II.F.2 "Instrumentation for Detection of Inadequate Core Cooling (ICC)," is incomplete. Therefore, the staff will require the applicant to provide the documentation required by Item II.F.2 of NUREG-0737. The ICC instrumentation consists of subcooling margin monitors, core exit thermocouple system, and reactor vessel level instrumenation system. The response should include evaluation of each ICC component against the design requirements. This is an open item. The staff will report its findings in a future SER supplement.

4.4.9 DNBR for Steamline Break

The RCS pressure during the steamline break (SLB) accident presented in Subsection 15.1.5 of the FSAR shows that the pressure drops below the range of pressure (1000 to 2300 psi) for which the W-3 correlation was originally developed. The staff requested more information (Q492.9) to justify the W-3 correlation for the SLB. The applicant's response was not extensive enough to resolve the staff's concern. This remains an open item. The staff will report on the resolution of this issue in a future supplement.

4.4.10 Conclusion

The thermal-hydraulic design of Vogtle was reviewed. The acceptance criteria used as the basis for the staff's evaluation are set forth in SRP Section 4.4, Part II, "Thermal and Hydraulic Design Acceptance Criteria" (NUREG-0800). The scope of the review included the design criteria, core design, and the steady-state analysis of the core thermal-hydraulic performance. The review concentrated on the differences between the proposed core design and those designs which have been previously reviewed and found acceptable by the staff. It was found that all such differences were acceptable except as noted below. The applicant's thermal-hydraulic design analyses were performed using analytical methods and correlations that have been previously reviewed by the staff and found acceptable.

The staff concludes that the initial core has been designed with appropriate margin to ensure that acceptable fuel design limits are not exceeded during steady-state operation and anticipated operational occurrences. The thermal-hydraulic design of the initial core, therefore, satisfies the requirements of GDC 10 (Appendix A of 10 CFR Part 50), and is acceptable. This conclusion is based on the applicant's analyses of the core thermal-hydraulic performance which the staff reviewed and found to be acceptable. The applicant has committed to a preoperational and initial startup test program in accordance with RG 1.68 to measure and confirm the thermal-hydraulic design aspects. For a discussion, refer to Section 14 of this report. ~~The staff has reviewed the applicant's preoperational and initial startup test program and has concluded that it is acceptable~~. However, before an operating license is issued, the staff requires the applicant to perform the following:

(1) Address the concerns regarding flow measurement capability with crud buildup as described in Section 4.4.4.2 of this SER.

(2) Address the concern on thermal-hydraulic design comparison as described in Section 4.4.5 of this SER.

(3) Address the concern regarding the N-1 loop operation as discussed in Section 4.4.6 of this SER.

(4) Address the concerns regarding the loose parts detection program as described in Section 4.4.7 of this SER.

(5) Supply the information for Item II.F.2 of NUREG-0737 as requested in Section 4.4.8 of this SER.

(6) Address the concern regarding DNBR for a steamline-break accident as described in Section 4.4.9 of this SER.

These issues will be addressed in . supplement to this SER.

4.5 Reactor Materials

4.5.1 Control Rod Drive Structural Materials

The staff concludes that the structural materials for the control rod drive mechanism are acceptable and satisfy the requirements of GDC 1, 14, and 26 and 10 CFR 50.55a. The applicant must confirm that the yield strength of austenitic stainless steels in these components does not exceed 90,000 psi.

This conclusion is based on the applicant having demonstrated that the properties of materials selected for components of the control rod drive mechanism exposed to the reactor coolant satisfy Appendix I of Section III of the ASME Code, and Parts A, B, and C of Section II of the Code. The applicant should confirm conformance with the staff position that the yield strength of cold-worked austenitic stainless steels does not exceed 90,000 psi. Conformance to the recommendations of RG 1.85 is discussed in Section 5.2.1.2.

The controls imposed upon the ferrite content of austenitic stainless steel filler materials satisfy the recommendations of RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," with the exception of the applicant's alternative approach of using chemical analysis in lieu of magnetic measurement devices to analyze the weld metal deposit to determine ferrite content. This approach has been discussed in WCAP 8324-A and was previously approved by the staff in a letter dated December 23, 1974, from D. B. Vassallo, NRC, to R. Salvatori, Westinghouse.

The controls imposed upon austenitic stainless steels to reduce sensitization satisfy, to the extent practical, the recommendations of RG 1.44, "Control of the Use of Sensitized Stainless Steel." The waiving of testing to show non-sensitization of fittings which do not have inaccessible cavities or chambers that would preclude rapid cooling when water is quenched or sprayed is an interpretation of the regulatory guide which the staff finds acceptable. The applicant has confirmed that the tempering temperatures and aging temperatures of heat-treatable materials in the control rod drive mechanism are specified to eliminate the susceptibility to stress corrosion cracking in reactor coolant. The fabrication and heat-treatment practices performed provide assurance that stress corrosion cracking will not occur during the design life of the components. The compatibility of all materials used in the control rod system in contact with the reactor coolant satisfies the criteria of Articles NB-2160 and NB-3120 of Section III of the Code.

Cleaning and cleanliness controls are in accordance, to the extent practical, with ANSI Std. N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," and RG 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated

Components of Water-Cooled Nuclear Power Plants." During final flushing, the applicant uses water of the same quality of the operating system's, except that the water is oxygen and nitrogen saturated. Since the applicant's systems which remove oxygen and nitrogen from the water will not be operating during final flushing, the staff concludes that the applicant has followed the recommendations of the regulatory guide and ANSI standard to the extent practical. The staff finds the applicant's approach acceptable. The applicant's use of controlled disposable materials and standard cleaning methods to control contamination levels of harmful elements and their compounds during construction follows the recommendations of the regulatory guide and ANSI standard to the extent practical and is thus acceptable to the staff.

4.5.2 Reactor Internal Materials

The staff concludes that the materials used for the construction of the reactor internal and core support structure are acceptable and meet the requirements of GDC 1 and 10 CFR 50.55a. The conclusion is based upon the following considerations:

The applicant has met the requirements of GDC 1 and 10 CFR 50.55a with respect to assuring that the design, fabrication, and testing of the materials used in the reactor internal and core support structure are of high-quality standards and adequate for structural integrity. The controls imposed upon components constructed of austenitic stainless steel satisfy, to the extent practical, the recommendations of RGs 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and 1.44, "Control of the Use of Sensitized Stainless Steel." Where the recommendations of these regulatory guides were not followed, the staff reviewed alternative approaches taken by the applicant and found them acceptable (see Section 4.5.1).

The materials used for construction of components of the reactor internal and core support structure have been identified by specification and found to be in conformance with the requirements of NG-2000 of Section III and Parts A, B, and C of Section II of the ASME Code. Conformance to the recommendations of RG 1.85, "Code Case Acceptability ASME Section III Materials," is discussed in

Section 5.2.1.2. As proven by extensive tests and satisfactory performance, the specified materials are compatible with the expected environment, and corrosion is expected to be negligible. The controls imposed on the reactor coolant chemistry provide reasonable assurance that the reactor internal and core support structure will be adequately protected during operation from conditions which could lead to stress corrosion of the materials and loss of component structural integrity.

The material selection fabrication practices, examination and testing procedures, and control practices performed in accordance with these recommendations provide reasonable assurance that the materials used for the reactor internal and core support structure are in a metallurgical condition to preclude inservice deterioration. Conformance with requirements of the ASME Code and the recommendations of the Regulatory Guides constitute an acceptable basis for meeting, in part, requirements of GDC 1 and 10 CFR 50.55a.

4.6 Functional Design of Reactivity Control Systems

Later

Table 4.1 Generic margins

Margin	% Reduction in rod bow penalties
The use of a design minimum DNBR of 1.30 instead of the 95/95 DNBR limit of 1.28.	1.6
A reduction in fuel rod pitch for the hot-channel analysis.	1.7
The use of a thermal diffusion coefficient (TDC) of 0.038 instead of a TDC of 0.051.	1.2
The addition of an extra grid in the design of the Westinghouse 17 x 17 fuel assembly relative to the 15 x 15 fuel design.	2.9
Insert 1/2 cR — The use of a 0.88 multiplier on the modified spacer factor (F_s) of the W-3 correlation instead of a 0.865 multiplier.	1.7
Maximum generic margin which may be claimed.	9.1

Table 4.2 Reactor design comparison

Characteristics	Vogtle	SNUPPS
<u>Performance Characteristics</u>		
Reactor core heat output (MWt)	3411	3411
System pressure, psia	2250	2250
Departure from nucleate boiling ratio		
Typical cell	2.07	2.07
Thimble cell	1.73	1.73
Minimum DNBR	1.30	1.30
Critical heat flux correlation	W-3	W-3
<u>Coolant Flow</u>		
Total flowrate (10^6 lb/hr)	140.3	140.3
Effective flowrate for heat transfer (10^6 lb/hr)	134.0	134.0
ft/sec | Average velocity along fuel rods (~~fps~~)	16.7	16.7
Average mass velocity (10^6 lb/hr-ft^2)	2.62	2.62
<u>Coolant Temperature, °F</u>		
Nominal reactor inlet	558.4	558.4
Average rise in core	60.1	60.1
<u>Heat Transfer, 100% Power</u>		
Active heat transfer surface area (ft^2)	59700	59700
Average heat flux (Btu/hr-ft^2)	189800	189800
Maximum heat flux (Btu/hr-ft^2)*	436500	440300
Average linear heat rate (kW/ft)	5.44	5.44
Maximum thermal output (kW/ft)	12.5	12.6

*This limit is associated with the value of ~~FQ~~ = 2.30 for Vogtle and F_Q = 2.32 for SNUPPS.

5 REACTOR COOLANT SYSTEM

5.1 Summary Description

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

During operation, the RCS 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 RCS at a flow rate and temperature consistent with achieving 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 RCS pressure boundary provides a second barrier against the release of radioactivity generated within the reactor and is designed to ensure a high degree of integrity throughout the life of the plant.

The RCS pressure changes during normal operation are controlled by the use of the pressurizer where water and steam are maintained at saturation conditions under operating conditions by electrical heaters and water sprays. Variations in pressure caused by reactor coolant contraction and expansion are minimized by the heaters forming steam or the pressurizer condensing steam as required. Spring-loaded safety valves and power-operated relief valves are mounted on the pressurizer and discharged to the pressurizer relief tank, when necessary, where steam is condensed and cooled by mixing with water.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance With ASME Code and Code Cases

Later

5.2.2 Overpressure Protection

Overpressure protection for Vogtle Electric Generating Plant, Units 1 and 2, has been reviewed in accordance with SRP Section 5.2.2 (NUREG-0800). Conformance with the acceptance criteria, except as noted, formed the basis for the staff's conclusion that the design of the facility for overpressure protection is acceptable.

The reactor coolant pressure boundary (RCPB) is protected from overpressurization by three pressurizer safety relief valves and two pressurizer power-operated relief valves in combination with the reactor protection system and operating procedures. For low-temperature operation, additional protection is afforded by the safety relief valve on each of the residual heat removal system (RHRS) suction lines. This combination of features provides overpressurization protection in accordance with the criteria of GDC 15 and GDC 31; the ASME Code, Section III; and 10 CFR 50, Appendix G. These criteria ensure RCPB overpressure protection for both power operation and low-temperature operation (startup and shutdown). Following is a discussion of overpressure protection for each mode of operation.

5.2.2.1 Overpressure Protection During Power Operation

Overpressure protection during power operation is provided by the pressurizer spray system, two power-operated relief valves (PORVs), and three spring-loaded safety relief valves (SRVs), all of which are connected to the pressurizer.

The pressurizer spray system is designed to maintain the reactor coolant system (RCS) pressure below the PORV relief setpoint of 2335 psig during normal design transients. The spray system flow is automatically modulated and can also be operated manually from the control room.

The two PORVs are solenoid-operated valves, each with a capacity of 210,000 lb of saturated steam per hour at 2385 psig. They are designed to limit the pressurizer pressure to a value below the high pressurizer pressure reactor trip setpoint. The PORVs also have the purpose of limiting challenges to the SRVs. However, the SRVs provide the final overpressure protection during power operation.

Both the SRVs and PORVs perform safety-related functions and thus have been designed to safety grade standards.

Credit is taken only for safety valves in analyzing operational transients and faulted conditions. Each pressurizer safety valve is spring loaded and has a relieving capacity of 420,000 lb mass per hour of saturated steam at 2485 psig. The combined capacity of two of these three safety valves is adequate to prevent the pressurizer pressure from exceeding the SRP criterion of 110% of design pressure and, therefore, the ASME Boiler and Pressure Vessel Code, Section III, limit for the pressure following the worst reactor coolant system pressure transient, identified to be a 100% load rejection resulting from a turbine trip with concurrent loss of main feedwater. This event was analyzed with no credit taken for operation of reactor coolant system PORVs, main steamline atmospheric steam dump valves, automatic rod control, auxiliary feedwater, condenser steam dump system, pressurizer level control system, and pressurizer spray system. Steam relief through the secondary safety valves is considered. In response to a staff concern, the applicant has stated that a time delay caused by discharge of the water in the safety valve loop seals was assumed in the analysis.

SRP Section 5.2.2 requires the applicant to demonstrate that adequate relief protection is provided, assuming the reactor trip is initiated by the second safety-grade signal from the reactor protection system. In the analysis the applicant has taken credit for a high pressurizer pressure trip (the first safety-grade trip signal from the reactor protection system). The evaluation is supported by a generic sensitivity study of required safety valve flow rate versus trip parameter presented in WCAP-7769. The applicant has also confirmed that the primary safety valves are sufficiently sized to maintain reactor coolant system pressure to within the ASME pressure limit without any reactor trip for the limiting overpressurization transient. Therefore, the SRP criterion has been met.

The above analyses were performed using the LOFTRAN Code, a digital simulation that includes point neutron kinetics, reactor coolant system including the reactor vessel, hot leg, primary side of the steam generator, cold leg, secondary side of the steam generator, pressurizer, and pressurizer surge line. The program computes pertinent plant variables, including temperatures, pressures and power level. The staff reviewed this code and found it acceptable. In addition, the applicant, in response to a staff concern, has stated that uncertainties in the design and operation of the plant were accounted for.

The applicant has provided assurance that the secondary safety valves can provide the required minimum relieving capacity assuming a single failure of one safety valve per loop. The staff has reviewed the assumed values of temperature and pressure, together with their assumed instrumentation channel errors that were used for the overpressure protection system design bases, and found them acceptable.

The safety valves are designed in accordance with ASME Code, Section III. Periodic testing and inspection of the PORVs and SRVs is required (10 CFR 50.55a(g)(4)(i)) to be performed in accordance with the edition of Section XI of this code that is in effect 12 months before the issuance of a license. Conformance to this requirement will be covered in a supplement to the SER in Section 3.9.6.

In FSAR Chapter 14, the applicant has described the preoperational test program, which includes testing of the pressure-relieving devices discussed in this SER section, and has indicated that these tests would be conducted in full compliance with the intent of RG 1.68, Revision 2. Additional testing of the SRVs and PORVs is required by NUREG-0737, Item II.D.1, "Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves." The applicant has stated that safety and relief valves similar to those at Vogtle have been tested within the Electric Power Research Institute safety and relief test program and have been found adequate for steamflow and waterflow. The evaluation of the applicant's compliance with Item II.D.1 is included in SER Section 3.9.3.

In response to NUREG-0737, Item II.D.3, the applicant has stated that positive position indication will be provided for the PORVs (magnetic reed switches) and the SRVs (stem-mounted limit switches). The evaluation of the applicant's compliance with Item II.D.3, "Direct Indication of Relief and Safety Valve Position," is included in SER Section 7.5.

The applicant states in FSAR Section 5.4.13.2 that it will be responsible for ensuring that any failure of PORVs or safety valves to close will be reported promptly to the NRC and that all challenges to PORVs and safety valves will be documented in the annual report. The staff concludes that the Vogtle procedures meet the criteria of TMI-2 Task Action Plan Item II.K.3.3, "Report Safety Valve and Power-Operated Relief Valve Challenges and Failures."

Check valves in the discharge side of the high-pressure safety injection, low-pressure safety injection, residual heat removal, and charging systems perform an isolation function in that they protect low-pressure systems from full reactor pressure. The staff requires that these check valves be classified ASME IWV-2000 Category AC, with the leak testing for this class of valve being performed to Code specifications. The applicant has stated that test lines and valves have been provided to ~~adequately check~~ assure that those check valves that serve to prevent backflow from the RCS into the safety injection system (including the RHR) will perform their isolation function.

5.2.2.2 Overpressure Protection During Low-Temperature Operation

The criteria for overpressure protection during low-temperature operation of the plant are in BTP RSB 5-2.

Low-temperature overpressure protection is primarily provided by the two pressurizer PORVs. These valves have their opening setpoints automatically adjusted as a function of reactor coolant temperature. In order to achieve this, the reactor coolant system wide-range temperature measurements will be auctioneered to obtain the lowest val~~v~~ue. This temperature will then be sent

value?

to a function generator that will have a PORV setpoint curve programmed into it. This PORV setpoint curve is to adequately account for the lag in the temperature change of the reactor vessel and for possible single failures in the auctioneering system. This function generator will produce a calculated maximum allowable pressure for the prevailing temperature. The calculated pressure is then compared to the indicated RCS pressure from a wide-range pressure channel. If the measured reactor coolant pressure approaches the maximum allowable pressure within a certain limit, an alarm is sounded on the main control board indicating a pressurization transient. If the reactor coolant pressure continues to increase, the PORVs are automatically opened to mitigate the pressure transient. Thus, the system pressure will always be below the maximum allowable pressure. This PORV setpoint curve shall be periodically updated, as shall be specified in the Bases for the Technical Specifications, to ensure that the stress intensity factors for the reactor vessel at any time in life are lower than the reference stress intensity factors as specified in 10 CFR 50, Appendix G. An alarm is provided to remind the operator to manually arm this system during cooldown. The cold overpressurization mitigation system (COMS) has been designed as two separate trains in order to meet the single-failure criteria.

The applicant has performed low-temperature overpressure transient analyses to determine the maximum pressure for the postulated worst-case mass input and heat input events that would challenge the cold overpressurization mitigation system, which includes the PORVs. The mass input transient analysis was performed assuming complete loss of letdown concurrent with full flow from one charging pump and spurious closure of the RHR inlet isolation valves. An event with the potential of even greater mass addition is the inadvertent actuation of a safety injection pump. The applicant has stated that this event will be mitigated by discharge through the RHR suction line relief valves.

The heat input analysis was performed for an inadvertent reactor coolant pump start assuming that the RCS was water solid at the initiation of the event and that a 50°F mismatch existed between the RCS (250°F) and the secondary side of the steam generators (300°F). These temperatures were assumed because at temperatures lower than these, the mass input case is limiting. The 50°F mismatch is based upon a Technical Specification requirement. The results of these analyses show that the allowable limits will not be exceeded. The applicant will provide PORV setpoint values later, and the staff will report its evaluation of these in a supplement to this SER.

These analyses for the worst case heat and mass input transients were presented in WCAP 10529 and as such, they differ from those presented in the FSAR. The applicant should revise the FSAR to confirm the assumptions and conclusions of this Westinghouse report. The staff will consider this a confirmatory issue until the FSAR is revised.

Inadvertent injection of an accumulator has been precluded as an overpressurization event because the isolation valve is closed with power locked out. Deliberate opening of the valve is not likely because the valves are not required to be tested at shutdown. Inadvertent injection from ~~to~~ a safety injection (SI) pump, which is periodically tested, has been considered and is relieved by other means as discussed later in this section.

The staff was concerned that the COMS may not adequately protect the reactor vessel for those events that resulted from temperature changes to the primary coolant in such a way that the auctioneered temperature differed from the vessel temperature. In Amendment 6 to the FSAR, the applicant has stated that nearly complete thermal mixing occurs in the RCS and that even failure of a temperature detector would not affect the COMS's ability to protect the reactor vessel since the detector is upstream of the SI injection location.

An acceptance criterion for Item II.G.1, "Emergency Power for Pressurizer Equipment," of NUREG-0737 is that the PORVs and associated block valves have safety-grade emergency power supplies. Section 8.3 of this report provides a discussion of Vogtle's compliance with this criterion.

As a backup to the low-temperature overpressure protection system, each of the two inlet suction lines to the residual heat removal system (RHRS) is equipped with a pressure relief valve with a capacity of 900 gpm at a setpoint pressure of 450 psig. The relieving capacity of each valve is adequate to relieve the combined flow of the two centrifugal charging pumps. The RHR suction relief valves provide overpressure protection after the RHR is put into operation and the RHR suction isolation valves are opened at RCS pressures less than 400 psig. As such, credit is taken for the RHR relief valves relieving pressure in the event of an inadvertent SI pump injecting into the primary system.

In performing the low-temperature overpressurization (LTOP) analysis, the applicant has assumed that the power to the SI pumps has been locked out, except for periodic testing, and that below 1000 psig and 425°F the isolation valves on the accumulators will be closed with power locked out. The staff requires technical specifications on the pressure in addition to technical specifications on the maximum permissible temperature mismatch between the secondary and the primary before a reactor coolant pump may be started.

5.2.2.3 Conclusions

Subject to (1) the generation of a conservative PORV setpoint curve, (2) a commitment to periodically update the PORV setpoints for LTOPs to account for radiation-induced embrittlement, (3) appropriate Technical Specifications, as discussed above to prevent LTOPs, and (4) a revision of the FSAR to reference the correct analysis for mass and heat input, the staff concludes that the overpressure protection system for both normal and low-temperature operation satisfies the relevant criteria of GDC 15 and 31 and is, therefore, acceptable. Conformance to Appendix G to 10 CFR 50 criteria will be confirmed when the PORV setpoint curve is found acceptable. This conclusion is based on the following:

The overpressure protection system prevents overpressurization of the RCPB under the most severe transients and limits reactor pressure during normal operational transients. Overpressurization protection is provided by three safety valves. These valves discharge to the pressurizer relief tank through a common header from the pressurizer. The pressurizer safety and power-operated relief valves and the RHR relief valves in the primary system, in conjunction with the steam generator safety and atmospheric steam dump valves in the secondary system, and the reactor protection system, will protect the primary system against overpressure.

The peak primary system pressure following the worst transient is limited to the ASME Code-allowable value with no credit taken for nonsafety-grade relief systems. The Vogtle plant was assumed to be operating at design conditions (102% of rated power) and the reactor is shut down by a high pressurizer pressure trip signal. The calculated pressure is less than 110% of design pressure.

Overpressure protection during low-temperature operation of the plant is provided by two PORVs and RHR suction relief valves in conjuntion with administrative controls.

The applicant has satisfied GDC 15 and 31. Appendix G criteria are expected to be ~~met~~ satisfied when the PORV setpoint curve is generated. In addition, the applicant has responded to TMI-2 Task Action Plan Items II.D.1 and II.D.3 of NUREG-0737.

Section 5.2.2 of the FSAR needs to be amended to bring it into conformance with the responses to staff questions as identified in Amendment 6 to the FSAR. The staff will consider this item confirmatory until the FSAR revisions are made. A Technical Specification for the maximum permissible temperature mismatch between the primary and secondary systems is also a confirmatory item.

5.2.3 Reactor Coolant Pressure Boundary Materials

The staff concludes that the plant's design is acceptable and satisfies the requirements of GDC 1, 4, 14, 30, and 31 of Appendix A of 10 CFR Part 50; the requirements of Appendices B and G of 10 CFR Part 50; and the requirements of 10 CFR 50.55a. This conclusion is based on the staff's review of the FSAR.

The materials used for construction of components of the RCPB have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code. Compliance with the above Code provisions for materials specifications satisfies the quality standards requirements of GDC 1, GDC 30, and 10 CFR 55.55a.

The materials of construction of the RCPB exposed to the reactor coolant have been identified and all of the materials are compatible with the primary coolant water, which is chemically controlled in accordance with appropriate technical specifications. This compatibility has been proven by extensive testing and satisfactory performance. This includes conforming to the recommendations of ~~Regulatory Guide~~ RG 1.44, "Control of the Use of Sensitized Stainless Steel," or the alternative approaches taken by the applicant which are acceptable to the staff (see Section 4.5.1).

General corrosion of all materials in contact with reactor coolant is negligible, and accordingly, general corrosion is not of concern. Compatibility of the materials with the coolant and compliance with the Code provisions satisfy the requirements of GDC 4 relative to compatibility of components with environmental conditions.

The materials of construction for the RCPB are compatible with the thermal insulation used in these areas. The thermal insulation used on the RCPB is either the reflective stainless steel type or is made of nonmetallic compounded materials that are in conformance with the recommendations of RG 1.36, "Non-metallic Thermal Insulation for Austenitic Stainless Steels." Conformance with the above recommendations satisfies the requirements of GDC 14 and GDC 31 relative to prevention of failure of the RCPB.

The ferritic steel tubular products and the tubular products fabricated from austenitic stainless steel have been found to be acceptable by non-destructive examinations in accordance with provisions of the ASME Code, Section III. Compliance with these Code requirements satisfies the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

The fracture toughness tests required by the ASME Code, augmented by Appendix G, 10 CFR Part 50, provide reasonable assurance that adequate safety margins against nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the RCPB. The use of Appendix G of the ASME Code, Section III, and the results of fracture toughness tests performed in accordance with the Code and NRC regulations in establishing safe operating procedures, provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and NRC regulations satisfies the requirements of GDC 31 and 10 CFR 50.55a regarding prevention of fracture of the RCPB.

The applicant has taken alternative approaches to the recommendations of RG 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels." The alternative approaches taken by the applicant are that welding procedures are qualified within the preheat temperature range rather than at the minimum preheat temperature, and preheat temperatures are maintained for an extended period of time rather than preheat temperatures maintained until the start of post-weld heat treatment. The staff concludes that these alternative approaches are adequate to prevent hydrogen cracking (the concern of this regulatory guide) and will not cause other hazards. Accordingly, the staff accepts these alternative approaches. The controls used provide reasonable assurance that components made from low-alloy steels will not crack during fabrication. If cracking does occur, the required Code inspections should detect such flaws. These controls satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a. RG 1.34, "Control of Electroslag Weld Properties," is not applicable because the electroslag welding process was not used when fabricating RCPB components.

The controls imposed on welding ferritic and austenitic steels under conditions of limited accessibility satisfy, to the extent practical, the recommendations of RG 1.71, "Welder Qualification for Areas of Limited Accessibility." As an alternative approach to Position C.1 of RG 1.71, the applicant's contractors maintain close supervisory control of the welders and welding situations in production are reoccur often enough to ensure that the most skilled welders are x
used in areas of limited accessibility. The staff concludes, that as such welds are inspected, qualification of the welders making acceptable welds occurs under the Code. These controls satisfy the quality standards requirements of GDC 1, GDC 50, and 10 CFR 50.55a. The controls imposed on weld cladding of low-alloy steel components by austenitic stainless steel are in accordance with the recommendations of RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components."

The controls to avoid stress corrosion cracking in RCPB components constructed of austenitic stainless steels satisfy, to the extent practical, the recommendations of RGs 1.44, "Control of the Use of Sensitized Stainless Steel," and

1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems Associated Components of Water-Cooled Nuclear Plants." The staff reviewed the alternative approaches taken by the applicant and found them acceptable (see Section 4.5.1).

The controls followed during material selection, fabrication, examination, protection, sensitization, and contamination, provide reasonable assurance that the RCPB components of austenitic stainless steels are in a metallurgical condition that minimizes susceptibility to stress corrosion cracking during service. These controls satisfy the requirements of GDC 4 relative to compatibility of components with environmental conditions and requirements of GDC 14 relative to prevention of leakage and failure of the RCPB.

The controls imposed during welding of austenitic stainless steels in the RCPB (1) satisfy, to the extent practical, the recommendations of RGs 1.31 and 1.71 or (2) the staff reviewed the alternative approaches taken by the applicant and found them acceptable (see Section 4.5.1 for RG 1.31, and this section for RG 1.71).

The controls provide reasonable assurance that welded components of austenitic stainless steel did not develop microfissures during welding and have high structural integrity. These controls satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a, and satisfy the requirements of GDC 14 relative to prevention of leakage and failure of the RCPB.

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

This section was prepared with the technical assistance of Department of Energy (DOE) contractors from the Idaho National Engineering Laboratory.

5.2.4.1 Compliance With the Standard Review Plan

The July 1981 Edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800) includes Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing." The Vogtle Electric Generating Plant Units 1 and 2 (Vogtle) review is continuing because the applicant has not completed the preservice inspection (PSI) program and examinations. The staff review to date was conducted in accordance with Standard Review Plan (SRP) Section 5.2.4, except as discussed below.

Paragraph II.4, "Acceptance Criteria, Inspection Intervals," has not been reviewed because this area applies only to inservice inspections (ISIs), not to PSIs. This subject will be addressed during review of the ISI program after licensing.

Paragraph II.5, "Acceptance Criteria, Evaluation of Examination Results," has been reviewed. The applicant committed in the FSAR to incorporate ASME Code Article IWB-3000, "Acceptance Standards for Flaw Indications," into the PSI program. However, ongoing NRC generic activities and research projects indicate that the presently specified ASME Code procedures may not always be capable of detecting the acceptable size flaws specified in the IWB-3000 acceptance standards. For example, ASME Code procedures specified for volumetric examination of the reactor vessel, bolts and studs, and piping have not proven to be capable of detecting the acceptable size flaws in all cases. The staff will continue to evaluate the development of new or improved procedures and will require that these improved procedures be made a part of the inservice examination requirements. The applicant's repair procedures based on ASME Code Article IWB-4000, "Repair Procedures," have not been reviewed. Repairs are not generally necessary in the PSI program. This subject will be addressed when the staff reviews the ISI program.

Paragraph II.7, "Acceptance Criteria, Code Exemptions," has not been completed because the applicant has not listed in the FSAR or the PSI program any Code exemptions as permitted by the criteria in IWB-1220. The SRP requires that the applicant list these exemptions, if used.

Paragraph II.8, "Acceptance Criteria, Relief Requests," has not been completed because the applicant has not identified all limitations to examination. Specific areas where ASME Code examination requirements cannot be met will be identified as performance of the PSI progresses. The complete evaluation of the PSI program will be presented in a supplement to this Safety Evaluation Report (SER) after the applicant submits the required examination information, identifies all plant-specific areas where ASME Code Section XI requirements cannot be met, and provides a supporting technical justification.

5.2.4.2 Examination Requirements

GDC 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A of 10 CFR Part 50, requires, in part, that components of the reactor coolant pressure boundary be designed to permit periodic examination and testing of important areas and features to assess their structural and leak-tight integrity. To ensure that no deleterious defects develop during service, selected welds and weld heat-affected-zones (HAZs) will be examined periodically at Vogtle.

The design of the ASME Code Class 1 and 2 components of the reactor coolant pressure boundary incorporates provisions for access for inservice examinations, as required by Subarticle IWA-1500 of Section XI of the ASME Code. Paragraph 50.55a(g), 10 CFR Part 50, defines the detailed requirements for the preservice and inservice programs for light-water-cooled nuclear power facility components. On the basis of the construction permit date of June 28, 1974, components (including supports) which are classified as ASME Code Class 1 and 2 shall meet the preservice examination requirements set forth in editions of Section XI of the ASME Code and addenda in effect 6 months before the date the construction permit is issued. The components (including supports) may meet the requirements set forth in subsequent editions of this Code and addenda which are incorporated by reference in Paragraph (b) of 10 CFR 50.55a, subject to the limitations and modifications listed therein. The initial ISI program must comply with the requirements of the latest edition and addenda of Section XI is issued, of the ASME Code in effect ~~twelve~~ 12 months before the date the operating license subject to the limitations and modifications listed in ~~Paragraph~~ 10 CFR 50.55a(b) ~~of 10 CFR Part 50~~.

5.2.4.3 Evaluation of Compliance With 10 CFR 50.55a(g)

Review has been completed on the information presented in the FSAR through Amendment 9, dated August 1984, and the "Preservice Inspection Program," Revision 0, dated April 19, 1984. The applicant states that on the basis of the construction permit date for Vogtle, the preservice inspection program is required to ~~meet~~ conform with ASME Code Section XI, 1971 Edition through the Winter 1972 Addenda. The applicant has voluntarily updated the preservice inspection program on the basis of ASME Code Section XI, 1980 Edition, with addenda through Winter 1980. The use of later referenced Code editions is acceptable as specified by 10 CFR 50.55a(g).

The staff has reviewed the PSI program for the reactor coolant pressure boundary systems and components. As the applicant stated in the FSAR, these systems and components are included for examinations per the applicable Code requirements. The staff established technical positions in the FSAR questions, some of which are resolved in the PSI program. The following items require further input or clarification from the applicant:

(1) The PSI program should contain a list of components subject to examination and a description of the welds exempted from examination in accordance with IWB-1220 and IWC-1220 and should include the criteria for exemption. In addition, the examination isometric drawings are necessary for the staff to determine the acceptability of the sample of welds required to be examined (Q250.1).

(2) In response to FSAR Questions 250.2 and 250.3, the applicant states that ultrasonic examination procedures have been or are being developed for the examination of (a) the reactor coolant pipe and fittings fabricated from SA351, Grade CF8A (centrifugal cast stainless steel) and (b) the reactor vessel which addresses the degree of compliance with RG 1.150 which discusses the near-surface examinations, the resolution with regard~~s~~ to detection of actual flaws, and the use of electronic gating as related to the volume of material near the surface of the reactor pressure vessel (RPV) that may not be examined. The staff requests that the applicant provide these examination procedures, so the staff can complete this review.

(3) The preservice examination preservice of the steam generator tubes should be performed in accordance with NUREG-0452, Revision 2, and RG 1.83, Revision 1, as discussed in Section 5.4.2.2 .2 of this SER.

After the examinations are performed, the specific areas where the Code requirements cannot be met will be identified. The applicant has committed to identify all plant-specific areas where the Code requirements cannot be met and to provide a supporting technical justification for requesting relief. The SER input will be completed after the applicant:

(1) dockets an acceptable resolution to the above issues

(2) dockets a complete and acceptable PSI program

(3) submits all relief requests with a supporting technical justification

The staff considers the review of Vogtle's PSI program an open item subject to the applicant providing the above items. Evaluation of the response will be reported in a supplement to the SER. (Refer also to Section 6.6 for additional discussion of the Vogtle PSI.)

The applicant has not submitted the initial inservice inspection program. The staff will evaluate this program after the applicable ASME Code edition and addenda can be determined based on ~~Paragraph~~ 10 CFR 50.55a(b) ~~of 10 CFR Part 50~~, but before ISI commences during the first refueling outage.

5.2.4.~~5~~4 Conclusions

The Vogtle preservice inspection program is an open item until the issues identified in Sections 5.2.4.3 and 6.6.3 are resolved.

The conduct of periodic examinations and hydrostatic testing of pressure-retaining components of the reactor coolant pressure boundary, in accordance with the requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and 10 CFR Part 50, will provide reasonable assurance that structural degradation or loss of leak-tight 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 preservice and inservice examinations required by the Code and by

10 CFR Part 50 constitutes an acceptable basis for satisfying the inspection requirements of GDC 32.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection
Later

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

The staff concludes that the reactor vessel materials are acceptable and satisfy the requirements of GDC 1, 4, 14, 30, 31, and 32 of Appendix A of 10 CFR Part 50; the material testing and monitoring requirements of Appendices B, G, and H of 10 CFR Part 50; and the requirements of 10 CFR 50.55a. This conclusion is based on the following:

The materials used for construction of the reactor vessel and its appurtenances have been identified by specification and found to be in conformance with 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 (see FSAR Section 5.3.1.1). Compliance with the above Code provisions for material specifications satisfies the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

5.3.1.1 Fracture Toughness

The staff has reviewed (1) the fracture toughness of ferritic reactor vessel and reactor coolant pressure boundary materials, and (2) the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references which are the basis for this evaluation are set forth in paragraphs II.5, II.6, and II.7 (Appendices G and H of 10 CFR Part 50) of SRP Section 5.3.1 in NUREG-0800, Revision 1, dated July 1981. A discussion of this review follows.

GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary" (Appendix A of 10 CFR Part 50) requires, in part, that the reactor coolant pressure boundary (RCPB) be designed with sufficient margin to ensure that, when stressed under operating, maintenance, and test conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

GDC 32, "Inspection of Reactor Coolant Pressure Boundary" (Appendix A of 10 CFR Part 50) requires, in part, that the RCPB be designed to permit an appropriate material surveillance program for the reactor pressure vessel.

The fracture toughness requirements for the ferritic materials of the reactor coolant pressure boundary are defined in Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Requirements" of 10 CFR Part 50.

Compliance With 10 CFR 50.55a

The Edition and Addenda of the ASME Code that are applicable to the design and fabrication of the reactor vessel and RCPB components are specified in 10 CFR 50.55a.

The ASME Code edition and addenda that apply are determined by the date the construction permit was issued. The Vogtle construction permits were issued on June 28, 1974. On the basis of that date, 10 CFR 50.55a requires that ferritic materials used for the Vogtle reactor vessels be designed and constructed to editions that are no earlier than the Winter 1971 Addenda to the 1971 ASME Code (hereafter Code) and that ferritic materials used in piping, pumps, and valves be constructed to editions that are no earlier than the Winter 1972 Addenda to the Code. The Vogtle ferritic materials meet all the above requirements.

Compliance With Appendix G, 10 CFR Part 50

The staff evaluated the Vogtle FSAR to determine the degree of compliance with the fracture toughness requirements of Appendix G, 10 CFR Part 50. The applicant has satisfied all the requirements of this appendix, except as discussed below.

Appendix G requires that for the reactor beltline materials the Charpy V-notch (C_v) impact tests shall be conducted at appropriate temperatures over a temperature range sufficient to define the C_v test curves (including the upper-shelf levels) in terms of both fracture energy and lateral expansion of specimens.

The Vogtle FSAR (Tables 5.3.1-2 and 5.3.1-3) contains upper-shelf Charpy V-notch (C_v) impact test data for the reactor beltline materials, but does not have the C_v curves in terms of fracture energy and lateral expansion. To fully comply with Appendix G, the applicant must supply the impact test data and the C_v curves for the beltline materials. Until the applicant provides the data and curves, the staff cannot complete its review of this item.

Compliance With Appendix H, 10 CFR Part 50

The materials surveillance program at Vogtle 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 as required by GDC 32, "Inspection of Reactor Coolant Pressure Boundary." The Vogtle surveillance program, which must be in compliance with Appendix H of 10 CFR Part 50 and ASTM E185, "Standard Recommended Practices for Surveillance Tests for Nuclear Reactor Vessels," requires that fracture toughness data 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 throughout its service life.

On the basis of the staff's review of the applicant's submittal that detailed the extent of the compliance of Vogtle with Appendix H of 10 CFR Part 50 and ASTM E185, the staff has determined that these requirements have been met, except as follows.

To determine the effect that neutron irradiation has on the reactor vessel, samples from the limiting material should be placed into the surveillance capsules. The limiting materials for the Vogtle Unit 2 reactor vessel beltline is plate material from heat B8628-1 and material from weld G-1.60. The materials in Vogtle Unit 2 surveillance capsules are from plate B8628-1 and weld E-3.23. Because the Vogtle Unit 2 surveillance weld material is not the most limiting material, the applicant's surveillance program will not completely monitor the extent of radiation damage to the weld material in Vogtle Unit 2.

Although the limiting weld metal is not contained in the applicant's surveillance program for Vogtle Unit 2, the applicant will be required to determine the effect of neutron irradiation damage on the limiting weld metal using RG 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The staff recommends that the applicant use the method predicting neutron irradiation damage that is in RG 1.99. The use of this guide to predict neutron irradiation damage is an acceptable alternative to testing the limiting weld metal as part of the surveillance program. This is an open item pending the applicant's commitment to comply with RG 1.99 and inclusion of this information in the Technical Specifications.

ASTM E185 also requires that the withdrawal schedules for the surveillance specimens be defined. Table 2 of Amendment 5 of the applicant's FSAR states that the withdrawal schedules will be in accordance with the Vogtle Technical Specification. Until the applicant defines the withdrawal schedules, the staff will not be able to complete its review of this part of the FSAR.

Until the applicant provides the information previously discussed, the staff cannot conclude there is reasonable assurance that the surveillance program will monitor the change in the beltline region material properties to the extent required for establishing pressure-temperature limits and to preserve the integrity of the vessel. This program must generate sufficient information to permit the determination of conditions under which the reactor vessel will be operated with an adequate margin of safety against rapidly propagating fracture throughout its service lifetime.

Conclusions for Compliance With Appendices G and H, 10 CFR Part 50

Appendix G, "Protection Against Non-Ductile Failures," Section III of the ASME Code, was used, together with the fracture toughness test results required by Appendices G and H, 10 CFR Part 50, to calculate the pressure-temperature limitations for the Vogtle reactor vessels.

The fracture toughness tests required by the ASME Code and by Appendix G of 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. The use of Appendix G, Section III of the ASME Code, as a guide in establishing 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 requirements of GDC 31.

The materials surveillance program required by Appendix H of 10 CFR Part 50 will provide information on the effects of irradiation on material properties so that changes in the fracture toughness of the material in the Vogtle reactor vessels beltline can be properly assessed, and adequate safety margins against the possibility of vessel failure can be provided.

Compliance with Appendix H of 10 CFR Part 50 and ASTM E185 ensures that the surveillance program will be capable of monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and satisfies the requirements of GDC 32.

There is reasonable assurance that the surveillance program will monitor the change in the beltline region material properties to the extent required for establishing pressure-temperature limits and to preserve the integrity of the vessel. The surveillance program will generate sufficient information to permit the determination of conditions under which the reactor vessel will be operated with an adequate margin of safety against rapidly propagating fracture throughout its service lifetime.

5.3.1.2 Fabrication

Ordinary processes were used for the manufacture, fabrication, welding, and nondestructive examinations of the reactor vessel and its appurtenances. Nondestructive examinations in addition to Code requirements were also performed. Since the applicant has certified that the requirements of Section III of the ASME Code have been complied with, the processes and examinations used are considered acceptable. Compliance with these Code provisions satisfies the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

When welding components of ferritic steels, Code controls are supplemented by conformance with the recommendations of regulatory guides as follows:

(1) The controls imposed on welding preheat temperatures are in conformance to the extent practical with the recommendations of RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel." The staff reviewed and found acceptable the alternative approaches taken by the applicant (see Section 5.2.3). These controls provide reasonable assurance that cracking of components made from low-alloy steels did not occur during fabrication and minimize the potential for subsequent cracking. These controls also satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

(2) RG 1.34, "Control of Electroslag Weld Properties," is not applicable because this process is not used in reactor vessel fabrication.

(3) The controls imposed during weld cladding of ferritic steel components are in conformance with the recommendations of RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." These controls provide assurance that underclad cracking did not occur during weld cladding of the reactor vessel.

When welding components of austenitic stainless steels, Code controls are supplemented by conformance with the recommendations of regulatory guides as follows:

(1) The controls imposed on delta ferrite in austenitic stainless steel welds satisfy most of the recommendations of RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." The staff reviewed and finds acceptable the alternate approaches taken by the applicant (see Section 4.5.1).

(2) RG 1.34, "Control of Electroslag Weld Properties" is not applicable because this process is not used in fabricating reactor vessels.

The controls (during all stages of welding) to avoid contamination and sensitization that could cause stress corrosion cracking in austenitic stainless steels conform with the recommendations of regulatory guides as follows:

(1) The controls to avoid contamination and excessive sensitization of austenitic stainless steel satisfy, to the extent practical the recommendations of RG 1.44, "Control of the Use of Sensitized Stainless Steel." The staff reviewed and finds acceptable the alternative approaches taken by the applicant (see Section 4.5.1). The controls used provide assurance that welded components were not contaminated or excessively sensitized before and during the welding process. These controls satisfy the quality standards requirement of GDC 1, GDC 30, and 10 CFR 50.55a, and the GDC 4 requirement relative to material compatibility.

(2) The controls regarding onsite cleaning and cleanliness controls of austenitic stainless steel are in conformance with the recommendations of RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" or the applicant's alternative approaches are acceptable to the staff as discussed in Section 4.5.1. These controls provide assurance that austenitic stainless steel components were properly cleaned onsite and satisfy Appendix B of 10 CFR Part 50 regarding controls for onsite cleaning of materials and components.

Integrity of the reactor vessel studs and fasteners is assured by conformance to the extent practical with the recommendations of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." The applicant's alternative approaches of (1) using a modified SA-540, Grade B-24 for closure stud material which is allowed by Code Case 1605 and (2) not specifying a maximum ultimate tensile strength and relying on the bolting material's low alloy steel chemistry, heat treatment, and toughness requirements to control ultimate tensile strength are acceptable to the staff. Compliance with these recommendations and the applicant's alternative approaches satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a; the prevention of fracture of the RCPB requirement of GDC 31; and the requirements of Appendix G, 10 CFR Part 50, as detailed in the provisions of the ASME Code, Sections II and III.

5.3.2 Pressure-Temperature Limits

The staff has reviewed the applicant's pressure-temperature limits for operating the reactor vessel. The acceptance criteria and list of references which are the basis for this evaluation are set forth in SRP Section 5.3.2 of NUREG-0800, Revision 1, dated July 1981. A discussion of this review follows.

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of 10 CFR Part 50, describe those conditions that require pressure-temperature limits and specify 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 the ASME Code, Section III, Appendix G, "Protection Against Non-Ductile Failures." Appendix G of 10 CFR Part 50 requires additional safety margins for the closure flange region materials and beltline materials whenever the reactor core is critical, except for low-level physics tests.

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

(1) preservice hydrostatic tests

(2) inservice leak and hydrostatic tests

(3) heatup and cooldown operations

(4) core operation.

Appendices G and H of 10 CFR Part 50, require the applicant to predict the amount of increase in reference temperature, RT_{NDT}, that will result from neutron irradiation. The shift in RT_{NDT} owing to neutron irradiation is then added to the initial RT_{NDT} to establish the adjusted reference temperature.

The applicant did not include pressure-temperature curves for the reactor vessel closure flange areas in its FSAR submittal or its response to a staff request for additional information. The applicant must revise its pressure-temperature curves to include the safety margins specified in Appendix G of 10 CFR Part 50 for the flange closure region. The staff cannot complete its evaluation of the pressure-temperature limits for Vogtle until the applicant revises the pressure-temperature curves to include the closure flange regions.

The pressure-temperature limits to be imposed on the reactor coolant system for all operating and testing conditions must have adequate safety margins against non-ductile or rapidly propagating failure, and must be in conformance with established criteria, codes, and standards. The use of operating limits based on these criteria, as defined by applicable regulations, codes, and standards, will provide reasonable assurance that non-ductile or rapidly propagating failure will not occur, and will constitute an acceptable basis for satisfying the applicable requirements of GDC 31.

5.3.3 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. In this section, the staff has reviewed the fracture toughness of ferritic reactor vessel and reactor coolant pressure boundary materials, the pressure-temperature limits for operation of the reactor vessel, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references which are the basis for the evaluation are set forth in paragraphs II.2, II.6, and II.7 (Appendices G and H of 10 CFR Part 50) of SRP Section 5.3.3 (NUREG 0800, Rev. 1, dated July 1981). A discussion of this review follows.

The staff has reviewed the information in each area to ensure that it is complete and that no inconsistencies exist that would reduce the certainty of vessel integrity. The areas reviewed (and the SER section in which the review can be found) are:

(1) design (5.3.1)

(2) materials of construction (5.3.1)

(3) fabrication methods (5.3.1)

(4) operating conditions (5.3.2).

The staff has reviewed the above factors contributing to the structural integrity of the reactor vessel and concludes that the applicant has complied with Appendices G and H of 10 CFR Part 50, except for the following items:

(1) The applicant has not reported the Charpy V-notch energy and mils lateral expansion data versus temperature for each reactor vessel beltline material.

(2) The applicant has not committed to follow RG 1.99 to predict neutron irradiation damage.

(3) The applicant has not reported the withdrawal schedule for the surveillance specimens.

(4) The applicant has not supplied pressure-temperature curves for the reactor vessel pressure closure flange regions.

Until the applicant supplies the information needed to complete the staff's evaluation of compliance with Appendices G and H of 10 CFR Part 50, the staff cannot complete its evaluation of the structural integrity of the Vogtle reactor vessels.

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps

5.4.1.1 Pump Flywheel Integrity

The safety objective of this review is to ensure that the integrity of the primary reactor coolant pump flywheel is maintained to prevent failure at normal operating speeds and speeds that might be reached under accident conditions, and thus preclude the generation of missiles.

The basis for review is outlined in SRP Section 5.4.1.1 and RG 1.14, which describe and recommend a method acceptable to the NRC staff in implementing GDC 4, "Environmental and Missile Design Bases," of Appendix A of 10 CFR Part 50, with regard to minimizing the potential for failure of flywheels of the reactor coolant pump.

Flywheels are fabricated from SA-533, Grade B, Class I steel and consist of two thick plates bolted together. The material is vacuum melted with a degassing process. The materials, as well as finished flywheels, are subjected to 100% volumetric ultrasonic inspection using procedures and acceptance standards specified in Section III of the ASME Code.

The nil-ductility transition temperature (NDTT) of the flywheel material is no higher than 10°F. The Charpy V-notch energy level is at least 50 ft-lb in the WP (transverse) orientation at 70°F. Hence, the RT_{NDT} of 10°F can be assumed.

The calculated stresses at the operating speed, that result from centrifugal forces, and the interference fit on the shaft are within RG 1.14 limits. The pump runs at 1,190 rpm and may operate briefly at overspeed of 109% during the loss of offsite electrical power. The design speed is 125% of the operating speed. The flywheels are also tested at 125% of the maximum synchronous speed of the motor. The combined stresses at the design overspeed, from interference fit and centrifugal forces, are within the Regulatory Guide limit.

The flywheels can be inspected by removing the cover. Hence, any crack that developed can be noticed. The critical crack length at the keyways, where the stress concentration is high, is about 6 in. at the design overspeed.

The staff has reviewed the material, fabrication, design, and inspection aspects of the pump flywheels for compliance with RG 1.14. The staff has concluded that the structural integrity of the flywheels is adequate to withstand the forces imposed by overspeed transients without the loss of function: periodic inspection will ensure that the integrity is maintained.

5.4.2 Steam Generators

5.4.2.1 Steam Generator Materials

The staff concludes that the steam generator materials specified are acceptable and meet the requirements of GDC 1, 14, 15, and 31, and Appendix B to 10 CFR Part 50. This conclusion is based on the following:

(1) The applicant has satisfied the requirements of GDC 1 with respect to codes and standards by assuring that the materials selected for use in Class 1 and Class 2 components were fabricated and inspected in conformance with codes, standards, and specifications acceptable to the staff. Welding qualification, fabrication, and inspection during manufacture and assembly of the steam generators were done in conformance with the requirements of Section III and IX of the ASME Code.

(2) The requirements of GDC 14 and 15 have been satisfied to ensure that the reactor coolant boundary and associated auxiliary systems have been designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapid failure and of gross rupture, during normal operation and anticipated operational occurrences.

The primary side of the steam generator is designed and fabricated to comply with ASME Class 1 criteria as required by the staff. The secondary side pressure boundary parts of the steam generator are designed, manufactured, and tested to ASME Code Class 2.

The crevice between the tubesheet and the inserted tube is minimal because the tube was expanded to the full depth of insertion of the tube in the tubesheet. The tube expansion and subsequent positive contact pressure between the tube and the tubesheet preclude a buildup of impurities from forming in the crevice region and reduce the probability of crevice boiling. The tubes are seal welded to the tubesheet to ensure maintenance of separate paths between the primary and secondary water flow.

The tube support plates will be manufactured from ferritic stainless material which has been shown in laboratory tests to be corrosion resistant to the operating environment. The tube support plates will be designed and manufactured with broached holes rather than drilled holes. The broached-hole design promotes high velocity flow along the tube, sweeping impurities away from the support plate locations.

(3) The requirements of GDC 31 have been satisfied with respect to the fracture toughness of the ferritic materials since the pressure boundary materials of ASME Class 1 components of the steam generators will comply with the fracture toughness requirements and tests of Subarticle NB-2300 of Section III of the Code.

The materials of the ASME Class 2 components of the steam generators will comply with the fracture toughness requirements of Subarticle NC-2300 of Section III of the Code.

(4) The requirements of Appendix B of 10 CFR Part 50 have been satisfied, since the onsite cleaning and cleanliness controls during fabrication conform to the recommendations of RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," or the applicant's alternative approaches are acceptable to the staff as discussed in Section 4.5.1. The controls placed on the secondary coolant chemistry are in discussed in Section 10.3.5.

Reasonable assurance of the satisfactory performance of the steam generator tubing and other generator materials is provided by the design provisions and the manufacturing requirements of the ASME Code and rigorous secondary water monitoring and control. The controls described above combined with conformance with applicable codes, standards, staff positions, and regulatory guides constitute an acceptable basis for satisfying in part the requirements of GDC 1, 14, 15, and 31, and Appendix B, 10 CFR Part 50.

5.4.2.2 Steam Generator Tube Inservice Inspection

This section was prepared with the technical assistance of DOE contractors from the Idaho National Engineering Laboratory.

5.4.2.2.1 Compliance With the Standard Review Plans

The July 1981 Edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800) includes Section 5.4.2.2 "Steam Generator Tube Inservice Inspection." The Vogtle FSAR through Amendment 9, dated August 1984 and the "Preservice Inspection Program," Revision 0, dated April 19, 1984, were reviewed in accordance with this section of the Standard Review Plan (SRP). The results of this review are summarized below.

The SRP Acceptance Criteria recommend that the applicant perform examinations based on RG 1.83 and the applicable Standard Technical Specifications. The applicant has not committed to examine the steam generator tubes in accordance with the recommendations of both RG 1.83 and the technical requirements of NUREG-0452, Revision 2.

5.4.2.2.2 Evaluation of the Inspection Program

GDC 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A of 10 CFR Part 50, requires, in part, that components of the reactor coolant pressure boundary be designed to permit periodic examination and testing of important areas and features to assess their structural and leak-tight integrity. The steam generators at Vogtle have been designed to meet the ASME Boiler and Pressure Vessel Code requirements for Class 1 and 2 components. Provisions also have been made to permit inservice inspection of the Class 1 and 2 components, including individual steam generator tubes. The design aspects that provide access for examination and the proposed inspection program must comply with the requirements of Section XI of the ASME Code, follow the recommendations of RG 1.83, Revision 1, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," and NUREG-0452, Revision 2, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," with respect to the examination methods to be used, provisions for a baseline examination, selection and sampling of tubes, inspection intervals, and actions to be taken in the event that defects are identified.

The applicant has not committed to perform a preservice examination of the full length of each tube in each steam generator using eddy-current techniques to establish the baseline condition of the tubing based on the Standard Technical Specification, NUREG-0452, Revision 2. The examination is to be performed before initial power operation using the equipment and techniques expected to be used during subsequent inservice examinations. The staff considers this an open item.

The Vogtle steam generators are the latest Westinghouse models, designated Model F, which incorporate multiple features to minimize operating problems associated with crevice corrosion, stress corrosion cracking, flow-induced vibration, and denting. These features include:

(1) Type 405 ferritic stainless steel quatrefoil tube support plate to eliminate denting

(2) thermally treated Inconel 600 tubing and stress relief of the innermost eight rows of the tube bundle to reduce the potential for stress corrosion cracking

(3) expansion of the tubes to the full depth of the tubesheet to eliminate crevices and potential for crevice corrosion

(4) a flow baffle plate above the tubesheet to direct lateral flow across the tubesheet surface and thus minimize the number of tubes exposed to sludge and potential for corrosion attack

(5) an improved blowdown system to increase blowdown capacity to minimize sludge buildup

Vibration-induced wear which has been experienced in the preheater section of Models D and E Westinghouse steam generators, prior to modifications, will not be a problem in Model F steam generators at Vogtle since this model does not have a preheater section.

The staff considers the examination of the steam generators an open issue, subject to the applicant committing to the applicable provisions of the Standard Technical Specification. The staff will complete the review and report the results in a supplement to the SER after the actual Technical Specifications are established.

5.4.2.2.3 Conclusions

Following resolution of the open item regarding the preservice examination of each steam generator tube, the staff will be able to conclude the following for Vogtle.

Conformance with RG 1.83, NUREG-0452, and the inspection requirements of Section XI of the ASME Code consitutes an acceptable basis for meeting, in part, the requirements of GDC 32.

5.4.3 Deleted*

5.4.~~3~~4 Deleted*

5.4.~~3~~5 Deleted*

5.4.6 Reactor Core Isolation Cooling System

This SRP section is not applicable to a PWR.

5.4.7 Residual Heat Removal System

The design of the residual heat removal system (RHRS) for Vogtle has been reviewed in accordance with SRP Section 5.4.7 and BTP RSB 5-1 of NUREG-0800.

*Deleted from the July 1981 edition of the Standard Review Plan (NUREG-0800).

Conformance with the acceptance criteria formed the basis for the staff's conclusion that the design of the RHRS is acceptable.

The RHRS has two independent cooling trains, which are designed for a pressure of 600 psig and a temperature of 400°F. Each train has a 3000-gpm pump and a heat exchanger that is designed to transfer 32.8 million Btu/hr to the component cooling water. The pumps, heat exchangers, and isolation and control valves are all located inside of containment. Each train of the RHRS is powered by a separate vital bus.

The RHRS operates in the following modes:

(1) <u>Emergency Core Cooling System (ECCS), Injection Mode</u>

Functions in conjunction with the high head portion of the ECCS to provide injection of borated water from the refueling water storage tank (RWST) into the RCS cold legs during the injection phase following a loss-of-coolant accident (LOCA).

(2) <u>Emergency Core Cooling System, Recirculation Mode</u>

Provides long-term cooling during the recirculation phase following a LOCA. This function is accomplished by aligning the RHRS to take fluid from the containment ~~p~~sump, cool it by circulation through the RHR heat exchangers, and supply it to the cold legs of the RCS. During this mode of operation, the RHRS discharge flow may be aligned to the suction of the charging pumps and safety injection pumps to provide water supplies for high head recirculation. Flow paths are also available for hot-leg injection during the long-term recirculation mode to prevent boron precipitation in the reactor core.

(3) <u>Refueling</u>

Used to transfer refueling water between the refueling cavity and the refueling water storage tank at the beginning and end of the refueling operations. During refueling, the RHRS is maintained in service to provide a heat-removal function to accommodate the heat load.

(4) Cold Shutdown

Removes fission product decay heat to maintain cold shutdown conditions.

(5) Startup

Connected to the chemical and volume control system (CVCS) via the low-pressure letdown line to control reactor coolant pressure.

The RHRS is designed to remove heat from the RCS after the system pressure and temperature have been reduced to approximately 400 psig and 350°F, respectively, by the steam and power conversion system. Under normal conditions, with two trains operating, it will take about 20 hr to get the reactor coolant temperature down to 140°F. If there is only one train operating it will take about 15 hr to get the reactor coolant temperature down to 212°F and another 10 hr to further reduce the temperature to 200°F.

Because of thermal stress considerations, the RCS cooldown is to be limited to 100°F/hr by Technical Specifications. The operator will be aided by both plant procedures and alarms on the safety parameter display systems so as not to exceed this limit.

5.4.7.1 Functional Requirements

BTP RSB 5-1 stipulates that the design of a plant shall be such that it can be taken to cold shutdown by using only safety-grade systems and that these systems shall satisfy GDC 1 through 5. In this regard Section 5.4.7.2.5 of the FSAR states that the entire RHRS for Vogtle is designed as Safety Class 2 with the exception of the suction isolation valves which form a part of the RCS pressure boundary and are designed as Safety Class 1. Compliance with GDC 1-5 criteria is as follows:

o GDC 1, quality assurance aspects of safety-grade systems, is evaluated in SER Section 17.1.

- o GDC 2, design bases for safety-grade system, is evaluated in SER Section 3.2.

- o GDC 3, fire protection of safety-grade systems, is evaluated in SER Section 9.5.1.

- o GDC 4, environmental and missile protection design for safety-grade systems, is evaluated in SER Sections 3.11 and 3.5.

- o GDC 5 is complied with because these RHRS's are not shared between units.

To comply with the redundancy criteria of GDC 34 the RHRS has two independent trains. Leak detection for the RHRS is discussed in Section 5.2.5 of this report. Isolation valve and power supply redundancy are discussed under separate topics in this section. The staff has reviewed the description of the RHRS and the piping and instrumentation diagrams to verify that the system can be operated with or without offsite power and assuming a single failure. The two RHR pumps are connected to separate buses that can be powered by separate diesel generators in the event of a loss of offsite power. Thus a single failure, such as that of a pump, valve, or heat exchanger, will still allow the operation of one train.

GDC 19 states that a control room shall be provided from which actions can be taken to maintain the plant in a safe condition under accident conditions, including loss-of-coolant accidents. SRP Section 5.4.7 stipulates that the control of the RHRS be such that the cooldown function can be performed from the control room assuming a single failure of any active component, with only either onsite or offsite electric power available. Any operation required outside of the control room is to be justified by the applicant.

The applicant states in FSAR Section 5.4.7.2.7 that the RHRS is designed to be fully operable from the control room for normal operation and in Section 5.4.7.2.3.5 that the RCS can be taken from no-load temperature and pressure to cold conditions using only safety-related systems with only onsite or offsite power available assuming the most-limiting single failure. It is also stated in Section 5.4.7.2.3 that as a backup to the isolation valves on the ECCS accumulators there are redundant, Class 1E, solenoid-operated valves to ensure that any accumulator may be vented, should it fail to be isolated from the RCS.

The auxiliary feedwater system, along with the steam generator safety and power-operated relief valves, provides a completely separate, independent, and diverse means of performing the safety function of removing residual heat. Decay heat removal is normally performed by the RHRS when RCS temperature is less than 350°F. The auxiliary feedwater system is capable of performing this function for an extended period of time following plant shutdown using the condensate storage tanks as a source of water. The applicant, in response to a staff concern, stated that the steam generator power-operated relief valves, their operators and power supplies are designed to safety-grade standards.

In FSAR Section 5.4.7.1 the applicant states that the RHRS is designed to reduce the temperature of the reactor coolant from 350°F to 140°F in approximately 20 hr. With only one train in service it will take approximately 15 hr to go from 350°F to 212°F. The cooldown time of 15 hr with one RHRS train is acceptable. With the stated 4-hr time for cooldown from standby to RHRS conditions, Vogtle can be brought to cold shutdown within a reasonable period of time with or without offsite power.

With the exception of the suitability of the leak detection, which is evaluated in Section 5.2.5, the RHRS has been reviewed and found to meet the functional requirements of BTP RSB 5-1.

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5.4.7.2 RHRS Isolation Requirements

THe RHRS valving arrangement is designed to provide adequate protection to the RHRS from overpressurization when the reactor coolant system is at high pressure.

There are two separate and redundant motor-operated isolation valves (MOVs) between each of the two RHRS pump suction lines and the RCS hot legs. These valves are separately, diversely, and independently interlocked to keep the valve from opening until the RCS pressure falls below 425 psig. If the valves are open, they are separately, diversely, and independently interlocked to close when the RCS pressure rises above 750 psig. Each one of the four RHRS suction MOVs is aligned to a separate motor control center. The four MOVs are powered from separate power trains. Thus a single failure will not prevent the isolation of the RHRS nor will it cause isolation of both trains of RHR. In addition, there is direct position indication of these valves in the control room.

Water, trapped between the two suction-side isolation valves at a low temperature, that is heated and thus expands will be relieved through a 3/4-in. line that extends from a point between the valves to the RCS.

There are two check valves and a normally open MOV on each RHR discharge line. The two check valves protect the system from the RCS pressure during normal plant operation. The applicant has provided design features to permit leak testing of the check valves with the RCS pressurized to fulfill the staff requirements for high/low-pressure isolation with two check valves. The check valves and the MOV are considered part of the ECCS.

The staff finds that the design of the RHRS isolation system satisfies the criteria of BTP RSB 5-1 and is acceptable.

5.4.7.3 RHR Pressure Relief Requirements

Overpressure protection of the RHRS is provided by four relief valves, one on each of the suction and discharge lines. Each suction line relief valve has a capacity of 900 gpm at 450 psig, which is sufficient to discharge the flow from a safety injection pump at the relief valve setpoint. The fluid discharged by these relief valves is collected in the pressurizer relief tank. Compliance of these valves with NUREG-0737 Item II.D.1 is evaluated in Section 3.9.3 of this report. Each discharge line from the RHRS to the RCS is protected from overpressurization by a pressure-relief valve in the ECCS. These valves, which have a relief flow capacity of 20 gpm at a set pressure of 600 psig, are to relieve the maximum possible back-leakage through the valves separating the RHRS from the RCS. The fluid discharged by these valves is collected in the recycle holdup tank of the boron recycle system.

In response to a staff request to discuss procedures available to the operator for responding to the lifting of an RHR relief valve, the applicant stated in FSAR Amendment 6 that specific procedures will be developed to diagnose a lifted relief valve and isolate the affected train. If the valve involved were to be the suction side relief valve, the operator would be alerted by level, temperature, and pressure indication on the pressurizer relief tank as well as pressurizer level. If the valve involved were to be the discharge side relief valve, the operator would be alerted by steadily increasing level in the recycle holdup tank and/or high pressure in the discharge line to the tank. This response is acceptable.

5.4.7.4 RHR Pump Protection

Each of the two RHR pumps has a miniflow bypass line to prevent overheating in the event of inadequate pump flow. A valve located in each miniflow line is regulated by a signal from the flow transmitters located in each pump discharge header. The control valve opens when the RHR pump discharge flow is less than approximately 500 gpm and closes when the flow exceeds approximately 1000 gpm.

A pressure sensor in each pump discharge header provides a signal for an indicator in the control room. A high-pressure alarm is also actuated by the pressure sensor.

In response to a staff concern regarding the mispositioning of the miniflow line valves, the applicant has stated in FSAR Amendment 6 that the valve position control switch has a spring return to automatic from both the open and closed postions. This is to prevent inadvertent operator action. Valve misalignment can be detected through use of the control room indications of valve position, RHR pump outlet temperature and pump discharge flow. Correct functioning of the valve is to be verified during RHR in-service testing and flow measurements in the miniflow lines are part of the preoperational testing program. The staff finds that these precautions and indications are acceptable.

The miniflow line valves are fast operating gate valves that will open or close in 10 sec or less. This precludes RHR pump damage from operator error that could result in closure of both the residual heat exchanger outlet and bypass flow control valves, which was another staff concern.

Cooling water is supplied to the RHR pump mechanical seal by the component cooling water (CCW) system. With regard to a staff question concerning loss of cooling to the seal, the applicant has stated that there are control room alarms to indicate low CCW pump discharge pressure, low flow from the RHR pump seal coolers, and high temperature in the RHR pump seal coolers. Furthermore, the applicant has stated that qualification testing has shown that the seals can operate for 96 hr at 300°F and for 24 hr at 350°F and 400 psig without any signs of detrimental effects to the seals. Considering (1) the capacity of each train to remove heat, (2) as noted earlier, the capability of the seals to sustain a loss of cooling over an extended period of time at high temperatures, (3) that degradation is temperature dependent, and (4) that there exists indication to alert the operator, loss of CCW to the seals is not a concern.

The staff asked for additional information concerning the possibility of air binding of the RHR pumps during and following periods of (1) maintenance when the RCS has been partially drained, (2) improper RCS level control, (3) partial loss of primary inventory or (4) operating the RHRS at an inadequate net pump suction head. In response, the applicant has stated that if the steam generator tubes need to be drained, the RCS inventory can be reduced without uncovering the inlets to the RHRS and that RCS level would be continuously monitored. Inventory makeup could, if need be, be performed by a charging pump. If the inlets were to be uncovered, the effect of air entrainment would be minimized by the location of the RHR pumps which provide positive head on the pump inlet and procedures calling for the minimum RHR flow necessary for decay heat removal. In the event that the pumps do become air bound, the suction line and the pump can be filled and vented by using control room capabilities to open the vent valves and the refueling water storage tank supply valves.

In response to a staff question, the applicant has stated that plant design precludes waterhammer from air entrapped in the RHR during startup. The design factors include suction piping that slope downward toward the pump to allow self-venting and vents at piping high points. This is to prevent vortexing and air entrainment. Procedures will ensure that the RHRS is completely vented following maintenance operations.

5.4.7.5 Tests, Operational Procedures, and Support Systems

The plant preoperational and startup test program provides for demonstrating the operation of the residual heat removal system in conformance with RG 1.68, as specified in SRP Section 5.4.7, Paragraph III.12.

The applicant was asked to demonstrate the adequacy of the mixing of borated water added to the RCS under natural circulation and the ability to cool down the Vogtle units with natural circulation. The applicant responded by referencing a program, described in a July 7, 1981, letter from Westinghouse to H. R. Denton. Although this program was approved by the staff, it was not approved as a means to satisfy BTP RSB 5-1 Position E. This program described a post-TMI-2 startup test procedure for Sequoyah Unit 1. Westinghouse has confirmed in subsequent discussion that the program was not implemented with the intent of conforming to BTP RSB 5-1 and, in fact, does not meet this ~~B~~ranch ~~T~~echnical ~~P~~osition. This item will remain open until the applicant commits to conduct boration and cooldown tests under natural circulation conditions that meet the criteria of BTP RSB 5-1 or to reference tests at a similar facility that have been accepted as satisfying BTP RSB 5-1. This commitment should state that BTP RSB 5-1 Position E will be satisfied by either alternative, before licensing. Test procedures and results obtained by the applicant, in order to show compliance, should be submitted to the NRC for review and approval.

The staff has reviewed the component cooling water system to ensure that sufficient cooling capability is available to the RHRS heat exchangers. The acceptability of this cooling capacity and its conformance to GDC 44, 45, and 46 are discussed in Section 9.2.2 of this report.

The applicant states that the RHRS is housed in a structure that is designed to withstand tornadoes, floods, and seismic phenomena, and there are no motor-operated valves in the RHRS which are subject to flooding after a LOCA or a steamline break. This area is addressed further in Section 3 of this report.

Leakages resulting from a passive failure of the RHRS piping will be collected by the floor drain system. THe operator will be alerted to such leaks by the control room alarms for the floor drain system, area radiation monitors, high-temperature room alarms and flood-retaining room system alarms. The faulted loop can be completely isolated with no impact on plant safety because the redundant loop would remain available. Single failures, active or passive, in the redundant loop have not been postulated. This is in accordance with BTP ASB 3-1.

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The staff questioned the applicant about the consequences of failing an RHR suction line valve closed during shutdown cooling. Consideration was to be given to either having the vessel head bolted or removed. In response, the applicant has stated that the valve closure may be reversible (closure caused by operator error), or not reversible. In any event, the operator would be instructed to start the non-operating RHR pump in the other train. Failing that (under some conditions only one RHR loop is required to be operable), the operator needs to find an alternate heat-removal path. With the head bolted on, the steam generators are available to remove decay heat. With the head off and the refueling canal full, boiling will begin in approximately 2.5 hr. A degraded water level due to boiloff can be restored by any of the water sources available to the operator (such as an accumulator). Considering the amount of decay heat and the water sources available, the operator has sufficient time to get at least one RHR loop into operation.

The RHRS capability to withstand pipe whip inside containment as required by GDC 4 and RG 1.46 is discussed in Section 3.6 of this report. Protection against piping failures outside of containment in accordance with GDC 4 is discussed in Section 3.6 of this report.

All RHR lines, including instrument lines, have containment isolation features; their satisfaction of the requirements of GDC 56, 57, and the criteria of RG 1.11 are discussed in Section 6.2.4 of this report.

The applicant, following SRP Section 5.4.7, Paragraph II.D.1, has demonstrated that suitable plant systems and procedures are available to place the plant in a cold shutdown condition with only offsite or onsite power available within a reasonable period of time following shutdown, assuming the most limiting single failure.

Paragraph 5.4.7.2.4 of the FSAR states that residual heat removal system (RHRS) suction side reliefs have a set pressure of 450 psig. It also states that the RHRS is not isolated from the reactor coolant system until a pressurizer bubble is formed and before increasing reactor coolant system pressure to 600 psig and that the isolation valves receive an automatic close signal at 750 psig. The applicant was asked to explain how the RHRS could be kept in service above 450 psig. This item will remain open until an acceptable response is received.

5.4.7.6 Conclusions

The RHR function is accomplished in two phases: the initial cooldown phase and the RHRS operation phase. In the event of loss of offsite power, the initial phase of cooldown is accomplished by use of the auxiliary feedwater system and the atmospheric dump valves. This equipment is used to reduce the reactor coolant system temperature and pressure to values that permit operation of the RHRS. The review of the initial cooldown phase is discussed in Section 10.3 of this report. The review of the RHRS operational phase is discussed below.

The RHRS removes core decay heat and provides long-term cooling following the initial phase of reactor cooldown. The scope of review of the RHRS included piping and instrumentation diagrams, failure modes and effects analysis, and design performance specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the RHRS and its analysis of the adequacy of conformance to these criteria and bases.

The staff concludes that the design of the RHRS is acceptable and meets the relevant criteria of GDC 2, 5, 19, and 34. This conclusion is based on the following:

(1) As stated in SER Section 3.2, the applicant has satisfied GDC 2 with respect to Position C.2 of RG 1.29 concerning the seismic design of

systems, structures, and components whose failure could cause an unacceptable reduction in the capability of the RHRS.

(2) The applicant has satisfied the criteria of GDC 5 with respect to sharing of structures, systems, and components by stating that the RHRS is not shared with another unit, i.e., each unit of ~~the~~ Vogtle ~~plant~~ has a separate RHRS.

(3) The applicant has satisfied GDC 19 with respect to the main control room requirements for normal operations and shutdown and GDC 34 which specifies requirements for the residual heat removal system by meeting the regulatory position in BTP RSB 5-1, except that the applicant is to demonstrate compliance with Position E of this Branch Technical Position.

The staff's resolution of NUREG-0660 Item II.E.3.2, as it relates to systems capability and reliability of shutdown heat removal systems under various transients, and Item II.E.3.3, as it relates to a coordinated study of shutdown heat removal requirements, will be contained in the resolution of Unresolved Safety Issue (USI)-45.

TMI-2 Task Action Plan Item III.D.1.1 of NUREG-0737 as it relate to primary coolant sources outside of containment is addressed in Section 15.9 of this report.

5.4.8 Reactor Water Cleanup System

This item does not apply to a PWR. ~~Later~~

5.4.9 Deleted*

5.4.10 Deleted*

5.4.11 Pressurizer Relief Tank

Later

*Deleted from the July 1981 edition of the Standard Review Plan (NUREG-0800).

5.4.12 Reactor Coolant System High Point Vents

10 CFR 50.44(c)(3)(iii) requires all light-water reactors to have high-point vents on the reactor coolant system and on the reactor vessel head. This requirement is supplemented by guidance in SRP Section 5.4.12 and NUREG-0737 Item II.B.1. The applicant has provided information on the RCS high-point vent system in FSAR Section 5.4.15 and in response to a staff request for further information which is included in Amendment 6 to the FSAR.

The Vogtle reactor vessel head vent system (RVHVS) consists of a single flow path with redundant isolation valves. The system is designed to mitigate a possible condition of inadequate core cooling or impaired natural circulation resulting from the accumulation of noncondensable gases in the RCS.

In FSAR Section 5.4.15 the applicant states the following about the RVHVS:

(1) The active portion of the system consists of four 1-in. open/close solenoid-operated isolation valves connected to a 1-in. vent pipe located near the center of the reactor vessel head.

(2) All piping and equipment from the vessel vent, up to and including the second isolation valve in each flow path, are designed and fabricated in accordance with ASME Section III, Class 1 requirements.

(3) The piping and equipment in the flow paths from the second isolation valves to the modulating valves and from the isolation valves to the excess letdown heat exchanger are designed and fabricated in accordance with ASME Section III, Class 2, requirements. The remainder of the piping is seismic Category I, non-nuclear safety.

(4) The isolation valves in one flow path are powered by one vital power supply and the valves in the second flow path are powered by a second vital power supply. The isolation valves are fail closed, normally closed valves.

(5) The system is operated from the control room or the shutdown panels. The isolation valves have stem position switches. The position indication from each valve is monitored in the control room by status lights.

The applicant states that a break of the RVHVS would result in a small LOCA of not greater than 1-in. diameter and would behave similarly to a hot-leg break. This event is bounded by the spectrum of pipe breaks considered in Section 15 of this report.

The applicant has evaluated the possibility of inadvertent actuation of the reactor vessel head vent system and states that no single active failure will preclude reactor vessel head venting or venting isolation. The staff has reviewed this design and concurs with this conclusion. However, the facility is to employ a Target Rock valve system which may be susceptible to common mode failure. The applicant is required to evaluate this susceptibility and implement any necessary corrective action. This item will remain open until the applicant confirms the results of this evaluation.

The pressurizer may be vented by opening one or both of the power-operated relief valves. The steam generator U-tubes are vented by fill and vent procedures during normal operations and by emergency procedures during off-normal conditions.

All remotely operated valves in the RVHVS can be tested during plant operation by means of the valve status lights located in the control room. A system flow test can be performed at low RCS pressures by observing the system flow indicators or by observing an increase in the level of the pressurizer relief tank.

The applicant has satisfied the requirements of 10 CFR 50.44(c)(3)(iii) by

(1) providing vent paths for the vessel head and pressurizer

(2) providing remote operation from the control room

(3) providing environmentally and seismically qualified components and power sources for the vent systems

(4) taking measures to provide a degree of redundancy to assure venting operation and minimize inadvertent or irreversible operation

The applicant has committed to including the RVHVS in the Vogtle inservice testing and inspection program.

The reactor systems aspects of the reactor coolant system vents have been reviewed, and the staff concludes that they satisfy the requirements in NUREG-0737 and are, therefore, acceptable.

Before the vent system is considered fully operational the applicant must

(1) Complete operating procedures based on staff approved operating guidelines.

(2) Adopt operability requirements for the vent system in the plant Technical Specifications. (This is a confirmatory item.)

(3) Evaluate the susceptibility of the system to common mode failures and to take any necessary corrective action. This evaluation should show that "the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident," which is the position of TMI-2 Task Action Plan Item II.B.1. The staff recommends that the RVHVS be equipped with a flow restrictor so as to limit the flow, from a pipe break or inadvertent actuation of the system, to the point where it can be fully compensated for by the normal makeup system. Regarding this item, the staff will report the results of the applicant's evaluation in a future supplement to this ~~SER~~. report.

6 ENGINEERED SAFETY FEATURES

6.1 Materials

6.1.1 Engineered Safety Features Materials

The staff concludes that the engineered safety features materials specified are acceptable and meet the requirements of GDC 1, 4, 14, 31, 35, and 41 of Appendix A of 10 CFR Part 50; Appendix B of 10 CFR Part 50; and 10 CFR 50.55a. This conclusion is based on the following:

(1) GDC 1, 14, and 31, and 10 CFR 50.55a have been satisfied with respect to assuring an extremely low probability of leakage, of rapidly propagating failure, and of gross rupture. The materials selected for engineered safety features satisfy Appendix I of Section III and Parts A, B, and C of Section II of the ASME Code, or equivalent American Society of Testing and Materials specifications. The applicant has complied with the staff position that the yield strength of cold-worked stainless steels shall be less than 90,000 psi.

The applicable code for the construction of Vogtle allowed waiving of impact testing of Class 2 and 3 systems. On the basis of the results of impact testing (by other applicants) of the same specification steels, and correlations of the metallurgical characterization of these steels with the fracture toughness data presented in NUREG-0577, the staff concludes that the fracture toughness properties of the ferritic materials in the engineered safety features have adequate margins against the possibility of nonductile behavior and rapidly propagating fracture.

The controls on the use and fabrication of the austenitic stainless steel of the systems satisfy most of the recommendations of RGs 1.31, "Control of Ferrite Content in Stainless Weld Metal," and 1.44, "Control of the Use of Sensitized Stainless Steel." The alternative approaches taken by the applicant have been reviewed and are acceptable to the staff (see Section 4.5.1). Fabrication and heat-treatment practices performed provide assurance that the probability of stress corrosion cracking will be reduced during the time interval of the postulated accident.

Conformance with the ~~Codes~~ codes and regulatory guides and with the staff positions mentioned above, constitute an acceptable basis for satisfying the requirements of GDC 1, 4, 14, 35, 41; Appendix B to 10 CFR Part 50, and 10 CFR 50.55a, in which the systems are to be designed, fabricated, and erected so that the systems can perform their functions as required.

(2) GDC 1, 14, 31, and Appendix B to 10 CFR Part 50 have been satisfied with respect to assuring that the reactor coolant pressure boundary and associated auxiliary systems have an extremely low probability of leakage, of rapidly propagating failure, and of gross rupture. The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on components of the engineered safety features (ESFs) are in accordance with the recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels," or the applicant's alternative approaches are acceptable to the staff as discussed in Section 5.2.3. Compliance with the recommendations of RG 1.36 forms a basis for satisfying the requirements of GDC 1, 14, and 31.

Protective coating systems are discussed in Section 6.1.2.

(3) The requirements of GDC 4, 35, 41 and Appendix B to 10 CFR Part 50 have been satisfied with respect to compatibility of ESF components with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

The controls on the ~~PH~~ pH and chemistry of the reactor containment sprays and the emergency core cooling water following a loss-of-coolant or design-basis accident are adequate to reduce the probability of stress-corrosion cracking of austenitic stainless steel components and welds x of the ESF systems in containment throughout the duration of the postulated accident to completion of cleanup.

Also, the controls of the pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, are in accordance with RG 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and provide assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution resulting from corrosion of containment metal or cause serious deterioration of the materials in containment.

The controls placed upon component and system cleaning are in accordance with the recommendations of RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," or the applicant's alternative approaches have been reviewed and approved by the staff as discussed in Section 4.5.1. These controls provide a basis for the finding that the components and systems have been protected against damage or deterioration by containment as stated in the cleaning requirements of Appendix B, 10 CFR Part 50.

Postaccident Emergency Cooling Water Chemistry

This review is related to providing and maintaining the proper pH of the containment sump water and recirculated containment spray water following a design-basis accident (DBA) to reduce the likelihood of stress-corrosion cracking of austenitic stainless steel.

During the containment spray injection phase, the applicant will educt 30% by weight sodium hydroxide into the containment spray solution which is supplied from the refueling water storage tank at a concentration between 1050 and 2050 ppm boron.

During the containment spray recirculation phase a final pH of 7.0 to 9.0 will be achieved in the sump once the borated water has thoroughly mixed with the educted sodium hydroxide.

The staff evaluated the pH of the containment sump water after it was mixed with the educted sodium hydroxide in the containment sump. The staff verified by independent calculations that enough sodium hydroxide is available to raise the pH of the containment sump water to between 7.0 and 9.0. This is consistent with the minimum pH of ≥7.0 required by Branch Technical Position (BTP) MTEB 6-1 to reduce the probability of stress-corrosion cracking of austenitic stainless steel components. The staff will include in the Technical Specifications surveillance requirements to verify that the 30% by weight sodium hydroxide does not deteriorate.

On the basis of the above evaluation, the staff concludes that the postaccident emergency core cooling water chemistry meets the requirements of SRP Section 6.1.1, BTP MTEB 6-1 and GDC 14 and is, therefore, acceptable.

6.1.2 Organic Materials

This evaluation is conducted to verify that protective coatings applied inside containment satisfy the testing requirements of ANSI N101.2(1972) and the quality assurance guidelines of RG 1.54. Compliance with these provides assurance that the protective coatings will not fail under DBA conditions, generating significant quantities of solid debris or combustible gas which could complicate the accident conditions.

In the FSAR the applicant states that paints and protective coatings applied to exposed surfaces will be applied in accordance with the quality assurance requirements of RG 1.54. Additionally, the applicant stated that protective coatings which are applied in the containment will satisfy the requirements of ANSI N101.2(1972) under simulated DBA conditions. X

On the basis of the applicant's compliance with the applicable regulatory guide and ANSI standard, the staff concludes that the protective coating systems and their applications are acceptable and satisfy the requirements of Appendix B to 10 CFR Part 50. This conclusion is based on the applicant having satisfied the quality assurance requirements of Appendix B to 10 CFR Part 50 since the coating systems and their applications satisfy the positions of RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to

Water-Cooled Nuclear Power Plants," and the requirements of ANSI N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities." These measures demonstrate their suitability to withstand a postulated design-basis-accident environment.

The control of combustible gases that can potentially be generated from the organic materials and from qualified and unqualified paints is reviewed in Section 6.2.5 of this report. The consequences of solid debris that can potentially be formed from unqualified paints are reviewed in Section 6.2.2.

On the basis of the above evaluation, the staff concludes that the organic materials satisfy the testing requirements of ANSI N101.2 and the positions of RG 1.54 and are, therefore, acceptable.

6.2 Containment Systems

Later

6.2.1 Containment Functional Design

Later

6.2.2 Containment Heat Removal Systems

Later

6.2.3 Containment Enclosure Emergency Cleanup System

Later

6.2.4 Containment Isolation System

Later

6.2.5 Combustible Gas Control System

Later

6.2.6 Containment Leakage Testing Program

Later

6.2.7 Fracture Prevention of Containment Pressure Boundary

The staff's safety evaluation review assessed the ferritic materials in the Vogtle containment system that constitute the containment pressure boundary to determine if the material fracture toughness is in compliance with the requirements of G~~E~~DC 51, "Fracture Prevention of Containment Pressure Boundary."

GDC 51 requires that under operating, maintenance, testing, and postulated accident conditions (1) the ferritic materials of the containment pressure boundary behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

The Vogtle containment is a reinforced concrete structure. A thin steel liner on the inside surface serves as a leaktight membrane. The ferritic materials of the containment pressure boundary which were considered in the staff's assessment are those that have been applied in the fabrication of the equipment hatch, personnel locks, penetrations, and fluid system components, including the valves required to isolate the system. These components are those parts of the containment system not backed by concrete that must sustain loads during the performance of the containment function under the conditions cited in GDC 51.

The staff has determined that the fracture toughness requirements contained in ASME Code editions and addenda typical of those used in the design of the Vogtle containment may not ensure compliance with GDC 51 for all areas of the containment pressure boundary. The staff has elected to apply in its licensing reviews of ferritic containment pressure boundary materials the criteria for Class 2 components identified in the Summer 1977 Addenda of Section III of the ASME Code. Because the fracture toughness criteria that have been applied in construction typically differ in Code classification and Code edition and addenda, the staff has chosen the criteria in the Summer 1977 Addenda of Section III of the Code to provide a uniform review, consistent with the safety function of the containment pressure boundary materials. Therefore, the staff has reviewed the materials of the components of the Vogtle containment pressure boundary according to the fracture toughness requirements of the Summer 1977 Addenda of Section III for Class 2 components.

Considered in the staff's review were 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 GDC 51. These components are the equipment hatch, personnel airlocks, penetrations, and elements of specific containment penetrating systems.

Staff assessment is based on the metallurigical characterization of these materials and fracture toughness data presented in the NRC report issued for comment, NUREG-0577 (Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," October 1979), and in ASME Code, Section III, Summer 1977 Addenda, Subsection NC.

The metallurigical characterization of these materials, with respect to their fracture toughness, was developed from a review of how these materials were fabricated and what thermal history they experienced during fabrication. The metallurigical characterization of these materials, when correlated with the data presented in NUREG-0577 and the Summer 1977 Addenda of the ASME Code, Section III, provides the technical basis for the staff's evaluation of compliance with the Code requirements.

On the basis of the staff's review of the available fracture toughness data and materials fabrication histories, and the use of correlations between metallurigical characteristics and material fracture toughness, the staff concludes, conditioned on the receipt of confirmatory information, that the ferritic components in the Vogtle containment pressure boundary satisfy the fracture toughness requirements that are specified for Class 2 components by the 1977 Addenda of Section III of the ASME Code. Compliance with these requirements provides reasonable assurance that the Vogtle reactor containment pressure boundary will behave in a nonbrittle manner, that the probability of rapidly propagating fracture will be minimized, and that the requirements of GDC 51 are satisfied.

6.3 Emergency Core Cooling System

The staff has reviewed the Vogtle emergency core cooling system (ECCS) in accordance with SRP Section 6.3 (NUREG-0800). Each of the four areas listed in the Areas of Review section of the SRP was reviewed according to the SRP Review Procedures. Conformance with the acceptance criteria, except as noted below, formed the basis for concluding that the design of the facility for emergency core cooling is acceptable.

As specified in the SRP, the design of the ECCS was reviewed to determine that it is capable of performing all of the functions stipulated in the design criteria. The ECCS is designed to provide core cooling as well as additional shutdown capability for accidents that result in significant depressurization of the reactor coolant system (RCS). These accidents include mechanical failure of the RCS piping up to and including the double-ended break of the largest pipe, rupture of a control rod drive mechanism, spurious relief valve operation in the primary and secondary fluid systems, and breaks in the steam piping.

The principal bases for the staff's acceptance of this system are conformance to 10 CFR 50.46, Appendix K to 10 CFR 50, and GDC 2, 5, 17, 27, 35, 36, and 37.

The applicant states that the criteria will be met even with minimum engineered safeguards available, such as the loss of one emergency power bus, with offsite power unavailable.

6.3.1 System Design

As specified in SRP Section 6.3.1.2, the design of the ECCS was reviewed to determine that it is capable of performing all of the functions required by the design bases. The ECCS design is based on the availability of a minimum of three accumulators, one charging pump, one safety injection pump, one RHR pump, and one RHR heat exchanger, together with their associated valves and piping.

Following a postulated LOCA, passive (accumulators) and active (injection pumps and associated valves) systems will operate. After the water inventory in the RWST has been depleted, long-term recirculation will be provided by taking suction from the containment sump and discharging to the RCS cold and/or hot legs. The low-pressure passive accumulator system consists of four pressure vessels partially filled with borated water and pressurized with nitrogen gas. Fluid level, boron concentration, and nitrogen pressure can be remotely monitored and adjusted in each tank. When RCS pressure is lower than the accumulator tank pressure, borated water is injected through the RCS cold legs. The accumulators are equipped with relief valves that the applicant has stated have sufficient capacity to relieve all RCS backleakage.

The high-head injection system consists of two centrifugal charging pumps and two centrifugal safety injection pumps that provide high-pressure injection of boric acid solution into the RCS. The high-head pumps are aligned to take suction from the RWST for the injection phase of their operation. Low-head injection is accomplished by two RHR pump subsystems taking suction from the RWST during the short-term ECCS injection phase and from the containment sump during long-term ECCS recirculation. The ECCS pumps are provided with overcurrent protection and monitoring of the bearing temperature and pump vibration.

The RWST is a seismic Category I structure with a nominal water inventory of 715,000 gal of 2000-ppm borated water. To maintain the RWST water above the temperature of boron precipitation and freezing, the applicant has provided the RWST with a heating system. The applicant has stated that this system will maintain a minimum water temperature of 50°F. In response to the staff's request for additional information, the applicant has further stated that there is RWST temperature indication and a low-temperature alarm in the control room. In addition, the piping leading from the RWST to the auxiliary building is heat traced by two redundant systems powered from two independent power sources. An alarm is provided in the control room in the event of a problem with the heat-tracing system. In response to another staff concern, the applicant has stated that the RWST is equipped with five radial vents that are symmetrically placed to prevent blockage of the vent on the leeward side in the event of a freezing rain. These responses, which are part of Amendment 8 to the FSAR, are acceptable, in that they fulfil the intent of SRP Section 6.3, Paragraph III.8.

The RWST has been sized to provide allowances for safety injection and manual switchover of ECCS and containment spray pumps to the containment sumps. Instrument errors in measuring the RWST level were considered in sizing the RWST as was the worst-case single active failure during switchover and level changes in the RWST caused by thermal variations. The rate of flow from the RWST was based upon having two RHR pumps, two SI pumps, two centrifugal charging pumps, and two containment spray pumps - all taking suction from the RWST. The containment and RCS pressures were assumed to be zero so as to obtain maximum pump flow. From the assumed allowances and flowrates, the applicant calculated that the minimum time to inject the design amount of water from the RWST before initiating the switchover process was approximately 19 min and the time to complete ECCS switchover was approximately 12 min once the low-low level alarm in the RWST is actuated. The switchover time was computed considering the worst-case single active failure (not isolating one RHR pump from the RWST). Operator actions were estimated to require less time than was available for switchover.

As specified in FSAR Section 6.3.3, the ECCS is initiated either manually or automatically on (1) low pressurizer pressure, (2) high containment pressure, or (3) low pressure in any steamline. This meets the requirements of GDC 20 and the position stated in SRP Section 6.3, Paragraph II. The ECCS may also be manually actuated and monitored from the control room as required by GDC 19. The ECCS is supplemented by instrumentation that will enable the operator to monitor and control the ECCS equipment following a LOCA, so that adequate core cooling may be maintained. This aspect of the postaccident monitoring system is evaluated in Section 7.5 of this report.

As recommended by SRP Section 6.3, Paragraph III.3, the available net positive suction head (NPSH) for all the pumps in the ECCS (centrifugal charging, safety injection, and RHR pumps) has been shown to provide adequate margin. The calculations were performed to meet the safety intent of RG 1.1, "Net Positive

Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps" as stated in FSAR Section 1.9.1.2.

As recommended in SRP Section 6.3, Paragraph III.11, the valve arrangement on the ECCS discharge lines has been reviewed with respect to determining adequate isolation between the RCS and the low-pressure ECCS. In some lines, this isolation is provided by two check valves in series with a normally closed isolation valve (high-head injection discharge and low-head injection discharge to the hot legs). Other discharge lines have only two check valves in series. The applicant has stated in Amendment 6 to the FSAR that test lines are provided for periodic leakage checks of reactor coolant past the check valves forming the reactor coolant system pressure boundaries and that these valves will be categorized as ASME IWV-2000 Category AC. The applicant has stated that these valves will be leak tested on a refueling outage basis. This isolation capability is acceptable. The frequency of the leak testing is reviewed in Section 3.9.6 of this report. (7)

Containment isolation features for all ECCS lines, including instrument lines (GDC 56 and the criteria in RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment") are discussed in Section 6.2.4 of this report.

The applicant has provided additional information to address the potential of having debris inside containment (including thermal insulation and construction materials) that may inhibit ECCS performance during the recirculation mode. The applicant has stated that a procedure has been identified to inspect the containment for debris. The applicant has also committed to periodically inspecting containment sump components such as screens and intake structures for visual evidence of structural distress or corrosion. Stainless steel reflective insulation is used for the reactor vessel, and glass fiber insulation with stainless steel jacketing is used for the remainder of the containment equipment including the primary piping. This fibrous insulation was approved for use in a letter from R. Baer, NRC, to G. Pinsky, Owens-Corning, dated December 8, 1978.

The applicant has stated, in response to staff concerns, that the containment sumps are provided with an inner 1/8-in. grating screen installed over the intake pipe. The purpose of the screen is to filter out those particles that have the potential to either damage ECCS equipment or to prevent flow through the most restrictive flow paths and to preclude vortex formation that could lead to air entrapment and pump cavitation. Full-scale test results were cited as evidence that the screen will prevent vortex formation. The tests were conducted by searching for vortices in the sump. With the grating screen removed, a spectrum of configurations of trash rack blockage were examined. Internal vortices began to appear when 61% of the trash rack was blocked. The most deleterious configuration, i.e., the configuration that had the most potential to degrade pump performance, was one in which 81% of the total trash rack flow area was blocked. This configuration was then tested with the grating screen in place. Under this test condition, the applicant states that no vortices, with a vapor core, formed inside the sump. The applicant has also stated that the required net positive suction head of the RHR pumps is still available even considering the intake losses due to the trash rack, grating screen and pipe intake.

The effects of primary coolant sources outside containment (NUREG-0737, Item III.D.1) are discussed in Section 15.9 of this report.

Proper ESF functioning can be verified by portions of the postaccident monitoring instrumentation (Table 7.5.2-1) of the FSAR. The applicant should commit to complying with TMI-2 Task Action Plan Item II.K.1.5. This item requires a review of all safety-related valve positions, positioning requirements, and positive controls to ensure the proper operation of the ESF. Procedures are also to be reviewed so that these valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes. This item is open.

In response to a staff concern, the applicant is to provide, at a later date, a clarification on compliance with Item II.K.3.10. The staff will evaluate compliance at that time. This item is open. The applicant has also responded that it has approved administrative procedures for indicating and recording the status of operable and inoperable plant safety systems in accordance with Item II.K.1.10. The staff finds this acceptable.

In FSAR Section 13.5.1.2.6, it is stated that procedures have been developed to report ECCS outage data to the NRC. The applicant states that these procedures are in accordance with TMI-2 Task Action Plan Item II.K.3.17.

The compliance of the combined control system capability associated with the ECCS with respect to RG 1.47 is evaluated in Section 7.5 in this report.

During normal operation, the ECCS lines will be maintained in a filled condition by the head of the RWST. High-point vents are provided, and administrative procedures will require that ECCS lines be returned to a filled condition following events such as maintenance that require draining of any of the lines. Maintaining these lines in a filled condition will minimize the likelihood of waterhammer occurring during system startup.

The safety injection lines are protected from intersystem leakage by relief valves in both suction header and discharge lines except for the hot-leg injection lines. The hot-leg injection lines are qualified to RCS pressures. Intersystem leakage detection is described in Section 5.2.5 of this report.

As specified in SRP Section 6.3, Paragraph II.B, no ECCS components are shared between units. This satisfies the requirement of GDC 5.

ECCS performance during the injection and recirculation modes can be monitored in the control room by observing the indications for the high-head and low-head safety-injection flows, ECCS valve status, RHR valve status, accumulator pressure, SI pump status, RWST level, containment sump level and RHR pump status. These indications are environmentally and seismically qualified.

Following a loss-of-coolant accident (LOCA), the ECCS pumps may be called upon to provide long-term core cooling. The applicant has stated in FSAR Amendment 8, that the pumps have been hydraulically and mechanically tested and that the pump seals have been endurance and leak tested under excessive conditions. Furthermore, the procurement specification for the pumps is that they be capable of performing their long-term cooling function for 1 yr. The applicant has been asked to verify that test data existed to confirm that this specification was satisfied. Section 3.10.2 of this report reviews the ability of the ECCS pumps to provide long-term cooling for 1 yr.

Various ECCS components are dependent upon plant auxiliaries in order to maintain the performance of their function. These auxiliary systems may provide the required ac power or cooling (primary auxiliaries) or provide proper control of the environments of the rooms in which the ECCS and support equipment is located (secondary auxiliaries). The applicant provided a list of these auxiliary systems in Amendment 8 and noted that with the exception of the electric recirculation heater for the RWST and the auxiliary gas system, these systems are safety grade and designed to satisfy applicable industry codes. The auxiliary gas system is normally isolated from the accumulators and the recirculation heater is isolated during ECCS operation. These support systems are either normally operating or receive automatic start signals when the components they support are actuated. The staff finds this acceptable. The review of these support systems (component cooling water, nuclear service water, containment heat removal, onsite ac power, auxiliary gas, diesel generator building ventilation, auxiliary building emergency ventilation, and essential chilled water) can be found in Sections 6, 8, and 9 of this report.

6.3.2 Evaluation of Single Failures

As recommended by SRP Section 6.3.II, the staff has reviewed the system description and piping and instrumentation diagrams to verify that sufficient core cooling will be provided during the initial injection phase with and without the availability of offsite power, assuming a single active failure. The accumulators, one in each cold leg, have normally open motor-operated isolation valves in the discharge lines. The applicant has stated in FSAR Section 6.3.2.2.16 that those valves whose spurious movement could result in degraded ECCS performance, including these accumulator isolation valves, have power lockout capability. In addition, the applicant further states that the accumulator isolation valves were the only valves identified to have their motor operators submerged in the event of a LOCA. Before startup, these valves are placed in the position necessary to mitigate the consequences of a LOCA (open) and then have power removed. The possible submergence of the accumulator isolation valve motor operators is therefore of no consequence.

The ECCS is a two-train system that is fully redundant except for the water source for the injection phase. There are no single active failures that can prevent the ECCS from taking suction from this source. The applicant states in FSAR Section 6.3.2.5 that each train of the ECCS is powered by an independent emergency bus. Each emergency bus can be powered from a separate diesel generator in the event of loss of offsite power, as required by GDC 17. At least one train will be operable in the event of loss of offsite power and failure of one diesel generator. The high-head injection systems contain parallel valves in the suction and discharge lines, thus ensuring operability of one train even if one valve fails to open. The low-head injection systems are normally aligned so that valve actuation is not required during the injection phase.

The engineered safety features actuation system (ESFAS) is designed to automatically perform the short-term injection phase; no operator actions are required. Two separate and redundant actuation trains are provided. Each actuation train is assigned to a corresponding electrical power train to ensure that, in the event of a single failure in the actuation logic, at least one emergency diesel generator, one RHR pump, one SI pump, and one charging pump would receive an actuation signal. There are also provisions for manual actuation and monitoring of the ECCS on the main control board. This complies with SRP Section 6.3 and is acceptable.

The applicant has proposed a partially automatic system to effect switchover of the low-head system from the injection to the recirculation mode. Operator action will be required to complete this switchover. Logic is provided to automatically open the containment sump isolation valves on low-low level in the refueling water storage tank (RWST) so as to provide a source of water to the RHR pumps. Manual actions are then required in order to isolate the RWST, isolate the SI miniflow lines, isolate the charging pump alternate miniflow lines, and to align the suction of the SI and charging pumps to the discharge of the RHR pumps. Switchover normally starts 19 minutes after ECCS initiation. The applicant has stated in Amendment 8 to the FSAR that once switchover begins, the operator has 22 minutes' worth of water in the RWST, considering the most limiting single failure, to complete the switchover, yet the manual actions only require about 12 minutes.

The applicant was asked to describe the consequences of failing to perform the manual actions properly, i.e., omitting a procedural step or performing the steps out of order. In Amendment 8 to the FSAR, the applicant stated that the charging pumps and the SI pumps can be damaged as a result of failing to change the position of particular valves. The applicant was then asked to clarify this response and to indicate whether the consequences were a result of a single failure. This item is open.

The staff has reviewed the plant's capability for hot-leg injection during the recirculation phase to preclude excessive buildup of boron concentration in the pressure vessel. The staff has concluded that there is sufficient redundancy in injection lines and pumps to ensure adequate hot-leg injection after 16 hr of cold-leg injection. This satisfies the requirements of SRP Section 6.3, Paragraph III.6.

During the long-term recirculation cooling phase of ECCS operation, leak detection is required to identify passive ECCS failures outside of containment, such as pump seal failures. The applicant has provided a system of water-level monitors and radiation detectors located in each compartment that contains ECCS components. The applicant states that with this system, the limiting leak (assumed to be 50 gpm) would be detected and isolated within 30 minutes. The applicant has calculated that the total leakage in 30 minutes would not compromise long-term cooling. Leak rates of less than 50 gpm would result in scenarios in which the detection (alarm) time would be longer, but the time available for operator response would also be longer. The staff finds the system acceptable because it provides a means of identifying and isolating a passive failure in the ECCS outside of containment. The limiting leak was assumed to be a pump shaft seal failure. Pipe breaks, causing leaks in excess of 50 gpm, are excluded by BTP MEB 3-1 of the SRP.

A concern was raised with the applicant over a single motor-operated isolation valve in the *common* miniflow path from the SI pumps to the RWST. Closure of this valve with the pumps running and with the RCS pressure high enough to preclude the use of the normal injection path could result in damage to both SI pumps. In response, the applicant has stated that the valve is normally in the open position with power locked out from the main control room. In addition, the valve has redundant safety-related position indication. The staff finds this acceptable.

On the basis of its review of the design features, and contingent upon resolution of the item discussed above, the staff concludes that the ECCS complies with the single-failure criterion of GDC 35.

6.3.3 Qualification of Emergency Core Cooling System

The ECCS design to seismic Category I criteria, in compliance with RG 1.29 is discussed in Section 3.2 of this report. The location of ECCS components in structures designed to withstand a safe-shutdown earthquake and other natural phenomena, per the criteria of GDC 2, is also discussed in Section 3.2, as is the compliance of the equipment to the guidance of ANSI N18.2a-1975 in lieu of RG 1.26.

The ECCS protection against missiles inside and outside containment by the design of suitable reinforced concrete barriers, which include reinforced concrete walls and slabs (conformance to GDC 4), is discussed in Section 3.5 of this report. The protection of the ECCS from pipe whip inside and outside of containment is discussed in Section 3.6 of this report.

The active components of the ECCS designed to function under the most severe duty loads, including safe-shutdown earthquake, are discussed in Sections 3.9 and 3.10 of this report. The ECCS design to permit periodic inspection in accordance with ASME Code, Section XI, which constitutes compliance with GDC 36, is discussed in Section 6.6 of this report. This satisfies the criteria set forth in SRP Section 6.3, Paragraph III.23.c.

The ECCS incorporates two subsystems that serve other functions. The RHRS provides for decay heat removal during reactor shutdown; at other times the RHRS is aligned for ECCS operation. The centrifugal charging pumps are used to maintain the required volume and water chemistry of primary fluid in the RCS. On an ECCS actuation signal, the system is aligned to ECCS operation, and the CVCS function is isolated. In neither case (RHR or centrifugal charging) does the normal system use impair its capability to function as an integral portion of the ECCS. In addition, the RWST serves as a source of water during refueling operations.

6.3.4 Testing

The applicant has committed to demonstrate the operability of the ECCS by subjecting all components to preoperational and periodic testing. The applicant has committed to meeting the intent of RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," and RG 1.79, "Preoperational Testing of Emergency Core Cooling System for Pressurized Water Reactors" for the ECCS. A program has been established for periodic testing that demonstrates compliance with GDC 37.

6.3.4.1 Preoperational Tests

Tests will be conducted to verify system actuation: namely, the operability of all ECCS valves initiated by the safety injection signal, the operability of all safeguard pump circuitry down through the pump breaker control circuits, and the proper operation of all valve interlocks. Operability of ECCS check valves will be verified.

Another test is to check the cold-leg accumulator system and injection line to verify that the lines are free of obstructions and that the accumulator check valves and isolation valves operate correctly. The applicant will perform a low-pressure blowdown of each accumulator to confirm that the line is clear and check the operation of the check valves.

The applicant will use the results of the preoperational tests to evaluate the hydraulic and mechanical performance of ECCS for delivering the flow for emergency core cooling. The pumps will be operated under both miniflow (through test lines) and full-flow (through the actual piping) conditions. This two-point test is to demonstrate compliance with the design specifications for the pump head-flow characteristic curve. The centrifugal charging and SI pumps will also be tested to demonstrate the time required to reach their rated flows.

net positive suction head

By measuring the flow in each pipe, the applicant will make the adjustments necessary to ensure that no one branch has an unacceptably low or high resistance. As part of the ECCS verification, the applicant will analyze the results to ensure there are sufficient total line resistances to prevent excessive runout of the pumps and adequate NPSH under the most limiting system alignment and RCS pressure. The applicant will verify that the maximum flow rate from the test results confirms the maximum flow rate used in the NPSH calculations under the most limiting conditions and will also confirm that the minimum acceptable flow used in the LOCA analysis is met by the measured total pump flow and the relative flow between the branch lines.

The RHRS response to simulated safety signals will be demonstrated to verify system alignment for the recirculation mode. RHR pump performance will be checked to verify head-flow characteristics and runout flow rates for hot- and cold-leg recirculation.

In Section 1.9 of the FSAR, the applicant has committed to compliance with RGs 1.68 and 1.79 with minor exceptions. The exception to RG 1.68 is not related to the ECCS. The exception taken to RG 1.79 is with respect to the accumulator isolation valve. The guide recommends that the valve be tested, with both normal and emergency power, to confirm that it will open under the maximum differential pressure conditions of zero RCS pressure and maximum expected accumulator precharge pressure. The applicant has stated that conditions at the valve motor are independent of power source so that testing with only normal power will meet the intent of the guide. The staff concludes that the preoperational test program for the ECCS conforms to the recommendations of RGs 1.68 and 1.79 and is acceptable pending successful completion of the program. Additional discussion of the preoperational test program is in Section 14 of this report.

6.3.4.2 Periodic Component Tests

Routine periodic testing of the ECCS components and all necessary support systems at power will be performed. All ECCS components can be tested on line or have power locked out. Valves that actuate after a LOCA are operated through a complete cycle. Pumps are operated individually in this test on their mini-flow lines including the charging pumps which can also be tested by their normal charging function. RHR pump operability is also verified during those times the RHR system is put into operation. Series check valves that form a pressure boundary are supplied with test lines to verify that the valves can independently sustain the differential pressure. A visual inspection of pump seals, valve packings, flange connections and relief valves will be made in order to detect leakage. Accumulator performance will be monitored by level and pressure instrumentation during plant operation. The Vogtle plant will have the capability to conduct an integrated test when the plant is cooled down and RHR is operating. This integrated test will demonstrate operability of the valves, pump circuit breakers, and automatic circuitry, including the starting and loading of the diesel generators. The applicant has stated that the ECCS components and systems are designed to satisfy the intent of ASME Code, Section XI.

6.3.5 Performance Evaluation

The ECCS has been designed to deliver fluid to the RCS to limit the maximum fuel cladding temperature following transients and accidents that require ECCS actuation. The ECCS is also designed to remove the decay and sensible heat during the recirculation mode. 10 CFR 50.46 lists the acceptance criteria for an ECCS. These criteria include the following:

(1) The calculated maximum fuel cladding temperature does not exceed 2200°F.

(2) The calculated total oxidation of the cladding does not exceed 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptable low value and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

In addition, 10 CFR 50.46 states

ECCS cooling performance shall be calculated in accordance with an acceptable model, and shall be calculated for a number of postulated loss-of-coolant accidents. Appendix K to 10 CFR 50, ECCS Evaluation Models, sets forth certain criteria and acceptable features of evaluation models.

6.3.5.1 Large-Break LOCA

The applicant has examined a spectrum of large breaks in RCS piping, and these analyses indicate that the most limiting event is a cold-leg double-ended guillotine break with a Moody discharge coefficient of 0.6. The applicant took credit for one train of the active ECCS components and three of the four accumulators in the analysis. In the large-break analysis, the worst-case break resulted in decreasing RCS pressure. ECCS was assumed to be initiated by the low pressurizer pressure trip. The analysis results demonstrated that adequate core cooling is provided assuming the worst single failure, with no credit taken for nonsafety-grade equipment.

During startup and shutdown, portions of the ECCS, such as the accumulators and the safety injection pumps, are intentionally isolated from the RCS. A double-ended guillotine break during these times was analyzed and found to be bounded by the full-power large-break LOCA. The applicant is to provide confirmation of this in a future amendment to the FSAR. In addition to this ECCS equipment being isolated, the signals providing ECCS actuation because of low pressurizer pressure or low compensated steamline pressure are blocked below the P-11 interlock (approximately 2000 psi RCS pressure). The applicant was asked to demonstrate there still existed adequate signals and alarms to detect a LOCA and to initiate mitigation actions. By FSAR Amendments 6 and 8, the applicant has stated that for a large-break LOCA at pressures below the

P-11 interlock, safety injection would be actuated on high containment pressure. In addition, the operator would be informed by safety-related indication of the pressurizer level (a low-level alarm is provided) and by rapid changes in the containment and RCS pressures. The applicant has stated that during startup and shutdown, the flow from one charging pump and one RHR pump would be sufficient to maintain the plant within the 10 CFR 50.46 criteria.

The large-break LOCA evaluation model used in this analysis is described in WCAP-9220 (1981 revision). This model was approved by NRC (letter from J. R. Miller, NRC, to E. P. Rahe, Westinghouse, dated April 29, 1978) and is used in large-break LOCA analyses for Westinghouse plants. Concerns expressed in NUREG-0630 about the conservatism of fuel-cladding swelling and rupture models used in LOCA analyses have been addressed in this revision of the WCAP.

Containment parameters are chosen to minimize containment pressure so that core reflood calculations are conservative. Fuel rod initial conditions are chosen to maximize cladding temperature and oxidation. Calculations of core geometry are carried out past the point where temperatures are decreasing. The most limiting break with respect to peak clad temperature is the double-ended guillotine break in the pump discharge leg with maximum safety injection and with a discharge coefficient (C_D) = 0.6 The peak clad temperature is 2171.9°F, which is below the 2200°F limit of 10 CFR 50.46. The limiting local and core-wide cladding oxidation values calculated by the applicant were 8.65% and less than 0.3%, respectively. These values are within the 17% and 1% limits of 10 CFR 50.46, respectively. Discharge coefficients of 0.4, 0.6, and 0.8 were considered. Appendix K of 10 CFR 50 requires that three valves of discharge coefficients be considered that span the range of 0.6 to 1.0. The applicant should therefore analyze the case for C_D = 1.0 or provide an evaluation that shows that this case is bounded by the previously analyzed cases. If the applicant chooses to analyze the C_D = 1.0 case, then it must be shown that the criteria of 10 CFR 50.46 are still satisfied. This is an open item.

In the LOCA analysis, an upper-head temperature equal to the cold-leg temperature was assumed. The applicant has predicted that there is sufficient bypass flow to the head region to validate this assumption. The analytical models used to predict the upper-head temperature were verified by testing at a Westinghouse facility. Further confirmation of the analytical models was obtained by head/fluid temperature measurements at a similar four-loop plant.

Excessive boric acid concentration can occur because of boiloff that results in boron precipitation. This will be prevented by switching over from cold leg recirculation to hot-leg recirculation. Backflushing ECCS water through the core prevents boron precipitation. The applicant has stated that if hot-leg recirculation is initiated 16 hr after the accident, the maximum allowable boric acid concentration will not be exceeded. At this time the boiloff rate (20 lb/sec) is exceeded by the worst-case hot-leg injection rate (82 lb/sec for the double-ended cold-leg guillotine break and loss of one ECCS train).

6.3.5.2 Small-Break LOCA

The LOCA sensitivity studies determined the limiting small break to be less than a 10-in.-diameter rupture of the RCS cold leg. A range of small-break analyses was presented that established the limiting break size. The analysis of this break has shown that the high-head portion of the ECCS, together with accumulators, provides sufficient core flooding to keep the calculated peak cladding temperature less than that calculated for a large break and below the limits of 10 CFR 50.46.

The applicant has submitted analyses for a spectrum of small-break LOCA analyses (3-in., 4-in, 6-in). These identify that the 4-in. break is the limiting small break in terms of calculated peak cladding temperature (1537°F). The maximum local zirconium-water reaction was calculated to be 0.78% for a 3-in. break and the core-wide zirconium-water reaction was calculated to be less than 0.3% for all break sizes.

The applicant was asked to demonstrate the adequacy of ECCS equipment and actuation signals during shutdown and startup in the event of a small-break LOCA. In Amendments 6 and 8 to the FSAR, the applicant has stated that, for very small LOCAs, the containment's high-pressure setpoint may not be reached. For these transients, less than a 2-in.-diameter break, the operator would have to observe the available indication, diagnose the situation, and manually initiate safety injection. The indication available to the operator is the loss of pressurizer level, decrease in RCS pressure and increase in containment pressure. In addition, radiation alarms and sump water level may also be available. The operator has 10 minutes before the initiation of core uncovery. This response is satisfactory, provided that the applicant can demonstrate that there are data to verify that the operator will act before the core exceeds Appendix K criteria for a very small LOCA inside and outside containment. This is a confirmatory item.

The applicant has analyzed the performance of the ECCS in accordance with the criteria set forth in 10 CFR 50.46 and Appendix K to 10 CFR 50. The staff has reviewed the applicant's evaluation, and concludes that it is acceptable with the exception that timely operator action for a very small LOCA during startup and shutdown shall be verified.

6.3.6 Conclusions

The ECCS includes the piping, valves, pumps, heat exchangers, instrumentation, and controls used to transfer heat from the core after a LOCA. The scope of review of the ECCS for the Vogtle plant included piping and instrumentation diagrams, equipment layout, failure modes and effects analyses, and design specifications for essential components. The review included the applicant's proposed design criteria and design bases for the ECCS and the manner in which the design conforms to these criteria and bases.

The staff concludes that the design of the ECCS is acceptable and satisfies the requirements of GDC 2, 5, 17, 27, 35, 36, and 37, except as noted. This conclusion is based on the following:

(1) As stated in Section 3.2 of this report, the applicant has satisfied ~~the criteria of~~ GDC 2 with regard to the seismic design of nonsafety systems or portions thereof that could have an adverse effect on ECCS by meeting Position C.2 of RG 1.29.

(2) The applicant has satisfied ~~the criteria of~~ GDC 5 with respect to sharing of structures, systems, and components by demonstrating that such sharing

does not significantly impair the ability of the ECCS to perform its safety function.

(3) The applicant has satisfied ~~the criteria of~~ GDC 17 with respect to providing sufficient capacity and capability to ensure that (a) specified acceptable fuel design limits and design conditions of the RCPB are not exceeded as a result of anticipated operational occurrences and (b) the core is cooled and vital functions are maintained in the event of postulated accidents.

(4) The applicant has satisfied ~~the criteria of~~ GDC 27 with regard to providing combined reactivity control system capability to ensure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained, and the applicant's design meets the guidelines of RG 1.47 except as noted in Section 7.5 of this report.

(5) The applicant has satisfied ~~the criteria of~~ GDC 35 in regard to abundant cooling capability for ECCS by providing redundant safety-grade systems that meet the recommendations of RG 1.1. However, the applicant must provide additional information to demonstrate adequate ECCS capability in light of single failures in operator actions during switchover.

(6) The applicant has satisfied ~~the criteria of~~ GDC 36 with respect to the design of ECCS to permit appropriate periodic inspection of important components of the system.

(7) The applicant has satisfied ~~the criteria of~~ GDC 37 with respect to designing the ECCS to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation. The plant Technical Specifications will need to be reviewed to confirm compliance to the criteria.

(8) The applicant has provided an analysis of the ECCS performance using an approved analysis model that satisfies the criteria of Appendix K to 10 CFR 50 and has shown the system performance satisfies the acceptance

criteria of 10 CFR 50.46 with the exception that a large-break LOCA must be evaluated with a discharge coefficient of 1.0. This includes a demonstration that the peak cladding temperature, maximum hydrogen generation, and long-term cooling, as calculated with an acceptable evaluation model, are in accordance with these criteria. The applicant must provide confirmation that the operator can take appropriate action within the stated time in the event of a very small LOCA during startup and shutdown.

6.4 Control Room Habitability

The requirements for protecting control room personnel under accident conditions are specified in GDC 19. The applicant proposes to fulfill these requirements by incorporating shielding; emergency heating, ventilating and air conditioning (HVAC) systems; and self-contained breathing apparatus in the control room habitability design. The habitability systems also provide storage for food and water, sanitary facilities, and fire protection that includes a remote shutdown capability.

The design of the control room habitability systems relative to the following areas is discussed in the SER sections indicated parenthetically:

(1) Explosion, fire, and toxic gas in vicinity of plant (2.2.1-2.2.3)

(2) Protection from wind and tornado effects (3.3)

(3) Flood design (3.5)

(4) Missile protection (3.5)

(5) Protection against dynamic effects associated with postulated ruptures of piping (3.6)

(6) Environmental qualification of equipment (3.11)

(7) Filter efficiencies (6.5.1)

(8) Radiation protection aspect of GDC 19 and NUREG-0737 Item II.B.2; shielding; TSC (12.3)

(9) HVAC systems analysis (9.4.1) (includes seismic review)

(10) Fire protection and remote shutdown capability (9.5.1)

(11) Human engineering, control room environment, and communications (18)

The staff evaluation indicates an inconsistency in the applicant's estimate of the control room leak rate. Relative to protection from toxic gas, the applicant states that the air leakage is no greater than 185 ft^3/min from all pathways based on 1/8-inch water gauge (WG) pressure differential (FSAR p. 6.4.2-6). This conflicts with the estimated 1500 ft^3/min air intake at the same pressure differential assumed in the evaluation of radiation doses to control room personnel following design-basis accidents (FSAR Amendment 8, Q450.5). Until this matter is resolved, habitability of the control room following radiation and toxic-gas release accidents is an open item.

In addition, information is needed from the applicant in two areas:

(1) Response to Question 450.3 on the data used to estimate the control room dose following a LOCA

(2) Toxic-gas evaluation for the chemicals listed in FSAR Table 2.2.3-18. The evaluation should include data described in Table C-3 of RG 1.78.

On the basis of the foregoing material, the staff concludes that the applicant has not demonstrated that the control room habitability systems will adequately protect the control room operators in accordance with the requirements of 10 CFR Part 50 (Appendix A, GDC 19) and, therefore, compliance with NUREG-0737, Item III.D.3.4, cannot be established.

6.5 Engineered Safety Feature Atmosphere Cleanup Systems

6.5.1 System Description and Evaluation

FSAR Section 6.5 contains information pertaining to engineered safety feature (ESF) atmosphere cleanup systems, their design bases, and applicable acceptance criteria.

The staff has reviewed the applicant's design, design criteria, and design bases for the ESF atmosphere cleanup systems for Vogtle, Units 1 and 2 (Vogtle). The acceptance criteria used as the basis for its evaluation are in Section II of SRP Section 6.5.1 (NUREG-0800). These acceptance criteria include the applicable GDC, ANSI N509-1980, ANSI N510-1980, RG 1.52, and other documents identified in Section II of the SRP. Conformance to the acceptance criteria provides the bases for concluding that the ESF atmosphere cleanup systems meet the requirements of 10 CFR Part 50.

The ESF atmosphere cleanup system at Vogtle consists of process equipment and instrumentation necessary to control the release of radioactive iodine and particulate material following a design-basis accident (DBA). At Vogtle the following four filtration systems have been designed for this purpose:

(1) control room heating, ventilation and air conditioning system described in FSAR Section 6.4 and Subsections 6.5.1 and 9.4.1;

(2) fuel handling building postaccident exhaust system described in FSAR Subsection 9.4.2;

(3) piping penetration filter exhaust system described in FSAR Subsection 9.4.3; and

(4) electrical penetration filter exhaust system described in FSAR Subsection 9.4.5.

Each system is designed to function automatically upon receipt of an ESF actuation system signal. Each of these systems was reviewed in accordance with the SRP. The results of these reviews are discussed below.

(1) Control Room Heating, Ventilation, and Air Conditioning System

The control room heating, ventilation and air conditioning system contains two 100% capacity essential air filtration systems, with each system designed to filter up to 25,000 ft^3/min of air. Each filtration system includes, in order, a demister, an electric heater, a high-efficiency particulate air (HEPA) filter, a 4-in.-deep charcoal adsorber, another HEPA filter, and a fan. The purpose of the control room heating, ventilation and air conditioning system is to limit the amount of radioactivity introduced into the control room following an accident by pressurizing the control room and by filtering the air entering the control room, and to filter radioactivity already in the control room so that doses to control room operators will be within the design criterion of GDC 19. A safety injection signal or the detection of high radiation levels in the control room outside air intake causes the initiation of the control room isolation signal. This signal causes activation of the essential air filtration units followed by closure of the isolation dampers between the normal and essential systems, which automatically trips the normal air handling units. One of the essential air filtration trains then may be manually transferred to the emergency standby mode. Air within the control room is recirculated continuously through the essential air filtration unit. The outside air required for pressurization is mixed with the return air upstream of the filtration unit. The system design provides no potential bypass pathways around the essential air filtration units while they are operating in this mode. The staff has credited the system with 99% removal efficiency for all forms of radioiodine, pending resolution of the open items discussed later regarding this subject.

(2) Fuel-Handling Building Postaccident Exhaust System

The fuel-handling building postaccident exhaust system consists of two 100% capacity filtration systems with each designed to filter up to

5,000 ft^3/min of air. Each filtration system includes a demister, an electric heater, a HEPA filter, a 4-in.-deep charcoal adsorber, another HEPA filter, and a fan. The fuel-handling building postaccident exhaust system is designed to maintain a minimum negative pressure within the fuel handling building and to filter exhaust air following a fuel handling accident, and thereby to minimize the release of airborne radioactivity to the outside atmosphere. This ensures that offsite radiation exposures are within the guidelines of 10 CFR Part 100 and exposures to operating personnel in the control room are within the design criterion of GDC 19. The system design provides no potential bypass pathways around the air filtration units while they are operating in this mode. The staff has credited the system with 99% removal efficiency for all forms of radioiodine, pending resolution of the open items discussed later regarding this system.

(3) Piping Penetration Filter Exhaust System

The piping penetration filter exhaust system consists of two 100% capacity filtration systems with each designed to filter up to 16,000 ft^3/min of air. Each filtration system includes a demister, an electric heater, a HEPA filter, a 4-in.-deep charcoal adsorber, another HEPA filter, and a fan. The system is designed to maintain a minimum negative pressure on the piping penetration area of the auxiliary building and to filter exhaust air following containment and penetration area leakage under accident conditions. This ensures that the offsite radiation exposures resulting from the postulated post-LOCA leakage in recirculation piping are within the guidelines of 10 CFR Part 100 and exposures to operating personnel in the control room are within the design criterion of GDC 19. It also ensures that the ECCS and containment spray pump rooms can be purged to allow access for equipment repair and maintenance. The system design provides no potential bypass pathways around the air filtration units while they are operating in this mode. The staff has credited the system with 99% removal efficiency for all forms of radioiodines, pending resolution of the open items discussed later regarding this system.

(4) Electrical Penetration Filter Exhaust System

The electrical penetration filter exhaust system consists of two 100% capacity filtration systems with each designed to filter up to 6,000 ft^3/min of air. Each filtration system includes a demister, an electric heater, a HEPA filter, a 4-in.-deep charcoal adsorber, another HEPA filter, and a fan. The system is designed to maintain a minimum negative pressure on the electrical penetration area of the control building and to minimize release of airborne radioactivity following postulated post-LOCA containment leakage by filtering recirculated and exhaust air. This ensures that the offsite radiation exposures resulting from these accidents are within the guidelines of 10 CFR Part 100 and exposures to operating personnel in the control room resulting from these accidents are within the design criterion of GDC 19. The system design provides no potential bypass pathways around the air filtration units while they are operating in this mode. The staff has credited the system with 99% removal efficiency for all forms of radioiodine, pending resolution of the open items discussed later regarding this system.

The ESF filtration systems were reviewed according to SRP Section 6.5.1 (NUREG-0800), RG 1.52, Revision 1, and GDCs 19, 41, 43, 61, and 64.

The applicant has provided a comparison of the design of the Vogtle ESF filtration systems with the acceptance criteria of the SRP in FSAR Subsection 6.5.1.7. The staff has determined that the applicant has proposed an exception to the SRP acceptance criterion concerning conformance to the guidelines of RG 1.52 and the recommendations of ANSI N509 in that the proposed minimum instrumentation, readout, recording, and alarm provisions for the ESF atmosphere cleanup systems are not in conformance with Table 6.5.1-1 of the SRP, as follows:

(1) no local indication is provided of unit inlet or outlet flow;

(2) no local high alarm signal is provided of the pressure drop across the prefilter (demister in the Vogtle design);

(3) no local status indication is provided for the electric heater;

for a temperature sensor located ~~before~~ between the heater and the first HEPA filter

(4) no local indication, high alarm, and low alarm signals are provided;

of the pressure drop across the first HEPA filter

(5) no local high alarm signal is provided;

(6) no local two-stage high alarm signal is provided for a temperature sensor located between the adsorber and the second HEPA filter;

(7) no local high alarm signal is provided for the pressure drop across the second HEPA filter;

(8) no local hand switch and status indication is provided for the deluge valves;

(9) no high alarm, low alarm, and trip-alarm signals are provided in the control room for a temperature sensor located between the heater and the first HEPA filter; and

(10) no recorded indication is provided in the control room of the pressure drop across the first HEPA filter.

The applicant has stated that the ESF filtration systems are designed to operate only during postaccident conditions and do not operate under normal conditions (as at some other plant), except during testing. Because the ESF filtration unit equipment rooms are potentially inaccessible because they may be in high radiation areas under post-accident conditions, the Vogtle design relies on control room instrumentation for monitoring of the filtration units. A high humidity alarm in the control room provides direct indication of high humidity rather than the indirect indication a low temperature alarm would provide. The other alarms are provided in the control room. This is acceptable because the local areas may be inaccessible after an accident. Therefore, based on its review, the staff concludes that the local instrumentation and the low temperature alarm in the control room identified above as not provided in the Vogtle design are not needed to assure that the ESF atmosphere cleanup systems will perform their design safety functions. This resolves items 1 through 8 above. The staff further concludes that the applicant has not provided justification for the other exceptions taken to the minimum instrumentation listed in Table 6.5.1-1 of the SRP (Items 9 and 10 above), namely:

(1) no high alarm and trip-alarm signals are provided in the control room for a temperature sensor located between the heater and the first HEPA filter; and

(2) no recorded indication is provided in the control room for the pressure drop across the first HEPA filter.

This, therefore, is an open item.

The applicant has proposed a further exception to the SRP acceptance criterion concerning conformance to the guidelines of RG 1.52 in that no cooling mechanism has been provided for the ESF filtration units charcoal adsorber sections which has been demonstrated to satisfy the single-failure criterion. The applicant has stated that an analysis was performed to conservatively model the heating of the charcoal due to a postulated loss-of-coolant accident; and that the results of this analysis showed that a cooling mechanism is not needed for the ESF filter systems. A water spray system is provided to allow flooding of the charcoal bed to prevent bed ignition. On the basis of its review, the staff concludes that the applicant has not provided adequate justification for not providing a cooling mechanism which satisfies the single-failure criteri~~a~~on. This, therefore, is an open item.

The staff concludes that the design of the ESF atmosphere cleanup systems, including the equipment and instrumentation to control the release of radioactive materials in gaseous effluents following a postulated DBA, is acceptable except as noted. This conclusion is based on the applicant having satisfied the requirements of GDC 19, 41 and 61 by providing ESF atmosphere cleanup systems on the control room habitability, containment, and associated systems. The applicant has satisfied the requirements of GDC 41, 43 and 64 by providing for the inspection and testing of the ESF atmosphere cleanup systems and monitoring for radioactive materials in effluents from these systems. In meeting these regulations, the applicant has demonstrated that the design of the ESF atmosphere cleanup systems meets the guidelines of RG 1.52 and the

ANSI N509 and N510 industry standards, as referenced in the SRP. The staff has reviewed the applicant's system descriptions and design criteria for the ESF atmosphere cleanup systems. On the basis of its evaluation, with respect to the SRP criteria, the staff finds the proposed ESF atmosphere cleanup systems acceptable, except as noted.

The filter efficiencies given in Table 2 of RG 1.52 are appropriate for use in accident analyses, pending resolution of the open items discussed above.
6.5.3, 6.5.4 Later

6.6 Inservice Inspection of Class 2 and 3 Components

This section was prepared with the technical assistance of DOE contractors from the Idaho National Engineering Laboratory.

6.6.1 Compliance With the Standard Review Plan

The July 1981 Edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800) includes Section 6.6, "Inservice Inspection of Class 2 and 3 Components." The Vogtle review is continuing because the applicant has not completed the PSI program and examinations. The staff review to date was conducted in accordance with SRP Section 6.6 except as discussed below.

Paragraph II.4, "Acceptance Criteria, Inspection Intervals," has not been reviewed because this area applies only to inservice inspection (ISI) not to PSI. This subject will be addressed during review of the ISI program after licensing.

Paragraph II.5, "Acceptance Criteria, Evaluation of Examination Results," has been reviewed. The applicant committed in the FSAR to incorporate ASME Code Article IWC-3000 "Acceptance Standards for Flaw Indications," into the PSI program. However, ongoing NRC generic activities and research projects indicate that the presently specified ASME Code procedures may not always be capable of detecting the acceptable size flaws specified in these standards. For example, ASME Code procedures specified for volumetric examination of vessels, bolts and studs, and piping have not proven to be capable of detecting acceptable size flaws in all cases. The staff will continue to evaluate the development of new or improved procedures and will require that these improved procedures be made a part of the inservice examination requirements. The applicant's repair procedures based on ASME Code Articles IWC-4000 and IWD-4000, "Repair Procedures," have not been reviewed. Repairs are not generally necessary in the PSI program. This subject will be addressed during review of the ISI program.

Paragraph II.8, "Acceptance Criteria, Code Exemptions," has not been completed because the applicant has not listed in the FSAR or the PSI program any Code exemptions as permitted by the criteria in IWC-1220. The SRP requires that the applicant list these exemptions, if used.

Paragraph II.9, "Acceptance Criteria, Relief Requests," has not been completed because the applicant has not identified the limitations to examination. Specific areas where ASME Code examination requirements cannot be met will be identified as the PSI progresses. The complete evaluation of the PSI program will be presented in a supplement to the Safety Evaluation Report (SER) after the applicant submits the required examination information and identifies all plant-specific areas where ASME Code Section XI requirements cannot be met and provides a supporting technical justification.

6.6.2 Examination Requirements

GDC 36, 39, 42, and 45 (Appendix A of 10 CFR Part 50), require, in part, that the Class 2 and 3 components be designed to permit appropriate periodic inspection of important components to ensure system integrity and capability. 10 CFR 50.55a(g) defines the detailed requirements for the preservice and inservice inspection programs for light-water-cooled nuclear power facility components.

On the basis of the construction permit date of June 28, 1974, this paragraph of the regulations requires that a preservice inspection program for Class 2 components (including supports) shall meet the preservice examination requirements set forth in editions of Section XI of the ASME Code and addenda in effect 6 months before the date the construction permit is issued. The components (including supports) may meet the requirements set forth in subsequent editions of this Code and addenda which are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein. The initial ISI program must comply with the requirements of the latest edition and addenda of Section XI of the ASME Code in effect 12 months before the operating license is issued, subject to the limitations and modifications listed in 10 CFR 50.55a(b).

6.6.4

6.6.3 Evaluation of Compliance With 10 CFR 50.55a(g)

Review has been completed on the information presented in the FSAR through Amendment 9 dated August 1984 and the "Preservice Inspection Program," Revision 0, dated April 19, 1984. The applicant stated that on the basis of the construction permit date for Vogtle, the preservice inspection program is required to meet ASME Code Section XI, 1971 Edition through Winter 1972 Addenda. The applicant has voluntarily updated the preservice inspection program based on ASME Code Section XI, 1980 Edition with Addenda through Winter 1980. The use of later referenced Code editions is acceptable as specified by 10 CFR 50.55a(g).

The PSI program for the Class 2 and 3 components has been reviewed. As the applicant stated in the FSAR, these systems and components are included for examination per the applicable Code requirements. The staff established technical positions in the FSAR questions, some of which are resolved in the PSI program. The following items require further input or clarification from the applicant:

(1) The PSI program should contain a list of components subject to examination and a description of the welds exempted from examination in accordance with IWC-1220 and include the criteria for exemption. In addition, the examination isometric drawings for ASME Class 2 components are necessary for the staff to determine the acceptability of the sample of welds required to be examined.

(2) 10 CFR 50.55a(b)(2)(iv) requires that ASME Code Class 2 piping welds in the residual heat removal (RHR) systems, emergency core cooling (ECCS)

systems, and containment heat removal (CHR) systems shall be examined. These systems should not be completely exempted from preservice volumetric examination based on Section XI exclusion criteria contained in IWC-1220. To satisfy the inspection requirements of GDC 36, 39, 42, and 45, the preservice inspection program must include volumetric examination of a representative sample of welds in the RHR, ECCS, and CHR systems.

The specific areas in which the Code requirements cannot be met will be identified after the examinations are performed. The applicant has committed to identify all plant-specific areas in which the Code requirements cannot be met and to provide a supporting technical justification for requesting relief. The SER input will be completed after the applicant:

(1) dockets an acceptable resolution to the above issues
(2) dockets a complete and acceptable PSI program
(3) submits all relief requests with ~~a~~ supporting technical justifications

The staff considers the review of Vogtle's PSI program an open item subject to the applicant providing the above items. The applicant's response will be evaluated in the SER. This open item concerning the Vogtle PSI program has been previously identified in Section 5.2.4.3.

The applicant has not submitted the initial inservice inspection program. This program will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b), but before inservice inspection commences during the first refueling outage.

6.6.~~4~~5 Conclusions

The Vogtle preservice inspection program is an open item until the issues identified in Sections 5.2.4.3 and 6.6.3 are resolved. Compliance with the preservice and inservice inspections required by the ASME Boiler and Pressure Vessel Code and 10 CFR Part 50 constitutes an acceptable basis for satisfying applicable requirements of GDC 36, 39, 42, and 45.

6.7 Main Steam Isolation Valve Leakage Control System

This SRP section is not applicable ~~does not apply~~ to a PWR.

7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

7.1.1 Acceptance Criteria

FSAR Section 7.1 contains information pertaining to safety-related instrumentation and control systems, their design bases, and applicable acceptance criteria. The staff has reviewed the applicant's design, design criteria, and design bases for the instrumentation and control systems for Vogtle Units 1 and 2. The acceptance criteria used as the basis for this evaluation are those identified in the Standard Review Plan (SRP) (NUREG-0800) in Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems Important to Safety," and Table 7-2, "TMI Action Plan Requirements for Instrumentation and Control Systems Important to Safety." These acceptance criteria include the applicable General Design Criteria and the Institute of Electrical and Electronics Engineers (IEEE) Standard 279 "Criteria for Protection System for Nuclear Power Generating Stations" (10 CFR 50.55a(h)). Guidelines for implementing the requirements of the acceptance criteria are provided in the IEEE standards, regulatory guides, and branch technical positions identified in SRP Section 7.1. Conformance to the acceptance criteria provides the bases for concluding that the instrumentation and control systems meet the requirements of 10 CFR Part 50.

7.1.2 Method of Review

Vogtle Units 1 and 2 use Westinghouse nuclear steam supply systems (NSSSs); Bechtel Power Corporation provides the balance of plant (BOP) design. Many safety-related instrumentation and control systems in the NSSS scope of supply are similar to those at Comanche Peak and McGuire and have been previously reviewed and approved by the staff. The staff concentrated its review on those areas where the Vogtle Units 1 and 2 design differs from previously reviewed designs and on those areas that have been of concern during reviews of other similar plants. A meeting was held with the applicant and the NSSS and BOP designers to clarify the design and to discuss concerns the staff has with the design. The staff reviewed detailed drawings--including piping and instrumentation diagrams, logic diagrams, control wiring diagrams, electrical one-line diagrams, and electrical schematic diagrams--during the audit.

7.1.3 General Conclusion

The applicant has identified the instrumentation and control systems important to safety and the acceptance criteria that are applicable to those systems as identified in the SRP. The applicant has also identified the guidelines--including the regulatory guides and the industry codes and standards--that are applicable to the systems as identified in FSAR Table 7.1.1-1.

On the basis of the review of FSAR Section 7.1, the staff concludes that the implementation of the identified acceptance criteria and guidelines satisfies the requirements of GDC 1, "Quality Standards and Records," with respect to the design, fabrication, erection, and testing to quality standards commensurate with the importance of the safety functions to be performed. The staff finds that the NSSS and the BOP instrumentation and control systems important to safety, addressed in FSAR Section 7.1, satisfy the requirements of GDC 1 and, therefore, are acceptable.

7.1.4 Specific Findings

7.1.4.1 Open Items

The staff's conclusions apply to the instrumentation and control systems important to safety with the exception of the open items listed below. The staff will review these items and report their resolution in a supplement to this report. The applicable sections of this report that address these items are indicated in parentheses following each open item.

(1) Design Modification for Automatic Reactor Trip Using Shunt Coil Trip Attachment (7.2.2.3)

(2) Level Measurement Errors Resulting From Environmental Temperature Effects on Level Instrument Reference Legs (7.3.3.4)

(3) Auxiliary Feedwater System (7.3.3.7)

(4) Override of Isolation Signals (7.3.3.8)

(5) Isolators Used in the BOP Design (7.3.3.9)

(6) Auxiliary Relays Used With No-Go Tested Slave Relays (7.3.3.10)

(7) Electrical Tunnel Ventilation System (7.3.3.11)

(8) Control Room Ventilation Isolation (7.3.3.12)

(9) Emergency Response Capability-RG 1.97, Rev. 2, Requirements (7.5.2.1)

(10) NUREG-0737, Item II.D.3, Direct Indication of Relief and Safety Valve Positions, Alarms in Conjunction With Valve Position Indication (7.5.2.3)

(11) Bypass and Inoperable Status Panel, Conformance to Position C.2 of RG 1.47 (7.5.2.4)

(12) IE Bulletin 79-27, Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation (7.5.2.5)

(13) Freeze Protection for Instrumentation Sensing and Sampling Lines (7.5.2.6)

(14) RCS Overpressure Protection During Low-Temperature Operation (7.6.2.1)

(15) NUREG-0737, Item II.K.3.1, Installation and Testing of Automatic Power-Operated Relief Valve Isolation System (7.6.2.2)

(16) Instrumentation for Process Measurements Used for Safety Functions (7.6.2.3)

(17) High-Energy-Line Breaks and Consequential Control System Failures (7.7.2.2)

(18) Control System Failure Caused by Malfunctions of Common Power Source or Instrument Line (7.7.2.3)

7.1.4.2 Confirmatory Items

In a number of cases, the applicant has committed to provide additional documentation to address concerns raised by the staff during its review. On the basis of information provided during meetings and discussions with the applicant, the technical issue has been resolved in an acceptable manner. However, the applicant must formally document its commitments for resolving these items. The sections of this SER that address these items are indicated in parentheses.

(1) Test of Engineered Safeguards P-4 Interlock (7.3.3.2)

(2) Steam Generator Level Instrumentation (7.3.3.5)

(3) IE Bulletin 80-06 Concerns (7.3.3.6)

(4) NUREG-0737, Item II.D.3, Direct Indication of Relief and Safety Valve Positions, FSAR Revision Regarding Power Supply For Primary Safety Valve Status (7.5.2.3)

(5) Bypass and Inoperable Status Panel, Fuel-Handling Building's ESF HVAC System (7.5.2.4)

7.1.4.3 Technical Specification Items

Items to be included in the plant Technical Specifications and information to be audited as part of the effort to issue Technical Specifications are discussed in the following sections:

(1) Lead, Lag, and Rate Time Constant Setpoints Used in Safety System Channels (7.2.2.1)

(2) Turbine Trip Following a Reactor Trip (7.2.2.2)

(3) Trip Setpoint and Margins (7.2.2.4)

(4) NUREG-0737, Item II.K.3.10, Proposed Anticipatory Trip Modification (7.2.2.5)

(5) Undetectable Failure in Online Circuitry for Engineered Safeguards Relays (7.3.3.3)

The above are all confirmatory items pending Technical Specification submittal and staff review.

7.1.4.4 Site Visit

A site review will be performed to confirm that the physical arrangement and installation of electrical equipment are in accordance with the design criteria and descriptive information reviewed by the staff. The site review will be completed before a license is issued; any problems found will be addressed in a supplement to this SER.

7.1.4.5 Fire Protection Review

The review of the emergency shutdown panel discussed in Section 7.4 of this report covered the compliance of this panel with GDC 19, "Control Room." The aspects of the emergency shutdown panel related to fire protection and the review for conformance to 10 CFR 50, Appendix R (safe shutdown analysis) are included in Section 9.5 of this SER.

7.1.5 TMI-2 Task Action Plan Items

Guidance on implementation of the TMI-2 Task Action Plan was provided to applicants in NUREG-0737. The items related to instrumentation and control systems are listed below. The specific section of the report addressing each item is indicated in parentheses.

(1) II.D.3, Direct Indication of Relief and Safety Valve Position (7.5.2.3)

(2) II.E.1.2, AFWS Automatic Initiation and Flow Indication (7.3.3.1)

(3) II.F.1, Accident Monitoring Instrumentation Positions (4), (5), and (6) (7.5.2.2)

(4) II.K.3.1, Installation and Testing of Automatic Power-Operated Relief Valve Isolation System (7.6.2.2)

(5) II.K.3.9, Proportional Integral Derivative Controller Modification (7.7.2.1)

(6) II.K.3.10, Proposed Anticipatory Trip Modification (7.2.2.5)

(7) II.K.3.12, Anticipatory Reactor Trip on Turbine Trip (7.2.2.6)

7.2 Reactor Trip System

7.2.1 Description

The reactor trip system (RTS) is designed to automatically limit reactor operation within the limits established in the safety analysis. This function is accomplished by tripping the reactor whenever predetermined safety limits are approached or reached. The RTS monitors variables that are directly related to system limitations or calculated from process variables. Whenever a variable exceeds a setpoint, the reactor is tripped by the insertion of control rods. The RTS initiates a turbine trip when a reactor trip occurs. The RTS consists of sensors and analog and digital circuitry arranged in coincidence logic for monitoring plant parameters. Signals from these channels are used in redundant logic trains. Each of the two trains opens a separate and independent reactor trip breaker. During normal power operation, a dc undervoltage coil in each reactor trip breaker holds the breaker closed. For a reactor trip, the removal of power to the undervoltage coils opens the breakers. Opening either of two series-connected breakers interrupts the power from the rod-drive motor generator sets, and the control rods fall by gravity into the core. The rods cannot be withdrawn until the trip breakers are manually reset, and the trip breakers cannot be manually reset until the abnormal condition that initiated the trip is corrected. Bypass breakers are provided to permit the testing of the primary breakers.

In addition to the automatic trip of the reactor described above, there is also provision for manual trip by the operator. The manual trip consists of two switches in the control room with two-train outputs on each switch, and two single-train switches on the remote shutdown panels. Actuation of these switches removes power from the undervoltage coils and energizes the shunt trip coils of the reactor trip breakers. The shunt trip coils are a diverse means for tripping the reactor trip breakers. The reactor will also be tripped by actuating either of the two manual switches for safety injection.

The generic implications of the Salem anticipated transient without scram (ATWS) events are discussed in Section 7.2.2.3 of this report.

The reactor trips listed below are provided in the design of Vogtle Units 1 and 2. The numbers in parentheses after each trip function indicate the coincident logic; for example, two-out-of-three (2/3).

(1) nuclear overpower trips
    (a) power range high neutron flux trip (2/4)
    (b) intermediate range high neutron flux trip (1/2)
    (c) source range high neutron flux trip (1/2)
    (d) power range high positive neutron flux rate trip (2/4)
    (e) power range high negative neutron flux rate trip (2/4)

(2) core thermal overpower trips
    (a) overtemperature ΔT trip (2/4)
    (b) overpower ΔT trip (2/4)

(3) reactor coolant system pressurizer pressure and water level trips

(a) pressurizer low-pressure trip (2/4)

(b) pressurizer high-pressure trip (2/4)

(c) pressurizer high-water-level trip (2/3)

(4) reactor coolant system low flow trips

(a) low reactor coolant flow (2/3 per loop)
above P-7 (2/4)
above P-8 (1/4)

(b) reactor coolant pump bus undervoltage (1/2 in both buses or 2/2 in either bus)

(c) reactor coolant pump bus underfrequency (1/2 in both buses or 2/2 in either bus)

(5) steam generator low-low-level trip (2/4 per loop)

(6) turbine trip (anticipatory)

(a) low auto stop oil pressure (2/3)

(b) turbine stop valves closed (4/4)

(7) safety injection signal actuation trip (see Section 7.3) coincident with actuation of safety injection (1/2)

(8) manual trip (1/2)

The power-range high-neutron-flux trip has two bistables for a high and a low trip setting. The high setting trip is active during all modes of operation. The low setting trip provides protection during reactor startup and shutdown when the reactor is below 10% power. The low-setting trip can be manually blocked above 10% power (P-10) and is automatically reinstated below the P-10 interlock.

The intermediate range trip provides protection during reactor startup and shutdown. This trip can be manually blocked above 10% power (P-10) and is automatically reinstated below the P-10 interlock.

The source-range trip provides protection during reactor startup and shutdown when the neutron flux channel is below the P-6 interlock (10×10^{-10} amp). This trip can be manually blocked above P-6 interlock and is automatically reinstated below the P-6 interlock. It is also automatically blocked above the P-10 interlock.

A power-range high-positive neutron flux rate trip occurs when a sudden abnormal increase in nuclear power is detected. This trip provides departure from nucleate boiling (DNB) protection against low-worth rod ejection accidents from midpower and is active during all modes of operation.

A power-range high-negative neutron flux rate trip occurs when a sudden abnormal decrease in nuclear power is detected. This trip provides protection against two or more dropped rods and is active during all modes of operation.

The overtemperature ΔT trip protects the core against a low departure from nucleate boiling ratio (DNBR). The setpoint for this trip is continuously calculated by analog circuits to compensate for the effects of temperature, pressure, and axial neutron flux difference on DNBR limits.

The overpower ΔT trip protects against excessive power (fuel rod rating protection). The setpoint for this trip is continuously calculated by analog circuits to compensate for the effects of temperature and axial neutron flux difference.

The pressurizer low-pressure trip is used to protect against low pressure that could lead to DNB. The reactor is tripped when the pressurizer pressure (compensated for rate of change) fails below a preset limit. This trip is automatically blocked below approximately 10% power (P-7 interlock) to allow startup and controlled shutdown.

The pressurizer high-pressure trip is used to protect the reactor coolant system against system overpressure. The same transmitters are used as for the pressurizer low-pressure trip except that separate bistables are used for the high-pressure trip. The reactor is tripped when pressurizer pressure exceeds a preset limit.

The pressurizer high-water-level trip is provided as a backup to the pressurizer high-pressure trip and serves to prevent water relief through the pressurizer safety valves. This trip is automatically blocked below approximately 10% of full power (P-7 interlock) to allow startup.

The low reactor-coolant-flow trip protects the core against DNB resulting from a loss of primary coolant flow. Above the P-7 setpoint (approximately 10% power), a reactor trip will occur if any two loops have low flow. Above the P-8 setpoint (approximately 48% power), a trip will occur if any one loop has low flow.

The reactor coolant pump bus undervoltage and underfrequency trips protect the reactor core from DNB. These trips are all automatically blocked below the P-7 setpoint to allow startup.

The steam generator low-low-water-level trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch.

A reactor trip on a turbine trip is actuated by trip fluid pressure switches (two-out-of-three) or by closed signals from the turbine steam stop valve position switches (four-out-of-four). A turbine trip causes a direct reactor trip above 50% power (P-9 interlock).

A safety injection signal initiates a reactor trip. This trip protects the core against a loss of reactor coolant or overcooling.

The manual trip from the control room consists of two switches. Operation of either switch de-energizes the undervoltage coils in each logic train. The breaker shunt coils in these breakers are energized at the same time, which provides a diverse means to ensure that the trip and bypass breakers are tripped.

The manual trip from the remote shutdown panels consists of two switches. Operation of a switch de-energizes the undervoltage coil and energizes the shunt coil in the breaker of the corresponding train.

The analog portion of the RTS consists of a portion of the process instrumentation system (PIS) and the nuclear instrumentation system (NIS). The PIS includes those devices that measure temperature, pressure, fluid flow, and level. The PIS also includes the power supplies, signal conditioning, and bistables that provide initiation of protective functions. The NIS includes the neutron-flux-monitoring instruments, including power supplies, signal conditioning, and bistables that provide initiation of protective functions.

The digital portion of the RTS consists of the solid-state logic protection system (SSLPS). The SSLPS takes binary inputs (voltage/no voltage) from the PIS and NIS channels corresponding to normal/trip conditions for plant parameters. The SSLPS uses these signals in the required logic combinations and generates trip signals (no voltage) to the undervoltage coils of the reactor trip circuit breakers. The system also provides annunciator, status light, and computer input signals that indicate the condition of the bistable output signals, partial and full trip conditions, and the status of various blocking, permissive, and actuation functions. In addition, the SSLPS includes the logic circuits for testing.

Analog signals derived from protection channels used for non-protective functions such as control, remote process indication, and computer monitoring are provided by isolation amplifiers located in the protective system cabinets. The isolation amplifiers are designed so that a short circuit, open circuit, or the application of credible fault voltages from within the cabinets on the isolated output portions of the circuit (non-protective side) will not affect the input signal. The signals obtained from the isolation amplifiers are not returned to the protective system cabinets.

7.2.2 Specific Findings

7.2.2.1 Lead, Lag, and Rate Time Constant Setpoints Used in Safety System Channels

Several safety system channels make use of lead, lag, or rate signal compensation to provide signal time responses consistent with assumptions in the analyses made in FSAR Chapter 15. The time constants for these signal compensations are adjustable setpoints within the analog portion of the safety system. The time constant setpoints will be incorporated into the plant Technical Specifications.

7.2.2.2 Turbine Trip Following a Reactor Trip

Credit is taken in the accident analysis for turbine trip on a reactor trip. The protection system trips the turbine following a reactor trip using the turbine emergency trip system. Redundant circuits used to trip the turbine are independently routed to and processed within the emergency trip system to provide two independent means of tripping the turbine. The circuits which traverse non-seismically qualified structures are isolated from the solid state protection system. The circuits are fully testable during full-power operation. The staff finds this design to be consistent with the functions importance to safety and, therefore, acceptable.

The staff will include in the plant Technical Specifications a requirement to periodically test these circuits.

7.2.2.3 Design Modification for Automatic Reactor Trip Using Shunt Coil Trip Attachment

The Westinghouse Owners Group (WOG) has submitted a generic design modification to provide automatic reactor trip system actuation of the breaker shunt trip attachments in response to Salem ATWS events. The staff has reviewed and accepted the generic design modification and has identified additional information required on a plant-specific basis. The applicant has not however, provided a response to Generic Letter 83-28 which established the requirements for this modification. The resolution of this matter will be addressed in a supplement to this report. This is an open item.

7.2.2.4 Trip Setpoint and Margins

The setpoints for the various functions in the reactor trip system are determined on the basis of the accident analysis requirements. As such, during any anticipated operational occurrence or accident, the reactor trip maintains system parameters with the following limits:

(1) minimum departure from nucleate boiling ratio of 1.30
(2) maximum system pressure of 2750 psi (absolute)
(3) fuel rod maximum linear power of 18.0 kW per foot

In performing its reviews, the staff audits detailed information on the methodology used to establish the Technical Specification trip setpoints and allowable values for the reactor protection system (including reactor trip and engineered safety feature channels) assumed to operate in the FSAR accident and transient analyses. The following information will be included:

(1) The trip setpoint and allowable value for the Technical Specifications.

(2) The safety limits necessary to protect the integrity of the physical barriers which guard against uncontrolled release of radioactivity.

(3) The values assigned to each component of the combined channel error allowance (e.g., modeling uncertainties, analytical uncertainties, transient overshoot, response time, trip unit setting accuracy, test equipment accuracy, primary element accuracy, sensor drift, nominal and harsh environmental allowances, trip unit drift), the basis for these values, and the method used to sum the individual errors. Where zero is assumed for an error, a justification that the error is negligible should be provided.

(4) The margin (i.e., the difference between the safety limit and the setpoint less the combined channel error allowance).

The detailed trip setpoint review will be performed as part of the staff's review of the plant Technical Specifications and will be completed before the operating license is issued.

7.2.2.5 NUREG-0737, Item II.K.3.10, Proposed Anticipatory Trip Modification

The design includes an anticipatory reactor trip upon turbine trip. Provisions are included to automatically block the reactor trip upon turbine trip at power levels below approximately 50% (P-9 interlock) where the condenser steam dump is capable of mitigating the reactor coolant system temperature and pressure transient without actuating pressurizer power-operated relief valves. A decision to trip the reactor following turbine trip at the 50% power level, noted in the TMI-2 Task Action Plan requirements, would involve only bistable setpoint changes and not instrument hardware changes. The staff finds that the design is, therefore, acceptable. The specific power level setpoint below which a reactor trip following a turbine trip is blocked will be reviewed and specified in the plant Technical Specifications.

7.2.2.6 NUREG-0737, Item II.K.3.12, Anticipatory Reactor Trip on Turbine Trip

As stated above, the design includes an anticipatory reactor trip on turbine trip. The staff has reviewed the design for conformance to BTP ICSB-26 and finds it acceptable.

7.2.3 Conclusions

Later.

7.3 <u>Engineered Safety Features Systems</u>

7.3.1 Engineered Safety Features Actuation System (ESFAS)

The ESFAS is a portion of the plant's protection system that monitors selected plant parameters and, on detection of out-of-limit conditions of these parameters, will initiate actuation of appropriate engineered safety feature (ESF) systems and essential auxiliary support systems equipment. The ESFAS includes both automatic and manual initiation of these systems. Also included with the ESF systems are the control systems that regulate operation of ESF systems following their initiation by the protection system.

The ESFAS is a functionally defined system and consists of:

(1) process instrumentation and control
(2) solid-state and relay logic

(3) ESF test circuits
(4) manual actuation circuits
(5) emergency generator load-sequence control logic

The ESFAS includes two distinct portions of circuitry: (1) an analog portion consisting of two to four redundant channels per parameter or variable to monitor various plant parameters such as reactor coolant and steam system pressures, temperatures, and flows and containment pressure, and (2) a digital portion consisting of two redundant logic trains that receive inputs from the analog protection channels and perform the logic to actuate the ESF equipment. The ESFAS is composed of NSSS circuits designed by Westinghouse and BOP circuits designed by Bechtel Power Corporation.

The actuation signals for each of the ESFAS functions are listed below. The numbers in parentheses after each actuation channel indicate the coincident logic; for example, two-out-of-four (2/4).

(1) Safety Injection

(a) manual (1/2)
(b) high-1 containment pressure (2/3)
(c) low compensated steamline pressure (2/3 in any line)
(d) low pressurizer pressure (2/4)

(2) Containment Spray and Containment Isolation (Phase B)

(a) manual (2/4)
(b) high-3 containment pressure (2/4)

(3) Containment Isolation (Phase A)

(a) safety injection (same as Item (1) above)
(b) high radiation (1/2)
(c) manual (1/2)

(4) Steam Line Isolation

(a) low compensated steamline pressure (2/3 in any line)
(b) high-2 containment pressure (2/3)
(c) high steamline pressure rate (2/3 in any line)
(d) manual (1/2 for all lines or 1/1 for each valve)

(5) Feedwater Line Isolation

(a) safety injection (same as Item (1) above)
(b) high steam generator level (2/3 in any generator)
(c) low T_{avg} (2/4) coincident with reactor trip

(6) Containment Ventilation Isolation

(a) safety injection (same as Item (1) above)
(b) high radiation (area monitors) (1/2)
(c) high radiation (particle and gas monitors) (1/3)
(d) manual
  (i) containment isolation (Phase A) (1/2)
  (ii) containment spray and isolation (Phase B) (2/4)

(7) Auxiliary Feedwater System Actuation

The motor-driven auxiliary feedwater pumps will be started on any of the following signals:

(a) safety injection (same as Item (1) above)
(b) low-low steam generator level (2/4 in any generator)
(c) loss of main feedwater pumps (2/2)
(d) station blackout signal (2/4 undervoltage at 4.16-kV bus)
(e) manual actuation (local or remote) (1/1)

The turbine-driven auxiliary feedwater pump will be started on any of the following signals:

(a) low-low steam generator level (2/4 in any steam generator)
(b) station blackout signal
(c) manual actuation (local or remote) (1/1)

(8) Control Room Isolation

(a) safety injection (same as Item (1) above)
(b) high intake radiation (1/2)
(c) high chlorine content (1/2)
(d) manual (1/2)

(9) Nuclear Service Cooling Water System Pump Start

(a) safety injection (same as Item (1) above)
(b) station blackout signal
(c) manual (1/1)

(10) Emergency Diesel Generator Startup

(a) safety injection (same as Item (1) above)
(b) station blackout signal
(c) manual (1/1)

(11) Containment Cooling System

(a) safety injection (same as Item (1) above)

(12) Containment Combustible Gas Control System

(a) manual (1/2)

(13) Component Cooling Water Pump Start

(a) safety injection (same as Item (1) above)
(b) station blackout signal
(c) manual (1/1)

(14) Control Building ESF Electrical Equipment Rooms HVAC and Auxiliary Relay Rooms ESF Air Conditioning

(a) safety injection (same as Item (1) above)
(b) manual

(15) Electrical Penetration Filter and Exhaust System

(a) containment ventilation isolation (same as Item (6) above)
(b) station blackout signal
(c) manual

(16) Auxiliary Building ESF HVAC System

(a) ESF room coolers

(i) safety injection (same as Item (1) above)
(ii) high room temperature (1/1)
(iii) manual

(b) piping penetration area filtration and exhaust

(i) containment ventilation isolation (same as Item (6) above)
(ii) manual

7.3.2 Engineered Safety Features (ESFs) and Essential Auxiliary Supporting (EAS) Systems Operation

7.3.2.1 Emergency Core Cooling System

The emergency core cooling system (ECCS) cools the reactor core and provides shutdown capability for pipe breaks in the reactor coolant system (RCS) that cause a loss of primary coolant greater than that which can be made up by the normal makeup system, for rod cluster control assembly ejection, for pipe breaks in the secondary coolant system and for steam generator tube failure. The primary function of the ECCS is to remove the stored and fission-product decay heat from the reactor core during accident conditions. The ECCS consists of the centrifugal charging pumps, residual heat removal (RHR) pumps, safety injection pumps, safety injection accumulators, boron injection and boron injection surge tanks, RHR heat exchangers, boron injection recirculation pumps, refueling water storage tank (RWST), the associated piping, valves, and instrumentation.

The ECCS provides shutdown capability for the accidents described above by injecting borated water into the RCS. The system's safety function can be performed with a single active failure (short term) or passive failure (long term). The emergency diesel generators supply power if offsite power is unavailable.

The safety injection signal will start the diesel generators and automatically initiate the following actions in the ECCS:

(1) starts centrifugal charging pumps
(2) opens RWST suction valves to charging pumps
(3) opens boron injection tank suction and discharge isolation valves
(4) closes normal charging path valves
(5) opens charging pump alternative miniflow valves
(6) closes charging pump miniflow valves
(7) starts safety injection and RHR pumps
(8) opens any closed accumulator isolation valves
(9) closes volume control tank outlet isolation valves
(10) stops boron injection recirculation pumps
(11) closes boron injection tank recirculation valves

The switchover from the injection mode to the recirculation mode is initiated automatically and completed manually by the operator from the main control room. During the injection mode, the RHR pumps deliver water to the reactor coolant system from the RWST. The water is taken from the containment sump during the recirculation mode. The transfer of the RHR pump suction to the containment sump is initiated automatically when the RWST level decreases below the low-level setpoint coincident with a safety injection signal. Four level measurement channels are provided and arranged in a two-out-of-four coincidence logic to open the two sump isolation valves. The operator then closes the RHR/RWST isolation valves. The RHR pumps continue to run during the switchover.

The two charging pumps and two safety injection pumps continue to take suction from the RWST following the automatic switchover described above. As part of the manual switchover procedure, the two charging pumps and the two safety injection pumps are realigned in series with the RHR pumps.

The four RWST level channels provide level indication in the control room and also generate low level alarms. The low level alarm coincident with the safety injection signal alerts the operator to complete the switchover as described above.

7.3.2.2 Containment Spray System

Two redundant trains of containment spray provide a spray of cold borated water from the upper regions of the containment to reduce containment pressure and temperature following a loss-of-coolant accident (LOCA), or a main steamline or a feedwater line break accident. Each train has an independent electrical power source backed up by a separate emergency diesel generator during the loss of offsite electrical power.

The containment spray system (CSS) operates in two sequential modes:

(1) spraying a portion of the contents of the refueling water storage tank mixed with NaOH from the spray additive tank into the containment atmosphere, using the containment spray pumps

(2) after the RWST has been drained, recirculating water from the containment sump through the containment spray pumps and heat exchangers back to the containment atmosphere

The CSS is provided with instrumentation and controls to permit the monitoring and actuation of the system from outside the containment. The containment spray pumps and valves are activated automatically by the containment high-3 pressure signal. Manual control switches are provided on the main control board. The status of pumps and valve positions are indicated in the control room. Abnormal conditions in the pump and valve operation and the spray water supply are alarmed on the main control board.

7.3.2.3 Containment Isolation System (Including Containment Ventilation Isolation)

The safety function of the containment isolation system is to automatically isolate the process lines penetrating the containment structure. The system is designed to limit the release of radioactive materials from the containment after an accident occurs.

The system is automatically actuated by signals (see Section 7.3.1) developed by the ESFAS. Containment isolation phase A isolates all process lines not required for ESF systems penetrating the containment. Containment ventilation signals isolate the containment purge system.

Containment isolation valves, which are equipped with power operators and are automatically actuated, may also be controlled individually by manual switches in the control room. Containment isolation valves with power operators are provided with an open/closed indication, which is displayed in the control room. All electric power supplies and equipment necessary for containment isolation are Class 1E.

7.3.2.4 Main Steamline Isolation

The main steamline isolation signal is generated on low steamline pressure, high-2 containment pressure or high steam pressure rate. A manual block permissive (P-11) is provided for the low steamline pressure signal for use during normal plant cooldowns and heatups. A high rate of decreasing steamline pressure is used to initiate main steamline isolation when the low steamline pressure signals are blocked. The main steam isolation valves are electro-hydraulic valves using stored gas (N_2) to drive a hydraulic piston for fast valve closure (5 seconds). Redundant isolation valves are provided for each steamline. Each main steam isolation valve is capable of being tested on-line by partial closure of the valve.

7.3.2.5 Feedwater Line Isolation

Feedwater line isolation is provided to terminate main feedwater following a pipe rupture or excessive feedwater event. Upon receipt of a safety injection (SI) signal or a steam generator high-high-level signal, all main feedwater pumps trip and all feedwater isolation and feedwater control and control bypass valves close. Upon receipt of either of the above two signals or low reactor coolant average temperature coincident with reactor trip, all main feedwater isolation valves, control valves, and control bypass valves close.

7.3.2.6 Combustible Gas Control System

The combustible gas control system controls the concentration of hydrogen gas inside the containment following a DBA. The system consists of (1) two redundant sets of hydrogen recombiners, (2) post-LOCA containment hydrogen purge system, (3) post-LOCA cavity hydrogen purge system, (4) containment hydrogen monitoring system, and (5) containment hydrogen mixing components.

Each hydrogen recombiner is powered from a separate safeguards bus and is manually controlled from a separate control panel located in the control building. The post-LOCA containment hydrogen purge system which backs up the recombiners is operated manually and is not safety related.

The post-LOCA cavity purge system prevents hydrogen pockets from forming in the reactor cavity. The system consists of two redundant fans, each powered from independent Class 1E power sources, which start on a safety injection signal.

The hydrogen monitoring system for the containment, consists of two redundant trains and provides for sampling and analyzing the containment environment for hydrogen concentration. The monitors are operated in standby during normal operation and provide indication and alarms in the control room.

7.3.2.7 Auxiliary Feedwater System

The function of the auxiliary feedwater system (AFWS) is to provide an adequate supply of water to the steam generators if the main feedwater system is not available. The AFWS consists of two motor-driven pumps and one turbine-driven pump with associated valves, controls, and instrumentation. Each motor-driven pump supplies water to two steam generators; the turbine-driven pump can be manually aligned to feed all four steam generators. Flow is controlled by a separate motor-operated flow control valve for each steam generator/pump combination. The auxiliary feedwater pumps are started automatically by the initiating conditions listed in Section 7.3.1, item (7). Each pump takes suction from either of two seismic Category I condensate storage tanks.

The motor-driven pumps can be operated manually from either the main control board or the shutdown panels. The turbine-driven pump can be operated manually from the main control board or from a local control panel. Also, the auxiliary feedwater flow can be controlled by adjusting the positions of the motor-operated control valves from either the control room or the shutdown panels.

7.3.2.8 Habitability Systems for the Control Room Envelope

The environmental habitability systems for the control room envelope (Unit 1 and Unit 2 control rooms) includes radiation shielding and monitoring; chlorine and smoke detection; air filtration and conditioning systems.

The function of the control room emergency ventilation and air-conditioning system is to maintain the environment in the control room envelope suitable for sustained occupancy throughout postaccident conditions. The system (for each unit) is divided into two redundant trains each connected to a separate and independent Class 1E power source. Each train consists of a moisture eliminator, an electric preheater, cooling coil, high-efficiency particulate air filters, charcoal absorbers, and fan.

High radiation or chlorine levels in the control room's outside air intakes or a safety injection signal will initiate the corresponding unit's system and isolate the non-emergency HVAC system. After initiation, one train may be placed manually in the standby mode.

The detection of smoke inside or outside of the control room is annunciated in the control room. Then the operator must isolate the control room manually.

7.3.2.9 Nuclear Service Cooling Water System

The nuclear service cooling water system provides cooling water for the containment coolers, control building essential chiller condensers, various ESF pump coolers, and component and auxiliary component cooling water heat exchangers. The nuclear service cooling water system consists of two redundant trains, each of which contains three 50% capacity pumps and associated piping and valves.

The third pump in each train is a backup pump for the other two normally operating pumps. This standby pump starts automatically on an auto-trip signal from any other pump in the same train or a low discharge header pressure with the two other pumps running in the same train. The same conditions provide confirmatory start signals to the normally operating pumps.

During normal operations and safe shutdown only one train is required. The nuclear service cooling water system is designed to meet the single-failure criterion.

On receipt of an SI or loss-of-offsite-power signal, all preferred pumps receive automatic start signals. If one of the preferred pumps does not start, the standby pump in the same train receives a subsequent start signal. An SI signal isolates the cooling tower blowdown lines, too. Manual initiation is also provided from the control room or from the remote shutdown panels.

7.3.2.10 Component Cooling Water System

The component cooling water system provides cooling water for the spent fuel pool during all operating modes and for the residual heat removal system during shutdown and emergency conditions. The system also serves as an intermediate system between the RCS and the nuclear service cooling water system. The component cooling water system consists of two redundant trains, each containing three 50% capacity pumps, one heat exchanger, one surge tank, one chemical addition tank, and associated piping, valves, and instrumentation.

The third pump in each train is a backup for the other two normally operating pumps. This standby pump starts automatically on an auto-trip signal from any pump in the same train or a low discharge header pressure with the two other pumps running in the same train. The same conditions provide confirmatory start signals to the normally operating pumps.

During normal operation, only one train is operated and only one train is required to bring the plant to cold shutdown conditions.

On receipt of an SI or loss-of-offsite power signal, all preferred pumps receive automatic start signals. If one of the preferred pumps does not start, the standby pump in the corresponding train receives a subsequent start signal. Manual initiation is also provided from the control room or from the remote shutdown panels.

Low-low-level switches on the surge tank, each set at a different level, stop one pump at a time when the level reaches the setpoints.

7.3.2.11 Containment Cooling System

The containment cooling system consists of eight separate 25% capacity fan cooler units located inside the containment. The system is divided into two trains of four fan cooler units each. Each train is supplied cooling water and electrical power from the corresponding train of nuclear service cooling water and Class 1E power systems. During normal operation only four fans operate at high speed.

Upon receipt of an SI signal, all fans are started at low speed. The containment air cooling units can be stopped or started from the control room or from the remote shutdown panels.

7.3.2.12 Emergency Diesel Generators

Each train of the 4,160-V ac essential auxiliary power system is supplied with emergency standby power from an independent diesel generator. Each diesel generator can be manually started for test and maintenance purposes from the control room or from the local diesel control panel.

When the diesel generators receive an emergency start signal, all manual modes of operation are overridden. If a diesel is in the maintenance mode, a starting signal is inhibited. An annunciator in the control room is provided to alert the operator whenever a diesel is in the maintenance mode. Protective trips also are provided for the diesels that are not bypassed by starting signals. These trips are annunciated in the control room.

7.3.2.13 Safety-Related Ventilation Systems

The applicant has identified that the following systems are safety related:

(1) control building ESF electrical equipment room HVAC
(2) control building electrical penetration area filtration and exhaust
(3) control building ES, HVAC equipment room ventilation
(4) control building auxiliary relay room ESF HVAC
(5) fuel-handling building ventilation
(6) auxiliary building emergency ventilation
(7) diesel generator building ventilation
(8) auxiliary feedwater pumphouse ESF HVAC
(9) electrical tunnel ESF HVAC

Several of these systems receive ESFAS signals as listed in Section 7.3.1. These systems are evaluated in Section 9.4 of this SER.

7.3.2.14 Emergency Onsite Power Supply System

The emergency onsite power supply system consists of two 4.16-kV diesel generators, two 4.16-kV buses, various 480-V buses, motor control centers, and 208/120-V power panels. There are four 120-V ac safety-related vital bus power supplies for safety-related vital instrumentation and control loads. The evaluation of the emergency onsite power supply system is addressed in Section 8.3 of this SER.

7.3.2.15 Emergency Diesel Generator Associated Systems

The applicant has identified that the emergency diesel generator cooling water, combustion air intake and exhaust, fuel oil storage and transfer, starting, and lubrication systems are safety-related systems. These systems are evaluated in Section 9.5 of this SER.

7.3.3 Specific Findings

7.3.3.1 NUREG-0737, Item II.E.1.2, AFWS Automatic Initiation and Flow Indication

The automatic system used to initiate the operation of the auxiliary feedwater system is part of ESFAS. The redundant actuation channels that provide signals to the pumps and valves are physically separated and electrically independent. Redundant trains are powered from independent Class 1E power sources. The initiation signals and circuits are testable during power operation, and the test requirements will be included in the plant Technical Specifications. Manual initiation and control can be performed from the main control board or the emergency shutdown panel. No single failure within the manual or automatic initiation system for the auxiliary feedwater system will prevent initiation of the system by manual or automatic means. The environmental qualification is addressed in Section 3.11 of this SER.

Redundant auxiliary feedwater flow instrument channels are provided for each steam generator. Each channel is powered from a separate Class 1E power source. Auxiliary feedwater flow indicators are located at the main control board and the emergency shutdown panel. The staff concludes that the design satisfies the requirements of NUREG-0737, Item II.E.1.2.

7.3.3.2 Test of Engineered Safeguards P-4 Interlock

On November 7, 1979, Westinghouse notified the Commission of a failure that could exist, undetected, in the engineered safeguards P-4 interlocks. Test procedures were developed to detect failures that might occur. The procedures require the use of voltage measurements at the terminal blocks of the reactor trip breaker cabinets.

To minimize the possibility of accidental shorting or grounding of safety system circuits during testing of the P-4 interlocks, the applicant is permanently installing a voltage indicator that has a switch across the terminals. The staff finds this acceptable pending discussion in the FSAR and confirmation of the installation. This is a confirmatory item.

7.3.3.3 Undetectable Failure in Online Testing Circuitry for Engineered Safeguards Relays

On August 6, 1982, Westinghouse notified the staff of a potentially undetectable failure in online test circuitry for the master relays in the engineered safeguards systems. The undetectable failure involves the output (slave) relay continuity proving lamps and their associated shunts which are provided by test pushbutton contacts. If after testing, a shunt is not provided for any proving lamp because a switch contact fails, any subsequent safeguards actuation could cause the lamp to burn open before its associated slave relay is energized. This would then prevent actuation of any associated safeguards devices on that slave relay. Westinghouse has provided test procedures that ensure that the slave relay circuits operate normally when testing of the master relays is completed.

Until an acceptable circuit modification is installed, the staff will require plant Technical Specifications to include monthly (in lieu of quarterly) testing of slave relays. These tests should be performed immediately following the monthly testing of associated master relays.

7.3.3.4 Level Measurement Errors Resulting From Environmental Temperature Effects on Level Instrument Reference Legs

The staff asked the applicant to evaluate the effects of high temperature in reference legs of water-level measurement systems from high-energy-line breaks. This issue was addressed for operating reactors in IE Bulletin 79-21. The applicant has not responded to this issue. This is an open item.

7.3.3.5 Steam Generator Level Instrumentation

During the staff's review of the steam generator level instrumentation, a conflict was found between the information provided by the applicant and FSAR

Figure 7.2.1-1 (Sheet 7) and Table 7.3.1-2. The FSAR shows the hi-hi steam generator level (P-14) logic to be two-out-of-three which should be, according to the applicant, two-out-of-four. This item is confirmatory, subject to FSAR revision to eliminate this conflict.

7.3.3.6 IE Bulletin 80-06 Concerns

IE Bulletin 80-06 requests a review of all systems serving safety-related functions to ensure that no device will change position solely because an ESF actuation signal is reset. The applicant was asked to respond to IE Bulletin 80-06 (FSAR Q420.3).

The staff has reviewed the applicant's response, contained in Amendment 8 of the FSAR, and finds that the applicant did not identify any component that would not remain in a safety state following reset. A test, to verify that the actual installed instrumentation and controls are in compliance with the requirements of IE Bulletin 80-06, will be conducted as part of the preoperational tests. On the basis of the applicant's commitment to such testing, the staff considers this issue resolved subject to confirmation that the test has been done. This is a confirmatory item.

7.3.3.7 Auxiliary Feedwater System

During the staff's review of the auxiliary feedwater system for Vogtle, several concerns were identified and are summarized as follows:

(1) Normally locked open valves may be manually closed locally during system testing to block flow to the steam generators. Valve position and bypassed/inoperable status indication are not provided in the control room for these valves.

(2) The trip and throttle valve may be manually closed, blocking steam flow to the turbine-driven pump. Bypassed/inoperable status indication is not provided in the control room for this valve.

This is an open item pending the applicant's response.

7.3.3.8 Override of Isolation Signals

Control room isolation signals are provided from radiation monitors and chlorine detectors. Containment isolation and containment ventilation isolation signals are provided from radiation monitors. Fuel-handling-building isolation signals are provided from radiation monitors and differential pressure channels. The design for these safety functions is based on one-out-of-two and one-out-of-three logic, and a single failure within an instrument channel can result in a spurious actuation of the system.

During the staff's review, a concern was raised that the use of the reset/override control switches to override an initiation signal from a faulty detector or channel would preclude subsequent initiation signals from non-failed channels. This item is open pending the applicant's response.

7.3.3.9 Isolators Used in the BOP Design

During the staff's review of the design criteria and tests performed on isolation devices used in BOP systems, information was unavailable for adequate review. In order for the staff to complete its review, the applicant must supply the following information:

(1) For each type of device used to accomplish electrical isolation in the BOP scope, describe the specific testing performed to demonstrate that the device is acceptable for its application(s). This description should include elementary diagrams, where necessary, to indicate the test configuration and how the maximum credible faults were applied to the devices.

(2) Data to verify that the maximum credible faults applied during the test were the maximum voltage/current to which the device could be exposed, and to define how the maximum voltage/current was determined.

(3) Data to verify that the maximum credible fault was applied to the output of the device in the transverse mode (between signal and return) and that other faults were considered (i.e., open and short circuits).

(4) Define the pass/fail acceptance criteria for each type of device.

This is an open item.

7.3.3.10 Auxiliary Relays Used With No-Go Tested Slave Relays

During the staff's review of schematics for automatic actuation circuitry for ESF systems, a concern was raised about the testability of final actuation devices or actuated equipment that has been assigned "No-Go" tested (continuity tested) slave relays which drive auxiliary relays in lieu of the final devices themselves. The staff has asked the applicant to review the testability of this type of circuit arrangement. This is an open item.

7.3.3.11 Electrical Tunnel Ventilation System

During the staff's review of Vogtle's environmental control systems, a concern was raised about the adequacy of a single temperature detector (per train) being used to initiate (on high temperature) the electrical tunnel ventilation system and to provide the high temperature alarm. In order for the staff to complete its review, the applicant must supply the following information:

(1) Section 9.4.9.2.5 of the FSAR states that electrical tunnel temperature indication is provided. The staff's review of system schematics indicates otherwise. To resolve this conflict, the applicant must provide detailed information on the indications available to the operator for assessing the environment in the electrical tunnel and the status of the actuated safety equipment.

(2) Information about the surveillance of the instrumentation used to initiate the electrical tunnel ventilation system.

(3) Information about the safety significance of the failure of the ventilation system in the electrical tunnel under high-temperature conditions.

This is an open item.

7.3.3.12 Control Room Ventilation Isolation

During the staff's review of the logic diagrams for control room ventilation isolation, conflicts were found between FSAR figures and information provided during discussions with the applicant. The following information must be provided for the staff to complete its review:

(1) Discuss the safety function of the smoke detector (1-ASH-12166) shown in FSAR Figure 7.3.6-1.

(2) Discuss the conflict between FSAR Figure 7.3.6-1 Sheet 1 and Sheet 3 as related to SI-A and its actuation of valve A-HY-12162A.

(3) Discuss the safety function provided by the hydrogen sulfide detectors shown in FSAR Figure 7.3.6-1 Sheet 7.

(4) Discuss the input "Control Room Isolation" shown in FSAR Figure 7.3.6-1 Sheet 15.

This is an open item.

7.3.4 Conclusions

Later.

7.4 <u>Systems Required for Safe Shutdown</u>

7.4.1 Systems Description

This section describes the equipment and associated controls and instrumentation of systems required for safe shutdown. It also describes controls and instrumentation outside the main control room that enable safe shutdown of the plant in case the main control room needs to be evacuated.

7.4.1.1 Safe Shutdown Systems

By appropriately aligning selected systems that serve a variety of operational functions under normal operating conditions, the plant can be placed and maintained in a safe shutdown condition. For safe shutdown the systems must

(1) prevent the reactor from achieving criticality

(2) provide an adequate heat sink so that the design and safety limits of the reactor coolant system temperature and pressure are not exceeded

To perform the above functions, the systems required for safe shutdown must have the following capabilities:

(1) boration
(2) adequate supply of auxiliary feedwater
(3) residual heat removal

In addition to the operation of systems required to provide the above functions to achieve and maintain safe shutdown, the following conditions are applicable:

(1) The turbine is tripped (in addition to automatic trip this can be accomplished manually at the turbine as well as from the control room).

(2) The reactor is tripped (in addition to automatic trip this can also be accomplished manually at the reactor trip switchgear as well as from the control room).

(3) All automatic protection and control systems are functioning (discussed in Sections 7.2 and 7.3).

The monitoring indicators for maintaining hot standby are as follows:

(1) water level for each steam generator
(2) pressure for each steam generator

(3) pressurizer water level
(4) pressurizer pressure
(5) primary coolant hot and cold leg temperatures
(6) auxiliary feedwater flow for each steam generator
(7) condensate storage tank level
(8) source range flux

These monitors indicate in the main control room and also on the shutdown panels.

The systems used for safe shutdown include the following:

(1) reactor coolant
(2) main steam
(3) auxiliary feedwater
(4) chemical and volume control
(5) component cooling water
(6) nuclear service cooling water
(7) residual heat removal
(8) containment fan coolers
(9) supportive HVAC

7.4.1.1.1 Reactor Coolant System

The reactor coolant system transfers residual heat from the core to the steam generators. The reactor core is at a lower elevation than the steam generators, ensuring that heat will flow from the reactor core to the steam generators via natural circulation.

7.4.1.1.2 Main Steam System

The main steam system consists of main steam piping, power-operated atmospheric steam relief valves (PORVs), safety valves, and main steam isolation valves. The system is used for maintaining a hot standby condition and for plant cooldown to the temperature and pressure at which the RHR can be placed in operation. Core residual heat and RCS sensible heat can be removed by use of the safety grade PORVs if the main condenser is not in service.

7.4.1.1.3 Auxiliary Feedwater System

See Section 7.3 for a discussion of the auxiliary feedwater system.

7.4.1.1.4 Chemical and Volume Control System

The CVCS is designed to:

(1) maintain a predetermined water level in the pressurizer

(2) maintain seal water injection flow to the reactor coolant pumps

(3) control reactor coolant water chemistry conditions, radioactivity level, and soluble chemical neutron absorber concentration

(4) provide emergency core cooling

(5) provide means for filling and draining the reactor coolant system

The safety-related part of the CVCS consists of the high-head SI/charging pumps and their associated valves and piping used for emergency core cooling. For safe shutdown the CVCS provides a safety-grade means to borate the RCS via the boric acid pumps and tanks and the charging pumps and their valves and piping. Additionally, the CVCS provides a safety-grade means for RCS inventory control with the aforementioned equipment and the reactor head letdown system.

7.4.1.1.5 Component Cooling Water System

See Section 7.3 for a discussion of the component cooling water system.

7.4.1.1.6 Nuclear Service Cooling Water System

See Section 7.3 for a discussion of the nuclear service cooling water system.

7.4.1.1.7 Residual Heat Removal System

The residual heat removal system (RHRS) tranfers heat from the RCS to the component cooling water system during plant cooldown from hot standby to cold shutdown and controls the temperature of the primary coolant during cold shutdown. The RHRS consists of two redundant, separate, and independent trains each of which is powered from a different Class 1E bus and is capable of maintaining the cooling function as designed, even if a major single failure (such as a failure of a pump, valve, or heat exchanger) occurs.

7.4.1.1.8 Containment Fan Coolers

See Section 7.3 for a discussion of the containment fan coolers.

7.4.1.1.9 Supportive HVAC Systems

See Section 9.4 for an evaluation of these systems.

7.4.1.2 Remote Shutdown Capability

If the control room ever needs to be evacuated, the operators can establish and maintain the plant in a hot-shutdown condition from outside the control room by using controls and indicators located at the shutdown panels and the auxiliary feedwater pump turbine control panel. Each of the two auxiliary shutdown panels is located in a separate locked room to restrict access. Selector switches on the shutdown panels allow the operator to transfer control of the equipment required for shutdown from the control room to the shutdown panels. Transfer of control is alarmed in the control room. A loss-of-control-room test will be conducted to demonstrate the remote shutdown capability. Cold-shutdown conditions can also be reached from outside the control room.

7.4.2 Conclusions

Later.

7.5 Information Systems Important to Safety

7.5.1 Systems Description

Indicators, annunciators, recorders, and lights are used to provide information to the operator during postaccident monitoring and normal operating conditions. The information is displayed on the operator's console, the various control boards in the control room, and the remote shutdown panels. This information is provided for systems that include the following functions:

(1) reactor trip
(2) engineered safety features
(3) safe shutdown

7.5.1.1 Safety-Related Display Instrumentation

The applicant has conducted an analysis to identify the appropriate set of variables so the operator can monitor conditions in the reactor coolant system, the secondary heat removal system, the containment, the engineered safety features systems, and the safe shutdown systems. The safety-related display instrumentation provides the information for the operator needs so that he can perform the required manual safety functions following a reactor trip It provides information for all operating conditions, including anticipated operational occurrences, accidents, and postaccident conditions.

Table 7.5.2-1 in the applicant's FSAR identifies the safety-related display instrumentation and includes the following information for each variable:

(1) instrument range
(2) environmental qualification
(3) seismic qualification
(4) display methodology
(5) type and category (according to the definition in RG 1.97, Rev. 2)

Environmental qualification is evaluated in Sections 3.10 and 3.11 of this SER.

7.5.1.2 Bypass and Inoperable Status Indication

Automatic bypass or inoperable status indication is provided in the control room for each redundant portion of a safety-related system. The indication of bypassed or inoperable status was designed following the guidance provided by RG 1.47 Revision 0.

The function bypass indicators receive their inputs from valve position limit switches, circuit breaker auxiliary contacts, switch contacts, relays, etc., indicative of function inoperability. Each bypass alarm can also be actuated manually in the control room.

Bypass indications are tested by a test switch that simulates operation of the remote contacts to verify proper operation of the circuits.

The design and installation of the bypass and inoperable status indication is such that a failure in an indication circuit will have no adverse affect on the function monitored or on any of the other functions monitored by the bypass panel.

Bypass indication is provided in the control room for each train of the following systems:

- residual heat removal
- auxiliary feedwater
- safety injection
- containment spray
- chemical and volume control
- essential chilled water
- auxiliary component cooling water
- nuclear service cooling water
- component cooling water
- spent fuel pit cooling
- control building/control room HVAC
- containment building air cooling
- auxiliary building ESF equipment room coolers

- control building ESF electrical equipment room HVAC
- diesel generator
- fuel-handling-building ESF HVAC
- piping penetration filtration and exhaust
- control building electrical penetration filtration and exhaust
- diesel generator standby power
- containment hydrogen recombiner
- bypassed/inoperable status indication

7.5.2 Specific Findings

7.5.2.1 Emergency Response Capability - RG 1.97, Revision 2, Requirements

Generic Letter No. 82-33 included additional clarification regarding RG 1.97, Revision 2, relating to the requirements for emergency response capability. The applicant's letter dated April 14, 1983, provided the response to the part of Generic Letter No. 82-33 pertaining to RG 1.97, Revision 2. The staff is reviewing the applicant's method of implementing RG 1.97, Revision 2, and the applicant's supporting technical justification for any proposed alternatives. The results of that review will be included in a supplement to this SER. This is an open item.

7.5.2.2 NUREG-0737, Item II.F.1, Accident-Monitoring Instrumentation Positions (4), (5), and (6)

Positions (4), (5), and (6) of this TMI-2 Task Action Plan item require installation of the extended-range containment pressure monitors, containment water-level monitors, and containment hydrogen-concentration monitors. Table 7.5.2-1 of the FSAR provides information for positions (4), (5), and (6). The staff has reviewed this information and finds that the indication provided for containment pressure, containment water level and containment hydrogen concentration satisfies the requirements of this TMI-2 Task Action Plan item and are, therefore, acceptable. This issue is considered closed.

7.5.2.3 NUREG-0737, Item II.D.3, Direct Indication of Relief and Safety Valve Positions

The applicant has provided detailed information on conformance to this TMI-2 Task Action Plan item in Table 7.5.2-1 of the FSAR. The staff has reviewed this information and finds that direct indication of the plant's primary relief and safety valve positions are in accordance with the requirements of this TMI-2 Task Action Plan item with the following exceptions:

(1) Information must be provided for the staff's review on what alarms have been provided in conjunction with valve position indication. This is an open item.

(2) Table 7.5.2-1 of the FSAR indicates that the power supply for the primary safety valve status is not Class 1E. The applicant has stated that the FSAR is in error and that it should state "Class 1E". This item is confirmatory pending FSAR revision to eliminate this error.

7.5.2.4 Bypass and Inoperable Status Panel

The applicant has provided detailed information on conformance to RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," and Branch Technical Position (BTP) ICSB-21, "Guidance for Application of Regulatory Guide 1.47," in FSAR Section 7.5.5. The staff has reviewed the information provided and finds that the design of Vogtle's bypassed and inoperable status panel conforms to the above guidance with the following exceptions:

(1) FSAR Section 7.5.5.2 states that the bypassed and inoperable status is automatically indicated for auxiliary or supporting systems that must be operable for the protection systems and the systems they actuate to perform their safety-related functions. Recent discussions with the applicant indicate that the status of supporting systems, when inoperable, are manually entered into the dependent system's inoperable status indication. The staff requests that the applicant provide descriptive information on conformance to Position C.2 of RG 1.47 to resolve this conflict. This is an open item.

(2) FSAR Section 7.5.5.3 states that the bypassed and inoperable status for the fuel-handling building's ESF HVAC system, which is shared by both Unit 1 and Unit 2, is only indicated in the Unit 1 control room. The staff asked the applicant to justify deviation from the guidance of Position B.2 of BTP ICSB 21. The applicant, in response, committed to have bypassed and inoperable status for this shared system indicate in both control rooms. On confirmation of installation and FSAR revision, the staff finds this acceptable.

7.5.2.5 IE Bulletin 79-27, Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation

The staff asked the applicant to review the adequacy of emergency operating procedures to be used by control room operators to attain cold shutdown on loss of any Class 1E or non-Class 1E bus supplying power to safety-related or nonsafety-related instrument and control systems. This issue was addressed for operating reactors in IE Bulletin 79-27.

The staff has not completed its review of the applicant's response, contained in Amendment 8 of the FSAR. This is an open item.

7.5.2.6 Freeze Protection for Instrumentation Sensing and Sampling Lines

In the past there have been many occurrences of frozen instrumentation and sampling lines. IE Bulletin 79-24 requested a review of plant designs to ensure that adequate measures had been taken to prevent safety-related process, instrument, and sampling lines from freezing during extremely cold weather. During the staff's review of Vogtle's environmental control systems which ensure that instrumentation sensing and sampling lines are protected from freezing, a concern was raised about the adequacy of the heat tracing used in the nuclear service cooling water system. The applicant has been asked to respond to this concern. This is an open item.

7.5.3 Conclusions

Later.

7.6 Interlock Systems Important to Safety

7.6.1 Systems Description

The systems described in this section operate to reduce the probability of occurrence of specific events or to maintain safety systems in a state to ensure their availability when required.

7.6.1.1 Residual Heat Removal Isolation Valve Interlocks

The RHRS consists of two residual heat exchangers, two pumps, and the associated piping, valves, and instrumentation necessary for operational control. The inlet lines to the RHRS are connected to the hot legs of two reactor coolant loops, and the return lines are connected to the cold legs.

The RHRS is a low-pressure system and is isolated during normal operation from the high-pressure reactor coolant system. The isolation is provided by two motor-operated valves in series in each of the two residual heat removal pump suction lines. Interlocks prevent the valves from opening until the reactor coolant system pressure is below a predetermined value (approximately 425 psig). Once opened, the valves will close automatically if the pressure increases above a preset value. The position of the valves is indicated on the main control board by lights actuated by the valve limit switches.

Two pressure transmitters, powered from separate emergency power sources and manufactured differently, are used to derive the isolation valve interlocks. Each of the four motor-operated valves are powered from a separate inverter.

7.6.1.2 Cold-Leg Accumulator, Motor-Operated Valve Interlocks

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. During normal operation each accumulator is isolated from the reactor coolant system by two check valves in series. Should the RCS pressure fall below the accumulator pressure, the check valves open and borated water is forced into RCS. To prevent injection of borated water at low-pressure operation during shutdown and startup, each of the accumulators is provided with a motor-operated isolation valve in series with the check valves. The operator closes the valve shortly after the RCS is depressurized below the safety injection unblock setpoint.

The motor-operated isolation valves are controlled by switches on the main control board and are interlocked as follows:

(1) They open automatically on receipt of a safety injection signal ("S").

(2) They open automatically whenever the RCS pressure is above the safety injection unblock pressure (P-11 interlock).

(3) They cannot be closed as long as an "S" signal is present.

After the RCS pressure is decreased during shutdown and the motor-operated isolation valves are closed, power to the valves is disconnected to prevent accidental operation. The power to the valves is also disconnected after the valves are opened during normal power operation to prevent accidental closing. Lights, actuated by the valve limit switches, provide valve position indication in the control room. Alarms, operated by the valve stem limit switch, are activated when a valve is not fully open and the pressure in the system is above the safety injection unblock level.

7.6.1.3 RCS Overpressure Protection During Low-Temperature Operation

Later.

7.6.2 Specific Findings

7.6.2.1 RCS Overpressure Protection During Low-Temperature Operation

During the staff's review of the automatic actuation logic for the pressurizer PORVs, conflicts were found between FSAR Figure 7.2.1-1 (Sheets 17 and 18) and corresponding Westinghouse drawings used for the Vogtle design. The applicant has been asked to correct the inconsistencies and to submit detailed information for the staff's review. This is an open item.

7.6.2.2 NUREG-0737, Item II.K.3.1, Installation and Testing of Automatic Power-Operated Relief Valve Isolation System

This TMI-2 Action Plan item requires all PWR licensees to provide a system that uses a PORV block valve to protect against a small-break loss-of-coolant accident. The system would automatically close the block valve when the reactor coolant system pressure decays after the PORV opens. Such a control system is not required if studies provided in response to Item II.K.3.2 show that the probability for the PORV sticking open is sufficiently small.

The applicant has installed automatic actuation logic to control the PORV block valves. Because of the conflicts in design documentation discussed in Section 7.6.2.1, the staff's review and evaluation are pending until the applicant provides final design information. This is an open item.

7.6.2.3 Instrumentation for Process Measurements Used for Safety Functions

During the staff's review of several interlock systems which isolate safety systems from nonsafety systems (or portions of systems) discussed in FSAR Section 7.6.6, several concerns were identified which relate to design features for certain instrumentation used to initiate safety interlock functions. In order for the staff to complete its review, the following information must be provided:

(1) alarms and/or indication that inform the operator that a high flow signal has initiated the isolation functions discussed in FSAR Sections 7.6.6.4 and 7.6.6.6

(2) the capability for testing each safety function listed in FSAR Sections 7.6.6.1, 7.6.6.2, 7.6.6.3, 7.6.6.4, and 7.6.6.6 during normal plant operation and confirmation that Vogtle's Technical Specifications will include periodic surveillance of these safety functions and their instrument channels

This is an open item.

7.6.3 Conclusions

Later.

7.7 Control Systems

The general design objectives of the plant control systems are:

(1) to establish and maintain power equilibrium of the primary and secondary system during steady-state unit operation

(2) to constrain operational transients so as to preclude the unit from tripping and to reestablish steady-state unit operation

(3) to provide the reactor operator with monitoring instrumentation that indicates all required input and output control parameters of the systems and provides the capability of assuming manual control of the system

7.7.1 System Descriptions

(1) Reactor Control System

The reactor control system enables the plant to accept a step load increase or decrease of 10% and a ramp increase or decrease of 5% per minute within the load range of 15% to 100% without reactor trip, steam dump, or pressurizer relief actuation (subject to possible xenon limitations). The system also maintains the reactor coolant average temperature within established limits by generating the demand signals for moving the control rods.

(2) Rod Control System

The rod control system modulates the reactor power by automatic or manual control of full-length control rod banks. The system receives rod speed and direction signals from the reactor control system. Manual control is provided to move a control bank in or out at a predetermined fixed speed. An interlock derived from measurements of turbine impulse chamber pressure prevents automatic control when the turbine load is below 15%.

The shutdown banks are moved to the fully withdrawn position by manual control before criticality is reached. These rods remain in that position during normal operation. The control banks are the only rods that are manipulated under automatic control. Each control bank is divided into two groups to obtain smaller incremental reactivity changes per step. All rod control cluster assemblies (RCCAs) in a group move simultaneously. Each rod cluster control assembly indicates individually.

(3) Plant Control Signals for Monitoring and Indication

(a) Nuclear Instrumentation Power Range System - Four channels are provided. Each of the channels uses a dual-section ionization chamber as a neutron-flux detector. The currents from the ionization chambers are used to measure the power level, axial flux imbalance, and radial flux imbalance.

(b) Rod Position Monitoring System - Two separate systems are provided, digital rod position indication and the demand position system. The digital rod position indication system measures the actual position of each rod. The demand position system counts pulses generated in the rod drive control system to provide a readout of the demanded bank position.

(c) Control Bank Rod Insertion Monitoring - This system warns the operator about excessive rod insertion. The "low" alarm alerts the operator to an approach to the rod insertion limits requiring boron addition by following normal procedures with the chemical and volume control system. The "low-low" alarm alerts the operator to a need for immediately adding boron by any one of several alternate methods.

(d) Rod Deviation Alarm - The rod deviation alarm is generated by the digital rod position indication system whenever any individual control rod position deviates from the position of other rods in the same bank by a preset limit.

(e) Rod Bottom Alarm - A "rod bottom rod drop" alarm is generated for each of the rods by the digital rod position indication system.

(4) Plant Control System Interlocks

(a) Rod Stops - Prevent abnormal power conditions which could result from excessive control rod withdrawal initiated by either a control system malfunction or operator violation of administrative procedures. The interlocks are generated by signals from the neutror flux, overtemperature ΔT, overpower ΔT, and turbine impulse chamber pressure measurement channels.

(b) Automatic Turbine Load Runback - Prevents high power operation which, if reached, would initiate reactor trip. Signals from overtemperature ΔT and overpower ΔT measurement channels are used to initiate automatic turbine load runback when an overpower or overtemperature condition is approached.

(c) Turbine Loading Stop - Limits turbine loading in a power transient resulting from a reduction in reactor coolant temperature. The interlock is cleared by an increase in coolant temperature that is accomplished by reducing the boron concentration in the coolant.

(5) Pressurizer Pressure Control

Pressure is controlled in the reactor coolant system by using either the heaters (in the water region) or the spray (in the steam region) of the pressurizer plus steam relief for large transients.

The electrical immersion heaters are located near the bottom of the pressurizer. A portion of the heater group is proportionally controlled to correct small pressure variations. These variations are caused by heat losses, including heat losses from a small continuous spray. The remaining (backup) heaters are turned on when the pressurizer pressure control signal demands approximately 100% proportional heater power.

The spray nozzles are located on the top of the pressurizer. A small continuous spray is normally maintained to reduce thermal stresses and thermal shock and

(6) Pressurizer Water Level Control

The pressurizer operates by maintaining a steam cushion over the reactor coolant. As the density of the reactor coolant adjusts to the various temperatures, the steam-water interface moves to absorb the variations with relatively small pressure disturbances.

The chemical and volume control system maintains a programmed pressurizer water level. During normal plant operation, the charging flow varies to produce the flow demanded by the pressurizer's water-level controller. The pressurizer's water level is programmed as a function of coolant average temperature; the highest average temperature (auctioneered) is used. The water level in the pressurizer decreases as the load is reduced from full load. This results when coolant contracts (upon following the programmed reduction of coolant temperature) as plant operation moves from full power to low power. The programmed level is designed to match as nearly as possible the changes in level that result from the reduced coolant temperature.

To control water level in the pressurizer during startup and shutdown operations, the charging flow is regulated manually from the main control room.

(7) Steam Generator Water Level Control

Each steam generator is equipped with a three-element feedwater-flow controller which maintains a programmed water level (a function of turbine load). The three-element feedwater controller regulates the feedwater valve by continuously comparing the feedwater flow signal, the water level signal, the programmed level, and the pressure-compensated steam flow signal.

Continued delivery of feedwater to the steam generators is required as a sink for the heat stored and generated in the reactor following a reactor trip and turbine trip. An override signal closes all feedwater valves when the average coolant temperature is below a set value and the reactor has tripped. The feedwater control system can be overridden manually at all times.

When the plant is at very low power, a secondary automatic control system is utilized. This system uses the setpoint programmed for the steam generator's water level in conjunction with the power range neutron flux signal to control the position of the bypass valves that parallel the main feedwater control valves. At approximately 25% power, the operator initiates switchover to this secondary system.

(8) Steam Dump Control System

The steam dump system, together with the rod control system, is designed to accept a 50% loss of net load without tripping the reactor. The system functions automatically by bypassing the steam directly to the condenser and atmosphere to maintain an artificial load on the primary system. The rod control system can then reduce the temperature in the reactor to a new equilibrium value without causing conditions of overtemperature and/or overpressure.

A demand signal for the load-rejection steam dump controller is generated if the difference between the reference average temperature based on turbine impulse chamber pressure and the lead/lag compensated auctioneered average temperature exceeds a preset value. To prevent actuation of steam dump on small load perturbations, an independent load rejection sensing circuit is provided. This circuit senses the rate of decrease in the turbine load as detected by the pressure in the turbine's impulse chamber and blocks the steam dump, unless the rate exceeds a preset value.

After a reactor trips, the load-rejection steam dump controller is deactivated and the plant-trip steam dump controller becomes active. The demand signal for this controller is generated if the difference between the lead/lag compensated auctioneered average temperature and the no-load reference average temperature exceeds a preset value. As the error signal reduces in magnitude after the dump valves trip, the dump valves are modulated by the plant-trip controller to regulate the rate of heat removal and thus gradually establish the equilibrium hot shutdown condition.

The residual heat is removed during a shutdown ~~is accomplished~~ by the steam-pressure controller which regulates the steam flow to the condensers according to measured steam pressure. This controller operates a portion of the same steam dump valves to the condenser which are used after load rejection or plant trip.

(9) Incore Instrumentation

The incore instrumentation system consists of chromel-alumel thermocouples at fixed core outlet positions and movable miniature neutron detectors at selected fuel assemblies. The thermocouple readings are monitored by the plant safety monitoring system. The movable detectors can perform flux mapping at various core quadrants to obtain a flux map for any region of the core.

7.7.2 Specific Findings

7.7.2.1 NUREG-0737, Item II.K.3.9, Proportional Integral Derivative Controller Modification

This TMI-2 Task Action Plan item calls for implementing a Westinghouse recommendation to modify the PORV's proportional integral derivative (PID) controller to prevent derivative action from opening the PORV. Two options were provided.

The applicant has satisfied this requirement by implementing the option of setting the derivative time constant equal to zero.

7.7.2.2 High-Energy-Line Breaks and Consequential Control System Failures

A concern was raised in IE Information Notice 79-22, issued September 19, 1979, that certain nonsafety-grade or control equipment, if subjected to the adverse environment of a high-energy-line break, could malfunction and cause plant conditions more severe than those analyzed in the safety analyses of FSAR Chapter 15. The applicant was asked to perform a review to determine what, if any, design changes or operator actions would be necessary to ensure that high-energy-line breaks will not cause control system failures to complicate the event beyond the FSAR Chapter 15 safety analyses.

The staff has reviewed the applicant's response, contained in FSAR Amendment 8, and finds it needs further information. The intent of NRC Question 420.4 was to require the applicant to review all possible control system malfunctions resulting from a high-energy-line break inside or outside of containment. It appears that the applicant only reviewed the four scenarios described in IE Information Notice 79-22 and limited those scenarios to inside containment.

This item is open pending staff review of further information from the applicant.

7.7.2.3 Control System Failure Caused by Malfunctions of Common Power Source or Instrument Line

To provide assurance that the FSAR Chapter 15 analyses adequately bound events initiated by a single credible failure or malfunction, the staff has asked (Q420.6) the applicant to identify any power sources or sensors that provide power or signals to two or more control functions, and to demonstrate that failures or malfunctions of these power sources or sensors will not result in consequences more severe than those of FSAR Chapter 15 analyses or beyond the capability of operator or safety systems.

The staff has reviewed the applicant's response, contained in Amendment 8 to the FSAR, and finds it needs further information in the following areas:

(1) The applicant's response should identify which control systems and sensor/instrument buses were analyzed.

(2) Analyses should be documented similar to a failure mode and effects analysis (FMEA) showing the cause of failure and then the corresponding effect on the system and its impact on FSAR Chapter 15 events.

This is an open item.

7.7.3 Conclusions

Later.

8 ELECTRIC POWER SYSTEMS

Later

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

Later

9.1.2 Spent Fuel Storage

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a subcritical array during all credible storage conditions. The staff reviewed the compatibility and chemical stability of the materials (except for the fuel assemblies) wetted by the pool water, in accordance with SRP Section 9.1.2 (NUREG-0800) and "Review and Acceptance of Spent Fuel Storage and Handling Application, April 1978."

The spent fuel racks will be constructed of Type 304 stainless steel. The spent fuel pool liner is constructed of stainless steel. The spent fuel storage rack configuration is composed of individual storage cells interconnected to form an integral structure.

The pool contains oxygen-saturated demineralized water that contains boric acid. The water chemistry control of the spent fuel pool has been reviewed elsewhere and meets staff recommendations.

The pool liner, rack lattice structure and fuel storage tubes are of stainless steel which is compatible with the storage pool environment. In this environment of oxygen-saturated borated water, the corrosive deterioration of the Type 304 stainless steel should not exceed a depth of 6.00×10^{-5} in. in 100 years, which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are, therefore, at similar potentials.

The staff finds that the corrosion that will occur in the spent fuel storage environment should be of little significance during the 40-year life of the plant. Components in the spent fuel storage pool are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion.

The staff also finds that the environmental compatibility and stability of the materials used in the spent fuel storage pool are adequate, based on test data and actual service experience in operating reactors.

On the basis of the above evaluation, the staff concludes that the applicant has selected appropriate materials of construction satisfying the requirements of (1) GDC 61 (10 CFR Part 50, Appendix A), insofar as the capability to permit appropriate periodic inspection and testing of components, and (2) GDC 62, preventing criticality by maintaining structural integrity of components and of the boron poison. The staff finds the materials acceptable.

9.1.3 Spent Fuel Pool Cleanup System

The spent fuel pool cleanup system is designed to remove radioactive species and other impurities for the purpose of maintaining area radiation levels as low as is reasonably achievable during fuel handling and other maintenance operations and to maintain water clarity for the purpose of facilitating the movement of fuel bundles. The fuel pool cleanup system services the spent fuel pool, refueling pool and fuel transfer canal. The system consists of two full design flow purification loops and one surface skimmer.

The fuel pool cleanup system will be used continuously during refueling operations to reduce radioactivity and to keep the water in the refueling pool clear. After refueling, the cleanup system will be manually operated intermittently to maintain clarity and reduce radioactivity of the spent fuel pool area to <2.5 mrem/hr. Manual intermittent use of the fuel pool cleanup system will usually be initiated concurrent with the fuel pool cooling system, but can be initiated when water in the spent fuel pool needs to be clarified or because high radiation (>2.5 mrem/hr) exists in the area.

During operation of the fuel pool cleanup system, samples will be taken for chemical analysis of boron, chloride, and fluoride, as well as for readings of pH and radioactivity, on an intermittent basis. Additionally, the system's decontamination factor and differential pressure for the filters and ion exchangers will be monitored to determine if filters or resin need to be replaced. The specific sampling and monitoring frequency will be based on how often the system is in use.

The staff has determined that the spent fuel pool cleanup system (1) can remove radioactive materials, corrosion products, and impurities from the pool water, and thus satisfies the requirements of GDC 61 (Appendix A to 10 CFR Part 50), as it relates to appropriate filtering systems for fuel storage; (2) is capable of reducing occupational exposure to radiation by removing radioactive products from the pool water, and thus satisfies the requirements of 10 CFR 20.1(c) as it relates to maintaining radiation exposures as low as is reasonably achievable; (3) confines radioactive materials in the pool water within the demineralizer and filters, and thus satisfies Position C.2.f(2) of RG 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," as it relates to reducing the spread of contaminants from the source; and (4) removes suspended impurities from the pool water by filters, and thus satisfies Position C.2.f(3) of RG 8.8, as it relates to removing crud through physical actions.

On the basis of the above evaluation, the staff concludes that the spent fuel pool cleanup system satisfies GDC 61, 10 CFR 20.1(c), and the appropriate sections of RG 8.8 and, therefore, is acceptable.

9.1.4, 9.1.5 Later

9.2 Water Systems

Later

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

Later

9.3.2 Sampling Systems

9.3.2.1 Process Sampling Systems

Process sampling is accomplished by a primary sampling system and a secondary sampling system. The primary sampling system is designed to collect water and gaseous samples contained in the reactor coolant system and associated auxiliary system process streams during all normal modes of operation. The secondary sampling system is designed to collect water and steam from the secondary cycle. Continuous secondary samples are analyzed automatically for pH, conductivity, dissolved oxygen, residual hydrazine, and sodium. Additionally, grab samples are obtained for confirmatory analysis and to test for other chemicals. Provisions are made to ensure that representative samples are obtained from well-mixed streams or volumes of effluent by the proper selection of sampling equipment, sampling points, and sampling procedures. The primary sample lines penetrating the containment are each equipped with two normally closed, pneumatically operated, isolation valves, which if open, close on a containment isolation actuation signal.

The staff's review included the provisions to sample all principal fluid process streams associated with plant operation and the applicant's design of these systems, including the location of sampling points, as shown on piping and instrumentation diagrams.

The staff has determined that the process sampling system satisfies:

(1) The requirements of GDC 13 (Appendix A to 10 CFR Part 50), because the system is capable of sampling boron concentration in the reactor coolant, the safety injection tanks, the refueling water storage tank, the boric acid mix tank, and the boron injection tank. (Boron can affect the fission

process for normal operation anticipated operational occurrences, and accident conditions.)

(2) The requirements of GDC 14, because the system is capable of sampling chemical impurities in the reactor coolant and the secondary coolant, to ensure that the reactor coolant pressure boundary will have a low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

(3) The requirements of GDC 26 by its capability to sample boron concentrations in the reactor coolant, the refueling water storage tank, and the boric acid mix tank for controlling the rate of reactivity changes.

(4) The requirements of GDC 63, because the system is able to sample radioactivity in the spent fuel pool and the gaseous radwaste storage tank to detect conditions that may result in excessive radiation levels.

(5) The requirements of GDC 64, because the system is capable of sampling for radioactivity that may be released from normal operations (including anticipated operational occurrences) and from postulated accidents, the reactor coolant, the pressurizer, the steam generator blowdown, the sump inside containment, the containment atmosphere, and the gaseous radwaste storage tank.

The staff has further determined that the process sampling system satisfies:

(1) The standards of ANSI N13.1-1969 for obtaining airborne radioactive samples

(2) The requirements of 10 CFR 20.1(c) and Positions 2.d(2), 2.f(3), 2.f(8), and 2.i(6) of RG 8.8, Revision 3, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," to maintain radiation exposures as low as is reasonably achievable, by providing

(a) ventilation systems and gaseous radwaste treatment systems to contain airborne radioactive materials

(b) liquid radwaste treatment system to contain radioactive material in fluids

(c) spent fuel pool cleanup system to remove radioactive contaminants in the spent fuel pool water

(d) remotely operated containment isolation valves to limit reactor coolant loss in the event of rupture of a sampling line

(3) The requirements of GDC 60 to control the release of radioactive materials to the environment by providing isolation valves that will fail in the closed position

(4) Positions C.1, C.2, and C.3 of RG 1.26, Revision 3, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and Positions C.1, C.2, C.3, and C.4 of RG 1.29, Revision 3, "Seismic Design Classification," by designing the sampling lines and components of the process sampling system to conform to the classification of the system up to and including the first isolation valves to which each sampling line and component is connected, and thus satisfies the quality standards requirements of GDC 1 and the seismic requirements of GDC 2.

On the basis of the above evaluation, the staff finds the proposed process sampling system acceptable.

9.3.2.2 Postaccident Sampling System (NUREG-0737, II.B.3)

After the TMI-2 incident, the staff recognized the need for an improved postaccident sampling system (PASS) to determine the extent of core degradation following a severe reactor accident. Criteria for an acceptable sampling and analysis system are specified in NUREG-0737, Item II.B.3. The system should have the carability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples without exposing any individual to radiation exceeding 5 rem to the whole body or 75 rem to the extremities (GDC 19) during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of severity of core damage (e.g, noble gases, isotopes of iodine and cesium, and nonvolatile isotopes), hydrogen in the containment atmosphere, and total dissolved gases or hydrogen, boron, and chloride in reactor coolant samples.

Criterion 1:

> The applicant shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hr or less from the time a decision is made to take a sample.

The applicant has provided in-line sampling and analysis capability to promptly obtain and analyze reactor coolant samples and containment atmosphere samples within 3 hr from the time a decision is made to take a sample. The staff finds that these provisions satisfy Criterion 1 and are, therefore, acceptable.

Criterion 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., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids
>
> (d) alternatively, have inline monitoring capabilities to perform all or part of the above analyses

The PASS provides the capability to collect diluted or undiluted liquid and gaseous reactor coolant and containment atmopshere grab samples that can be transported to the onsite radiological and chemical laboratory for hydrogen, oxygen, pH, conductivity, boron, chloride, and radionuclide analyses. Arrangements have been made with an offsite laboratory for backup and supplemental analyses. The staff finds that these provisions partially satisfy Criterion 2. The applicant should provide a plant-specific core-damage estimating procedure which takes into consideration plant parameters such as core exit temperature, water level in reactor vessel, containment radiation monitors, and hydrogen analyses.

Criterion 3:

> Reactor coolant and containment atmosphere sampling during post-accident conditions shall not require an isolated auxiliary system (e.g., the letdown system, reactor water cleanup system (RWCUS)) to be placed in operation in order to use the sampling system.

[Reactor coolant and containment atmosphere sampling during postaccident conditions does not require an isolated auxiliary system to be placed in operation in order to perform the sampling function. The PASS valves which are not accessible after an accident have been selected to withstand the specified service environment. These provisions satisfy Criterion 3 and are, therefore, acceptable.]

Criterion 4:

> Pressurized reactor coolant samples are not required if the applicant can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

The PASS can quantify both dissolved oxygen and dissolved hydrogen with unpressurized reactor coolant samples using a hydrogen gas chromatograph and an

Orbisphere oxygen analyzer. Dissolved oxygen can be measured to less than 0.1 ppm. The staff has determined that these provisions satisfy Criterion 4 of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Criterion 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 applicant shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the applicant shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

The cooling water is neither seawater nor brackish water, and more than one barrier is provided between primary containment systems and the cooling water; therefore, chloride analysis will be performed within 4 days. The staff has determined that these provisions satisfy Criterion 5 of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Criterion 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 GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 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 the GDC 19 criterion (October 30, 1979, letter from H. R. Denton to all licensees).

The PASS is designed to permit in-line analysis of the reactor coolant, emergency sumps, and containment atmosphere in accordance with the exposure criteria of GDC 19. Remote in-line sampling is performed from the radiochemistry laboratory. In addition, the PASS has the capability to obtain both diluted and undiluted backup grab samples. The staff has determined that these provisions satisfy Criterion 6 of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Criterion 7:

> The analysis of primary coolant samples for boron is required for PWRs.

The PASS has the capability to perform boron analysis in primary coolant samples. The staff has determined that these provisions satisfy Criterion 7 of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Criterion 8:

> If in-line 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 week until the accident condition no longer exists.

In-line monitoring is used as the primary method of performing the required analysis. The PASS has the capability to obtain both diluted and undiluted backup grab samples. In the event both in-line monitoring and grab sampling analysis capability fail, arrangements have been made to send the samples to Oak Ridge National Laboratories in a licensed shipping cask.

Provisions for inline monitor flushing have been made to reduce plateout, crud buildup, and radiation exposure of components; the panel tubing and monitors are flushed after every panel exercise. The staff has determined that these provisions satisfy Criterion 8 of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Criterion 9:

The applicant's radiological and chemical sample analysis capability shall include provisions to:

(a) Identify and quantify isotopes of the nuclide categories discussed above to levels corresponding to the source term given in (1) RG 1.3, or (2) 1.4 and 1.7. Where necessary and practicable, the ability 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 1 μCi/g to 10 Ci/g.

(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 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

The radionuclides in both the primary coolant and the containment atmosphere will be identified and quantified. Provisions are available for diluted reactor coolant samples to minimize personnel exposure. The PASS can perform radioisotope analyses at the levels corresponding to the source term given in RG 1.4, Revision 2. Radiation background levels will be restricted by shielding. Radiological and chemical analysis facilities are provided to obtain results within an acceptably small error (approximately a factor of 2). The staff finds that these provisions satisfy Criterion 8 and are, therefore, acceptable.

Criterion 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.

The applicant stated that the PASS is designed to provide accuracy, range, and sensitivity necessary to allow the operator to determine the radiological and chemical status of the sample. The staff has determined that these provisions do not satisfy Criterion 10. The applicant should provide information about accuracy, range, and sensitivity of the PASS instruments and analytical procedures in the postaccident water chemistry and radiation environment, consistent with the recommendations of RG 1.97, Revision 3, and the clarifications of NUREG-0737, Item II.B.3, "Postaccident Sampling Capability," transmitted to the applicant on March 13, 1984. In addition, the frequency for demonstrating operability of procedures and instrumentation and retraining of operators on a minimum semi-annual basis should be provided.

Criterion 11:

In the design of the postaccident sampling and analysis capability, consideration should be given to the following items:

(a) Provisions should be made 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 RCS 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 postaccident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere 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 returned 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 (HEPA) filters.

The applicant has addressed provisions for (1) purging lines to ensure samples are representative, (2) size of sample line, (3) isolation valves to limit reactor coolant loss from a failure of the sample line, and (4) ventilation exhaust from PASS to be filtered through charcoal adsorbers and HEPA filters. To limit iodine plateout, the containment air sample line is heat traced. The postaccident reactor coolant and containment atmosphere samples will be representative of the reactor coolant in the core area and the containment atmosphere. The staff has determined that these provisions satisfy Criterion 11 of Item II.B.3 of NUREG-0737, and are, therefore, acceptable.

Conclusion

The staff concludes that the postaccident sampling system satisfies nine of the eleven criteria in Item II.B.3 of NUREG-0737. Additional information is needed to complete the staff's review of the following two criteria:

(2) Provide a plant-specific procedure to estimate the extent of core damage.

(10) Provide information on the accuracies, sensitivities, and performance of the PASS instrumentation and analytical procedures in the postaccident water chemistry and radiation environment. Provide the frequency for demonstrating operability of procedures, and instrumentation and retraining of operators on semiannual basis.

Conformance to Item II.B.3 of NUREG-0737 is an open item.

9.3.3 Equipment and Floor Drainage System

Later

9.3.4 Chemical and Volume Control System

The chemical and volume control system (CVCS) is designed to control and maintain reactor coolant inventory and to control the boron concentration in the reactor coolant through the process of charging (makeup) and letdown (drawing off). The CVCS purifies the primary coolant by passing letdown flow through heat exchangers and purification ion exchangers. The CVCS is also designed to provide reactor coolant pump seal injection flow and to collect the controlled bleedoff from the seals. Three charging pumps (one positive displacement and two centrifugal), supply high-pressure injection (charging) of borated water into the reactor coolant for normal and emergency boration. The volume control tank serves as a surge for the reactor coolant system, to provide for (1) control of hydrogen concentration in the reactor coolant, and (2) a reservoir of makeup for the charging pumps. The boric acid makeup system in conjunction with the boron thermal regeneration system, provides for (1) boron additions to compensate for reactivity changes and (2) a shutdown margin for maintenance and refueling operations or emergencies. Therefore, the charging portion of the system is designed to seismic Category I requirements and contains redundant active components and an alternate flow path in order to meet the single failure criteria.

The CVCS, including the boron recovery system, consists of components and piping associated with the system from the letdown line of the primary system to the charging lines that provide makeup to the primary system and the reactor coolant pump seal water system.

The basis for acceptance in the staff's review has been conformance of the applicant's design of the CVCS system with the following regulations and regulatory guides: (1) the requirements of GDC 1 and the guidelines of RG 1.26 by assigning quality group classifications to system components in accordance with the importance of the safety function to be performed; (2) the requirements of GDC 2 and the guidelines of RG 1.29 by designing safety-related portions of the system to seismic Category I requirements; (3) the requirements of GDC 14 by maintaining reactor coolant purity and material compatibility to reduce corrosion and thus reduce the probability of abnormal leakage, rapid propagating failure, or gross rupture of the reactor coolant pressure boundary; (4) the requirements of GDC 29 as relate to the reliability of the CVCS to provide negative reactivity to the reactor by supplying borated water to the reactor coolant system in the event of anticipated operational occurrences; and (5) the requirements of GDC 60 and 61 with respect to confining radioactivity by venting and collecting drainage from the CVCS components through closed systems.

On the basis of the above review, the staff concludes that the design of the chemical and volume control system and supporting systems is acceptable and satisfies the requirements of GDC 1, 2, 14, 29, 60, and 61, and is, therefore, acceptable.

9.4 Heating, Ventilation, and Air Conditioning Systems

Later

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

The staff has reviewed the fire protection program, as described in the FSAR through Amendment 10 for conformance with SRP Section 9.5.1 (NUREG-0800, July 1981) which contains in Branch Technical Position (BTP) CMEB 9.5-1, the technical requirements of Appendix A to BTP APCSB 9.5-1, and Appendix R to 10 CFR 50. Because the applicant has compared its program to the latter guidelines, this report also references these guidelines.

In response to the staff's request for an evaluation of the fire protection program at Vogtle against the guidelines of Appendix A to BTP APCSB 9.5-1, the applicant transmitted its report evaluating fire protection at the plant. At the same time, the applicant also provided an evaluation against the requirements of Appendix R to 10 CFR 50 and BTP CMEB 9.5-1 (NUREG-0800).

As part of this review, the staff will visit the plant site 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 site has not yet been visited because the construction has not progressed to a level where such a visit would be meaningful.

The staff's review included an evaluation of the automatic and manually operated water and gas suppression systems, the fire detection systems, fire barriers, fire doors and dampers, fire protection administrative controls, and the size of the fire brigade. The objective of the review is to ensure that in the event of a fire, personnel and plant equipment would be adequate to safely shut down the reactor, to maintain the plant in a safe shutdown condition, and to minimize the release of radioactive material to the environment.

The staff's consultant, Rolf Jensen and Associates, Inc., participated in the review of the fire protection program.

The staff will require that the fire protection program be operational before initial fuel loading.

9.5.1.1 Fire Protection Program Requirements

Fire Protection Program

The applicant's fire protection program is described in FSAR Section 9.5-1 and in the "Fire Protection Evaluation Report." On the basis of its review, the staff concludes that the applicant's program will conform to the technical requirements in BTP CMEB 9.5-1, Section C.1, and, therefore, will be acceptable pending resolution of the open items discussed in this section .

Fire Hazards Analysis

The applicant's fire hazard analysis specified the combustible materials present in fire areas, identified safety-related equipment, determined the consequences of a fire on safe shutdown capability, and summarized available fire protection in accordance with BTP CMEB 9.5-1, Section C.1.b. The staff evaluates the identified fire hazards in the paragraphs that follow. Alternative shutdown capability has been provided for the control room and cable spreading room. That capability also is evaluated below.

GDC 3 (Appendix A to 10 CFR Part 50) requires that "Fire fighting systems shall be designed to assure that rupture or inadvertent operation does not significantly impair the safety capability of those structures, systems and components." To satisfy this requirement, the applicant has designed the components required for hot shutdown so that the rupture or inadvertent operation of fire suppression systems will not adversely affect the operability of these components. Where necessary, appropriate protection is provided to prevent impingement of water spray on components required for hot shutdown. Redundant trains of components that are susceptible to damage from water spray are physically separated so that manual fire suppression activities will not adversely affect the operability of components not involved in the postulated fire. However, the staff is concerned that the mechanism by which fire and firefighting systems may cause the simultaneous failure of redundant or diverse trains has been adequately considered in the design. The staff will require that the applicant identify such mechanisms that were considered in its fire hazards analysis and the measures taken to preclude the fire or fire suppressant induced failure of redundant or diverse safety trains. This is an open item.

9.5.1.2 Administrative Controls

The administrative controls for fire protection consist of the fire protection program and organization, the fire brigade training, the controls over combustibles and ignition sources, the prefire plans and procedures for fighting fires, and quality assurance. On the basis of its review, the staff finds that the administrative controls conform to the guidelines in BTP CMEB 9.5-1, Item C.2, and are, therefore, acceptable.

9.5.1.3 Fire Brigade and Fire Brigade Training

The applicant has committed to provide a fire brigade which will be composed of five members per shift. The applicant has provided a description of the plant fire brigade, including equipment and training, to satisfy the guidelines contained in BTP CMEB 9.5-1, Section C.3.

The fire brigade leader is the only person identified as being knowledgeable about safety-related systems. This does not meet staff guidelines. The staff will require the applicant to provide training in accordance with Section C.3 of BTP CMEB 9.5-1 which states that the fire brigade leader and at least two other brigade members should have sufficient training in or knowledge of plant safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability. This item has been discussed and identified as open in Section 13.2.2.1.

9.5.1.4 General Plant Guidelines

Building Design

Fire areas are defined by walls and floor/ceiling assemblies. Walls that separate buildings and walls between rooms containing safe shutdown systems are 3-hr fire-rated assemblies. In cases where the fire rating is less than 3 hr, combustible

the staff has evaluated each area with respect to its combustible fuel ~~load~~, fire suppression and detection systems, and proximity to safe shutdown equipment to determine if the fire-rated assemblies provided are adequate to protect the areas affected and to meet the guidelines in Section C.5.a of BTP CMEB 9.5-1. On the basis of this evaluation, the staff finds that the protection provided satisfies staff guidelines.

The applicant will provide penetration seals for all penetrations of fire-rated walls or floor/ceiling assemblies. The penetration seals have been subjected to qualification tests using the time-temperature curve specified by ASTM Std. E-119, "Fire Test of Building Construction and Materials." A maximum temperature rise of 325°F will be used as acceptance criteria in accordance with Section C.5.a of staff guidelines. The applicant will provide masonry 2-hr-rated fire barriers for the plant's enclosed stairwells. This complies with Section C.5.c of BTP CMEB 9.5-1 and is, therefore, acceptable.

Door openings in fire barriers will be protected with equivalently rated doors, frames and hardware except for watertight doors (which are not fire rated by UL), pressure-retaining doors (which will have a certificate of UL label construction applied by the vendor), and security doors (which will have a certificate of UL label construction applied by the vendor). The applicant also states that only those fire doors that serve a security function are electrically supervised and monitored in the main control room. Doors designed to remain open to maintain postaccident (pipe break) pressures within design limits are normally held open and released only when smoke is detected. Other doors are kept closed. On the basis of its review, the staff will require the applicant to indicate the type of door and the method of supervision provided for each door opening in the plant that is not a labeled fire door assembly. The applicant should also justify the adequacy of the special-purpose doors when used in rated fire barriers. This information is needed for the staff ~~us~~ to independently determine that door openings in fire barriers satisfy the guidelines of BTP CMEB 9.5-1, Section C.5.a(5). This is an open item.

Ventilation ducts that penetrate fire barriers are provided with fire dampers. The applicant states that 17 oversize fire dampers are used that do not bear a UL label. This does not comply with Section C.5.a of BTP CMEB 9.5-1. The staff will require the applicant to provide fire dampers that are tested and approved by a nationally recognized testing laboratory for all HVAC penetrations of fire barriers. This is an open item.

All transformers installed inside buildings are of the dry type. Oil-filled outdoor transformers are located more than 50 ft away from buildings containing safety-related equipment, and confinement dikes are provided. On the basis of its evaluation, the staff concludes that the installation of the transformers conforms to the guidelines of BTP CMEB 9.5-1, Sections C.5.a(12) and (13) and is, therefore, acceptable.

Interior wall and structural components, thermal insulation and radiation shielding materials are noncombustible. The FSAR indicates that interior finish materials have flame-spread, fuel-contributed, and smoke-developed ratings of 25 or less. The applicant has not stated the qualifications of soundproofing materials. The staff will require the applicant to verify that soundproofing materials also conform to the guidelines of BTP CMEB 9.5-1, Section C.5.a(9). This is an open item.

Metal roof deck construction is FM-listed Class 1. The staff concludes this conforms to the guidelines in Section C.5.a(10) of BTP CMEB 9.5-1, and is, therefore, acceptable.

Floor drains are provided to remove fire protection water from all safety-related areas. Drains in areas of combustible liquids have provisions for preventing backflow of combustible liquids to safety-related areas through interconnecting drain piping. The staff concludes that the plant's floor drain system conforms to the guidelines in Section C.5.a(14) of BTP CMEB 9.5-1 and is, therefore, acceptable.

Safe Shutdown Capability

The staff's review of safe shutdown capability is continuing and will be addressed in a supplement to this SER. It is an open item.

Alternate or Dedicated Shutdown Capability

The staff's review of alternate or dedicated shutdown capability is continuing and will be addressed in a supplement to this SER. Therefore, this is an open item.

Control of Combustibles

The storage of flammable liquids complies with NFPA Std. 30. Compressed gases are stored either outdoors or in nonsafety-related structures, except for small quantities of compressed gases used in the laboratory. Hydrogen lines in safety-related areas are designed to seismic Category I requirements. On the basis of its evaluation, the staff concludes that control of combustibles conforms to the guidelines of BTP CMEB 9.5-1, Section C.5.d and is, therefore, acceptable.

Electrical Cable Construction, Cable Trays, and Cable Penetrations

All cable trays are made of noncombustible materials. All cables, except some inside cabinets and those in the turbine generator, are specified to pass, as a minimum, the IEEE 383 flame test.

Line-type detectors are provided in the containment, and other detector types are provided elsewhere. Automatic sprinkler protection is provided for areas containing concentrations of cable trays. On the basis of its review, the staff concludes that electrical cable construction, cable trays and penetrations conform to the guidelines of BTP CMEB 9.5-1, Section C.5.e, and are, therefore, acceptable.

Ventilation

Air conditioning, ventilation or exhaust systems which could be utilized for discharging smoke or gas directly to the atmosphere are provided. Release paths for potentially contaminated smoke and gas are continuously monitored.

Power supplies and controls for mechanical ventilation systems will be located outside the fire area served, where practical. This does not conform to the guidelines in Section C.5.f. of BTP CMEB 9.5-1. The staff will require the applicant to demonstrate that a single fire will not disable both trains of ventilation needed for safety-related areas. This is an open item.

Air intake and exhaust ventilation dampers in areas protected by total flooding gas extinguishing systems are provided with mechanisms that will close them upon actuation of the suppression system. Stairwells are provided with self-closing doors designed to minimize smoke infiltration during a fire. Charcoal filters are protected in accordance with RG 1.52. The staff finds this acceptable.

Lighting and Communication

Fixed, self-contained lighting units with individual 8-hr battery power supplies are installed in all areas that will be manned for shutdown and for access and egress routes thereto.

Suitable battery-powered portable hand lights will be provided for emergency use. Fixed emergency communications, independent of the normal plant communications system, are installed at pre-selected stations. If repeaters are necessary for operation of the portable radio system, they will be provided. The staff finds this acceptable.

On the basis of its review, the staff concludes that lighting and communication systems are provided in accordance with the guidelines in Section C.5.g of BTP CMEB 9.5-1, and are, therefore, acceptable.

9.5.1.5 Fire Detection and Suppression

Fire Detection

A fire detection system is provided for all areas containing safety-related equipment and for all areas in which safety-related equipment can be exposed to fire.

The system complies with NFPA Std. 72D for a Class A system, except for the circuits that actuate the pre-action sprinkler system valves. These circuits are Class B circuits. If a fault occurs in these circuits, a trouble signal occurs that automatically trips the pre-action valve. The staff finds this an acceptable level of protection in lieu of providing Class A supervision, as it will permit automatic operation of the pre-action systems after a fault occurs.

A backup power supply exists for the fire detection system with access to the Class 1E diesel, switching manually, and a 4-hr battery backup power supply for the suppression actuation system with no access to the Class 1E diesels.

The applicant did not provide enough information for the staff to verify that this is in accordance with staff guidelines. The staff will require the applicant to confirm that primary and secondary power supplies for the fire detection system and for electrically operated control valves for automatic suppression systems conform to the guidelines of BTP CMEB 9.5-1, Section C.6.a(6). This is an open item.

The applicant states that it will be guided by NFPA Std. 72E for selecting fire detectors and installing them in the plant, but does not indicate how the detectors might deviate from NFPA Std. 72E. The staff will require the applicant to confirm that the fire detectors meet the guidelines of BTP CMEB 9.5-1, Section C.6.a(3).

Fire Protection Water Supply System

The fire protection water supply consists of two 100%-capacity diesel-driven fire pumps in the north fire pump house, and one 100%-capacity electrically driven fire pump in the south fire pump house. Each pump can satisfy 100% of the water demand for fire protection. The two diesel-driven pumps share a common discharge line. The electrically driven pump utilizes 4160-V switchgear instead of an NFPA Std. 20 pump controller. The applicant provided a line-by-line comparison of the 4160-V controller to the requirements of NFPA Std. 20. The staff has evaluated the comparison and concludes that the 4160-V controller is an acceptable deviation from Section C.6.b of BTP CMEB 9.5-1. The diesel-driven fire pumps and the electric pump are separately connected to a buried water main loop around the plant, thereby providing two independent pumping sources. Jockey pumps maintain pressure on the water supply system used for fire protection.

Each pump has a capacity of 2500 gpm.

The greatest water demand for the fixed fire-suppression system is 2000 gpm. Coupled with 500 gpm for hose streams, this creates a total water demand of 2500 gpm. The staff finds that the water supply system can deliver the required water demand with one pump out of service.

The source of water for the fire protection system is two 300,000-gal-capacity water tanks.

Hydrants are provided in the yard at intervals of less than 250 ft along the fire protection water supply loop. The lateral to each yard hydrant is provided with a key-operated isolation valve to facilitate maintaining and repairing the hydrants without shutting down any part of the system. Standard hose houses are provided in accordance with NFPA Std. 24.

Approved postindicator sectional control valves are provided to isolate portions of the underground main for maintenance or repair without shutting off the supply to primary and backup fire suppression systems that serve areas containing or exposing safety-related systems.

Not all valves in the fire protection water-supply system are supervised in accordance with staff guidelines and NFPA Std. 26. The staff will require the applicant to conform to the guidelines in Section C.6.c of BTP CMEB 9.5-1. This is an open item.

Sprinkler and Standpipe Systems

The applicant states that NFPA Stds. 13 and 15 have been used as guidance in the design of wet pipe sprinkler systems, deluge systems, and pre-action systems, but has not indicated how these systems might vary from the applicable standards. The staff will require that the applicant conform to NFPA Stds. 13 and 15 or identify and justify any deviations from Section C.6.c of staff guidelines. This is an open item.

Each automatic sprinkler system and interior hose standpipe is supplied through separate connections from the yard main or from the internal cross-connections through buildings to ensure that no single failure in the water-supply system will impair both the primary and backup fire protection in building areas. Each sprinkler and standpipe system connection to the distribution system is equipped with an indicating gate valve so that groups of sprinkler systems and/or manual hose stations can be isolated without interrupting the supply to other sprinkler systems and manual hose stations connected to the same header.

The applicant states that NFPA Std. 14 was used for guidance on manual hose stations located throughout the plant, but has not indicated how the standpipes and hose stations might vary from NFPA Std. 14. The staff will require the applicant to comply with NFPA Std. 14 or to identify and justify any deviations from Section C.6.c of BTP CMEB 9.5-1. This is an open item.

A seismic Category I, dry standpipe is provided for the control building, containment, auxiliary building, and diesel generator buildings. The staff finds this acceptable.

Halon Suppression Systems

Total flooding Halon 1301 systems are provided for the two shutdown panel rooms, the computer room, the cable-spreading-room termination cabinets, and five nonsafety-related areas in the control building.

The applicant states that the use of Halon systems is guided by NFPA Std. 12A, but has not indicated how it might vary from NFPA Std. 12A. The staff will require the applicant to comply with NFPA Std. 12A or to identify and justify any deviations from Section C.6.f of BTP CMEB 9.5-1. This is an open item.

Portable Extinguishers

Portable fire extinguishers are provided to conform with the guidelines of NFPA Std. 10. ~~The staff finds this acceptable.~~

On the basis of its review, the staff concludes that the extinguishers conform with the guidelines of BTP CMEB 9.5-1, Section C.6.f, and are, therefore, acceptable.

9.5.1.6 Fire Protection of Specific Plant Areas

Containment

The reactor coolant pumps will be equipped with an oil-collection system so designed and installed that failure will not lead to fire during normal or design-basis accident conditions, offering reasonable assurance that the system will withstand a safe shutdown earthquake.

The collection systems will be capable of collecting lube oil from all potential pressurized and unpressurized leakage sites in the reactor coolant pump's lube oil systems.

Fire detection is provided at major fire hazards, including the reactor coolant pumps, charcoal filters, and safety-related cable trays.

Hose stations are provided inside the containment. Each hose station is equipped with 100 ft of 1½-in.-diameter hose. During normal operation, the standpipe is dry. Automatically actuated deluge systems are provided for charcoal filters. Normally, dry pre-action sprinkler systems are provided below the reactor coolant pumps and in areas of high cable tray concentrations.

On the basis of it review, the staff concludes that the fire protection provided for containment conforms to the guidelines in Section C.7.a of BTP CMEB 9.5-1, and is, therefore, acceptable.

Control Room Complex

The control room complex is separated from all other areas of the plant by 3-hr-rated assemblies. Peripheral rooms are separated from the main control room by 3-hr-rated barriers. Automatic suppression has not been provided in all peripheral rooms. This does not comply with staff guidelines. The staff will require these rooms to be provided with automatic suppression and detection in accordance with Section C.7.b of BTP CMEP 9.5-1. This is an open item.

All cables entering the control room terminate there. No cables are routed through the control room from one area to another.

The only cables in the ceiling area are needed for the control room lighting system.

Ionization smoke detectors have been installed in the control room but not inside the individual cabinets and consoles within the control room. This does not comply with staff guidelines. The staff will require the applicant to provide smoke detectors in the control room cabinets and consoles in accordance with the guidelines in Section C.7.b of BTP CMEB 9.5-1. This is an open item.

The applicant has provided an alternate shutdown system for the control room. The alternate shutdown system is reviewed in Section 9.5.1.4 of this report.

The outside air intakes for the control room ventilation system are equipped with smoke detectors that alarm in the control room. In the event of a fire, the smoke venting systems can be manually initiated to purge smoke from the control room, or isolated to preclude smoke from entering the control room. The staff finds this acceptable.

Cable Spreading Rooms

The cable spreading rooms are separated from each other and from the balance of the plant by 3-hr fire-rated walls and floor/ceiling assemblies. All penetrations through fire-rated barriers are fitted with 3-hr fire-rated dampers and/or 3-hr fire-rated penetration seals.

Each cable spreading room contains only one division of safe shutdown cables.

Automatic pre-action sprinkler systems and smoke detection systems are the primary fire-suppression system for each cable spreading room. The primary fire-suppression system for termination cabinets is an automatic Halon 1301 system actuated by smoke detectors in the cabinets. Backup fire suppression is provided by water hose stations located in stairwells adjacent to the cable spreading room. The ventilation system is designed to isolate the room upon actuation of the fire suppression system. Smoke can be vented manually.

On the basis of its review, the staff concludes that the cable spreading rooms are designed in accordance with staff guidelines in Section C.7.c of BTP CMEB 9.5-1, and are, therefore, acceptable.

Switchgear Rooms

The switchgear rooms are separated from each other and from other plant areas by 3-hr fire-rated walls and floor/ceiling assemblies. Each switchgear room contains only one division of equipment and cables.

Automatic fire detection is provided. Manual protection is provided by standpipe hose stations and portable extinguishers. Floor drains have been provided in the switchgear rooms. On the basis of its review, the staff concludes that the protection provided for the switchgear rooms has been designed in accordance with staff guidelines in Section C.7.e of BTP CMEB 9.5-1 and is, therefore, acceptable.

Remote Safety-Related Panels

Redundant safety-related panels remote from the main control room are separated by barriers having a minimum fire rating of 3 hr. On the basis of its review, the staff concludes that the protection provided for remote safety-related panels complies with the guidelines in Section C.7.f of BTP CMEB 9.5-1 and is, therefore, acceptable.

Safety-Related Battery Rooms

The safety-related battery rooms are separated from each other and from the balance of the plant by 3-hr fire-rated barriers. A smoke detection system is provided in each battery room. Hose stations and portable fire extinguishers are available in adjacent areas. The ventilation system, consisting of two

100%-capacity exhaust fans, is designed to maintain the hydrogen concentration in each room below 2% by volume. Air-flow monitors that alarm in the control room, annunciate the loss of fan operation.

On the basis of its evaluation, the staff concludes that the fire protection for the safety-related battery rooms complies with the guidelines of BTP CMEB 9.5-1, Section C.7.g, and is, therefore, acceptable.

Emergency Diesel Generator Rooms

The emergency diesel generators are in individual rooms separated from each other and from other areas of the plant by fire barriers having a fire rating of 3 hr. The primary fire-suppression systems for each diesel generator room is an automatic pre-action sprinkler system. Detection is provided in each diesel generator enclosure.

A diesel-fuel-oil day tank of 1250-gal capacity is located in a separately enclosed area of each diesel generator room, separated from the diesel generators by 3-hr-rated walls. Each enclosure has a capacity of 200% of tank volume to compensate for oil leakage and accumulated firefighting water. The capacity of the day tank exceeds the staff's recommended capacity of 1100 gal for inside storage. Because of the protection provided, the staff finds this acceptable.

On the basis of its review, the staff concludes that the protection provided for the diesel generator rooms, with the approved deviation, is in accordance with the staff guidelines in Section C.7.i of BTP CMEB 9.5-1 and is, therefore, acceptable.

Other Plant Areas

The applicant's fire hazards analysis addressed other plant areas not specifically discussed in this report. The staff finds that the fire protection for these areas is in accordance with the guidelines of BTP CMEB 9.5-1 and is, therefore, acceptable.

Conclusion

The following are the open fire-protection items and the section in BTP CMEB 9.5-1 that states staff guidelines for the item:

(1)	fire hazards analysis	(C.5.b)
(2)	fire brigade	(C.3)
(3)	fire doors	(C.5.a)
(4)	fire dampers	(C.5.a)
(5)	soundproofing materials	(C.5.a)
(6)	safe shutdown	(C.5.b)
(7)	alternate shutdown	(C.5.c)
(8)	power supplies for ventilation	(C.5.f)
(9)	fire detection	(C.6.a)
(10)	valve supervision	(C.6.c)
(11)	automatic sprinkler systems	(C.6.c)
(12)	standpipes	(C.6.c)
(13)	Halon 1301 systems	(C.6.d)
(14)	control room complex	(C.7.b)

The following items deviate from BTP CMEB 9.5-1:

(1)	fire pump controllers	(C.6.c)
(2)	diesel generator day tank capacity	(C.7.g)

9.5.2 Communications Systems

Later

9.5.3 Lighting Systems

Later

9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System

Later

9.5.5 Emergency Diesel Engine Cooling Water System

Later

9.5.6 Emergency Diesel Engine Starting System

Later

9.5.7 Emergency Diesel Engine Lubrication System

Later

9.5.8 Emergency Diesel Engine Combustion Air Intake and Exhaust System

Later

10 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion system is designed to remove heat energy from the primary reactor coolant loop via four steam generators and to generate electric power in the turbine generator. After the steam passes through the high- and low-pressure turbines, the main condenser deaerates the condensate and transfers the rejected heat to the closed-cycle circulating water system, which uses a natural draft cooling tower to dissipate the rejected heat. The condensate is reheated and returned as feedwater to the steam generator. The entire system is designed for the maximum expected energy from the nuclear steam supply system. As noted in the sections that follow, the staff performed its safety evaluation in accordance with the applicable requirements of the Standard Review Plan (SRP) (NUREG-0800).

A turbine steam dump (bypass) system is provided to discharge directly to the condenser up to 40% of the main steam flow around the turbine during transient conditions. This bypass capacity, together with a 10% reactor automatic step load reduction capability, is sufficient to withstand a 50% generator load loss without tripping the turbine or tripping the reactor, and a 100% generator load loss without lifting safety valves.

10.2 Turbine Generator

Later

10.2.1 Deleted*

10.2.2 Deleted*

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*Deleted from the July 1981 edition of the Standard Review Plan (NUREG-0800).

10.2.3 Turbine Disk Integrity

Later

10.3 Main Steam Supply System

10.3.1 Later

10.3.2 Later

10.3.3 Deleted*

10.3.4 Deleted*

10.3.5 Secondary Water Chemistry

In late 1975, the staff 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 plants that have been issued an operating license from 1974 until 1979 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 components such as tubes and tube-support plates.

In a number of instances, the plant 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. On the basis of this experience and the knowledge gained in recent years, the staff has concluded that technical specification limits

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*Deleted from the July 1981 edition of the Standard Review Plan (NUREG-0800).

are not the most effective way of assuring that steam generator degradation will be minimized.

Because the corrosion phenomena involved are so complex, and considering the state of the art as it exists today, the staff finds that, instead of specifying limiting conditions in the plant Technical Specifications, a more effective approach would be to specify a technical specification that required the implementation of secondary water chemistry monitoring and control program containing appropriate procedures and administrative controls. This has been the approach for control of secondary water programs since 1979.

Applicants/licensees and reactor vendors or other consultants are to develop the required program and procedures to account for site- and plant-specific factors that affect water chemistry conditions in the steam generators. In the staff's view, plant operation following such procedures would provide adequate assurance that applicants/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.

The applicant did not provide details of a secondary water chemistry monitoring and control program. The applicant stated in Amendment 5 that the requested information would be provided at a later date. To complete its review, the staff needs the following information:

A summary of operative procedures to be used for the steam generator secondary water chemistry control and monitoring program, should address the following:

(1) Identify the sampling schedule for the critical chemical and other parameters and the control points or limits for these parameters for each operating mode of the plant, i.e., dry layup, cold shutdown, hot standby/shutdown, and power operation.

(2) Identify the procedures used to measure the values of the critical parameters, i.e., standard identifiable procedures and/or instruments.

(3) Identify the sampling points, considering as a minimum the steam generator blowdown, the hot-well discharge, the feedwater, and the demineralizer effluent. The staff recommends using a process flow chart similar to that in the Electric Power Research Institute (EPRI) report, NP-2704-AR, "PWR Secondary Water Chemistry Guidelines."

(4) State the procedures for recording and managing data, defining corrective actions for various out-of-specification parameters. The procedures should define the allowable time for correcting out-of-specification parameters. The staff recommends multiple levels of time be allowed for providing correction based upon the amount of out-of-specification of the variable. (See EPRI report NP-2704-SR above.)

Because of the significance of condenser in-leakage, the chemistry program should include a corrective action provision so that a condenser inservice inspection program will be initiated if condenser leakage is of such a magnitude that power must be reduced (action level 2 of the EPRI/SGOG guidelines) more than once per three-month period.

(5) Identify (a) the authority responsible for interpreting the data and initiating action, and (b) the sequence and timing of administrative events required to initiate corrective action.

10.3.6 Main Steam and Feedwater Materials

The staff concludes that the main steam and feedwater system materials are acceptable and satisfy the relevant requirements of 10 CFR 50, 10 CFR 50.55a, GDC 1, and Appendix B to 10 CFR 50. This conclusion is based on the following:

The applicant selected materials for Class 2 and 3 components of the steam and feedwater systems that satisfy Appendix I of Section III of the ASME Boiler and Pressure Vessel Code, as well as Parts A, B, and C of Section II of the Code. Conformance to the recommendations of RG 1.85 is discussed in Section 5.2.1.2.

In this time frame, the Code allowed impact testing of main steam and feedwater materials to be waived. However, on the basis of the results of impact testing by other applicants of the same specification steels, and correlations of the metallurgical characterization of these steels with the fracture toughness data presented in NUREG-0577, the staff concludes that the fracture toughness properties of the ferritic materials in the main steam and feedwater systems which were not impact tested have adequate safety margins against the possibility of nonductile behavior and rapidly propagating fracture.

The applicant has conformed (to the extent practical) to the recommendations of RG 1.71, "Welder Qualification for Areas of Limited Accessibility" by adhering to the regulatory positions in RG 1.71 or by adhering to alternative approaches which the staff has reviewed and found to be acceptable (see Section 4.5.1). The onsite cleaning and cleanliness controls during fabrication conform to the positions given in RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the requirements of ANSI Std. N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," or the staff has reviewed the applicant's alternative approaches and finds them acceptable, as discussed in Section 4.5.1 of this report.

10.4 Other Features

10.4.1 Main Condensers

Later

10.4.2 Main Condenser Evacuation System

FSAR Section 10.4.2 contains information pertaining to the main condenser evacuation (air removal) system, the system design bases, and the applicable acceptance criteria. The staff has reviewed the applicant's design, design criteria, and design bases for the main condenser evacuation system (MCES) for Vogtle in accordance with Section II of SRP Section 10.4.2 (NUREG-0800). The SRP acceptance criteria include GDC 60 and 64 and Heat Exchanger Institute Standard, "Standards for Steam Surface Condensers." Guidelines for implementation of the requirements of the acceptance criteria are provided in the regulatory guides referenced in Section II of the SRP. Conformance to the acceptance criteria of the SRP provides the bases for concluding that the MCES meets the requirements of 10 CFR 50.

The MCES is designed to establish and maintain main condenser vacuum by removing noncondensible gases from the main condenser. Two two-stage steam jet air ejectors and two mechanical vacuum pumps are provided to remove noncondensibles and hold vacuum during normal operation. The steam jet air ejectors are provided with water-cooled inter- and after-condensers. The mechanical vacuum pumps are used to draw initial condenser vacuum during plant startups and may also be used during normal operation. Air and noncondensable gases removed from the main condenser shells by the steam jet air ejectors and mechanical vacuum pumps are continuously monitored for radioactivity before discharge through the turbine building vent. The exhaust gases are routed through the condenser vacuum exhaust filter system prior to discharge whenever a high level of radiation is detected by the radiation monitoring system.

The applicant's provisions for quality assurance for the design, construction, and operational phases of the MCES should conform to RGs 1.33 and 1.123, as provided in the SRP. The applicant has proposed an exception to the criteria in that the MCES is not safety related and RGs 1.33 and 1.123 establish requirements only for safety-related structures, systems, and components. However, the applicant has provided no further information regarding quality assurance for the design, construction, and operational phases for the main condenser evacuation system. Therefore, this is an open item. Equipment quality group classifications were reviewed to determine conformance with RG 1.26, as provided in the SRP. No exceptions were noted. The MCES capacity was reviewed to determine conformance with Heat Exchanger Institute Standard, "Standards for Steam Surface Condensers," as provided in SRP Section 10.4. No exceptions were noted.

The MCES includes equipment and instruments to establish and maintain condenser vacuum and to prevent an uncontrolled release of gaseous radioactive material to the environment. The scope of the staff's review included the system's capability to transfer radioactive gases to the ventilation exhaust systems and the design provisions incorporated to monitor and control releases of radioactive materials in effluents. The staff has reviewed the applicant's system descriptions, piping and instrumentation diagrams, and design criteria for the MCES components in accordance with the SRP. It concludes that the MCES design is acceptable except regarding its quality assurance.

10.4.3 Turbine Gland Sealing System

FSAR Section 10.4.3 contains information pertaining to the turbine gland sealing system, the design bases, and applicable acceptance criteria.

The staff has reviewed the applicant's design, design criteria, and design bases for the turbine gland sealing system for Vogtle in accordance with Section II of SRP Section 10.4.3 (NUREG-0800). The acceptance criteria include GDC 60 and 64. Guidelines for implementation of the requirements of the acceptance criteria are provided in the regulatory guides identified in Section II of the SRP. Conformance to the acceptance criteria provides the bases for concluding that the turbine gland sealing system meets the requirements of 10 CFR Part 50.

The turbine gland sealing system provides sealing steam to the main turbine generator shaft to prevent the leakage of air into the turbine casings and the potential escape of radioactive steam into the turbine building. The turbine gland sealing system uses three steam sources: main, auxiliary, and/or extraction steam. The steam supply is passed through the turbine gland seals and condensed in the steam packing exhauster condenser. The condensate is returned to the main condenser and noncondensable gases are continuously monitored for radioactivity before discharge through the turbine building vent. The exhaust gases are routed through the steam packing exhauster filtration unit prior to discharge whenever a high level of radiation is detected by the radiation monitoring system.

The applicant's provisions for quality assurance for the design, construction and operational phases of the turbine gland sealing system should conform to RGs 1.33 and 1.123, as provided in the SRP. The applicant has proposed an exception to the criteria in that the turbine gland sealing system is not safety related and RGs 1.33 and 1.123 establish requirements only for safety-related structures, systems and components. However, the applicant has provided no further information regarding quality assurance for the design, construction, and operational phases for the turbine gland sealing system. Therefore, this is an open item.

The staff concludes that the turbine gland sealing system design is acceptable except regarding its quality assurance.

10.4.4 Turbine Bypass System

Later

10.4.5 Circulating Water System

Later

10.4.6 Condensate Cleanup System

The purpose of the condensate cleanup system is to remove dissolved and suspended solids from the condensate in order to maintain a high quality of the feedwater being supplied to the steam generators under all normal plant conditions (startup, shutdown, hot standby, power operation). This is accomplished by directing the full flow of condensate to a set of filter mixed-bed demineralizer units. Since the demineralizers need periodic resin regeneration, spare units are provided in the system to replace units taken out of service. The system provides final polishing of the secondary cycle condensate water.

The condensate cleanup system (CCS) is designed to assist in the control of the secondary side water chemistry and is part of the total control system.

The condensate cleanup system includes all components and equipment necessary for removing dissolved and suspended impurities which may be present in the condensate.

The staff has reviewed the CCS equipment design, materials and system operation in accordance with SRP Section 10.4.6 (NUREG-0800). The system satisfies the requirements for condensate cleanup capacity, and contains adequate instrumentation to monitor the effectiveness of the system.

The staff has reviewed the sampling equipment, sampling locations, and instrumentation to monitor and control the CCS process parameters. On the basis of this review, the staff finds that the instrumentation and sampling equipment provided is adequate to monitor and control process parameters.

On the basis of its review of the applicant's criteria and design bases for the condensate cleanup system and the requirements for operation of the system, the staff concludes that the design of the condensate cleanup system and supporting systems conforms to staff guidelines and is, therefore, acceptable.

The secondary water chemistry monitoring and control program is evaluated in Section 10.3.5 of this SER.

On the basis of its review, the staff concludes that the design of the condensate cleanup system supporting systems is acceptable and satisfies the primary boundary, integrity requirements of GDC 14. This conclusion is based on the applicant's having ~~meeting~~ met the requirements of GDC 14 as it relates to maintaining acceptable chemistry control for PWR secondary coolant during normal operation and anticipated operational occurrences by reducing corrosion of PWR steam generator tubes and materials, thereby reducing the likelihood and magnitude of reactor piping failures and of primary-to-secondary coolant leakage. The applicant's design of the CCS satisfies BTP MTEB 5-3 for PWRs.

On the basis of the foregoing, the staff concludes that the condensate cleanup system conforms to staff guidelines and is, therefore, acceptable.

10.4.7 Condensate and Feedwater System

Later

10.4.8 Steam Generator Blowdown System

The steam generator blowdown system (SGBS) is used in conjunction with the condensate demineralizer, chemical addition, and sample systems to control the chemical composition of the water in the secondary side of the steam generator ~~water~~ within specified limits during all operating modes. The blowdown fluids are directed to a set of heat exchangers to cool the fluids and then to a series of filter/mixed-bed demineralizers, the blowdown tank, and then to the condenser.

The SGBS controls the concentration of chemical impurities and radioactive materials in the secondary coolant. The scope of review of the SGBS included piping and instrumentation diagrams, seismic and quality group classifications, design process parameters, and instrumentation and process controls.

The portion of the steam generator blowdown system up to and including the containment isolation valves is seismic Category I and designated ASME II, Class 2. All other piping and equipment in the SGBS is not safety related and is designed and built to ANSI B31.1 requirements. Thus, the SGBS satisfies the quality standards requirements of GDC 1 and seismic requirements of GDC 2.

Instrumentation and automatic controls are provided to monitor and control the operation of the blowdown system, with provision for sampling of the blowdown, in conformance with the guidelines of BTP MTEB 5-3.

On the basis of the staff's evaluation, the SGBS satisfies the primary boundary material integrity requirements of GDC 14 as it relates to maintaining acceptable secondary water chemistry control during normal operation and anticipated operational occurrences by reducing corrosion of steam generator tubes and materials, thereby reducing the likelihood and magnitude of primary-to-secondary coolant leakage. On the basis of the foregoing evaluation, the staff concludes that the proposed steam generator blowdown system conforms to staff guidelines and, is therefore, acceptable.

Two items remain open.

(A) <u>Postaccident Sampling System (SER Section 9.3.2.2, II.B.3)</u>

(2) Provide plant-specific procedures to estimate core damage.

(10) Provide information on accuracy, sensitivity, and performance of PASS instrumentation and procedures.

(B) <u>Secondary Water Chemistry (SER Section 10.3.5)</u>

(1) sampling schedule and control limits
(2) procedures
(3) sampling points
(4) data management
(5) responsible authority

10.4.9 Auxiliary Feedwater System

Later

11 RADIOACTIVE WASTE MANAGEMENT

11.1 Introduction

The radioactive waste management systems for Vogtle, Units 1 and 2 (Vogtle) are designed to provide for the controlled handling and treatment of liquid, gaseous and solid wastes. The liquid radioactive waste management system processes wastes from equipment and floor drains, sample wastes, decontamination and laboratory wastes, and chemical regeneration wastes. The gaseous radioactive waste management system provides (1) waste gas decay tanks to allow decay of short-lived noble gases, and (2) treatment of ventilation exhausts through high-efficiency particulate air (HEPA) filters and carbon adsorbers, as necessary, to reduce releases of radioactive materials to as low as is reasonably achievable (ALARA) levels in accordance with 10 CFR Parts 20 and 50.34a. The solid radioactive waste management system provides volume reduction by drying and incineration and solidification by using cement and polymer binders. The radioactive waste management review area also includes the process and effluent radiological monitoring and sampling system provided for the detection and measurement of radioactive materials in plant process and effluent streams.

11.1.1 Acceptance Criteria

The staff has reviewed the applicant's design, design criteria and design bases for the radioactive waste management systems for Vogtle. The acceptance criteria used as the basis for staff evaluation are in SRP Sections 11.1, 11.2, 11.3, 11.4, and 11.5 (NUREG-0800). These acceptance criteria include the applicable GDC (Appendix A to 10 CFR 50), 10 CFR 20.106, Appendix I to 10 CFR 50, and American National Standards Institute (ANSI) Standard N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities." Guidelines for implementation of the requirements of the acceptance criteria are provided in the ANSI standards, regulatory guides, and other documents identified in SRP Section II. Conformance to the acceptance criteria provides the bases for concluding that the radioactive waste management systems meet the requirements of 10 CFR Parts 20 and 50.

11.1.2 Liquid and Gaseous Effluent Source Terms

The applicant provided the expected annual radioactive releases from Vogtle in FSAR Tables 11.2.3-1 and 11.3.3-2. The staff has performed an independent calculation of the primary and secondary coolant concentrations and of the release rates of radioactive materials using the information supplied in the FSAR, the GALE computer program, and the methodology presented in NUREG-0017. Table 11.1 presents the principal parameters that were used in this independent calculation of source terms. These source terms were used to calculate individual doses in Sections 11.2 and 11.3 for Vogtle~~, Units 1 and 2,~~ in accordance with the mathematical models and guidance contained in RG 1.109, "Calculation of Annual Average Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50, Appendix I."

11.2 Liquid Waste Management System

11.2.1 System Description and Review

The liquid radioactive waste (radwaste) management system consists of process equipment and instrumentation necessary to collect, process, monitor, and recycle or dispose of radioactive liquid wastes from the operation of Vogtle. The liquid radwaste system is designed to collect and process wastes according to the source, activity, and composition of the fluids.

The liquid waste processing system treats ~~All~~ liquid waste ~~will be processed~~ on a batch basis to permit optimum control and disposal of radwaste. Before these wastes are released, samples will be analyzed to determine the types and amounts of radioactivity present. On the basis of the results of the analyses, the waste will be recycled for eventual reuse in the plant, retained for further processing, or released under controlled conditions. The released waste is combined with the effluent from the cooling tower blowdown sumps and the combined flow is then discharged to the river through the discharge pipe. A radiation monitor in the discharge line will automatically terminate liquid waste discharges if radiation measurements exceed a predetermined level.

The liquid waste management system consists of the liquid waste processing system (FSAR Section 11.2), the boron recycle system (FSAR Subsection 9.3.4.2), the steam generator blowdown processing system (FSAR Subsection 10.4.8), and the turbine building floor drain system (FSAR Subsection 9.3.3).

11.2.1.1 Liquid Waste Processing System

The liquid waste processing system consists of (1) the reactor coolant drain tank subsystem which collects nonaerated, reactor grade effluent from sources inside the containment for recycling; (2) drain channel A which collects aerated, reactor grade effluent that normally can be recycled; and (3) drain channel B which processes all effluent that is normally to be discharged to the environment and is not suitable for recycling. In addition, the liquid waste processing system provides capability for handling and storage of spent ion exchange resins.

The reactor coolant drain tank subsystem of the liquid waste processing system collects valve leakoffs from the No. 3 seal leakoffs, reactor coolant pumps, reactor vessel flange leakoff and other deaerated tritiated water sources inside the containment. This liquid is normally processed by the boron recycle system for reuse; otherwise, it is sent to the liquid waste processing system drain channel A for processing. The staff estimated that the reactor coolant drain subsystem waste input flow will be approximately 300 gpd (0.21 gpm) per generating unit and assumed that 25% of the treated process stream will be released to the river through the discharge pipe. The remainder will be recycled for reuse within the plant. A separate reactor coolant drain tank subsystem is provided for each generating unit. The design capacities of the reactor coolant drain subsystem pumps and heat exchanger are each approximately 100 gpm. The reactor coolant drain tank usable volume is 350 gal. The difference between the expected flow and the design capacities provides adequate reserve for processing surge flows.

Drain channel A of the liquid waste processing system collects, through lines connected to the waste holdup tank, liquids from accumulator drains (via reactor coolant drain tank pump suction); sample room sink drains (excess primary sample volume only); ion exchanger, filter, pump, and other equipment drains; and condensate from the volume reduction system. If the quality of the water in the containment sump or auxiliary building sump is acceptable for recycling, it may be directed to the waste holdup tank. Otherwise, it will be directed to the floor drain tank for disposal. The wastes will be processed through the waste evaporator and, if necessary, through the waste evaporator condensate demineralizer. If further processing is required, the condensate can be returned to the waste holdup tank for additional evaporation. The staff estimated that the drain channel A waste input flow will be approximately 775 gpd per generating unit and assumed that 25% of the treated process stream will be released to the river through the discharge pipe. The remainder will be recycled for reuse within the plant. A separate drain channel A subsystem is provided for each of the two generating units.

Drain channel B of the liquid waste processing system collects and processes wastes from floor drains, equipment drains containing nonreactor grade water, laundry and hot shower drains, and other nonreactor grade sources. Water may enter the floor drain tank from leaks inside the containment through the containment sump, from leaks in the auxiliary building through auxiliary building sumps and floor drains, and from chemical laboratory drains. If necessary, the floor drain tank liquid can be processed by the waste evaporator or by the waste monitor tank demineralizer prior to release. The staff estimated that the floor drain tank waste input flow will be approximately 2050 gpd per generating unit and assumed that 100% of the treated process stream will be released to the river through the discharge pipe. A separate floor drain tank and associated equipment are provided for each generating unit.

The laundry and hot shower tank is provided to collect and process waste effluents from the plant laundry and personnel decontamination showers and hand sinks. If necessary, the water can be directed through the Unit 1 or Unit 2 waste monitor tank demineralizer for cleanup before discharge to the river through the discharge pipe.

The design capacity of the drain channel A and drain Channel B subsystems combined (based on the design flow of the waste evaporator) is 15 gpm (21,600 gpd), which corresponds to 10,800 gpd per generating unit. The difference between the expected flow and the design flow provides adequate reserve for processing surge flows.

~~11.2.1.1 Liquid Wsate Processing System~~

11.2.1.2 Boron Recycle System

The boron recycle system, which is shared between the two generating units, processes reactor coolant effluent that can be readily reused as makeup. The system processes the effluent by mixed bed demineralizers, an evaporator with gas stripping and a polishing anion demineralizer. The system collects and processes reactor coolant system effluent, most of which is from the letdown and process drains. The staff estimated that the boron recovery subsystem waste input will be approximately 2000 gallons per day (gpd) per generating unit, of which 1700 gpd will come from the letdown line in the chemical and volume control system, and 300 gpd will come from the reactor coolant drain tank. The staff assumed that 25% of the processed water will be released to the river. The remainder will be recycled for reuse within the plant. The design capacity of the system (based on the design flow of the recycle evaporator package) is 15 gpm (21,600 gpd). The difference between the expected flow and the design flow provides adequate reserve for processing surge flows.

11.2.1.3 Steam Generator Blowdown Processing System

The steam generator blowdown processing system (1 per generating unit) accepts water from each steam generator blowdown line, processes the water as may be required by mixed-bed demineralizers, and delivers the processed water to the condensate system or to the waste water retention basin. The staff estimated that the steam generator blowdown processing system waste input will be approximately 54,000 lb/hr (108 gpm) for the four steam generators and assumed that 10% of the treated process stream will be discharged to the river via the waste water retention basins, cooling tower blowdown sump, and discharge pipe. The remainder will be recycled for reuse within the plant. The design flow capacity of the steam generator blowdown processing system is 360 gpm, based on a maximum flow of 90 gpm from each of the four steam generators. The difference between the expected flow and design flow provides adequate reserve for processing surge flows.

11.2.1.4 Turbine Building Floor Drain System

The turbine building floor drain system collects in sumps the normally non-radioactive turbine building floor drains, equipment drains, sampling wastes, and other miscellaneous drains. The collected fluid is usually sent to the oil separator prior to discharge to the waste water retention basins. If the fluid becomes radioactive, it is directed to the turbine building drain tank, from which the waste water is pumped to an oil separator, an activated charcoal filter, and demineralizers to remove oil and radioactive materials prior to discharge to the waste water retention basins.

11.2.2 Acceptance Criteria

In its evaluation of the liquid radioactive waste management system, the staff considered (1) the capability of the system to maintain releases below the limits in 10 CFR 20 during periods of fission product leakage (at design levels) from the fuel, (2) the capability of the system to meet the ALARA criterion in accordance with 10 CFR 50, Appendix I, Sections II.A and II.D, (3) the system design objectives for equipment necessary to control releases of radioactive effluents to the environment in accordance with 10 CFR 50.34a, (4) the system design to ensure adequate safety under normal and postulated accident conditions in accordance with GDC 61, and (5) the design features that are incorporated to control and monitor the releases of radioactive materials in accordance with GDC 60 and 64.

The estimated releases of radioactive materials in liquid effluents were calculated using the PWR-GALE Code described in NUREG-0017. The PWR-GALE Code is a computerized, mathematical model for calculating the routine releases of radioactive material in effluents from pressurized-water reactors (PWRs). The basic code has been in use since 1976 for all PWR licensing reviews. The calculations in the code are based on (1) data generated from operating reactors, (2) field and laboratory tests, (3) standardized coolant activities derived from American Nuclear Society (ANS) 18.1 Working Group recommendations,

(4) release and transport mechanisms that result in the appearance of radioactive material in liquid streams, and (5) the Vogtle, Units 1 and 2, radwaste system design features used to reduce the quantities of radioactive materials ultimately released to the environment. The principal parameters used in these calculations are given in Table 11.1 of this SER.

11.2.~~1~~3 Evaluation Findings

The liquid radwaste system includes the equipment necessary to control the releases of radioactive materials in liquid effluents in accordance with GDC 60 and 64. Capacities of principal components considered in the liquid waste processing system evaluation are listed in Table 11.2. The staff concludes that the design of the liquid waste management system is acceptable and meets the requirements of 10 CFR 20.106, 10 CFR 50.34a, Appendix I of 10 CFR 50, and GDC 60, 61 and 64, as referenced in the SRP. This conclusion is based on the following:

(1) The applicant has satisfied the requirements of 10 CFR 20.106. The staff has considered the potential consequences resulting from reactor operation and has determined that the concentrations of radioactive materials in liquid effluents in unrestricted areas will be a small fraction of the limits in 10 CFR 20, Appendix B, Table II, Column 2.

(2) The applicant has satisfied the requirements of Section II.A of Appendix I of 10 CFR 50 with respect to dose-limiting objectives by proposing a liquid radwaste treatment system that is capable of maintaining releases of radioactive materials in liquid effluents so that the calculated individual doses in an unrestricted area from all pathways of exposure are less than 3 mrem to the total body and 10 mrem to any organ. In its evaluation, the staff considered releases of radioactive materials in liquid effluents for normal operation, including anticipated operational occurrences, based on expected radwaste inputs over the life of the plant for Vogtle in accordance with SRP Section 11.1.

The applicant has satisfied the requirements of the Commission's Sep~~t~~-tember 4, 1975 Annex to Appendix I to 10 CFR 50 with respect to meeting the

ALARA criterion, and, therefore, need not perform a cost-benefit analysis as otherwise would be required by Section II.D of Appendix I to 10 CFR 50.

(3) The staff has reviewed the applicant's quality assurance provisions for the liquid radwaste systems, the quality group classifications used for system components, and the seismic design applied to structures housing these systems. The design of the systems and structures housing these systems meets the intent of the criteria given in RG 1.143. The staff has reviewed the provisions incorporated in the applicant's design to control the release of radioactive materials in liquids resulting from inadvertent tank overflows and concludes that the measures proposed by the applicant are consistent with the criteria given in RG 1.143.

(4) The applicant has satisfied the requirements of GDC 60, 61 and 64 with respect to controlling and monitoring the releases of radioactive material to the environment. The staff has considered the capabilities of the proposed liquid radwaste treatment system to meet the demands of the plant resulting from anticipated operational occurrences and has concluded that the system's capacity and design flexibility are adequate to meet the anticipated needs of the plant.

11.3 Gaseous Waste Management System

11.3.1 System Description and Review

The gaseous waste processing and plant ventilation systems are designed to collect, store, process, monitor, and discharge potentially radioactive gaseous wastes that are 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 processing system and ventilation exhausts from the containment, auxiliary, fuel handling, radwaste solidification, radwaste transfer, and turbine buildings.

The gaseous waste processing system is designed to collect, process and store gaseous wastes generated by normal plant operations including anticipated operational occurrences. The system consists mainly of two closed loops, each associated with one of the generating units and comprised of a waste gas compressor, a catalytic hydrogen recombiner, and seven waste gas decay tanks. Waste gas is pumped by the waste gas compressor to the hydrogen recombiner where oxygen is added to oxidize the hydrogen to water vapor. After removal of the water vapor, the gas stream is circulated to a waste gas decay tank and then back to the waste gas compressor suction. A waste gas decay tank is valved into the recirculation loop for 1 to 2 days after which it is isolated and another tank valved into service. Hydrogen is continuously removed in the recombiner and therefore does not build up in the system. The largest contributors to the nonradioactive gas accumulation are helium generated from boron in the reactor and impurities in the hydrogen and oxygen supplies. With continued plant operation, the gas pressure in the system will gradually increase as the nonremovable gases accumulate in the system. At pressures over 20 psig, the valves are aligned so that the gases flow from the compressor to the decay tanks and then to the recombiner and back to the compressor suction. This arrangement is suitable for pressure up to 100 psig. Although the system is designed to accommodate continuous operation without atmospheric releases, the system design permits controlled discharge of gas from the waste gas decay tanks to the plant vent.

Ventilation air from the containment building is filtered prior to exhausting through the plant vent. The exhaust air flow rate is 15,000 ft^3/min for normal purge operations during refueling and 5,000 ft^3/min for a minipurge during power access periods.

The auxiliary building normal ventilation system is designed to maintain the building at a negative pressure to prevent release of radioactivity to the atmosphere. The air is filtered before exhausting through the plant vent.

The fuel-handling building ventilation system normal subsystems are designed to maintain the building at a negative pressure and to filter the air before exhausting through the Unit 1 plant vent.

The radwaste building ventilation system filters potentially contaminated air from the radwaste solidification building ~~prior to~~ before exhausting through the radwaste solidification building stack and ducts potentially contaminated air from the radwaste transfer building to the auxiliary building normal exhaust filtration system before exhausting through the plant vent.

In the turbine building, gases from the condenser vacuum exhaust and from the steam jet air ejectors are routed through the condenser vacuum exhaust filter system when a high level of radiation is detected by the radiation monitoring system. Also in the turbine building, gases released from the steam packing exhauster are routed through the steam packing exhauster filter system when a high level of radiation is detected by the radiation monitoring system. Otherwise, no filtration is provided before exhausting gases from the turbine building ventilation system through the turbine building roof exhaust fans.

11.3.2 Acceptance Criteria

In its evaluation of the gaseous radwaste management system, the staff considered the following SRP criteria: (1) the capability of the system to meet the processing demands of the station during anticipated operational occurrences, (2) the quality group and seismic design classification applied to the equipment and components and structures housing the system, (3) the design features that are incorporated to control and monitor the releases of radioactive materials in accordance with GDC 60 and 64, (4) the potential for gaseous releases resulting from hydrogen explosion in the gaseous radwaste system, and (5) the capability of the system design to meet the ALARA criterion in accordance with 10 CFR 50, Appendix I, Sections II.B, II.C, and II.D.

as discussed in Section 11.2.2 of this report.

The estimated releases of radioactive materials in gaseous effluents were calculated using the PWR-GALE Code described in NUREG-0017. The PWR-GALE Code is a computerized mathematical model for calculating the routine releases of radioactive material in effluents from PWRs. The basic code has been in use since 1976 for all PWR licensing reviews. The calculations in the code are based on (1) data generated from operating reactors, (2) field and laboratory tests, (3) standardized coolant activities derived from ANS 18.1 Working Group recommendations, (4) release and transport mechanisms that result in the appearance of radioactive material in gaseous streams, and (5) the Vogtle radwaste system design features used to reduce the quantities of radioactive materials ultimately released to the environment. The principal parameters ~~used in these calculations are given in Table 11.1 of this SER.~~

The staff has reviewed the applicant's quality assurance provisions for the gaseous radwaste system, the quality group classifications used for system components, the seismic design criteria applied to the design of the system and structures housing the radwaste system. The design of the system and structures housing this system meets the intent of the criteria given in RG 1.143 and referenced in the SRP.

The staff has reviewed the provisions incorporated in the applicant's design to control releases resulting from hydrogen explosions in the gaseous radwaste system and concludes that the measures proposed by the applicant are adequate to prevent the occurrence of an explosion.

The staff has reviewed the provisions incorporated in the applicant's design to control and monitor radioactive materials in the normal ventilation exhaust systems during normal plant operation, including anticipated operational occurrences, and concludes that the system design is adequate to control and monitor airborne radioactivity.

11.3.3 Evaluation Findings

The staff concludes that the design of the gaseous waste management system is acceptable and meets the requirements of 10 CFR 20.106; 10 CFR 50.34a; GDC 3, 60, 61, and 64; and 10 CFR 50, Appendix I, as referenced in the SRP. This conclusion is based on the following findings:

(1) The applicant has satisfied the requirements of GDC 60 and 64 with respect to controlling releases of radioactive material to the environment by ensuring that the design of the gaseous waste management system includes the equipment and instruments necessary to detect and control the release of radioactive materials in gaseous effluents. Capacities of principal components considered in the gaseous waste processing system evaluation are listed in Table 11.2.

(2) The applicant has satisfied the requirements of Appendix I of 10 CFR 50 by meeting the ALARA criterion as follows:

(a) Regarding Sections II.B and II.C of Appendix I, the staff has considered releases of radioactive material (noble gases, radioiodines, and particulates) in gaseous effluents for normal operation, including anticipated operational occurrences, based on expected radwaste inputs over the life of the plant. The staff has determined that the proposed gaseous waste management system is capable of limiting releases of radioactive materials in gaseous effluents so that the calculated individual doses from releases of radioiodine and radioactive material in particulate form in an unrestricted area from all pathways of exposure are less than 5 mrems to the total body, 15 mrems to the skin, and 15 mrems to any organ.

(b) The applicant has satisfied the requirements of the Commission's September 4, 1975 Annex to Appendix I to 10 CFR 50 with respect to meeting the ALARA criterion, and, therefore, need not perform a cost-benefit analysis as otherwise would be required by Section II.D of Appendix I to 10 CFR 50.

(3) The applicant has satisfied the requirements of 10 CFR 20. The staff has considered the potential consequences resulting from reactor operation and determined that the concentrations of radioactive materials in gaseous effluents in unrestricted areas will be a small fraction of the limits specified in 10 CFR 20, Appendix B, Table II, Column 1.

(4) The staff has considered the capabilities of the proposed gaseous waste management system to meet the demands of the plant resulting from anticipated operational occurrences and has concluded that the system's capacity and design flexibility are adequate to meet these demands.

(5) The staff has reviewed the applicant's quality assurance provisions for the gaseous waste management system, the quality group classifications used for system components, and the seismic design applied to the design of the system and of structures housing the radwaste system. The design

of the system and of structures housing the system satisfied the criteria given in RG 1.143.

(6) The staff has reviewed the provisions incorporated in the applicant's design to control releases resulting from hydrogen explosions in the gaseous waste management system and concludes that the measures proposed by the applicant are adequate to prevent the occurrence of an explosion in accordance with GDC 3.

11.4 Solid Waste Management System

11.4.1 System Description and Review

The solid waste management system consists of equipment and instrumentation necessary for the solidification or packaging of radioactive waste resulting from operation of the reactor water letdown purification system, the condensate demineralizer system, the liquid and gaseous radwaste systems, and the miscellaneous debris resulting from normal operation and maintenance of the plant.

The solid waste management system consists of the following subsystems:

(1) resin transfer system;
(2) backflushable filter system;
(3) backflushable filter crud transfer system;
(4) liquid/slurry waste solidification system;
(5) dry waste system;
(6) volume reduction system;
(7) filter handling system;
(8) dry product transfer system; and
(9) dry product solidification system.

The resin transfer system provides for the remote transfer of spent regenerative resin from the radwaste transfer building to the radwaste solidification building. The backflushable filter system provides for the removal and delivery of radioactive crud from certain process streams to the backflushable filter crud transfer system, which provides for the remote transfer of the radioactive crud from the radwaste transfer building to the radwaste solidification building.

The liquid/slurry waste solidification system provides for the solidification of spent resins and backflushable filter crud and, as a backup to the volume reduction system, provides for the solidification of chemical drain wastes and evaporator concentrates. A process control program will be used to ensure complete solidification in a cement binder.

The dry waste system provides for the collection of dry wastes, shredding of combustible waste, compaction of compressible noncombustible dry wastes, handling of activated components and equipment, and packaging and storage of compressed dry wastes and ~~handling of~~ activated components and equipment. The volume reduction system consists of two subsystems; the fluid bed dryer processes evaporator concentrates or chemical drain tank wastes, and the fluid bed dryer waste processor incinerates combustible dry wastes, contaminated oil and low activity spent resin from the condensate polishing demineralizer system and the steam generator blowdown system. The system is described in detail in the topical report, "Radioactive Waste Volume Reduction System, Topical Report No. AECC-3-NP," Aerojet Energy Conversion Company, December 1981. The system produces a dry solid waste product, a liquid condensate process stream which is returned to the liquid radwaste system for processing, and an effluent gas process stream which is filtered prior to discharge through the radwaste solidification building vent.

The filter handling system provides for the semiremote removal of spent radioactive cartridge filters and filter housings and their placement in shielded drums for transport to the radwaste solidification system. The dry product transfer system provides holdup capacity for the dry product from the volume reduction system and provides for the transfer of the dry product to the dry product solidification system, which solidifies the particulate and ash from the volume reduction system. A process control program will be used to ensure complete solidification of the dry product in a polymer binder.

Solidified wastes, spent filter cartridges, and solid compactible wastes will be packaged in 55-gal drums and stored onsite until they are shipped in shielded casks for offsite disposal.

11.4.2 Acceptance Criteria

The review of the solid waste management system, which was conducted in accordance with the SRP, included line diagrams of the system, piping and instrumentation diagrams (P&IDs), and descriptive information on the solid waste management system and those auxiliary supporting systems that are essential to its operation. The applicant's proposed design criteria and design bases for the solid waste management system and the applicant's analysis of these criteria and bases were reviewed and compared with those of the SRP. The staff also reviewed (1) the capability of the proposed system to process the types and volumes of wastes expected during normal operation and anticipated operational occurrences in accordance with GDC 60, (2) the provisions for the processing and packaging of wastes relative to the requirements of 10 CFR 20, 61, and 71 and applicable Department of Transportation (DOT) regulations, (3) the applicant's quality group classification and seismic design relative to RG 1.143, and (4) provisions for onsite storage before shipment. The basis for acceptance in the staff's review was conformance of the applicant's designs, design criteria, and design bases for the solid radwaste management system to the regulations, guides, staff technical positions, and industry standards referenced in the SRP.

11.4.3 Evaluation Findings

The annual quantities of solid wastes without volume reduction are estimated to be approximately 16,000 ft^3 of solidified wet wastes, containing approximately 10,000 Ci of activity, and approximately 7500 ft^3 of dry waste. With volume reduction, the volume of solidified wet wastes will be reduced to approximately 4,000 ft^3 and the volume of dry waste will be reduced due to the incineration of combustible dry wastes. The onsite storage capacity exceeds the expected quantity of drummed waste for 1 year of plant operation. Because the staff's guidance specifies storage space for 1 month's capacity of waste, the staff finds the storage volume adequate for meeting the demands of the plant.

In the FSAR, the applicant stated that the process control program (PCP) for the liquid/slurry waste solidification system and the dry product solidification system will be available prior to fuel load. This is a confirmatory item.

The staff has not completed its review of the topical report referenced in Section 11.4.1 concerning the volume reduction system, as referenced in the FSAR. Therefore, the following conclusions are pending the completion of staff review. This is an open item.

The staff concludes that the design of the solid waste management system is acceptable and meets the requirements of 10 CFR 20.106; 10 CFR 50.34a; GDC 60, 63, and 64; and 10 CFR 71, as referenced in SRP Section 11.4. This conclusion is based on the applicant demonstrating that the solid waste management system includes the equipment and instrumentation used for the processing, packaging, and storing of radwastes before shipment offsite for burial.

The basis for acceptance in the staff's review has been conformance of the applicant's designs, design criteria, and design bases for the solid radwaste system to the regulations and guides referenced above and in SRP Section 11.4, as well as to staff technical positions and industry standards. On the basis of the foregoing evaluation and the condition that the staff's review of Topical Report AECC-3-NP is satisfactory and that the applicant provides an acceptable process control program, which includes a compliance program to meet 10 CFR 20.311 and 10 CFR 61 for waste classification and waste minimum and stability requirements, the staff concludes that the proposed solid radwaste system is acceptable.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.1 System Description and Review

The process and effluent radiological monitoring systems are designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in plant discharges to the environment.

Table 11.3 provides the proposed locations of continuous monitors. Monitors on certain effluent release lines will automatically terminate discharges if radiation levels exceed a predetermined value. Systems that are not amenable to continuous monitoring or for which detailed isotopic analyses are required will be periodically sampled, and the samples will be analyzed in the plant laboratory. The potential airborne radioactive releases to the environment from Vogtle are from the following normal release points:

(1) Unit 1 plant vent
(2) Unit 2 plant vent
(3) Unit 1 turbine building exhaust
(4) Unit 2 turbine building exhaust
(5) radwaste solidification building vent

The plant vent includes discharges from the containment purge system, gaseous radwaste system, fuel-handling building HVAC system, and auxiliary building HVAC system. Condenser air ejector and steam packing exhaust is released through the turbine building exhaust.

The plant vent effluent radiogas, air particulate, and iodine monitors are designed to provide representative data on the gaseous activity, particulate activity, and gaseous iodine activity released to the plant environs. Th~~i~~ese data ~~is~~ are collected and displayed on a CRT and hardcopy printer on demand. Alarms will be annunciated on high radiation signals.

The containment vent effluent radiogas, air particulate, and iodine monitors measure radiogas, air particulate, and gaseous iodine activity in the containment purge vent. The monitors initiate automatic closure of the containment purge supply and exhaust valves for high radiation levels.

The waste gas processing system effluent radiogas monitor isolates the waste gas processing system on a high radiation alarm signal. The fuel-handling building effluent radiogas monitor initiates, on a high radiation signal, the switching of the fuel-handling building ventilation system from the normal operating mode to the accident mode.

Gases released from the condenser vacuum exhaust and from the steam jet air ejectors in the turbine building are routed through the filtration system when a high level of radiation is detected by the condenser air ejector and steam packing exhauster radiogas monitor.

The waste solidification building effluent radiogas, air particulate, and iodine monitors are designed to provide representative data on radioactive releases to the environs. There is no provision, however, to isolate the volume reduction system on a high radiation signal. The staff has not completed its analysis of the need for such an automatic control function to maintain releases below the limits of 10 CFR 20. Therefore, this is part of the open item identified in Section 11.4.3.

The potential radioactive liquid effluent normal release points are as follows:

(1) liquid waste discharge line into the discharge pipe, which discharges into the river

(2) steam generator blowdown liquid process discharge line into the water retention basins, which discharge into the main condenser cooling tower blowdown sump, which discharges through the discharge pipe into the river

(3) turbine building drain liquid line into the waste water retention basins

(4) nuclear service water system cooling tower blowdown discharge line into the main condenser cooling tower blowdown sump

(5) control building sump discharge line

The waste liquid effluent monitor, on a high alarm signal, initiates automatic valve closure on the liquid waste discharge line. The steam generator blowdown liquid process monitor, on a high alarm signal, automatically closes the steam generator blowdown processing system isolation valves and discharge lines. The turbine building drain liquid effluent monitor, on a high alarm signal, stops the flow from the turbine building drain system to the waste water retention basin. The nuclear service water process monitor provides indication of leakage from equipment processing radioactive liquid into the nuclear service cooling water. The control building sump effluent monitor, on a high alarm signal, initiates automatic isolation of the discharge line.

11.5.2 Evaluation Findings

The following evaluation findings are pending the resolution of the open item regarding isolation of the volume reduction ~~education~~ system on a high radiation signal (which is part of the open item identified in Section 11.4.3) previously noted. The staff concludes that the process and effluent radiological monitoring instrumentation and sampling system for the liquid and solid radwastes are acceptable and meet the relevant requirements of 10 CFR 20.106 and GDC 60, 63, and 64. The process and effluent radiological monitoring and sampling systems for the liquid and solid radwaste include the instrumentation for monitoring and sampling radioactivity in contaminated liquid and solid waste process and effluent streams. The staff's review included (1) the provisions proposed to sample and monitor all liquid effluents in accordance with GDC 64; (2) the provisions proposed to provide automatic termination of liquid effluent releases and ensure control over discharges in accordance with GDC 60; (3) the provisions proposed for sampling and monitoring plant waste process streams for process control in accordance with GDC 63; (4) the provisions for conducting sampling and analytical programs in accordance with the guidelines in RGs 1.21 and 4.15; and (5) the provisions for sampling and monitoring process and effluent streams during postulated accidents in accordance with the guidelines in RG ~~s~~ 1.97, Revision 2. The review included P&IDs and process flow diagrams for the liquid, gaseous, and solid radwaste systems and ventilation systems, and the location of monitoring points relative to effluent release points shown on the site plot diagrams.

On the basis of its review, the staff has determined that the applicant's designs, design criteria, and design bases for the process and effluent radiological monitoring instrumentation and sampling systems for the liquid and solid radwastes meet the guidelines of SRP Appendix 11.5-A and industry standards and concludes that the systems are acceptable.

Table 11.1 Principal parameters and conditions used in calculating releases of radioactive material in liquid and gaseous effluents from Vogtle Electric Generating Plant, Units 1 and 2

Parameter	Value/unit
Reactor power level (MWt)	3,565
Plant capacity factor	0.80
Failed fuel (%)	0.12*
Primary system	
Mass of coolant (lb)	5.1×10^5
Letdown rate (gpm)	75
Shim bleed rate (gpd)	1.7×10^3
Leakage to secondary system (lb/day)	100
Leakage to containment building (lb/day)	**
Leakage to auxiliary building (lb/day	160
Frequency of degassing for cold shutdowns (times/yr)	2
Letdown cation demineralizer flow (gpm)	7.5
Secondary system	
Steam flow rate (lb/hr)	1.51×10^7
Mass of liquid/steam generator (lb)	1.11×10^5
Mass of steam/steam generator (lb)	6.4×10^3
Secondary coolant mass (lb)	2.0×10^6
Rate of steam leakage to turbine area (lb/hr)	1.7×10^3
Containment building volume (ft^3)	2.75×10^6
Frequency of containment purges (times/yr)	4
Containment low volume purge rate (ft^3/min)	5000
Iodine partition factors (gas/liquid)	
Leakage to auxiliary building	0.0075
Leakage to turbine area	1.0
Main condenser/air ejector (volatile species)	0.15

LIQUID RADWASTE SYSTEM DECONTAMINATION FACTORS

Material	Boron recycle system	Liquid waste processing (clean waste)	Floor drains (dirty waste)
Iodine	1×10^5	1×10^3	1×10^3
Cesium	2×10^3	1×10^4	1×10^4
Other	1×10^4	1×10^4	1×10^4

*This value is constant and corresponds to 0.12% of the operating power product source term as given in NUREG-0017 (April 1976).

**1%/day of the primary coolant noble gas inventory and 0.001%/day of the primary coolant iodine inventory.

Table 11.1 (Continued)

INDIVIDUAL EQUIPMENT DECONTAMINATION FACTORS

(1) Evaporator

System	All nuclides except iodine	Iodine
Radwaste evaporator	10^4	10^3
Boron evaporator	10^3	10^2

(2) Demineralizers

System	Anions	~~Cesium,~~ ~~R~~ubidium	~~Other~~ ~~N~~uclides]Fix
Boron recycle system feed demineralizer	10	2	10
Boron recycle evaporator polishing demineralizer	10^2	1	1

LIQUID WASTE INPUTS

System	Flow rate (gpd)	Fraction of PCA	Fraction discharged	Collection time (days)	Decay time (days)
Shim bleed rate	1,700	1.0	0.25	4.2	2.1
Equipment drains	300	1.0	0.25	4.2	2.1
Clean wastes	713	0.127	0.25	5.6	0.19
Dirty wastes	2,050	0.019	1.0	1.95	0.19

GASEOUS WASTE INPUTS***

Holdup time for xenon (days)	90
Holdup time for krypton (days)	90
Fill time of decay tanks (days)	0

***There is no continuous stripping of full letdown flow.

Table 11.2 Design parameters of principal components considered in the evaluation of liquid, gaseous, and solid radioactive waste treatment systems for Vogtle Electric Generating Plant, Units 1 and 2

Component	Number	Capacity, each
LIQUID SYSTEMS*		
Waste holdup tank (per unit)	2	10,000 gal
Waste evaporator condensate tank (per unit)	1	5,000 gal
Flow drain tank (per unit)	1	10,000 gal
Waste monitor tank (per unit)	2	5,000 gal
Chemical drain tank (shared)	1	600 gal
Laundry and hot shower tank (shared)	1	10,000 gal
Waste evaporator concentrates holdup tank (shared)	1	2,000 gal
Waste evaporator (per unit)	1	15 gpm
Waste evaporator condensate demineralizer (per unit)	1	35 gpm
Waste monitor tank demineralizer (per unit)	1	35 gpm
Boron recycle holdup tank (shared)	2	112,000 gal
Boron recycle evaporator (shared)	1	15 gpm
Boron recycle feed demineralizer (shared)	2	120 gpm
Boron recycle evaporator polishing demineralizer (shared)	1	120 gpm
GASEOUS SYSTEMS*		
Waste gas compressor (2 units)	4	40 scfm
Waste gas decay tank (7 per unit plus 2 shared)	16	600 ft^3, 100 psig
Catalytic hydrogen recombiner (2 units)	3	50 scfm, 3 scfm H_2
SOLID SYSTEMS*		
Polymer storage tank	1	6,000 gal
Cement storage tank	1	2,400 gal
Steam generator blowdown spent resin transfer tank	1	5,980 gal
Waste processing system liquid spent resin transfer tank	1	5,980 gal
Backflushable filter crud transfer tank	1	2,185 gal
Product storage hopper	1	80 ft^3
Volume reduction system		
- spent secondary resin	1	18.4 gal/h
- evaporator concentrates chemical drain wastes	1	35 gal/h
- dry combustible wastes	1	95 lb/hr

*Quality group and seismic design in accordance with RG ~~Regulatory Guide~~ 1.143.

Table 11.3 Process and effluent radiation monitoring systems for Vogtle Electric Generating Plant, Units 1 and 2

Stream monitored	Detector type	Detectable range, μCi/cc
<u>LIQUID</u>		
Waste liquid effluent	Gamma scintillation	10^{-6} to 10^{-1}
Steam generator liquid	Gamma scintillation	4×10^{-7} to 4×10^{-2}
Nuclear service water	Gamma scintillation	4×10^{-7} to 4×10^{-2}
Steam generator blowdown	Gamma scintillation	4×10^{-7} to 4×10^{-2}
Component cooling water	Gamma scintillation	4×10^{-7} to 4×10^{-2}
Turbine building drain	Gamma scintillation	4×10^{-7} to 4×10^{-2}
Control building sump effluent	Gamma scintillation	4×10^{-7} to 4×10^{-2}
<u>GASEOUS</u>		
Plant vent effluent (low range)		
- Particulate	Beta scintillation	10^{-11} to 10^{-6}
- Iodine	Gamma scintillation	10^{-11} to 10^{-6}
- Radiogas	Beta scintillation	5×10^{-7} to 5×10^{-2}
Containment vent effluent		
- Particulate	Beta scintillation	10^{-11} to 10^{-6}
- Iodine	Gamma scintillation	10^{-11} to 10^{-6}
- Radiogas	Beta scintillation	5×10^{-7} to 5×10^{-2}
Radwaste solidification building effluent		
- Particulate	Beta scintillation	10^{-11} to 10^{-6}
- Iodine	Gamma scintillation	10^{-11} to 10^{-6}
- Radiogas	Beta scintillation	10^{-8} to 10^{-3}
Waste gas processing system effluent	G-M tube	10^{-1} to 10^{4}
Fuel handling building effluent	Thin-walled G-M tube	10^{-6} to 10^{-1}
Plant vent (high range)		
- Particulate	Passive particulate filter	NA
- Iodine	Passive iodine filter	NA
- Radiogas	Beta scintillation	10^{-6} to 10^{4} for Xe-133

Table 11.3 (Continued)

Stream monitored	Detector type	Detectable range, μCi/cc
GASEOUS (Continued)		
Main steam line	Strap-on gamma	10^{-1} to 10^{3}
Condenser air ejector and steam packing exhauster		
- Particulate	Passive particulate filter	NA
- Iodine	Passive iodine filter	NA
- Radiogas	Beta scintillation	5×10^{-7} to 10^{5} for Xe-133

12 RADIATION PROTECTION

The staff has evaluated the proposed radiation protection program presented in FSAR Chapter 12 against the criteria set forth in the Standard Review Plan (SRP) (NUREG-0800). The radiation protection measures at Vogtle are intended to ensure that internal and external radiation exposure to station personnel, contractors, and the general population as a result of station conditions, including anticipated operational occurrences, will be within applicable limits of 10 CFR 20, and will be as low as is reasonably achievable (ALARA).

The basis of the staff acceptance of the Vogtle radiation protection program is that doses to personnel will be maintained within the limits of 10 CFR 20, "Standard for Protection Against Radiation." The applicant's radiation protection designs and program features are consistent with the guidelines of Regulatory Guide (RG) 8.8, "Information Relevant to Ensuring That Occupational Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable." Some of the radiation protection measures which the applicant will use at Vogtle include: using remote handling equipment; coating concrete shields for ease of decontamination; locating radiation components in separately shielded cubicles; and training personnel in radiation protection. The applicant's use of these and other radiation protection features will help to ensure that occupational radiation exposures are maintained ALARA, both during plant operation and during decommissioning.

On the basis of its review of the Vogtle Final Safety Analysis Report (FSAR), the staff has concluded that the radiation protection measures incorporated in the design will provide a reasonable assurance that occupational doses will be maintained ALARA and below the limits of 10 CFR 20. These radiation design features are consistent with the guidelines of RG 8.8.

12.1 Ensuring That Occupational Radiation Exposures Are ALARA

12.1.1 Policy Considerations

The applicant provides a management commitment to ensure that the Vogtle plant will be designed, constructed, and operated in a manner consistent with RGs 8.8; 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable"; and 1.8, "Personnel Selection and Training" (Revision 1). The ALARA philosophy was applied during the initial design of the plant. Since then some of the applicant's experienced operating personnel have continued to review, update, and modify the plant's design and construction on the basis of exposure data and experience gained from operating nuclear power plants. This is done to ensure that occupational exposures will be kept as low as is reasonably achievable in accordance with RG 8.8 criteria. Therefore, the policy considerations are acceptable.

12.1.2 Design Considerations

The objective of the plant's radiation protection design is to maintain individual doses and total person-rem doses to plant workers, including construction workers, and to members of the general public ALARA, and to maintain individual doses within the limits of 10 CFR 20. Within restricted areas, all plant sources of direct radiation and airborne radioactive contamination are considered in the staff review.

To reduce radiation exposure, the applicant has utilized feedback information obtained from plants currently operational. Some examples of design improvements directly attributed to experiences and operations are listed below.

(1) An adequate number of equipment decontamination areas have been included to reduce congestion and reduce maintenance time.

(2) Concrete shield walls, floors, and ceiling are coated with a nonporous coating to enhance decontamination whenever a potential exists for radioactive material to leak or spill on these surfaces.

(3) To reduce the amount of radioactive sources in valve aisles, pipes through which radioactive materials run (to and from valves aisles) are routed, whenever practical, behind the shield wall separating the piece of equipment from the valves.

(4) A spent fuel pool cleanup system is utilized to maintain the radiation level of the fuel pool area at less than 2.5 mrem/hr.

(5) Instrument readouts are designed and located to minimize the time and exposure necessary to take a reading.

Utilizing the feedback information from operating plants and following RG 8.8, the applicant has incorporated the equipment design considerations listed below to satisfy Vogtle's radiation protection design objectives.

(1) Equipment that processes fluids with low radioactivity is located in cubicles that are separate from equipment that processes highly radioactive fluids.

(2) Equipment is located in accessible parts of cubicles. Equipment frequently changed in whole or in part is readily accessible.

(3) Localized shielding or space and adequate structure for localized shielding are provided as part of the shielding design.

(4) Unmortared removable block walls or easily removable floor or wall plugs minimize the radiation exposure to workers who need to remove highly radioactive components.

These design considerations conform with the guidelines of RG 8.8 and are acceptable.

12.1.3 Operational Considerations

Operational considerations at Vogtle included developing a radiological training program, a radiation zoning and access control system, and general guidelines for workers who perform maintenance in high-radiation areas. These procedures will be part of the plant's commitment to develop a radiation protection program using the recommendations of NUREG-0761, Section 5, "Radiation Protection Plan for Nuclear Power Reactor Licensees." These operational considerations ensure that operating and maintenance personnel follow specific plans and procedures in order to ensure that ALARA goals are achieved in the operation of the plant. High-radiation-exposure operations are carefully preplanned and performed by personnel well trained in radiation protection and using proper equipment. During such maintenance activities, personnel are monitored for exposure to radiation and contamination. When major maintenance jobs have been completed, radiation exposures are evaluated and compared with predicted person-rem exposures. The results are used to make changes in job procedures and techniques. The station's management personnel annually review radiation exposure trends to determine (1) major changes in problem areas, and (2) which worker groups are accumulating the highest exposures. The station personnel use these reports to recommend design modifications or changes in plant procedures. The operational considerations conform to RGs 8.8 and 8.10 and are acceptable.

The staff concludes that the policy considerations, design considerations, and operational considerations at the Vogtle facility are adequate to ensure that occupational radiation exposures will be ALARA in accordance with RGs 8.8 and 8.10 and are acceptable.

12.2 Radiation Sources

Section 12.2 of the FSAR 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. Additional information on source terms is described in Chapter 11 of the FSAR.

12.2.1 Contained Sources and Airborne Radioactive Material Sources

Inside the containment during power operation, the greatest potential for personnel dose during operation results from nitrogen-16, noble gases, and neutron

Outside the containment and after shutdown inside the containment, the primary sources of personnel exposure are fission products from fuel cladding defects and activation products, including activation corrosion products. Almost all of the airborne radioactivity within the plant results from equipment leaks. The source terms are based on 1% fuel cladding defects. The coolant and corrosion activation product source terms are based on operating experience from reactors of similar design; allowances are included for the buildup of activated corrosion products. Neutron and prompt gamma source terms are based on reactor-core-physics calculations and operating experience from reactors of similar design. The source terms presented are comparable to estimates by other applicants and are acceptable.

The applicant has tabulated the maximum expected radioactive airborne concentration in equipment cubicles, corridors, and operating areas, from equipment leaks. As the basis for these calculations, the applicant was consistent with RG 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Power Reactors," and these calculations are acceptable.

The ventilation system will be designed to change enough air each hour in those occupied areas that may contain significant airborne radioactivity, to maintain exposure to plant personnel ALARA. Air will be routed from areas of low potential airborne contamination to areas of increasing potential airborne contamination. The resulting estimated airborne radioactivity concentrations in frequently occupied areas will be below 10 CFR 20.103 limits and are acceptable. The source terms used to develop these airborne concentration values are comparable to estimates by other applicants with similar designs and are acceptable.

12.3 Radiation Protection Design Features

Section 12.3 of the FSAR describes the features that are included in the radiation protection design of the plant to maintain exposures ALARA. Separate descriptions are presented for the categories of facility design features, shielding, ventilation, and area-radiation and airborne-radioactivity monitoring instrumentation.

12.3.1 Facility Design Features

The applicant has provided evidence that the dose-accumulating functions performed by workers have been considered in the plant design. Features have been included in the design to help maintain exposure ALARA in the performance of those functions. These features will give easier access to work areas, reduce or allow the reduction of source intensity, reduce the time personnel must stay in the radiation fields, and provide for portable shielding and remote handling tools. The applicant's facility design features are consistent with the guidance of RG 8.8. Therefore, the staff concludes that the facility design features are acceptable.

The applicant has provided five major radiation zones as a basis for classifying occupancy and access restrictions on various areas within the plant. On this basis, maximum design dose rates are established for each zone are used as input for shielding of the respective zones. The areas that predictably will be occupied during normal operations and anticipated occurrences are zoned so that exposures are below the limits of 10 CFR 20, and will be ALARA. The zoning system and access control features will also meet the posting entry requirements of 10 CFR 20.203 or standard NRC technical specifications and are consistent with RG 8.8.

Several features included in the plant design and operational program will minimize the buildup of activated corrosion products, a major contribution to occupational doses. Examples include

(1) Tanks containing radioactive material have sloped bottoms wherever practical, so that the accumulation of sludge is minimized and it is easier to drain the tanks.

(2) Equipment is selected and the design is reviewed to ensure that there are no obvious ledges or pockets where radioactive material may become trapped or may accumulate.

(3) The flushing capability of radioactive service equipment is used to minimize retention of radioactive crud or sludge in the equipment before maintenance takes place or the equipment is removed.

(4) Chemistry in the primary system is controlled.

The applicant's corrosion product control features are consistent with the guidance of RG 8.8 and are acceptable.

The design features incorporated by the applicant for maintaining occupational radiation doses ALARA during plant operation and maintenance will also serve to maintain radiation doses ALARA during decommissioning operations and are, therefore, acceptable.

12.3.2 Shielding

The objectives of the plant's radiation shielding are to provide protection against radiation for operating personnel, both inside and outside the plant, and for the general public, during normal operation (including anticipated operational occurrences) and during reactor accidents. The shielding was designed to meet the requirements of the radiation dose rate zone system discussed above. Following are several of the shielding design features incorporated into the Vogtle plant.

(1) Access labyrinths are provided for rooms housing equipment that contains high-radiation sources to preclude a direct-radiation path from the equipment to accessible areas.

(2) Radioactive piping is routed through high-radiation areas where practical, or in shielded pipe chases in low-radiation areas.

(3) Shielding is provided for all equipment which is anticipated to be normally radioactive.

(4) Piping penetrations, ducts, and voids in radiation shield walls are located to preclude the possibility of streaming from a high-radiation area to a low-radiation area.

(5) Shielding is designed to be removable where required, to provide access to personnel who need to inspect, service, maintain, or replace plant equipment.

These shielding techniques are designed to maintain personnel radiation exposure ALARA. Therefore, the staff concludes that the shielding design objectives are acceptable.

The applicant's shielding design methods included the use of well-known computer codes, such as ANISN and QAD. The applicant also used shielding information from operating nuclear plants as input data for its shield design calculations. All concrete shielding in the plant will be constructed in general compliance with RG 1.69, "Concrete Radiation Shields for Nuclear Power Plants." Therefore, the staff concludes that the shielding design methodology presented is acceptable.

II.B.2 Plant Shielding for Vital Area Access

In accordance with the criteria of NUREG-0737, Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Postaccident Operations," the applicant has performed a design review of station shielding to allow access to plant areas after an accident.

The applicant has identified the systems that could be required to operate following an accident. Dose rates were calculated for vital access areas by using well-known computer codes and superimposing the effects of all sources to obtain the maximum expected dose rate throughout the plant. The radiation environment was evaluated for 1 hour, 1 day, and 1 week following the reactor shutdown after a loss-of-coolant accident (LOCA) with significant core damage. Dose rate zone maps were provided for each relevant area.

Vital areas requiring accessibility following an accident were identified with respect to location, occupancy requirements, and maximum dose levels. Vital areas include: the control room; remote shutdown panel; technical support center; safety-related motor control center and switchgear in the control building, auxiliary building, diesel generator building, auxiliary feedwater pump house, and radiochemistry laboratory.

The applicants' shielding design review showed that Vogtle meets the less-than-15-mrem/hr criterion outlined in NUREG-0737 for vital areas that require extended or continuous occupancy. Additionally, General Design Criterion (GDC) 19 limits are met for those vital areas requiring only infrequent access.

On the basis of its review, the staff has concluded that the applicant has performed a radiation and shielding design review for access to vital areas in accordance with Item II.B.2 of NUREG-0737.

12.3.3 Ventilation

The ventilation system at Vogtle will be designed to protect personnel and equipment from extreme thermal environmental conditions and ensure that plant personnel are not inadvertently exposed to airborne contaminants exceeding those given in 10 CFR 20.103. The applicant intends to maintain personnel exposures ALARA by

(1) maintaining airflow from areas of potentially low airborne contamination to areas of higher potential concentrations,

(2) ensuring negative or positive pressures to prevent exfiltration or infiltration of potential contaminants, respectively,

(3) locating ventilation system intakes so that intake of potentially contaminated air from other building exhaust points is minimized.

These design criteria are in accordance with the guidelines of RG 8.8. Some examples of features in the ventilation system that will reduce exposure are:

(1) Shielding is provided as necessary around heating, ventilating, and air conditioning (HVAC) filters (for example, charcoal filters) to minimize

radiation exposure to plant personnel as a result of source buildup on the filter.

(2) Dampers provided before and after the filter train, isolate the train during filter changes.

The staff concludes that the applicant's ventilation system will keep personnel exposure below 10 CFR 20 values and is, therefore, acceptable.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring Instrumentation

The applicant's area radiation monitoring system is designed to

(1) provide operating personnel in the main control room with an indication and record of gamma-radiation levels at selected locations within the various plant buildings,

(2) contribute radiation information to the control room so that correct decisions may be made with respect to deployment of personnel in the event of a radiation release,

(3) help detect unauthorized or inadvertent movement of radioactive materials in the plant, including the radwaste area,

(4) supplement other systems in detecting abnormal migrations and radioactive material in or from the process streams,

(5) provide local alarms at key points where a substantial change in radiation levels might be of immediate importance to personnel frequenting the area,

(6) provide a continuous record of radiation levels at key locations throughout the plant.

To meet these objectives, the applicant plans to use monitors in areas where personnel may be present and where radiation levels could become significant. The staff has evaluated the plant's area radiation monitoring system against the criteria of NUREG-0800, Section 12.3, and finds the system acceptable.

The applicant has indicated in its FSAR that a request for an exemption from the requirements of 10 CFR 70.24 will be made for two criticality accident monitors around the fuel storage area. The NRC staff will review the exemption request when it is formally submitted. This is a confirmatory item.

To meet the criteria of NUREG-0737, Item II.F.1(3), "Additional Accident-Monitoring Instrumentation: Containment High-Range Radiation Monitor," the applicant has committed to install and calibrate two high-range gamma monitors at Vogtle in each of the units that meets the criteria of Table II.F.1(3) of NUREG-0737. As part of the FSAR, the applicant has submitted drawings showing where monitors are located in the plant.

On the basis of its review, the staff has concluded that the applicant meets the staff's position in Item II.F.1(3) of NUREG-0737.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

The design objectives of the applicant's airborne radioactivity monitoring system are

(1) to assist in maintaining occupational radiation exposure to airborne contaminants ALARA,

(2) to check on the integrity of those systems being monitored that contain radioactivity,

(3) to warn of inadvertent release of airborne radioactivity to protect plant personnel from overexposure.

The applicant will install fixed continuous airborne monitoring systems (CAMS) that will be dedicated to monitoring a particular ventilation pathway for high airborne radioactivity. Visual and audible alarms are provided in the main control room for these monitors so that appropriate administrative actions may be taken when this radioactivity is detected. The fixed CAMS have the capability to detect 10 MPC*-hours of particulate and iodine radioactivity in any compartment that has a possibility of containing airborne radioactivity and that may be occupied by personnel. The applicant will provide portable continuous air monitors when they are needed to monitor air in areas not provided with fixed airborne-radioactivity monitors. Airborne-radioactivity monitors will be calibrated at regular time intervals.

The objective and location criteria of Vogtle area- and airborne-radiation monitoring systems are in conformance with 10 CFR 20 and RGs 8.2 and 8.8, and are acceptable.

The staff concludes that the equipment and facility design features, shielding, ventilation, area-radiation monitoring, and airborne-radioactivity monitoring systems at Vogtle are sufficient to ensure that radiation exposures are ALARA and are acceptable.

12.4 Dose Assessment

Section 12.4 of the FSAR describes the applicant's assessment of (1) the collective radiation-dose-equivalent workers will receive in operating and maintaining the facility, and (2) the airborne radioactivity to which workers will be exposed in operating the facility. The way in which the assessment of collective dose from plant operation was factored into the radiation protection design is also described.

The applicant presented its estimate of the collective radiation dose equivalent that workers will receive from the operation of the facility. The estimate is 418 person-rems per year per unit; this estimate includes the staff and contractor personnel at the station. The method used by the applicant is not consistent with the acceptance criteria in NUREG-0800, that is, listing the methods outlined in RG 8.19. The applicant's assessment was not based on an analysis

*maximum permissible concentration

of the tasks involved in the operation of the plant, the expected radiation dose rates, and the personnel required to perform those tasks. However, the applicant based the estimate on dose data from operating pressurized-water reactors (PWRs). The applicant has examined operating experience at several operating PWRs to determine those activities that contribute significantly to the collective occupational dose equivalent. Then, the applicant proceeded to examine possible design and procedure changes to reduce doses in those activities. Through this process the applicant has made several improvements in the Vogtle design to reduce doses.

Therefore, the method used by the applicant served the purpose of a dose assessment identification of instances where additional dose-reduction features are justified. Although the person-rem estimate was not made in accordance with RG 8.19, the staff concludes that the applicant's dose assessment is acceptable, because it meets the intent of the regulatory guide, in that it identified potential unnecessary exposures, minimized foreseen doses, and led to an examination of methods and techniques for reducing doses.

The applicant has estimated the potential exposure of individual workers to airborne radioactivity in various parts of the plant. These estimates, in most cases, are a small fraction of the allowable exposures given in 10 CFR 20.103. Therefore, the staff concludes that the assessments of exposure to airborne radioactivity are acceptable.

The applicant has also estimated the potential whole-body exposure to construction workers during Unit 2 construction because of the operation of Unit 1. These estimates are only a small percentage of the allowable whole-body exposure given in 10 CFR 20.105 and are, therefore, acceptable.

12.5 Operational Radiation Protection Program

FSAR Section 12.5 describes the applicant's health physics program. The description includes the radiation protection organization, equipment, instrumentation, facilities, and the procedures for radiation protection. Chapter 13 of the FSAR gives additional information on the plant's organization.

The objectives of the health physics program provide reasonable assurance that the limits of 10 CFR 20 are not exceeded, further reduce unavoidable exposures, and ensure that individual and total occupational radiation exposures are maintained ALARA.

12.5.1 Organization

The Health Physics Superintendent is the Radiation Protection Manager (RPM) at Vogtle and is responsible for implementing and enforcing the plant's health physics program. However, the ultimate responsibility of the health physics program lies with the Plant Manager. The Health Physics Superintendent is a member of the plant's ALARA review committee. The Health Physics Supervisor or a health physicist will act as the backup to the RPM if the RPM is away from the plant.

The Vogtle radiation-protection organization has been evaluated in accordance with the position of NUREG-0731, "Criteria for Utility Management and Technical Organization," and RG 8.8 (Section C.1.b(2) and (3)).

The paragraphs that follow present an evaluation of how the health physics organization for Vogtle coordinates with staff positions that involve plant organization and management criteria.

(1) The organization description for Vogtle shows that the Health Physics Superintendent (RPM) reports to an Assistant Plant Manager and has direct access to the Plant Manager in all radiation protection matters. In addition, the RPM has access to other station supervisors through regular contact at morning meetings. This satisfies the requirement of RG 8.8 and is acceptable.

(2) The radiation protection section and chemistry section are currently separated at Vogtle. This meets the recommendation of RG 8.8 and NUREG-0731, and is acceptable.

(3) The applicant has shown that the qualifications of Vogtle's Health Physics Superintendent meet the requirements of RG 1.8, "Personnel Selection and Training," and the incumbent is acceptable as the plant's RPM.

(4) The backup to the RPM during his absence from the station will be selected from the positions of Station Health Physicist or Health Physicist. The applicant has committed to using the criteria of ANSI 3.1, December 1979 draft, in selecting the individual temporarily filling the RPM position. This satisfies the requirements of NUREG-0731 and is acceptable.

(5) The applicant has committed to having at least one rad/chem technician on site at all times. This satisfies the requirements of NUREG-0731 and is acceptable.

Vogtle has shown that the current health physics organization meets staff criteria as stated in NUREG-0731 and RG 8.8 for an acceptable radiation protection organization. The staff finds the organization acceptable.

~~12.5.2 Health Physics Facilities~~

~~Later~~

12.5.~~1~~2 Equipment, Instrumentation, and Facilities

The radiation-protection features at Vogtle include a counting room, hot laboratory, cold laboratory, radiation-protection office, calibration facility, laundry room, personnel-decontamination room, and a mask-cleaning facility. These facilities are sufficient to maintain occupational radiation exposure ALARA and are consistent with the provisions of RG 8.8. Equipment to be used for radiation-protection purposes includes portable radiation-survey instruments, personnel-monitoring equipment, fixed and portable area- and airborne-radioactivity monitors, laboratory equipment, air samples, respiratory-protective equipment, and protective clothing. The number and types of equipment to be used are adequate and provide reasonable assurance that the applicant will be able to maintain occupational exposure ALARA.

In order to meet the criteria of NUREG-0737, Item III.D.3.3, "Improved Inplant Iodine Instrumentation Under Accident Conditions," the applicant has committed to having the capability to sample and determine airborne-radioiodine concentration by using portable air samplers using silver zeolite as a sample medium. If entrapped noble gases interfere with the radioiodine analysis, clean air or nitrogen flushing will be performed.

Low-background-counting facilities for postaccident analysis are available.

The use of sampling equipment and analysis systems for the determination of radioiodine during an accident situation has been incorporated into the Vogtle plant's training program.

The postaccident radioiodine sampling and analysis provisions described for the Vogtle plant meet the staff's position as outlined in NUREG-0737 and are acceptable.

12.5.~~4~~3 Procedures

All permanent plant personnel entering controlled areas (where radioactivity may be present) will be assigned thermoluminescent dosimeter (TLD) badges and pencil dosimeters. The TLDs will be processed monthly, or more frequently if significant exposures are suspected, and the pencil dosimeters will be read daily. In accordance with the requirements of 10 CFR 20, ~~special~~ neutron surveys will be performed when plant personnel enter areas that may ~~be contaminated with neutrons~~ have neutron radiation. Whole-body counts of all plant personnel will be conducted on a scheduled basis and other bioassays will be provided when the plant's health physics staff, using the guidance of RGs 8.9 and 8.26, find such counts are needed. All radiation-exposure information will be processed and recorded in accordance with 10 CFR 20.

Maintenance, repair, surveillance, and refueling procedures and methods used by the applicant are reviewed to ensure that all of the plant's radiation-protection procedures, practices, and criteria have been considered, to ensure that occupational radiation exposures will be ALARA and in accordance with RG 8.8. Procedures will have been developed to meet the requirements of

RG 1.33, "Quality Assurance Program Requirements (Operation)." The staff has reviewed the applicant's commitment to provide plant radiation-protection procedures and finds it acceptable.

12.6 Conclusions

On the basis of information presented in the FSAR and the applicant's responses to the staff's questions, the staff concludes, pending resolution of the confirmatory items, that the applicant intends to implement a radiation-protection program that will maintain inplant radiation exposures within the applicable limits of 10 CFR 20 and will maintain exposures ALARA in accordance with RG 8.8. The confirmatory item regards the applicant's exemption request from the requirements of 10 CFR 70.24 as discussed in Section 12.3.4.1 of this draft SER.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Operations

Later

13.2 Training

The applicant's training programs for licensed reactor operators and nonlicensed plant staff were reviewed according to the Standard Review Plan (NUREG-0800), Section 13.2. The staff acceptance criteria included applicable portions of 10 CFR Parts 19, 50, and 55, and Regulatory Guides 1.8 and 1.149, the clarification of TMI Action Plan (NUREG-0737), and H. R. Denton's letter of March 28, 1980, to all power reactor applicants and licensees as well as ANSI/ANS 18.1-1971.

13.2.1 Licensed Operator Training Program

The training program for Vogtle Electric Generating Plant's (Vogtle's) licensed reactor operators and senior reactor operators has been implemented to develop and maintain an organization fully qualified to operate the plant and to maintain plant safety. The initial and requalification programs are designed to meet the requirements of 10 CFR Parts 50 and 55 and TMI Action Plan-related requirements, and are based on the individual employee's level of education, experience, and skills, as well as on the level of assigned responsibility and intended position. The applicant's training programs are the result of initial position task analysis for Vogtle staffing and include methods to obtain feedback to update training plans.

13.2.1.1 Initial Training Program

The initial training program for personnel who will be licensed consists of the following segments:

(A) Nuclear Power Plant Theory

Four nuclear power plant theory programs of varying durations are utilized to teach personnel: (1) a 1-week program for personnel with commercial reactor license or NRC certification, (2) a 3-week program for personnel with one year of military PWR (pressurized-water-reactor) experience, (3) a 5-week program for personnel with a degree in engineering or applicable science, and (4) a 12-week program for personnel with no previous reactor experience. These programs for licensed operators provide training in fundamentals of reactor theory, general core design, radiological safety and, for personnel with no previous experience, training in materials, thermodynamics, fluid mechanics, and heat transfer. The methods used are classroom instruction or self-study, for any supervisory, staff, or operator position.

(B) Vogtle Systems and Procedures

This segment of licensed operator training is provided in four different programs for different experience groups: (1) 1 week of training for personnel with commercial Westinghouse reactor license or NRC certification, (2) 4 weeks of training for personnel with commercial reactor license, NRC certification, or 1 year of military PWR experience, (3) 6 weeks of training for personnel with a degree in engineering or applicable science and (4) 12 weeks of training for personnel with no previous experience. This portion of licensed operator training covers procedures for design and operating changes, reactor coolant system mechanical design, reactivity control mechanisms and indications, reactor safety systems, emergency and reserve systems, containment and shielding, radiation monitoring system, auxiliary systems, and radioactive waste for any supervisory, staff, or operator position. Radioactive waste is not included as a topic in training for plant operator or assistant plant operator training. Either classroom instruction or self-study is employed.

(C) Vogtle License and Technical Specifications

This segment of training for licensed plant staff consists of a 1-week program of instruction for personnel with commercial reactor license or NRC certification, personnel with 1 year of military PWR experience as a reactor operator, and personnel with a degree in engineering or applicable sciences. The training includes license conditions and limitations, design and operating limitations, and procedures for design and operating changes. Classroom training is employed for any supervisory, staff, or operator position.

(D) Fuel Handling and Core Alteration

This 3-day classroom training program is offered for any supervisory, staff, or operator position for personnel with a degree in engineering or applicable science or with some commercial or military operating experience. The training covers facilities and procedures for fuel handling and core alterations.

(E) Control Room Operations

This segment of licensed operator training consists of three programs: (1) a 4-week program (including 80 hours on a Vogtle simulator), for personnel with commercial Westinghouse reactor license or NRC certification, (2) a 5-week program (including 100 hours on a Vogtle simulator) for personnel with a commercial reactor license or NRC certification, 1 year of military PWR experience, or a degree in engineering or applicable science, and (3) a 6-week program (including 100 hours on a Vogtle simulator) for personnel with no previous experience. Training includes general operating characteristics, specific operating characteristics; load changes; operating limitations; standard, emergency, and plant procedures; control manipulation; mitigating core damage (incore instrumentation, excore instrumentation, vital instrumentation, primary chemistry, radiation monitoring, gas generation); and transients. The program consists of training on a simulator and in the classroom covering all topics, and is offered for any supervisory, staff, operator, plant operator, or assistant plant operator position.

(F) Vogtle Walkthrough Training

Three programs are utilized for plant walkthrough training which is designed to familiarize the trainee with plant layout, systems, and equipment. For personnel who have a commercial Westinghouse reactor license or NRC certification, the walkthrough program consists of 3 weeks' training at Vogtle (previous senior reactor operator license, or reactor operator license), or 3 months on-the-job training at a similar operating reactor combined with training at Vogtle after fuel load (upgrade to senior reactor operator from reactor operator license or certification). For personnel with a commercial reactor license or NRC certification, (previous senior reactor operator license or reactor operator license) training consists of 5 weeks' walkthrough at Vogtle or 3 months on-the-job training (upgrade to senior reactor operator from reactor operator license or certification) at a similar operating reactor combined with training at Vogtle after fuel load. For personnel with 1 year of military PWR experience as a reactor operator, or a degree in engineering or applicable sciences, the training program consists of a Vogtle walkthrough for 5 weeks. Walkthrough training for personnel with no previous experience is combined with the final review and audit training.

(G) On-the-Job Training

On-the-job training is provided for personnel with (1) 1 year of military PWR experience, (2) a degree in engineering or applicable sciences, and (3) no previous experience. The on-the-job training is a 3-month program, including assignment at a similar operating reactor combined with shift experience at Vogtle after fuel load, that will provide for regular participation in day-to-day operations, including manipulation of reactor and reactor equipment controls.

(H) Review and Audit

The final review and audit portion of training consists of 1 week of training for all personnel with previous commercial or military experience or with a degree in engineering or an applicable science. For personnel with no previous experience, the review and audit training is combined with Vogtle walkthrough training in a 6-week program that utilizes the simulator, the classroom, and self-study.

Conclusions

The applicant has not submitted the details of the simulator training program. The description of the control room operations portion of the training program includes both classroom and simulator topics. Therefore, the staff cannot conclude that the simulator training program is acceptable.

The applicant has not described the kind of training that trainees will receive in the walkthrough training program, including the objectives to be accomplished in this segment of training.

The applicant has not described the objective and content of the review and audit segment of licensed reactor operator and senior operator training nor the method for ensuring successful completion of the training review.

The applicant has not provided a separate description of training for licensed senior reactor operators. Topics as described in 10 CFR 55.22 for senior reactor operator examinations are included in various segments of the training curriculum as described. The applicant should ensure that appropriate segments of training are specified for licensed reactor operators and licensed senior reactor operators.

The initial qualifications training for personnel with commercial or military operating experience, NRC certification, or an engineering degree, as described in FSAR Tables 13.2.1-1 through 13.2.1-4, does not provide the required description of training as described in Enclosure 1 of H. R. Denton's March 28, 1980, letter. The staff will review the program when it is submitted and report its findings in the SER.

The applicant has not provided evidence that formal segments of the training program as described above will be substantially completed when the preoperational test program begins. The schedule for licensed operator training programs as shown in FSAR Figure 13.2.1-2 does not provide sufficient detail and breakdown on training segments to show that training can be accomplished before the scheduled preoperational test program begins. FSAR figures 13.2.1-2 and 13.2.1-3 should be clarified to show training segments to be completed before preoperational test and to reflect other training times required before examinations.

Insufficient information is given to determine whether the number of plant personnel for whom training is planned in preparation for operator and senior operator examinations before criticality is sufficient to meet applicable Technical Specification conditions.

Until the above information is provided in amendments to the FSAR, the staff cannot conclude that the applicant's initial training program for reactor operators and senior reactor operators is acceptable. As discussed above, this item is open.

13.2.1.2 Licensed Operator Requalification Training Program

The applicant has described a licensed operator requalification training program which includes classroom study, on-the-job training, reactivity control manipulations and evaluation by annual observation and written examination for all licensed personnel.

The following programs comprise the licensed operator requalification program at Vogtle:

(A) <u>Classroom Study</u>

Topics for classroom study include those subjects that training feedback indicates need additional training, as well as the following topics:

- theory and principles
- general and specific plant operating characteristics
- plant instruments and controls
- plant protection systems
- engineered safety systems
- procedures
- radiation control and safety
- technical Specifications

- applicable portions of 10 CFR
- quality assurance for operations
- major upcoming events
- heat transfer, fluid flow, and thermodynamics
- mitigation of accidents involving a degraded core

(B) Reactivity Controls

The following control manipulations will be performed on the Vogtle simulator (the starred items shall be performed on an annual basis and all other items shall be performed on a 2-year cycle):

(1)* plant or reactor startups to include a range within which reactivity feedback from nuclear heat addition is noticeable and heatup rate is established

(2) plant shutdown

(3)* manual control of steam generators and/or feedwater during startup and shutdown

(4) boration and/or dilution during power operation

(5)* any significant (10%) power changes in manual rod control

(6)* loss of coolant including:

(a) significant pressurized water reactor steam generator leaks
(b) inside primary containment
(c) large and small, including leak rate determination
(d) saturated reactor coolant response

(7) loss of instrument air

(8) loss of electrical power (and/or degraded power sources)

(9)* loss of core coolant flow/natural circulation

(10) loss of condenser vacuum

(11) loss of service water

(12) loss of shutdown cooling

(13) loss of component cooling system or cooling to an individual component

(14) loss of normal feedwater or normal feedwater system failure

(15)* loss of all feedwater (normal and emergency)

(16) loss of protective system channel

(17) mispositioned control rod or rods (or rod drops)

(18) inability to drive control rods

(19) conditions requiring use of emergency boration

(20) fuel cladding failure or high activity in reactor coolant or offgas

(21) turbine or generator trip

(22) malfunction of automatic control system(s) which affects reactivity

(23) malfunction of reactor coolant pressure/volume control system

(24) reactor trip

(25) main steamline break (inside or outside containment)

(26) nuclear instrumentation failure(s)

Training will include reviews of plant design changes and procedure changes. In addition, each licensed operator will demonstrate in the performance of assigned duties, satisfactory understanding of the operation of systems, apparatus, and operating procedures. Criteria for requiring a licensed individual to participate in accelerated requalification are consistent with passing grades for issuance of a license.

The applicant has described an evaluation procedure for a requalification training program by which annual written examinations are given to all licensed personnel to determine areas in which requalification training is needed. Attendance is required at requalification lectures for any section grade less than 80% but not less than 70% on an annual examination.

Accelerated requalification training will be given to operators and senior operators who receive an overall grade of less than 80% on an annual requalification examination, a section grade of less than 70%, or an unsatisfactory performance evaluation. Individuals will be relieved of operating duties to participate in accelerated requalification training in appropriate areas and must receive a grade of not less than 80% on examinations.

Conclusions

The applicant has not stated that the requalification program will be implemented within 3 months after issuance of an operating license as required by 10 CFR 50.54(i-1). The applicant should provide clarification to ensure implementation of the requalification training program as required. The applicant has not described procedures for retaining records of the requalification training program to document the participation of each licensed operator and senior operator in the requalification program.

With respect to instructions in topics of fluid flow, thermodynamics, and heat transfer, the staff requires the applicant to provide a program in accordance with the guidelines as outlined in Enclosure 2 of H. R. Denton's March 28, 1980, letter. The applicant should describe the requalification training for heat transfer, fluid flow, and thermodynamics for the staff's review.

With respect to training for mitigating core damage, the staff requires the applicant to provide a requalification program in accordance with the guidelines as outlined in Enclosure 3 of H. R. Denton's March 28, 1980, letter. The staff will review the program when it is submitted and will report its findings in the SER.

The applicant has not included a review of all abnormal and emergency procedures on a regularly scheduled basis as part of the requalification program.

As indicated above, the staff finds that the applicant's requalification training program for licensed reactor operators and senior reactor operators does not fully conform to the requirements as specified in 10 CFR Part 55, Appendix A, and in the letter from H. R. Denton to all power reactor applicants and licensees dated March 28, 1980. Therefore, the staff cannot conclude that the applicant's requalification training program is acceptable, and it is an open item.

13.2.1.3 TMI-Related Requirements for New Operating License

I.A.2.1 Immediate Upgrading of Operator and Senior Reactor Operator Training and Qualification

The applicant has established a program to ensure that all reactor operator and senior reactor operator license candidates have the prescribed experience, qualification, and training.

Before applying for the examination, each candidate for operator license will be certified competent to take the NRC license examination. As an operating license applicant, Vogtle Electric Generating Plant is not subject to the one-year experience requirement for cold license senior reactor operator candidates. However, after one year of station operation, individuals applying for a senior reactor operator license will be required to comply with the one-year experience requirement for hot license senior reactor operator applicants, unless previously experienced in an equivalent position at another nuclear plant or at a military propulsion reactor. The experience of license applicants will be documented by the applicant on a case-by-case basis in sufficient detail so that the staff can make a finding regarding equivalency. Applicants for the senior reactor operator license who possess a degree in engineering or applicable science are considered to meet the one-year experience requirement as a reactor operator provided that they: (1) satisfy the requirements set forth in Sections A.1.a and A.2 of Enclosure 1 to the March 28, 1980, letter from H. R. Denton to all power reactor applicants and licensees and (2) have participated in a training program equivalent to the one that applicants for cold senior reactor operator licenses must take.

The requirement for three months on-shift experience for control room operators and senior reactor operator candidates as an extra person on shift will be met by Vogtle cold license candidates by a combination of training at a similar operating reactor and shift assignment at Vogtle after initial fuel load. Vogtle Electric Generating Plant will be required to provide the required shift experience for hot license candidates three months after the plant begins operation.

Conclusions

The applicant's training program has not included training in heat transfer, fluid flow, and thermodynamics for all candidates for an operator's license.

Reactor and plant transient training is performed by each license applicant on the Vogtle simulator which conforms with RG 1.149 with one exception. The applicant has not committed to provide for retesting of simulator response as specified in §5.4 of ANSI/ANS-3.5-1981. The purpose of the simulator performance testing is to verify software capability and modeling of plant dynamics based on analyses of plant transients and accidents. Therefore, this aspect of the applicant's simulator is not found to be fully in compliance with provisions of RG 1.149. The staff cannot conclude at this time that the applicant's training program for licensed reactor operators and licensed senior reactor operators fully satisfies the requirements of NUREG-0737 Item I.A.2.1, and it is therefore, open.

I.A.2.3 Administration of Training Program

The applicant has described an instructor qualification program which includes certification of instructors who teach systems, integrated response, transients, and simulator courses. However, the applicant has not committed to successful completion of an NRC-administered senior reactor operator examination for certification of instructors. The NRC will administer examinations for certifying instructors upon request. This requirement does not prevent noncertified members of the training staff from teaching special subjects such as reactor theory, heat transfer, fluid mechanics, instrumentation, health physics, and chemistry. The applicant has described a program for requalifying certified instructors. The applicant should clarify the position regarding NRC certification by examination before the staff can find the administration of training programs to be acceptable. This is an open item.

II.B.4 Training for Mitigating Core Damage

The applicant has indicated that licensed personnel, certified instructors, and shift technical advisors will receive training for mitigating core damage. The applicant should ensure that plant managers as well as managers and technicians in the health physics, instrumentation and control, and chemistry departments will receive training for mitigating core damage commensurate with their responsibilities as required.

In FSAR Table 13.2.2-1 the applicant has described a program for training for mitigating core damage. The description does not include the required topics described in H. R. Denton's March 28, 1980, letter to all power reactor applicants and licensees. Although the course syllabi for license training in Tables 13.2.1-1 through 13.2.1-5 do describe the required topics, the applicant should resolve the discrepancy to ensure that the program for training to mitigate core damage fully satisfies the requirements per NUREG-0737. The staff will review the applicant's clarification which should be submitted in an amendment ~~supplement~~ to the FSAR.

13.2.2 Training for Nonlicensed Plant Staff

In the FSAR the applicant has described the training given to nonlicensed personnel. The training program for nonlicensed personnel includes training for health physics and radiochemistry personnel (foremen and supervisors), instrumentation and control personnel (foremen and supervisors), mechanical maintenance personnel (foremen and supervisors), electrical maintenance personnel (foremen and supervisors), shift technical advisors, nonlicensed operators, instructors, general employee training, fire brigade training, independent review boards, quality control personnel, and engineering and technical support personnel.

The applicant has described initial, continuing, and requalification training programs and provided curriculum outlines for the training programs of the above categories of personnel. The applicant has not described the organization teaching the course or supervising instruction for nonlicensed staff nor the distribution of classroom and on-the-job training, before and after initial fuel loading for nonlicensed personnel.

The mechanical and electrical maintenance training programs' initial training does not include health physics training. The applicant should clarify the extent to which the health physics training is contained in the general employee training and whether additional health physics training should be provided for electrical and mechanical maintenance personnel for compliance with 10 CFR 19.12.

The requalification training for nonlicensed personnel as described in FSAR Sections 13.2.2.1.1 through 13.2.2.1.4 does not include refresher instruction in administrative procedures or radiation protection, emergency, and security procedures.

On the basis of the above evaluation, the staff cannot find the applicant's training for nonlicensed plant staff acceptable. Until the information requested above is provided, this item shall remain open.

13.2.2.1 Fire Protection Training

The applicant's description of fire brigade training includes initial training consisting of four days' instruction in organization and responsibilities, fire behavior, fire suppression systems, plant equipment layout and operation, fire fighting strategies, type and location of potential fires, forceable entry, fire fighting equipment, field practice demonstration, and fire protection system. In addition, a quarterly training series for fire brigade members includes a review of changes in the fire protection program, refresher training, and drills. Each fire brigade member will participate in at least two drills each year. Each year, fire brigade members will practice putting out fires and using emergency breathing apparatus; they will also perform a drill on a back-shift.

The applicant's description of fire brigade training does not include instruction in the fire fighting plan in the curriculum for initial fire brigade training provided in FSAR Section 13.2.2.1.9. FSAR Section 13.2.2.1.9 also does not include or reference the specific items of BTP CMEB 9.5-1 items (a) through (j), regarding the content of the initial classroom instruction for all fire brigade members. Where applicable, FSAR Section 13.2.2.1.9 should cross-reference relevant subsections of FSAR 9.5 which describe portions of fire brigade training. The applicant has not addressed instruction by qualified individuals, use of regularly planned drills, meetings every three months, drills, and refresher classroom training for fire brigade members in FSAR Section 13.2.2.1.9 or cross-referenced FSAR sections.

The applicant has not described fire protection training for other plant employees (instruction and drills) nor training for the fire protection staff. Therefore, the staff cannot conclude that the applicant's fire protection training program is acceptable; thus, this is an open item.

13.2.2.2 Shift Technical Advisor Training

The applicant has described the qualifications and training for the Shift Technical Advisor (STA). The STA will have a bachelor's degree in a scientific or engineering discipline and one year of nuclear power plant experience performing reactor operator or senior reactor operator duties for that type of reactor, or the candidate will receive one month of on-the-job training as an extra STA.

The STA training program includes 16 weeks of training in the classroom and plant, covering reactor theory, design characteristics, transient analysis, administrative controls, and leadership, as well as a 4-week operations training program covering normal, transient, and accident plant conditions, including a minimum of 80 hours of simulator manipulations. Shift Technical Advisors will participate in the same requalification program as licensed operators. Persons not actively performing STA duties for 30 days or more will receive training before assuming responsibilities of the position. Persons not performing STA duties for 6 months or more will undergo requalification training before resuming STA duties.

The applicant has not identified the training for mitigating core damage as part of the STA training program, as described in FSAR Section 13.2.2.1.5. Required topics which must be included, according to H. R. Denton's March 28, 1980, letter to all power reactor applicants and licensees are:

- o incore instrumentation
- o excore nuclear instrumentation
- o vital instrumentation
- o primary chemistry
- o radiation monitoring
- o gas generation

The applicant has not provided a comparison between the STA training program and Institute of Nuclear Power Operations (INPO) recommendations for Shift Technical Advisor training (NUREG-0737, Item I.A.1.1 and Appendix C).

FSAR Figure 13.2.2-1 and Section 13.2.2.5 do not indicate the schedule of each segment of the training programs for nonlicensed personnel and the extent to which the training programs have been accomplished relative to FSAR submittal. The applicant has not provided adequate information to determine whether the number of people for whom training is planned before criticality is reached is sufficient to ensure satisfying applicable Technical Specification conditions.

On the basis of the above discussion, the staff concludes that the Shift Technical Advisor program is not acceptable, and is, therefore, an open item. Additional information and clarification is needed to meet licensing conditions as specified in 10 CFR 19.12, 10 CFR 50.34(b), and 10 CFR 50.40(b) as they relate to training as an integral part of technical qualifications of personnel.

13.3 Emergency Planning

Later

13.4 Operational Review

Later

13.5 Station Procedures

13.5.1 Administrative Procedures

Later

13.5.2 Operating and Maintenance Procedures

The staff has reviewed the applicant's plan for development and implementation of operating and maintenance procedures according to SRP Section 13.5.2. The review was conducted to determine the adequacy of the applicant's program for assuring that routine operating, offnormal, and emergency activities will be conducted in a safe manner. The review was based on information in the FSAR, including the Amendment 8 response to a staff request for additional information and other correspondence from the applicant dated April 14 and May 1, 1984.

The staff evaluated (1) the applicant's classification system for procedures that are performed by licensed operators in the control room, and for other operating and maintenance procedures; (2) the applicant's plan for completing operating and maintenance procedures during the initial plant testing program to allow for correction before fuel loading; (3) the applicant's program for complying with RG 1.33, Revision 2, "Quality Assurance Program Requirements," regarding the minimum procedural requirements for safety-related operations; (4) conformance with ANSI N18.7-1976/ANS 3.2, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants;" and (5) the applicant's program for complying with the requirements of Supplement 1 to NUREG-0737.

In the FSAR, the applicant has committed to a program in which activities important to safety are to be conducted in accordance with detailed written and approved procedures that will satisfy RG 1.33, Revision 2, and ANSI/ANS 3.2-1978.

As described in the FSAR, the applicant will use the following procedure categories for those operations performed by the plant operating staff:

- plant/unit
- system
- surveillance
- annunciator response
- abnormal
- emergency
- maintenance
- health physics
- laboratory
- refueling
- emergency plan implementation

The staff has concluded that the applicant's program for use of operating and maintenance procedures satisfies the relevant requirements of 10 CFR 50.34 and is consistent with the guidance provided in RG 1.33, Revision 2, and ANSI/ANS 3.2-1978.

The applicant's operating and maintenance procedures include the plant emergency operating procedures (EOPs). In a submittal dated April 14, 1983, the applicant committed to develop EOPs as required by Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability." In this submittal, the applicant agreed to provide the staff its procedures generation package (PGP) approximately 28 months before the scheduled fuel load date of September 1986. The applicant also stated that the EOPs written in accordance with the PGP will be developed and implemented approximately 24 months before fuel load. To preclude the possibility of training operators on procedures developed using an unacceptable PGP, the PGP should be submitted 3 months before the start of operator training. In a letter dated May 1, 1984, from D. O. Foster to Elinor G. Adensam, the applicant submitted its PGP in accordance with the above commitments. The staff's evaluation of that submittal follows. The staff finds the applicant's commitment to Supplement 1 to NUREG-0737 to be acceptable.

13.5.2.1 Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

13.5.2.1.1 Introduction

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, the staff required licensees of operating plants, applicants for operating licenses, and licensees of plants under construction to (1) perform analyses of transients and accidents, (2) prepare EOP guidelines, (3) upgrade EOPs, and (4) conduct operator retraining (see also NUREG-0737, Item I.A.2.1). EOPs are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Clarification of the scope of the task and appropriate schedule revisions were included in NUREG-0737 (Item I.C.1) and Supplement 1 to NUREG-0737, which require development and submittal of PGPs to NRC. The PGP describes how the Vogtle EOPs will be written using the generic Emergency Response Guidelines (ERGs) developed by the Westinghouse Owners Group.

The NRC staff reviewed the proposed Westinghouse Owners Group ERGs as described in Westinghouse Owners Group letters of November 30, 1981; July 21, 1982; January 4, 1983; November 30, 1983; and May 4, 1984; and the material accompanying those letters. The staff determined that the guidelines are based on reanalysis of transients and accidents and concluded that the guidelines are acceptable for implementation as discussed in Generic Letter 83-22, "Safety Evaluation of 'Emergency Response Guidelines.'"

In accordance with NUREG-0737 Item I.C.7, "NSSS Vendor Review of Procedures," NSSS vendor review of low-power testing, power ascension, and EOPs was necessary to further verify adequacy of the procedures. Because the applicant has committed to implement procedures based on the NRC-approved Westinghouse ERGs, the staff does not consider an additional NSSS vendor review of the EOPs necessary. In addition, because the NSSS vendor will review preoperational and initial startup testing procedures (Section 14.2.2.2 of the FSAR), the staff considers NUREG-0737 Item I.C.7 resolved.

Since the incident at TMI-2, applicants have been required to satisfy NUREG-0737, Item I.C.8, "Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants." This pilot monitoring program was used on an interim basis for evaluating applicants' EOPs before the staff approved generic technical guidelines and developed the long-term program for upgrading EOPs. This is no longer necessary because the NRC has approved the Westinghouse ERGs and the applicant has committed to develop EOPs based on those ERGs. Therefore, the staff considers TMI-2 Task Action Plan Item I.C.8 resolved.

In addition to the Westinghouse Owners Group efforts, the staff has published guidelines for long-term upgrading of EOPs (NUREG-0899) in accordance with TMI Action Plan Item I.C.9, "Long-term Program Plan for Upgrading of Procedures." These guidelines should be used in preparing the Vogtle EOPs.

In a letter dated May 1, 1984, the applicant submitted a PGP for Vogtle in accordance with the requirements of Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability" (Generic Letter 82-33). The generic letter requires each licensee and applicant for an operating license to submit to the NRC a PGP, which includes:

(i) plant-specific technical guidelines
(ii) a writer's guide
(iii) a description of the program to be used for the validation of EOPs
(iv) a description of the training program for the use of upgraded EOPs.

On the basis of its review, the staff has determined that the procedure generation program for Vogtle is acceptable except for the items identified in the following sections. The applicant should address these items in a revision to the PGP, or should justify why such revisions are not necessary. The staff's review of the applicant's response to these items will be included in a supplement to this SER.

13.5.2.1.2 Evaluation and Findings

In a letter dated May 1, 1984, from D. O. Foster to the Director of the Office of Nuclear Reactor Regulation, Attention: Ms. Elinor G. Adensam, the applicant submitted its PGP. The PGP contained the following sections:

(1) Method of Developing Plant-Specific Emergency Operating Procedures From the Generic Guidelines

(2) Writer's Guide

(3) Validation and Verification Program

(4) Training Program Description.

The staff reviewed the Vogtle PGP to determine the adequacy of the program for preparing and implementing EOPs. Criteria for the review of a PGP are not currently in the Standard Review Plan. Therefore, this review was based on NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," the reference document for the EOP upgrade portion of Supplement 1 to NUREG-0737 (Generic Letter 82-33). Review criteria based on this guidance will be developed for the next SRP revision.

Method of Developing Plant-Specific Emergency Operating Procedures From the Generic Guidelines

This section of the PGP was reviewed to determine if it provided an adequate means of accomplishing the objectives of NUREG-0899. The applicant has elected to develop plant-specific EOPs using the generic Westinghouse Owners Group ERGs, Revision 1.

The applicant submitted Procedure 10013-C, "Writing Emergency Operating Procedures from the Westinghouse Emergency Response Guidelines." This procedure is the administrative control for the instruction and direction of EOP writers. It prescribes three phases: preparation, generation, and documentation. The EOPs are to follow the ERGs step by step, except when plant-specific design is not consistent with ERGs. Furthermore, steps should not be added or deleted except as required to conform with plant design.

The applicant identified ten differences between the Vogtle plant and the reference plant used for developing the Westinghouse Owners Group ERGs. The

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

applicant concludes that these differences do not preclude the use of the ERGs as a basis for writing the Vogtle EOPs. The staff concurs with this conclusion since the Westinghouse Owners Group wrote the ERGs to encompass the basic Westinghouse design with provisions for plant-unique differences in the guideline structure. Nevertheless, these differences will require some deviations between the EOPs and the ERGs. Such deviations from the approved ERGs having safety significance should be submitted for staff review. In addition, any analysis needed to ensure that these deviations are acceptable should be provided. Each procedural step that was changed because of equipment differences need not be described, but how the differences affect specific EOPs, any procedural strategy changes, and the analysis or evaluation to support acceptability should be provided.

The applicant did not provide the process for using the generic guidelines and background documentation to identify the characteristics of needed instrumentation and controls. For the information of this type that is not available from the ERG and background documentation, the applicant should submit a description of the process to be used to generate this information (e.g., from transient and accident analyses) to derive instrumentation and control characteristics. This process can be described in either the revised PGP or the Detailed Control Room Design Review (DCRDR) Program Plan with appropriate cross-referencing.

For potentially safety-significant, plant-specific deviations from the ERG instrumentation and control needs, a list of the deviations and their justification should also be provided.

With adequate resolution of the above items, the staff concludes that the applicant's Procedure 10013-C, "Writing Emergency Operating Procedures From the Westinghouse Emergency Response Guidelines," will provide adequate guidance to accomplish the objectives stated in NUREG-0899 for incorporating guidance provided in the ERGs into plant EOPs. The staff will confirm that the applicant adequately addresses the above items and will report on its review in a supplement to this SER.

Writer's Guide

This section of the PGP was reviewed to determine if it provided acceptable methods for accomplishing the objectives stated in NUREG-0899. The applicant selected a two-column EOP format. The left column was designated for user actions and expected responses, and the right column was designated for contingency actions to be taken if the performed actions were not performed or could not be performed. The staff's review of the Writer's Guide identified the following concerns:

(1) Since EOPs are often used under circumstances which could produce significant stress on operators, placekeeping aid(s) should be provided or the applicant should justify not providing such aids.

(2) In Section 5.0 of the Writer's Guide, several important directions are provided for writing instructional steps. However, directions are not provided for writing the following types of steps.

   (a) Verification Steps - Verification steps are used to determine whether the objective of a task or a sequence of actions has been achieved. There are three common methods for verification:

      o Checking that an action has resulted in a command signal to a piece of equipment. A more positive method should be used.

      o Checking that an action has resulted in a positive indication that the equipment has responded to a command.

      o Checking that an operator has correctly performed an action or has carried out a series of steps.

   These types of verification steps should be used where appropriate in the procedures to ensure that equipment responses and operator actions have occurred and are correct.

(b) Time-dependent Steps - Time-dependent steps are those that are required of the operator at some specified time interval, or some time after an action has taken place. A means should be provided to assist the operator in performing the step(s) within the required time frame.

(c) Diagnostic Steps - Diagnostic steps are those that lead the operator to the appropriate section of the EOPs. These steps should assist the operator in diagnosis, and should provide clear and unambiguous guidance leading to the diagnostic decision, as well as clear and unambiguous referencing to the appropriate section of the EOP. These steps may include the use of flow diagrams, graphs, or other operator aids.

Methods should be described in the guide for writing the above types of steps.

(3) Excellent guidance is provided for constructing a procedure and presenting instructional information and attachments. Guidance should also be included for assuring the quality of procedure copies. Good copy quality or "legibility" of EOPs is essential, so the operator using the EOP during an emergency has no question about its contents.

With adequate resolution of the above items, the staff concludes that the Vogtle Writer's Guide provides adequate guidance for translating the technical guidelines into EOPs which should be usable, accurate, complete, readable, convenient to use, and acceptable to control room operators. The staff will confirm that the applicant adequately addresses these items and will report on its review in a supplement to this SER.

Validation and Verification (V&V) Programs

The description of the applicant's validation/verification programs were reviewed to determine if they acceptably address the objectives stated in NUREG-0899. The V&V programs have two purposes: (1) to ensure that the procedures are written consistent with the Writer's Guide, and (2) to ensure that procedures are technically correct. The applicant's stated basis for all validation criteria is to ensure that the EOPs provide directions to place the plant in a safe, stable condition regardless of imposed structural and equipment failures. As stated by the applicant, "safe" means that the reactor is shut down, and has adequate cooling; "stable" means either that conditions are in equilibrium, or are changing in response to operator control. During its review of the V&V, the staff noted the following items:

(1) A control room walkthrough of each procedure in the program should be conducted. The programs state that a demonstration of correspondence between the control room and hardware must exist; however, no method to accomplish this objective is specified.

(2) The programs do not provide criteria for selecting scenarios to be used in simulator exercises for V&V.

(3) The programs do not indicate that the simulator exercises will include multiple failures (simultaneous and sequential).

(4) Page 9, Step 2.1.1 of "Verification of EOPs" states, "The Operation Procedures Coordinator shall designate personnel to verify EOPs." The verification team should include subject matter experts as well as members of the operating crews. Team composition should be identified by discipline.

With the addition of appropriate guidance on the preceding items, the V&V programs would satisfy the objectives of NUREG-0899 and, therefore, would be acceptable. The staff will confirm that the applicant addresses these items and will report on its review in a supplement to this SER.

Training Program Description

The applicant's description of the program for training operators on the EOPs was reviewed to determine if it acceptably addresses the objectives stated in NUREG-0899. The training program as described in the PGP consists of classroom and simulator instructions. Criteria for evaluating trainees are included in the program. The program does not require each trainee to exercise all EOPs on the simulator. This requirement should be clarified.

With adequate resolution of this item, the staff finds that the training plan will adequately address the objectives stated in NUREG-0899 and should result in appropriate operator training on the EOPs.

13.5.2.1.3 Conclusions

The PGP that the applicant submitted for Vogtle adequately addresses the guidance of NUREG-0899, except for the items noted in Section 13.5.2.1.2 of this SER. The PGP must be revised to address these items and the revision must then be submitted to the staff for review. The results of the staff's review will be reported in a supplement to this SER.

On the basis of its review, the staff concludes that, with the exceptions noted, the Vogtle PGP meets the requirements of Supplement 1 to NUREG-0737 and describes acceptable methods for accomplishing the objectives stated in NUREG-0899. The staff, therefore, has reasonable assurance that EOPs developed and implemented in accordance with the program described in the applicant's PGP should be adequate for control room personnel to effectively mitigate the consequences of a broad range of accidents and multiple equipment failures. Future changes to the PGP shall be documented in accordance with 10 CFR 50.59.

13.6 Physical Security

13.6.1 Introduction

The Georgia Power Company (applicant) has filed security program plans with the Nuclear Regulatory Commission for the Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle).

This evaluation summarizes how the applicant has provided for satisfying the requirements of 10 CFR Part 73. The evaluation is composed of a basic analysis that is available for public review, and a protected appendix (not available for public review).

13.6.2 Staff Evaluation

13.6.2.1 Physical Security Organization

To satisfy the requirements of 10 CFR 73.55(b) the applicant has provided a physical security organization that includes a Nuclear Security Shift Supervisor who is on site at all times and who has the authority to direct the physical protection activities. To implement the commitments made in the physical security plan, nuclear training and qualification plan, and the contingency plan, written security procedures have been developed specifying the duties of members of the security organization, and are available for inspection.

The training program and tasks and duties critical to performance of personnel employed in the security organization are defined in the "Vogtle Electric Generating Plant Units 1 and 2 Nuclear Training and Qualification Plan" which satisfies the requirements of 10 CFR Part 73, Appendix B, for training, equipping and requalifying members of the security organization. The physical security plan and the training program provide commitments that preclude the assignment of any individual to a security-related duty or task before that individual has been trained, equipped, and qualified to perform the assigned duty in accordance with the approved guard training and qualification plan.

13.6.2.2 Physical Barriers

In satisfying the requirements of 10 CFR 73.55(c), the applicant has provided a barrier to the protected area as is defined in 10 CFR 73.2(f)(1). A 20-ft-wide isolation zone along both sides of the barrier, except at the plant entry and security building, permits observation of activities at the perimeter. Isolation zones are kept clear of foliage, other materials, and structures, except for items listed in the appendix. Illumination of 0.2 footcandles (ftc) is maintained for the isolation zones, protected area barriers, and external portions of the protected area.

The staff has reviewed those locations and determined that the security measures in place are satisfactory insofar as the requirements of 10 CFR 73.55(c).

13.6.2.3 Identification of Vital Areas

The appendix contains a discussion of the applicant's vital area program and identifies those areas and items of equipment determined to be vital for protection purposes. Vital equipment is located within vital areas that exist within the protected area; passage through at least two barriers, as defined in 10 CFR 73.2(f)(1) and (2), is required to gain access to the vital equipment. Barriers to vital areas are separated from barriers to protected areas.

The control room and central alarm station are provided with bullet-resistant walls, doors, ceilings, and floors. On the basis of these findings and the analysis set forth in paragraph C of the appendix, the staff has concluded that the applicant's program for identifying and protecting vital equipment satisfies the intent of the regulation. However, this program is subject to onsite validation by the staff, and to subsequent changes, if changes prove necessary.

13.6.2.4 Access Requirements

In accordance with 10 CFR 73.55(d), all points of access to the protected area are controlled for personnel and vehicles passing through them. The individual responsible for controlling the final point of access into the protected area is stationed in a bullet-resistant structure. As part of the access control program, vehicles (except under emergency conditions), personnel, packages, and materials entering the protected area are searched for explosives, firearms, and incendiary devices by electronic search equipment and/or physical search.

Vehicles admitted to the protected area, except for vehicles designated by the applicant, are controlled by escorts when in operation. Applicant-designated vehicles are limited to onsite station functions and remain in the protected area except for operational maintenance, repair, security, and emergency purposes. Either personnel authorized to use the vehicles or escort personnel maintain positive control over the vehicles. A picture-badge/key-card system, utilizing encoded information, identifies individuals who are authorized unescorted access to protected and vital areas, and is used to control access to these areas. Individuals not authorized unescorted access are issued badges without pictures; these indicate that an escort is required. Only people who need to go into protected and vital areas to perform their duties are given access authorization.

Unoccupied vital areas are locked and alarmed. During periods of refueling or major maintenance, access to the reactor containment is controlled by a member of the security organization to ensure that only authorized individuals and materials are permitted to enter. In addition, all doors and personnel/equipment hatches into the reactor containment are locked and alarmed. Keys, locks, and related equipment are changed every year. In addition, when an individual's access authorization has been terminated because (1) he is not reliable or trustworthy, or ~~because~~ (2) he performs poorly at his job, the keys, locks, and related equipment to which that person had access are changed.

13.6.2.5 Detection Aids

In satisfying the requirements of 10 CFR 73.55(e), the applicant has installed intrusion detection systems at the barriers to the protected area, at entrances to vital areas, and at all emergency exits. Alarms from the intrusion detection system annunciate both inside the continuously manned central alarm station and at a secondary alarm station located inside the protected area. The interior of the central alarm station is not visible from outside the perimeter of the protected area. In addition, walls, floors, ceilings, doors, and windows of the central station are bullet-resistant. The alarm stations are located and designed so that a single act cannot stop a call for assistance or a response to an alarm. The central alarm station has no other functions or duties that would interfere with its alarm-response function. Transmission lines to the intrusion detection system are self-checking and tamper-indicating, as is alarm annunciation hardware associated with the transmission lines. When activated, alarm annunciators indicate the type and location of the alarm. When the alarm system is on standby power, an automatic indication is provided in the central alarm station.

13.6.2.6 Communications

As required in 10 CFR 73.55(f), the applicant has provided the capability of continuous communications between the central and secondary alarm station operators, guards, and armed response personnel by a conventional telephone system and a security radio system. In addition, direct communication with the local law enforcement authorities is maintained through the use of a conventional telephone system and two-way VHF radio links. All permanently installed communication links, except the conventional telephone system, are provided with an emergency power source that can not be interrupted.

13.6.2.7 Test and Maintenance Requirements

In satisfying the requirements of 10 CFR 73.55(g), the applicant has established a program for testing and maintaining all intrusion alarms, emergency alarms, communication equipment, physical barriers, and other devices and equipment related to maintaining security. Equipment and devices that do not meet the design performance criteria or have failed to operate will be compensated for by appropriate measures as defined in the "Vogtle Electric Generating Plant Units 1 and 2 Physical Security Plan and Contingency Plan" and in procedures at the site. The compensatory measures defined in these plans will ensure that the effectiveness of the security system is not reduced by failures or other contingencies affecting the operation of the security-related equipment or structures. Systems to detect intrusion are tested at least once every seven days; such systems are also tested at the beginning and end of any period that they are used.

Communication systems for onsite use are tested at the beginning of each security shift. Offsite communications are tested at least once each day.

The security program is audited once every 12 months by personnel independent of site security management and supervision. The audits, focusing on the effectiveness of the physical protection provided by the onsite security organization implementing the approved security program plans, include, but are not limited to: a review of the security procedures and practices; system local law enforcement

testing and maintenance programs; and assistance agreements with authorities. A report is prepared documenting audit findings and recommendations and is submitted to the plant management.

13.6.2.8 Response Requirements

In satisfying the requirements of 10 CFR 73.55(h), the applicant has provided that armed personnel are immediately available to respond at all times should the need arise. The appendix details considerations that apply insofar as considerations used in support of this number of personnel available for armed response. In addition, liaison has been established with local law enforcement authorities to provide additional response support if more people are ever needed to maintain security; the applicant has documented this agreement.

The applicant's safeguards contingency plan for dealing with thefts, threats, and radiological sabotage satisfies the requirements of 10 CFR Part 73, Appendix C. The plan identifies security events which could initiate a radiological sabotage event and identifies the applicant's preplanning, response resources, safeguards contingency participants, and coordination activities for each identified event. Through this plan, upon the detection of abnormal presence or activities within the protected or vital areas, response activities using the available resources would be initiated. The response activities and objectives include neutralizing the existing threat by requiring the response force members to place themselves between the adversary and the objective, instructions to use force commensurate with that used by the adversary, and authority to request sufficient assistance from the local law enforcement authorities to maintain control over the situation.

To assist in the assessment/response activities the security organization will use a closed-circuit television system; this system will provide the capability to observe the entire perimeter of the protected area, the isolation zones, and most of the protected area.

13.6.2.9 Employee Screening Program

In satisfying the requirements of 10 CFR 73.55(a) to protect against the design-basis threat of radiological sabotage performed by an "insider," including a person employed at the plant (10 CFR 73.1(a)(1)(ii)), the applicant has provided an employee screening program. Personnel who successfully pass the screening or its equivalent may be granted unescorted access to protected and vital areas at the Vogtle site. All other personnel requiring access to the site are accompanied by persons authorized and trained for escort duties and who have passed the employee screening process. The employee screening program is based upon accepted industry standards and includes a background investigation, a psychological evaluation, and a continuing observation program. In addition, the applicant may waive its own screening program, in cases where (1) an employee has passed a screening program of another nuclear utility or contractor, and (2) the applicant has determined the screening programs are similar (comparability review). The present plan provides for a "grandfather clause" which equates a certain period of trustworthy service with the utility or contractor as equivalent to passing the overall employee screening program. The staff has reviewed the applicant's screening program against the accepted industry standards (ANSI N18.17-1973) and finds the program acceptable.

14 INITIAL TEST PROGRAM

The initial test program at Vogtle encompasses the scope of events that begins when construction is finished and ends when power ascension testing is complete. The initial test program consists of the preoperational and the startup test programs. At the conclusion of these test programs a unit is ready for routine power operation.

The preoperational test program begins with system/component turnover from the construction organization to the operations organization, and ends with the beginning of fuel loading. These tests demonstrate, to the extent practicable, the capability of structures, systems, and components to meet performance requirements, and to satisfy design requirements. To the extent practicable, the objectives of the preoperational test program are:

(1) to document the performance and operability of equipment and systems,

(2) to provide baseline test and operating data on equipment and systems,

(3) to operate new equipment for a sufficient time to identify and correct design, manufacturing, and installation defects,

(4) to ensure integrated systems operation,

(5) to familiarize plant operating, technical, and maintenance personnel with facility operation, and

(6) to confirm the adequacy of normal and emergency operating procedures.

The startup test program begins with fuel loading, is followed by zero-power and low-power testing, and ends with the completion of power ascension testing. These tests confirm the design bases, and demonstrate to the extent practicable that the plant operates and responds to transients as designed. Startup testing is sequenced to ensure that the safety of the plant is not dependent upon the performance of untested structures, systems, or components. The objectives of the startup test program are:

(1) to accomplish a controlled, orderly, and safe initial core loading,

(2) to accomplish a controlled, orderly, and safe initial criticality,

(3) to conduct low-power testing sufficient to ensure that design parameters are satisfied, and that safety analysis assumptions are correct or conservative,

(4) to perform a controlled, orderly, and safe power ascension with requisite testing, terminating at plant-rated conditions, and

(5) to confirm codes and analytical models used in the reactor design.

The staff review of FSAR Chapter 14 concentrated on the administration of the test program and the completeness of the proposed preoperational and startup tests. The review consisted of the following parts. The Safety Evaluation Report-Construction Permit stage (SER-CP) was reexamined to determine the principal design criteria for the plant and to identify any specific concerns or unique design features that would warrant special test consideration. FSAR Chapters 1 through 12 were examined so the NRC staff could become familiar with the facility's design and nomenclature. FSAR Chapter 15 was examined to identify assumptions pertaining to performance characteristics that should be verified by testing and to identify all structures, systems, components, and design features that were assumed to function (either explicitly or implicitly) in the accident analyses. Licensee Event Report Summaries for operating reactors of similar design were examined to identify potentially serious events and chronic or generic problems that might warrant special test consideration. The Standard Technical Specifications were examined to identify all structures, systems, and components that would be relied upon for establishing conformance with safety limits or limiting conditions for operation. Post-TMI-related testing requirements were examined in conformance with: NUREG-0660, "NRC Action Plan Developed As a Result of the TMI-2 Accident"; NUREG-0694, "TMI-Related Requirements for

New Operating Licenses"; and NUREG-0737, "Clarification of TMI Action Plan Requirements." And finally, startup test reports for other similar type plants were examined to identify problem areas that should be emphasized in the initial test program.

In determining the acceptability of the applicant's test program, the criteria of SRP Section 14.2 were used. The staff review process includes verifying the following features of the initial test program:

(1) The applicant will develop test procedures using input from the nuclear steam supply system (NSSS) vendor, the architect-engineer, the applicant's engineering staff, and equipment suppliers and contractors. Operating experiences at similar plants will be considered in the development of the test procedures for Vogtle.

(2) The applicant will conduct tests using approved test procedures. Administrative controls will cover (a) the completion of test prerequisites, (b) the completion of necessary data sheets and other documentation, and (c) the review and approval of modifications to test procedures. Administrative procedures will also cover implementation of modifications or repair requirements identified as being required by the tests and by any necessary retesting.

(3) The applicant will have review groups (including the NSSS vendor and architect-engineer, as appropriate) review the results of each test for technical adequacy and completeness. Preoperational test results will be reviewed before fuel loading and the startup test results from each test condition or power level will be reviewed before proceeding to the next test condition or power level.

(4) The applicant will use its normal operating procedures and emergency operating procedures to the extent practicable in performing the initial test program, thereby verifying the correctness of the procedures.

(5) The applicant's schedule for conducting the initial test program allows adequate time to conduct all preoperational and startup tests. Preoperational test procedures will be available for NRC regional personnel to review at least 60 days before scheduled implementation. Startup test procedures will be available for review not less than 60 days before the scheduled fuel loading date.

(6) An abstract of each test is presented in FSAR Chapter 14. The staff verified that there are test abstracts for those structures, systems, components, and design features that: (a) will be used for shutdown and cooldown of the reactor under normal, transient, and accident conditions and for maintaining the reactor in a safe shutdown condition for an extended period of time; (b) will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility's Technical Specifications; (c) are classified as engineered safety features or will be relied on to support or ensure the operations of engineered safety features within design limits; (d) are assumed to function or for which credit is taken in the accident analysis of the facility, as described in the FSAR; or (e) will be used to process, store, control, or limit the release of radioactive materials.

(7) The test objectives, prerequisites, test methods, and acceptance criteria for each test abstract are in sufficient detail to establish that the functional adequacy of the structures, systems, components, and design features will be demonstrated.

(8) Exceptions to RG 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," Revision 2, are identified and adequately justified.

The staff was unable to verify all of the above; therefore, the staff sought additional information from the applicant (letter to applicant, April 30, 1984). The applicant responded to this request in FSAR Amendments 5 through 9. Since some of the responses were incomplete, open items remain (see Table 14.1).

On the basis of its review, the staff has concluded that once the open items identified in Table 14.1 are resolved, the initial test program described in the FSAR will satisfy the acceptance criteria of SRP Section 14.2 and the regulatory requirements of:

(1) 10 CFR 50.34(b)(6)(iii) that requires inclusion of plans for preoperational testing and initial operations in the FSAR,

(2) 10 CFR 50, Appendix B, Section XI, that requires a test program to ensure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents, and

(3) NUREG-0737, Item I.G.1, which requires additional testing and training during the initial test program.

This review and evaluation was performed with the assistance of Battelle Pacific Northwest Laboratories personnel.

Table 14.1 Open items

NRC Question No.	Status
640.03	The Service Air System Preoperational Test (FSAR Subsection 14.2.8.1.56), the Instrument Air System Preoperational Test (FSAR Subsection 14.2.8.1.57), or other test abstracts should be modified to demonstrate that the valves identified in FSAR Table 9.3.1-2 (Safety-Related Pneumatically Operated Valves) fail to the safe position on loss of instrument air in accordance with Position C.8 of RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems."
640.04	The response to Q640.04, Part B, should reference the Pressurizer Heater and Spray Capability and Continuous Spray Flow Verification Test (FSAR Subsection 14.2.8.2.2) since part of the tests referenced in the response are accomplished in this test abstract.
640.07	The Power Coefficient Determination Test (FSAR Subsection 14.2.8.2.26) should be modified to provide appropriate test prerequisites.
640.08	A. Appropriate acceptance criteria should be provided for the following: 1. Preoperational test abstract numbers (FSAR Subsection 14.2.8.1): 2, 7, 8, 11, 14-26, 28, 31, 32, 34, 41, 47-50, 53, 59, 63, 64, 71, 74, 76, 80, 84-86, 89-91, 93, 95-100, and 102-110 (2, 47, and 80 are to be provided later). 2. Startup test abstract numbers (FSAR Subsection 14.2.8.2): 2-5, 7, 9, 10, 14, 15, 24, 25, 27, 35, 36, 43-48, 52, and 53. C. Preoperational test modifications to FSAR Paragraphs 14.2.8.1: 1, 2, 27, 33, 35-40, 42, 44-47, 51, 52, 54, 58, 78, 79, 82, 83, 87, 92, and 94 to be made later.
640.09	Response to be provided later.
640.11	Response to be provided later.
640.12	The Ventilation Capability Test (FSAR Subsection 14.2.8.2.58), or other test abstracts, should be modified in accordance with the response to this item.
640.14	Response to be provided later.

Table 14.1 (Continued)

NRC Question No.	Status
640.20	B. The 125-V dc preoperational test abstracts (FSAR Subsections 14.2.8.1.69 and 14.2.8.1.74) should be modified to demonstrate that testing of dc loads necessary for safe shutdown is conducted at minimum dc system voltage or that the voltage drop at load to these components is measured to verify that the dc loads are supplied with appropriate voltage under minimum battery voltage conditions.
640.22	Response to be provided later.
640.30	1. Operability of main steam bypass valves will be described in a future amendment.
	2. The Reactor Coolant System (RCS) Preoperational Test (FSAR Subsection 14.2.8.1.7) should provide a test method and acceptance criteria regarding testing of pressurizer safety valves.
	3. Testing should verify that the capacity of pressurizer and steam generator power-operated relief valves are in accordance with the accident analysis assumptions for maximum valve capacity (FSAR Subsections 15.6.1 and 15.1.4).
640.32	FSAR Subsection 14.2.8 (Individual Test Descriptions) should be expanded to address the following systems listed in RG 1.68, Revision 2, Appendix A:
	1. Main stop, control, intercept, and intercept stop valves will be addressed in a future amendment ((1.e.6).
	2. Feedwater heater temperature, level, and bypass control systems will be addressed in a future amendment (1.j.17).
	3. Isolation features for steam generator blowdown will be addressed in a future amendment (1.1.4).
	4. Isolation features for several ventilation systems will be addressed in a future amendment (1.1.6).
	5. The Fuel Building Hoists and Elevator Preoperational Test (FSAR Subsection 14.2.8.1.60) and the Fuel Transfer System Preoperational Test (FSAR Subsection 14.2.8.1.61) should be modified to include verification of the performance of the dynamic (100%) and static (125%) load tests of cranes and hoists associated with fuel handling and storage in accordance with the response to this item (1.m.4).
	6. Turbine plant water sampling systems will be addressed in a future amendment (1.n.5).

Table 14.1 (Continued)

NRC Question No.	Status
	7. The Fuel Handling and Vessel Servicing Preoperational Test (FSAR Subsection 14.2.8.1.59) should be modified to include verification of the performance of the dynamic (100%) load tests of the polar crane.
	8. Containment ventilation operability with the reactor coolant system at rated temperature will be addressed in a future amendment (4.j).
	9. Operability of the gross failed fuel detector should be demonstrated at 25% and 100% power (5.q).
	10. Operability of the gaseous and liquid radwaste systems should be demonstrated during the power ascension test phase (5.c.c).
	11. Ventilation and air conditioning systems will be addressed in a future amendment (5.f.f). The response to Part 5.f.f should reference the Ventilation Capability Test (FSAR Subsection 14.2.8.2.58), since this abstract addresses the concerns of this item.
	12. FSAR Subsection 1.9.68.2 should be modified to provide justification for exception to the loss of or bypass of feedwater heaters test (5.k.k).

Other Items Status	
	1. The RCS Leak Rate Preoperational Test (FSAR Subsection 14.2.8.1.10) was modified to delete the test objectives relating to testing of the containment cooler condensate measuring system. The test objectives are now part of the containment's, Auxiliary, Control, and Fuel Handling Building Drains System Preoperational Test (FSAR Subsection 14.2.8.1.54). The test objectives, test method, and acceptance criteria for the containment cooler condensate system should be included as a whole in either test abstract, rather than arbitrarily separated.
	2. FSAR Tables 14.2.1-1 (Preoperational Test Procedures) and 14.2.1-2 (Startup Test Procedures) should be updated to include FSAR Subsection 14.2.8.1.107-110 and 14.2.8.2.57-58, respectively.

15 ACCIDENT ANALYSES

To evaluate the effectiveness of the engineered safety features (ESFs) proposed for the Vogtle plant and to ensure that the distances to the exclusion area boundary (EAB) and low-population zone (LPZ) are adequate, the applicant and the staff have analyzed the radiological consequences of a number of accidents, selected to challenge specific design features of the plant. The severity of the hypothetical accidents analyzed ranges from minor leakage of coolant to a release to the containment of a fraction of the core's fission products associated with substantial core melting, as required by 10 CFR 100.11. The calculated doses for these accidents are given in Table 15.1.

The accident analyses for Vogtle Units 1 and 2 (Vogtle) have been reviewed in accordance with Section 15 of the Standard Review Plan (SRP) (NUREG-0800). Conformance with the acceptance criteria, except as noted for each of the sections, formed the basis for concluding that the design of the facility for each of the areas reviewed is acceptable.

In accordance with SRP Section 15.1.1, paragraph I, the applicant evaluated the ability of Vogtle to withstand anticipated operational occurrences and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. The results of these analyses are used to show conformance with GDC 10, 15, 27, 28, 31, and 35.

For each event analyzed, the worst operating conditions and the most limiting single failure were assumed, and credit was taken for minimum engineered safeguards response, such as maximum delay times and minimum pump performance. The staff has asked the applicant to show the effect of a loss of offsite power on all anticipated operational occurrences and postulated accidents. The applicant has stated that all the the design-basis accidents and anticipated operational occurrences have been analyzed with and without offsite power available.

Parameters specific to individual events were conservatively selected. Two types of events were analyzed:

(1) those incidents that might be expected to occur during the lifetime of the reactor

(2) those incidents not expected to occur that have the potential to result in significant radioactive material release (accidents)

The nuclear feedback coefficients were conservatively chosen to produce the most adverse core response. The reactivity insertion curve, used to represent the control rod insertion, accounts for a stuck rod; it complies with GDC 26.

For transients and accidents, the applicant used a method that conservatively bounds the consequences of the event by accounting for fabrication and operating uncertainties directly in the calculations. Departures from nucleate boiling ratios (DNBRs) were calculated using the W-3 correlation with a modified spacer factor R, with a minimum DNBR of 1.3 used as the threshold for fuel failure.

The applicant accounts for variations in the initial conditions by making the following assumptions as appropriate for the event being considered:

- core power, 3425 MWt, +2%
- average reactor vessel temperature (T_{avg}), 588.5 ± 4.0°F
- pressure (at pressurizer), 2250 ± 30 psi

The staff asked the applicant to discuss the degree of conservatism in the initial pressurizer volume and to justify why this parameter is to be excluded from the plant Technical Specifications. In response, the applicant stated that the pressurizer volume assumed in the accident analyses was a nominal value that included allowances for uncertainties such as measurement error and control dead band and that the level is maintained by a control system. In addition, the applicant stated that the values for the process variables in the Technical Specifications, other than those specifically noted, may be treated as indicated values without consideration for instrument uncertainties. The rationale presented is that the typical measurement uncertainties are negligible in comparison to the conservatisms in the plant design and safety analyses. Although the staff concurs that safety margin may exist in FSAR safety analyses to cover these instrument uncertainties, such safety margin has not been explicitly quantified.

The existence and magnitude of the safety margins will be confirmed as part of efforts to address the recommendations of NUREG-1024. Specifically, the issue of indicated vs. actual values will be incorporated into the NUREG-1024 efforts. Moreover, the applicant has indicated it will pursue this with the staff, along with issues identified in NUREG-1024, on a generic basis. Given the commitment to achieve confirmation of this issue, the staff finds the applicant's response acceptable.

The staff concludes that the assumptions for initial conditions are acceptable because they are conservatively applied to produce the most adverse effects. These assumed values will form the basis for the technical specification limits. For transients and accidents used to verify the engineered safety feature (ESF) design, the applicant used the safeguards power design value of 3579 MWt.

In analyzing the transients and accidents, it was assumed that the pressurizer heaters were not energized. The applicant was asked to demonstrate the conservatism of this assumption or to quantify the effects to show that they are negligible. In response, the applicant has demonstrated that for cooldown transients or departure from nucleate boiling (DNB) limited transients, the transients occur so quickly that the pressurization caused by the heaters is negligible. For steam generator tube ruptures and small-break LOCAs, the limiting assumption of loss of offsite power precludes the energization of the heaters. For the large LOCAs, the energy release and heat stored in the piping significantly dominate the heat input from the pressurizer heaters. The break flow controls the system pressure, and again, the heaters have a negligible effect. For long-term heatup transients, the applicant cited WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture," as evidence to show that the heaters had a negligible effect.

At this time staff review of WCAP-9230 indicates reasonable assurance that the conclusions of the Westinghouse submittal will not be appreciably changed by completion of the review. If the final results of the review indicate that revisions to the applicant's analyses are necessary, the applicant will be required to implement the results of such changes. The staff does not consider this an open item.

The applicant has also analyzed several events expected to occur one or more times in the life of the plant. A number of transients can be expected to occur with moderate frequency as a result of equipment malfunctions or operator errors in the course of the various operating modes during the plant lifetime.

Specific events were reviewed to ensure conformance with the acceptance criteria provided in the SRP.

The acceptance criteria for transients of moderate frequency in the SRP include the following:

(1) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (derived from Section III of the ASME Boiler and Pressure Vessel Code).

(2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNB will remain above the 95/95 DNBR limit for PWRs. (The 95/95 criterion discussed in Section 4.4 of this SER provides a 95% probability, at a 95% confidence level, that no fuel rod in the core experiences a DNB.)

(3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

(4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission-product barrier, other than fuel

element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

The ~~staff~~ staff noted that the loss of nonemergency ac power to the station auxiliaries, a Condition II event (any transient conditions arising from faults of moderate frequency), is the initiator for the complete loss of forced reactor coolant flow, which is classified as a Condition III event (any transient conditions arising from infrequent faults). The applicant, in response to a staff request to clarify this issue, stated that although the complete loss of forced reactor coolant flow is classified as a Condition III event in the FSAR, it was analyzed and found to meet the Condition II acceptance criteria. This is acceptable. *

Conformance with the SRP acceptance criteria for anticipated operational occurrences constitutes compliance with GDC 10, 15, and 26 of Appendix A to 10 CFR 50. See Section 10.4.9 of this SER for a discussion of auxiliary feedwater system conformance to TMI-2 Task Action Plan Item II.E.1.1 and Section 7.3.3.1 for a discussion of compliance with TMI-2 Task Action Plan Item II.E.1.2. These items address the adequacy of the AFWS design to remove decay heat.

In response to TMI-2 Task Action Plan Item II.K.2.17 (Potential for Voiding in the Reactor Coolant System During Transients), the applicant has stated that Westinghouse has performed a study that addresses the potential for void formation in Westinghouse-designed nuclear steam supply systems (NSSSs) during natural circulation cooldown/depressurization transients. The Westinghouse Owners Group submitted this study to the NRC. The results of this study have been accepted by the staff for all Westinghouse 2, 3, and 4 loop plants (letter from T.M. Novak, NRC to O.W. Dixon, South Carolina Electric and Gas Company dated February 15, 1984 on V.C. Summer, Unit 1).

Sequential auxiliary feedwater flow criteria are only of concern to once-through steam generator designs. Since Westinghouse has inverted U-tube steam generator designs, the analysis requested by TMI-2 Task Action Plan Item II.K.2.19 (Sequential Auxiliary Flow Analysis) is not needed for Vogtle.

The transients analyzed are protected by the following reactor trips:

(1) power range high neutron flux (high and low settings)
(2) high pressurizer pressure
(3) low pressurizer pressure
(4) overpower ΔT
(5) overtemperature ΔT
(6) low reactor coolant flow
(7) reactor coolant pump undervoltage
(8) low-low steam generator water level
(9) high steam generator water level

The reactor may also trip on other variables for which credit has not been taken in the accident analyses. This includes source and intermediate range neutron flux, pressurizer water level, turbine trip, safety injection, and reactor coolant pump underfrequency. The operators will also have the capability to trip the reactor manually.

Time delays to trip, calculated for each trip signal; emergency core cooling system (ECCS) actuation times, including the times for the ECCS pumps to reach rated flow; and diesel generator startup times are included in the analyses. See Section 4.6 of this SER for a discussion of the staff review of reactivity control system functional design.

All of the events that are expected to occur with moderate frequency can be grouped according to the following plant process disturbances: changes in heat removal by the secondary system, changes in reactor coolant flow rate, changes in reactivity and power distribution, and changes in reactor coolant inventory. Design-basis accidents have been evaluated separately and are discussed at the end of this section and in Section 6.3 of the SER.

The fuel design is such that the fuel rod internal pressure may exceed primary system pressure. The applicant was asked to demonstrate that this was considered because this higher internal pressure may cause additional fuel rod ballooning and failures. The applicant cited WCAP-8963-P-A, "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis." By letter of May 19, 1978, to T. M. Anderson, Westinghouse, from J. F. Stolz, NRC, this report was found acceptable.

The applicant was asked to discuss the loss of instrument air as a plant transient. In response, it was demonstrated that the cause for a complete loss of instrument air would be the loss of all nonemergency ac power to the plant auxiliaries which is an analyzed event (FSAR Section 15.2.6).

Limited operator action may be required following some transients. Some of these actions occur after the plant conditions have been stabilized and an orderly shutdown is undertaken. Operator actions here would be similar to those for normal shutdown. Other operator actions are required for switchover from injection to recirculation. These operator actions are discussed in Section 6.3 of this SER. Other operation actions would include identifying and isolating a faulted steam generator from auxiliary feedwater flow in the event of a steamline or feedwater line break. For these two events, the operator will also need to manually control the repressurization of the reactor coolant system (RCS). This is performed by modulating the flow from the charging and safety injection (SI) pumps. Relying on safety-related indications and controls, the operator has enough time to accomplish these actions.

15.1 Increase in Heat Removal by the Secondary System

The applicant's analysis of events that produced increased heat removal by the secondary system is addressed in the following paragraphs.

15.1.1 Decrease in Feedwater Temperature

The cause of this transient was assumed to be the isolation of one string of low-pressure feedwater heaters. Reactor trip on either neutron overpower,

overpower ΔT, or overtemperature ΔT prevents any power excursion which could lead to a DNB of less than 1.30, according to the applicant. The consequences of a decrease in feedwater temperature transient are bounded by those in FSAR Sections 15.1.2 and 15.1.3. This is considered a Condition II event.

15.1.2 Increase in Feedwater Flow

Increases in feedwater flow can result from the full opening of a feedwater control valve when a system malfunctions or an operator makes a mistake. This will decrease the temperature of the reactor coolant water. Because of the negative moderator temperature coefficient, this will insert positive reactivity and increase core power.

In Section 15.1.2.1 of the FSAR, the applicant states that for these events the high neutron flux trip, overtemperature ΔT trip, and overpower ΔT trip prevent any power increase which could lead to a DNBR less than the limit value of 1.30. The analytical results presented for these events are those where a steam generator (SG) high-high level trip closes all feedwater control and isolation valves, trips the main feedwater pumps, and trips the turbine. The applicant states that the steam generator high-high level trip prevents continuous addition of feedwater.

This transient was analyzed by LOFTRAN which is a digital code that simulates a multiloop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, SG, and SG safety valves. Two sets of initial conditions were analyzed. One set assumed the transient initiating with the reactor just critical at zero-load conditions and that the malfunction resulted in a step increase in feedwater flow from 0 to 225% of the nominal full-load value for one SG. The other set of initial conditions assumed full-load conditions and a step increase to 157% of the nominal feedwater flow to one SG.

The analysis shows that at no-load conditions, the maximum reactivity insertion rate from an increase in feedwater flow is less than the maximum value calculated for an inadvertent control rod withdrawal, which is evaluated in

Section 15.4 of this SER. The full-power case results in the largest power increase. However, for this case, reactor trip is initiated by SG level high-high and the applicant states that the DNBR remains above 1.30.

15.1.3 Increase in Steam Flow

Increases in steam flow in excess of the capability of the reactor control system can be caused by an administrative violation such as excessive loading or an equipment malfunction in the steam dump control or turbine speed control. Four sets of initial conditions were analyzed. These were: reactor control in manual or automatic with either minimum or maximum moderator reactivity feedback. LOFTRAN was used to analyze this transient. Full-power operation with a 10% step increase in steam demand is assumed.

Protection against the transient could be afforded by either the power range high neutron flux, overpower ΔT or overtemperature ΔT reactor trip. However, for the cases involving automatic reactor control, no credit was taken for the ΔT trips. In fact, the analyses performed by the applicant show that the reactor would not trip but would reach a stabilized condition at the higher power level. This is qualified for the automatic control cases since the uncertainties in the setpoints may result in reactor trip. In any case, all analyses show the reactor achieving a stabilized condition with the DNBR remaining above 1.30 at all times.

15.1.4 Inadvertent Opening of a Steam Generator Relief Valve or Safety Valve

In FSAR Section 15.1.4.1 the applicant states that the most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The suddenly increased steam demand causes a reactor power increase which results in a reactor trip from high neutron flux, overtemperature, or overpower signals or because the trip occurs in conjunction with safety injection. The continued steam flow through the open valve will cause additional cooldown which will, because of the negative moderator temperature coefficient, result in positive reactivity. The safety injection system (SIS) will inject borated water from the boron injection tank into the primary coolant system on either two-out-of-four pressurizer low pressure signals, or two-out-of-three low steamline pressure signals in any one loop. This ensures the reactor will be shut down during any subsequent cooldown. The normal steam generator feedwater would be isolated automatically upon SIS initiation, and then the plant would be gradually cooled down with only safety-grade equipment. DNB does not occur during this transient.

This transient was analyzed using LOFTRAN. The initial conditions included having the reactor at just critical (no load) with an end-of-life shutdown margin. The most reactive rod cluster control assembly was assumed to be stuck in the fully withdrawn position and the single failure assumed was one SI train so as to minimize the boric acid injection. The transient was assumed initiated by the valve with the highest rated steam flow capacity that relieves to outside of the secondary system.

The no-load condition, in conjunction with offsite power available, all reactor coolant pumps running, and maximum cold auxiliary feedwater flow, have been assumed in order to maximize the cooldown transient that follows the valve opening.

The applicant has stated, in response to a staff concern, that although the pressurizer empties during this transient, void formation in the RCS will not occur because the coolant enthalpy will remain well below the saturation enthalpy corresponding to the prevailing RCS pressure.

The applicant's analyses show that for transient events leading to an increase in heat removal by the secondary system (with or without single failure), the minimum DNBR remains above the design-basis limit of 1.3. Thus, no fuel failure is predicted to occur, core geometry and control rod insertability are maintained with no loss of core cooling capability, and the maximum RCS pressure remains below 110% of design pressure. The staff finds the results of these analyses in conformance with the acceptance criteria of SRP Section 15.1.1 through 15.1.4, and, therefore, acceptable.

15.1.5 System Piping Failures Inside and Outside Containment

15.1.5.1 Steamline Rupture Accident

The applicant has submitted analyses of postulated steamline breaks that show no fuel failures attributed to the accident. These results are similar to those obtained for previously reviewed Westinghouse four-loop plants.

A postulated double-ended rupture at hot shutdown power was analyzed as the worst case. The applicant referenced WCAP-9226 as justification for this selection. The staff is currently reviewing WCAP-9226. The applicant has stated that the steam generators have integral flow restrictors with a 1.4-ft^2 throat area; since any rupture with a break area greater than 1.4 ft^2, regardless of location, will have the same effect on the system as a 1.4-ft^2 break, this was assumed in the analysis. The doubled-ended rupture would cause the reactor to increase in power because of the decrease in reactor coolant temperature.

The reactor would be tripped by either reactor overpower ΔT, by high neutron flux or by the actuation of the SIS. The SIS will be actuated by any of the following: two-out-of-four low pressurizer pressure signals; two-out-of-three high containment pressure signals; or two-out-of-three low steamline pressure signals in any one loop. The transient is terminated using only safety-grade equipment. The injection of highly borated water ensures the reactor is maintained in a shutdown condition.

The applicant analyzed this transient with and without offsite power available. As a result, both full RCS flow and loss of flow were considered with the full-flow case determined to be more limiting. The initial conditions included end-of-life shutdown margin, no load, and the most reactive rod cluster control assembly stuck out. A single failure was chosen so as to minimize boron injection.

In addition to assuming offsite power available, and thus full RCS coolant flow, other assumptions were made so as to maximize the cooldown transient that follows the postulated steamline break. These assumptions included a maximum auxiliary feedwater (AFW) flow, minimum enthalpy, and immediate delivery of auxiliary feedwater to the steam generators. Operator action at 30 min was assumed to isolate the AFW from the faulted steam generator.

In analyzing this event, the applicant utilized the W-3 correlation beyond the range for which it has been accepted. This is an open item as discussed in Section 4.4 of this SER.

Compliance with TMI-2 Task Action Plan Item II.K.3.5 for a non-LOCA event is addressed in a Westinghouse Owners Group (WOG) report (Letter OG-110, December 1983). This report is in response to Generic Letter 83-10C which delineates the staff resolution of Item II.K.3.5. The WOG response is under staff review.

Staff review at this time indicates reasonable assurance that the conclusions contained in the WOG submittal will not be appreciably changed by completion of the review. If final results of the review indicate that revisions to the WOG response are necessary, the applicant will be required to implement the results of such changes. The staff does not consider this an open item.

Compliance to TMI-2 Task Action Plan Item II.K.3.25 is addressed in FSAR Section 9.2.8. The intent of the item is to prevent excessive loss of RCS inventory as a result of reactor coolant pump (RCP) seal failure because of a loss of cooling water. The applicant has stated that in the event of a loss of offsite power, the RCP motor is de-energized and the auxiliary component cooling water (ACCW) pump, which provides the cooling water to the seals, is loaded onto the diesel generator's pump, unless there is a coincident ESF signal. Thus, unless there is an ESF signal, seal cooling water would be restored within seconds. The applicant states that the pumps will incur no damage if ACCW flow is interrupted for 10 minutes. In the event of an ESF signal, the operator will thus have 10 minutes in which to manually load the ACCW onto the diesel generators. The RCPs can also be tripped manually, if so desired. The operator has safety-related indication of ACCW flow, pressure, surge tank level, and valve position that can be relied upon in reaching a decision. The staff finds the applicant to be in compliance with Item II.K.3.25.

The staff concludes that the consequences of postulated steamline breaks conform with the relevant criteria in GDC 27, 28, 31, and 35 regarding control rod insertability and core coolability and TMI-2 Task Action Plan items. This conclusion is based on the following:

(1) The applicant has conformed with the criteria of GDC 27 and 28 by demonstrating that fuel damage, if any, is such that control rod insertability will be maintained, and there will be no loss of core cooling capability. The minimum DNBR experienced by any fuel rod was >1.30, resulting in none of the fuel elements being predicted to experience cladding perforation.

(2) The applicant has conformed with the criteria of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(3) The applicant has conformed with the criteria of GDC 35 with respect to demonstrating the adequacy of the emergency cooling systems to provide abundant core cooling and reactivity control (via boron injection). (Refer to Section 6.3.)

(4) A mathematical model, which accounts for incomplete coolant mixing in the reactor vessel, has been reviewed and found acceptable by the staff. This model was used to analyze the effects of steamline breaks inside and outside of containment, during various modes of operation, with and without offsite power.

(5) The parameters used as input to this model were reviewed and found to be suitably conservative.

15.1.5.2 Radiological Consequences of a Main Steamline Break Outside Containment

Both the staff and the applicant have evaluated the radiological consequences of a postulated steamline-break accident occurring outside containment and upstream of the main steam 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, the staff has conservatively assumed that the entire release throughout the course of the accident occurs at ground level, rather than at an elevated release point. During the course of the accident, the shell side of the affected steam generator was assumed to stay dry, since auxiliary feedwater flow to the affected steam generator would be blocked off under the conditions of this accident. Because of the dryout condition in the affected steam generator, all iodine transported to the secondary side by leakage (1 gpm) was assumed available for release to the atmosphere with no reduction from holdup or attenuation.

The staff investigated three scenarios. For case 1, the most reactive control rod was assumed to be stuck in the fully withdrawn position. The applicant has indicated, and the staff agrees, that no departure from nucleate boiling is expected to occur and, therefore, no fuel-cladding failure was assumed in the calculation. With no additional fuel failures occurring, case 1 becomes identical to case 2, and no radiological consequences are presented for case 1.

For case 2, the staff assumed that a temporary increase in the coolant iodine concentration (iodine spike) occurred as a result of the power pressure transient caused by the accident. Before the accident, the plant was assumed to be operating at the Westinghouse Standard Technical Specification equilibrium primary coolant limit of 1 μCi/gm dose equivalent iodine-131 (DEI-131). The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate results in an increasing iodine concentration in the primary coolant during the course of the accident. The radiological consequences for this case have been calculated using assumptions given in Table 15.3 and the consequence values are given in Table 15.1 of this section.

For case 3, the staff assumed that previous reactor operation had resulted in a primary coolant concentration equal to the maximum transient full-power Westinghouse Standard Technical Specification limit (60 μCi/gm DEI-131). As in case 2, the radiological consequences were calculated using assumptions found in Table 15.3 and the consequence values are given in Table 15.1.

On the basis of its findings, the staff concludes that there is reasonable assurance that the calculated radiological consequences of a postulated main steamline failure outside the containment of the Vogtle plant can be controlled by Technical Specifications to remain within (1) the dose guidelines of 10 CFR 100.11 for the case in which the failure occurs with a primary coolant iodine concentration corresponding to a preaccident iodine spike; and (2) 10% of these guidelines for the case in the failure occurs with a primary coolant activity corresponding to the equilibrium concentration of the Westinghouse Standard Technical Specifications.

These conclusions are based on (1) the staff's review of the plant design and the applicant's analysis of this postulated accident, (2) the staff's independent calculation using appropriately conservative assumptions (including atmospheric diffusion factors as discussed in Section 2 of this SER), and (3) the specific Technical Specifications for the iodine concentration in the reactor coolant (which consists of a maximum allowable limit and a limit for the equilibrium concentration for continued plant operation) and the limit on primary-to-secondary leakage in the steam generators. The staff will review the Vogtle plant's Technical Specifications to ensure that these operating restraints are incorporated.

15.2 Decrease in Heat Removal by the Secondary System

The applicant's analyses of events that result in a decrease in heat removal by the secondary system are presented below.

15.2.1 Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow

In Section 15.2.1 of the FSAR the applicant states that there are no steam pressure regulators whose failure or malfunction could cause a steam-flow transient. The staff concurs with this statement

15.2.2 Loss of External Load

In a loss of external load event, an electrical disturbance can cause a loss of a significant portion of the generator load. This loss of load situation differs from the loss of ac power condition considered in FSAR Section 15.2.6 in that offsite power remains available to operate such station auxiliaries as the reactor coolant pumps. The onsite diesel generators are, therefore, not required for this transient. The applicant states that in the event that a safety limit is approached, the reactor will be tripped on high pressurizer pressure, high pressurizer level, or overtemperature ΔT.

In addition, in Section 15.2.2.1 of the FSAR the applicant states that the results of the turbine trip event analysis are more severe than those expected for the loss of external load. The reason given is that a turbine trip actuates the turbine stop valves whereas a loss of external load actuates only the turbine control valves. Since the stop valve can more suddenly cut off the steam flow to the turbine, ~~this~~ the turbine trip is a more severe "decreased heat removal" transient than the loss of external load as discussed in the following section.

15.2.3 Turbine Trip

The applicant analyzed the turbine trip event for a complete loss of steam load from full power without a direct reactor trip (on turbine trip) and with only the pressurizer and steam generator safety valves assumed for pressure relief. The applicant states that RCS temperatures and pressures do not increase significantly if the turbine bypass system and pressurizer pressure control systems function properly. However, loss of the condenser would result in loss of main feedwater and could result in lifting the SG safety valves. Reactor protection would be provided by the high pressurizer pressure, high pressurizer level, low-low SG water level, and overtemperature ΔT trips. The applicant took credit for auxiliary feedwater only for long-term recovery so as to maximize the primary site pressure transient. The applicant did not take credit for the turbine bypass system or the SG power-operated relief valves. The transient was analyzed with both minimum and maximum reactivity feedback. These two cases were analyzed with and without credit taken for pressurizer spray and pressurizer PORVs. The FSAR results show that the RCS peak pressure for all of the cases was below 2550 psia, which is well below the SRP limit of 110% of design pressure. For these assumptions, the minimum DNBR is 1.50 which is above the minimum limiting value of 1.30. The staff finds this acceptable.

The consequences of a turbine trip without offsite power available are discussed in Section 15.2.6 of this SER.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

Inadvertent closure of the main steam isolation valves would result in a turbine trip without the turbine bypass system. This event is identical to the turbine trip which is described in Section 15.2.3 of this SER.

15.2.5 Loss of Condenser Vacuum and Other Events Resulting in a Turbine Trip

Loss of the condenser vacuum is an event that can cause a turbine trip. In addition, loss of the condenser will preclude the use of the steam dump. The turbine trip analysis does not take credit for the steam dump; hence, this event is identical to the turbine trip transient which is described in Section 15.2.3 of this SER. The applicant has stated that other causes of a turbine trip are also covered in FSAR Section 15.2.3 and are, therefore, evaluated as turbine trip events.

15.2.6 Loss of Nonemergency AC Power to the Plant Auxiliaries

The loss of the nonemergency ac power can be caused by a complete loss of the offsite grid followed by a turbine trip or a loss of the onsite ac distribution grid.

A loss-of-nonemergency-ac-power event is more limiting than the turbine-trip-initiated decrease in secondary heat removal without loss of ac power because the reactor coolant pumps are lost and the subsequent flow coastdown further reduces the rate of heat removal from the core. In this transient, the loss of offsite power is closely followed by turbine trip and reactor trip. The reactor trip is assumed to come from either the turbine trip, loss of power to the control rod drive mechanisms, or from one of the trip setpoints in the primary or secondary systems that would be reached as a result of the flow coastdown and decrease in secondary heat removal. The auxiliary feedwater system is automatically started on low-low level in any steam generator. With one electric-motor-driven pump assumed to the ~~Only~~ other ~~one~~ electric-motor-driven pump is assumed to be feeding only two ~~all three~~ steam generators.

The overall SG heat transfer coefficient assumed in the analysis following coastdown was that associated with natural circulation. The applicant has stated that this coefficient will be checked during the initial test program.

The applicant's LOFTRAN analysis shows that the natural circulation flow available adequately transfers the decay heat from the core to steam generators, which are being fed with emergency feedwater flow. The steam which is generated is assumed to be relieved through the SG safety valves. The primary system relief valves are assumed not to function.

The variations over time for this transient, as calculated by the applicant, show that DNBR remains above 1.30 and that the primary and secondary pressures remain below 110% of their design pressure. The staff agrees with the applicant's calculation and finds this acceptable.

15.2.7 Loss of Normal Feedwater Flow

A loss of normal feedwater flow can be caused by failures in the main feedwater system such as pump or valve malfunctions or by the loss of ac power. If the event is initiated by a loss of ac power, then the consequences are identical to those of the loss-of-nonemergency-ac-power event that is discussed in Section 15.2.6, above. The applicant has analyzed this event for the case where it is initiated by a pump or valve failure. The result is that there will be a reduction in the capability of the secondary side to remove heat from the primary side. In order to maximize the consequences of the event, the applicant's assumptions were made so as to minimize the heat-removal capability and to maximize the initial energy in the core. The assumptions included an auxiliary feedwater flow from one motor-driven pump, a heat-transfer coefficient associated with natural circulation of the RCS, the plant operating at 102% of design, the initial T_{avg} being 4°F over the nominal value, and auxiliary feedwater initiated by SG low-low level.

The analysis shows that a low-low level in an SG will initiate reactor trip. Since the condenser has been assumed to be unavailable, the secondary side pressure rise will be limited by lifting the secondary relief and safety valves. However, the applicant only took credit for the safety valves.

In FSAR Section 15.2.7.1, the applicant notes that a small secondary system break could affect normal feedwater control causing low SG levels before protective actions for this break. The applicant has committed to reanalyze this event and has stated that this reanalysis will be part of the response to IE Information Notice 79-22. The evaluation of the applicant's response to IE Information Notice 79-22 is included in Section 7.7 of this SER.

The applicant's results show that primary pressure remains at or below 2500 psia, that secondary pressure remains at or below 1250 psia and that DNBR drops to about 1.40 (60 seconds after event initiation) and then increases. These results are within the SRP criteria. The staff agrees with the applicant's calculations and finds this acceptable.

15.2.8 Feedwater System Pipe Breaks

In FSAR Section 15.2.8.1, the applicant states that pipe breaks upstream of the feedwater line check valve would preclude AFW flow to the faulted SG and would affect the plant only as a loss of feedwater and is covered by the evaluations in FSAR Sections 15.2.6 and 15.2.7. The applicant also states that, depending upon the size of the break and the operating conditions, the break could cause an RCS cooldown or heatup. The potential cooldown transient resulting from a feedwater line break is evaluated in Section 15.1.5 of this SER. As a result, the applicant analyzed the heatup case for this event and thus used assumptions that minimized secondary system heat-removal capability and maximized heat addition to the primary system coolant.

Sensitivity studies, as presented in WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture," have shown that the most limiting feedwater line break is a double-ended rupture of the largest feedwater line. The staff is reviewing WCAP-9230. Staff review at this time indicates reasonable assurance that the conclusions of the Westinghouse submittal will not be appreciably changed by completion of the review. If the final results of the review indicate that revisions to the applicant's analyses are necessary, the applicant will be required to implement the results of such changes. The staff does not consider this an open item.

The applicant has analyzed this event at full power with and without loss of offsite power and did not take credit for the pressurizer PORVs, pressurizer spray or for reactor trip on high pressurizer pressure, high pressurizer level, containment pressure, or overtemperature ΔT.

The analysis assumed reactor trip on SG low-low level. The auxiliary feedwater system was assumed to be initiated also on this signal and AFW is assumed to be provided by one motor-driven pump to the three intact SGs. The turbine-driven AFW pump was assumed to fail and the other motor-driven pump was assumed to deliver all of its flow out the break.

There is sufficient feedwater flow to adequately remove the residual heat after reactor shutdown. The use of safety-grade equipment will mitigate this accident. The applicant has provided the results of the analysis that shows that peak RCS pressure and peak secondary pressure remain below 110% of their design values and that DNBR remains above 1.30. The staff considers this acceptable.

15.3 Decrease in Reactor Coolant Flow Rate

15.3.1/15.3.2 Loss of Forced Reactor Coolant Flow, Including Trip of Pump and Flow Controller Malfunctions

A partial loss of coolant flow may be caused by a mechanical or electrical failure in an RCP motor, a fault in the power supply to the RCP, or a pump trip. A complete loss of flow may result from the simultaneous loss of electrical power to all pump motors. The loss of coolant flow, if the reactor is at power, will result in a rapid increase in coolant temperature.

Core protection against the partial and complete loss-of-coolant-flow events is provided by reactor trip on low primary coolant flow. Above permissive P8, low flow in any one loop will initiate a reactor trip; a reactor trip will be actuated on low flow in any two loops if the power level is below P8 but above P7. Also above P7, the undervoltage reactor trip is available. Frequency disturbances in the power grid can result in a reactor trip on RCP underfrequency.

The partial loss of reactor coolant flow was analyzed for a loss of two pumps with four loops in operation. The complete loss-of-coolant-flow event was analyzed with loss of all four pumps with all four loops in operation. These events were reviewed with the procedures and acceptance criteria set forth in SRP Sections 15.3.1 and 15.3.2.

Results provided by the applicant show that primary pressure remains well below the 110% of design pressure criteria. For both cases, the results show a decrease in the margin to DNB with the complete loss of coolant flow being more limiting. However, even for the limiting case, the minimum DNB, which is reached about 3 seconds into the transient, remains above 1.30 (approximately 1.33). Therefore, this is acceptable.

15.3.3/15.3.4 Reactor Coolant Pump Rotor Seizure and Shaft-Break Accident

The applicant has analyzed the reactor coolant pump (RCP) rotor seizure and shaft-break events with the LOFTRAN and FACTRAN computer codes. Since the initial rate of reduction of coolant flow is greater after an RCP rotor seizure, this is the limiting event. The locked rotor (RCP rotor seizure) was analyzed, both with a loss of offsite power and with offsite power available with four loops in operation. A rapid buildup in the coolant temperature results in expansion of the coolant into the pressurizer, causing a pressure increase in the RCS. In the analysis, the applicant states that credit was not taken for the pressurizer PORVs or for the pressurizer spray. The pressurizer safety valves are taken credit for in order to maintain pressure below the 110% of design limitation. The results show that the cladding goes into DNB and that the peak RCS pressure, maximum clad temperature, and zirconium-steam reaction are independent of whether or not offsite power is available. The maximum pressure was calculated to be 2548 psia.

The applicant's results show that fuel cooling enters into the nucleate boiling regime (i.e., DNB) within 1 second. Those fuel rods that were computed to have entered DNB were assumed to have failed for computing the radiological consequences. ~~consequences.~~ The radiological consequences of this event have been analyzed by the staff and found to fall within 10 CFR Part 100 guidelines (the acceptance criteria of the SRP).

15.4 Reactivity and Power Distribution Anomalies

In the following sections, the staff addresses the applicant's evaluation of events that result in reactivity and power distribution anomalies.

15.4.1 Uncontrolled Rod Cluster Control Assembly (Rod) Bank Withdrawal From Zero Power Conditions

The consequences of an uncontrolled rod cluster control assembly bank withdrawal at zero power have been analyzed. Such a transient can be caused by a failure of the reactor control rod control systems. The analysis assumed a conservatively small (in absolute magnitude) negative Doppler coefficient and a conservative moderator coefficient. Further, hot zero power initial conditions with the reactor just critical are chosen because they are known to maximize the calculated consequences. The reactivity insertion rate is assumed to be equivalent to the simultaneous withdrawal of the two highest worth banks at maximum speed (45 in./min).

Reactor trip is assumed to occur on the low setting of the power range neutron flux channel at 35% of full power (a 10% uncertainty has been added to the setpoint value). The maximum heat flux is much less than the full-power value and average fuel temperature increase to a value lower than the nominal full power value. The minimum DNBR at all times remains above the limiting value of 1.30.

The staff has reviewed this event according to SRP Section 15.4.1 (NUREG-0800).

The possibilities for single failures of the reactor control system which could result in uncontrolled withdrawal of control rods under low power startup conditions have been reviewed. The scope of the review has included investigations of initial conditions and control rod reactivity worths, the course of the resulting transients or steady-state conditions, and the instrument response to the transient or power maldistribution. The methods used to determine the peak fuel rod response, and the input into the analysis, such as power distributions and reactivity feedback effects due to moderator and fuel temperature changes, have been examined.

The staff concludes that the requirements of GDC 10, 20, and 25 have been satisfied. This conclusion is based on the following:

The applicant has satisfied the requirement of GDC 10 that the specified acceptable fuel design limits are not exceeded, GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and GDC 25 that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded. These requirements have been met by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures, and cladding strain limits should not be exceeded), to ensure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that the applicant's analyses of the maximum transients for single error control rod withdrawal from a subcritical or low-power condition have been confirmed, that the analytical methods and input data are reasonably conservative, and that specified acceptable fuel design limits will not be exceeded.

15.4.2 Uncontrolled Rod Cluster Control Assembly (Rod) Bank Withdrawal at Power

The consequences of uncontrolled withdrawal of a rod bank in the power operating range have been analyzed. The effect of such an event is an increase in coolant temperature (due to the core-turbine power mismatch) which must be terminated before exceeding fuel design limits.

The analysis is performed as a function of reactivity insertion rates, reactivity feedback coefficients, and core power level. Protection is provided by the high neutron flux trip, the overtemperature ΔT and overpower ΔT trips, and pressurizer pressure and pressurizer water level trips. In no case does the departure from nucleate boiling ratio fall below 1.30. Adequate fuel cooling is therefore maintained. The maximum heat flux reached, including uncertainties, does not exceed 118% of full power, thus precluding fuel centerline melting.

The staff has reviewed this event according to SRP Section 15.4.2 (NUREG-0800).

The possibilities for single failures of the reactor control system which could result in uncontrolled withdrawal of control rods beyond normal limits under power operation conditions have been reviewed. The scope of the review has included investigations of initial conditions and the range of reactivity insertions, the course of the resulting transients, and the instrumentation response to the transient. The methods used in determining the peak fuel rod response, and the input into the analysis, such as power distributions, rod reactivities, and reactivity feedback effects of moderator and fuel temperature changes, have been examined.

The staff concludes that the requirements of GDC 10, 20, and 25 have been satisfied. This conclusion is based on the following:

The applicant has satisfied the requirements of GDC 10 that the specified acceptable fuel design limits are not exceeded, GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and GDC 25 that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded. The requirements have been satisfied by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures, and cladding strain limits should not be exceeded), to ensure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that the applicant's analysis of maximum transients for single-error control rod malfunctions have been confirmed, the analytical methods and input data are reasonably conservative, and specified acceptable fuel design limits will not be exceeded.

15.4.3 Rod Cluster Control Assembly Malfunctions

The applicant has analyzed rod cluster control assembly misalignment incidents, including a dropped full-length assembly, a dropped full-length bank, a misaligned full-length assembly, and the withdrawal of a single assembly while operating at power. Misaligned rods are detectable by: (1) asymmetric power distributions sensed by excore nuclear instrumentation or core exit thermocouples, (2) rod deviation alarm, and (3) rod position indicators. A deviation of a rod from its bank by about 15 in., or twice the resolution of the rod position indicator, will not cause the power distribution to exceed design limits. Additional surveillance will be required to ensure rod alignment if one or more rod position channels are out of service.

In the event of a dropped assembly or group of assemblies, the reactor will typically scram on a neutron flux negative rate trip, and analysis indicates that thermal limits will not be exceeded for the event. If the rod locations are such that the reactor does not scram, however, the automatic controller may return the reactor to full power and the control could result in a power overshoot. An analysis methodology for this event has been developed by Westinghouse and reported in WCAP-10297-P, "Dropped Rod Methodology for Negative Flux Rate Trip Plants," January 1982. The staff reviewed and approved this methodology in a letter dated March 31, 1983 from C. O. Thomas, NRC, to E. P. Rahe, Westinghouse. Generally, detailed analyses for most reactors, for most cycles, show that if this event occurs, thermal limits will be exceeded.

However, the analysis is reactor and cycle specific, and the applicant has not yet completed or submitted analyses for Vogtle for cycle 1. The staff has also accepted an interim position for operating reactors which consists of a restriction on operations above 90% power, so that either the reactor is in manual control or rods are required to be out more than 215 steps. This restriction will be applied to Vogtle in the event that calculations for cycle 1 operation are not completed in time for initial operations. With this restriction, thermal limits will not be exceeded. When the analysis specific to Vogtle for cycle 1 is approved, the restriction will be removed. Similar analysis will also be needed for each subsequent reload cycle.

For cases where a group is inserted to its insertion limit with a single rod in the group stuck in the fully withdrawn position, analysis indicates that departure from nucleate boiling (DNB) will not occur. The staff has reviewed the calculated estimates of the expected reactivity and power distribution changes that accompany postulated misalignments of representative assemblies. The staff has concluded that the values used in this analysis conservatively bound the expected values including calculational uncertainties.

The inadvertent withdrawal of a single assembly requires multiple failures in the control system, multiple operator errors, or deliberate operator actions combined with a single failure of the control system. As a result, the single assembly withdrawal is classified as an infrequent occurrence. The resulting transient is similar to that from a bank withdrawal, but the increased peaking factor may cause DNB to occur in the region surrounding the withdrawn assembly. Less than 5% of the rods in the core experience DNB for such a transient.

The staff has reviewed this event according to SRP Section 15.4.3 (NUREG-0800).

The possibilities for single failures of the reactor control system which could result in a movement or malposition of control rods beyond normal limits have been reviewed. The scope of the review has included investigations of possible rod malposition configurations, the course of the resulting transients or steady-state conditions, and the instrumentation response to the transient or power maldistribution. The methods used to determine the peak fuel rod response, and the input to that analysis, such as power distribution changes, rod reactivities, and reactivity feedback effects due to moderator and fuel temperature changes, have been examined.

The staff concludes that the requirements of GDC 10, 20, and 25 have been satisfied. This conclusion is based on the following:

The applicant has satisfied the requirements of GDC 10 that the specified acceptable fuel design limits are not exceeded, GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and GDC 25 that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded. These requirements have been fulfilled by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures, and cladding strain limits should not be exceeded), to ensure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that maximum configurations and transients for single error control rod malfunctions have been analyzed, that the analysis methods and input data are reasonably conservative, and that specified acceptable fuel design limits will not be exceeded.

15.4.4/15.4.5 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

In FSAR Section 15.4.4, the applicant provides the results of an analysis for startup of an inactive reactor coolant pump event. This event was reviewed with the procedures and acceptance criteria set forth in SRP Section 15.4.4.

The applicant assumed that the transient began at a power level of 72%.

During the first part of the transient, the increase in core flow with cold water results in an increase in nuclear power and a decrease in core average temperature. Reactivity addition for the inactive loop startup event is the result of the decrease in core inlet water temperature. This transient was evaluated by the applicant using a mathematical model that has been reviewed and found acceptable to the staff. The maximum calculated RCS pressure is below 2300 psia and the minimum DNBR is above 1.30 throughout the transient.

15.4.6 Inadvertent Boron Dilution

Unborated water can be added to the reactor coolant system, via the chemical and volume control system (CVCS), to increase core reactivity. This may happen inadvertently, because of operator error or CVCS malfunction, and cause an unwanted increase in reactivity and a decrease in shutdown margin. The applicant has analyzed this event during all modes of operation. Operator action is required to mitigate this event and time limitations for this action are set forth in SRP Section 15.4.6.

The boron dilution that was analyzed by the applicant was caused by the flow from the reactor water makeup system (part of the CVCS) into the RCS. The applicant has stated that the maximum flow from this system is 242 gpm of unborated water.

For the boron dilution event during power operations, the reactor is tripped by overtemperature ΔT. According to the applicant, if the reactor is in a manual reactor control mode, then the alarms associated with this trip alert the operator to take action. If the reactor is in automatic, then the operator is alerted to the event by the rod insertion limit alarms. During startup, the applicant states that the event will cause a trip on power range neutron flux and the alarms associated with this trip alert the operator to take action. For these modes of operation, the applicant has demonstrated that sufficient time exists for the operator to take action. Furthermore, the applicant states, in response to a staff concern, that the transient prior to reactor trip for these two modes is bounded by the rod withdrawal at power event.

The applicant discounts the boron dilution event during refueling since, by administrative procedure, the RCS is isolated from unborated water sources by locking closed key valves. This item will remain open until Technical Specifications are provided that would require these valves (175, 176, 177, and 183) to be locked closed during refueling.

Preliminary results of the applicant's analysis show that the time available to the operator to take mitigative steps for this event during hot standby, hot shutdown, and cold shutdown, is insufficient. The applicant has committed to provide supplemental information in the FSAR on this topic. Previous staff concerns, stated in the May 31, 1984, letter to the applicant, regarding DNBR, RCS pressure, and redundant alarms, should also be addressed. This item is open.

The applicant was asked to describe the potential for boron dilution because of the chemical addition portion of the CVCS and because of dilution sources other than the CVCS. In response, the applicant has stated that either these other sources are precluded because they would require more than one failure to bring about RCS dilution or that they are bounded by the dilution source assumed in the FSAR. The staff considers this response acceptable.

15.4.7 Inadvertent Loading of a Fuel Assembly Into Improper Position

Strict administrative controls in the form of previously approved established procedures and startup testing are followed during fuel loadings to prevent operation with a fuel assembly in an improper location or a misloaded burnable poison assembly. Nevertheless, an analysis of the consequences of a loading error has been performed.

The applicant presented comparisons of power distributions calculated for the nominal fuel loading pattern and those calculated for five loadings with misplaced fuel assemblies or burnable poison assemblies. The selected non-normal loadings represent the spectrum of potential inadvertent fuel misplacement. Calculations included, in particular, the power in assemblies which contain provisions for monitoring with incore detectors.

As part of the required startup testing, the incore detector system is used to detect misloaded fuel before operating at power. The analysis described above shows that all but one of the above misloading events would be detected by this test. In the excepted case, an interchange of region 1 and 2 assemblies near the center of the core, the increase in the power peaking is approximately equal to the uncertainty in the measurement of this quantity (5%). This uncertainty is allowed for in analyses so that this misloading event does not result in unacceptable consequences.

The staff has reviewed this event according to SRP Section 15.4.7 (NUREG-0800).

The staff has evaluated the consequences of a spectrum of postulated fuel loading errors. The staff concludes that the analyses provided by the applicant have shown for each case considered that either the error is detectable by the available instrumentation (and hence remediable) or the error is undetectable but the offsite consequences of any fuel rod failures are a small fraction of 10 CFR Part 100 guidelines. The applicant affirms that the available incore instrumentation will be used before the start of a fuel cycle to search for fuel-loading errors.

The staff concludes that the requirements of GDC 13 and 10 CFR Part 100 have been satisfied. This conclusion is based on the following:

The applicant has satisfied the requirements of GDC 13 with respect to providing adequate provisions to minimize the potential of a misloaded fuel assembly going undetected and satisfies 10 CFR Part 100 with respect to mitigating the consequences of reactor operations with a misloaded fuel assembly. These requirements have been fulfilled by providing acceptable procedures and design features that will minimize the likelihood of loading fuel in a location other than its designated place.

15.4.8 Spectrum of Rod Ejection Accidents

15.4.8.1 Rod Cluster Control Assembly Ejection

The mechanical failure of a control rod mechanism pressure housing would result in the ejection of a rod cluster control assembly. For assemblies initially inserted, the consequences would be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Although mechanical provisions have been made to make this accident extremely unlikely, the applicant has analyzed the consequences of such an event.

Methods used in the analysis are reported in WCAP-7588, Revision 2, "An Evaluation of the Rod Ejection Accident in Westinghouse Reactors Using Spatial Kinetics Methods," which has been reviewed and accepted by the staff. This report demonstrated that the model used in the accident analysis is conservative relative to a three-dimensional kinetics calculation.

The applicant's criteria for gross damage of fuel are a maximum cladding temperature of 2700°F and an energy deposition of 200 cal/gm in the hottest pellet. These criteria are more conservative* than those proposed in RG 1.77. Therefore, they are acceptable.

Four cases were analyzed: beginning-of-cycle at 102% and zero power and end-of-cycle at 102% and zero power. The highest cladding temperatures, 2426°F, and the highest fuel enthalpy, 172 cal/gm, were reached in the beginning-of-cycle zero power and beginning-of-cycle full-power cases, respectively. The analysis also shows that less than 10% of the fuel experiences DNB and less than 10% of the hot pellet melts. Analyses have been performed to show that the pressure surge produced by the rod ejection is mild and will not approach the reactor coolant system emergency limits. Further analyses have shown that a cascade effect, i.e., the ejection of a further rod because of the ejection of the first one, is not credible.

The staff has reviewed this event according to SRP Section 15.4.8 (NUREG-0800).

The staff concludes that the analysis of the rod ejection accident is acceptable and satisfies the requirements of GDC 28. This conclusion is based on the following:

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*RG 1.77 has an acceptance criterion of 280 cal/gm energy deposition and no criterion for cladding temperature other than that implicit in requirements for fuel and pressure vessel damage.

The applicant satisfied the requirements of GDC 28 with respect to preventing postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary greater than limited local yielding, or cause sufficient damage that would significantly impair the capability to cool the core. The requirements have been satisfied by demonstrating that the regulatory positions of RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for PWRs" are complied with. The staff has evaluated the applicant's analysis of the assumed control rod ejection accident and finds the assumptions, calculation techniques, and consequences acceptable. Since the calculations resulted in peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. The pressure surge was, therefore, calculated on the basis of conventional heat transfer from the fuel and resulted in a pressure increase below "Service Limit C" (as defined in Section III, "Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code) for the maximum control rod worths assumed. The staff finds that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained.

15.4.8.2 Radiological Consequences of a Rod Ejection Accident

A nonmechanistic rupture of a control rod drive housing is postulated. Because of the resultant opening in the pressure vessel, primary coolant is lost to the containment with concurrent rapid depressurization of the reactor pressure vessel. Reactor trip, initiated by one of several trip signals, is assumed to occur rapidly.

Ejection of a control rod results in rapid reactivity insertion. The applicant has conservatively assumed that 10% of the fuel elements will experience cladding failure, releasing the volatile fission products in the fuel-cladding gap. In addition, the applicant stated that 0.25% of the fuel rods may experience fuel melting. The fission products released as a result of this damage to the fuel are assumed to be released with the primary coolant. The release to the environment may occur to either of two pathways. The first pathway involves a release of coolant carrying fission products to the primary containment, which is then assumed to leak to the atmosphere at the design leak rate of the containment (0.2% per day). In the second pathway, activity would reach the secondary coolant via steam generator tube leaks. A maximum of 1-gpm primary-to-secondary leak rate is assumed (as limited by Technical Specifications). With loss of offsite power (assumed to occur as a result of reactor/turbine trip) and subsequent steam venting, some of the iodine transferred to the shell (secondary) side is available for leakage to the environment.

In considering the consequences of this postulated event, the staff calculated the doses from the activity available for release separately for each of the above pathways. The staff would expect the actual consequences to be some combination of these pathways. The assumptions used in calculating the radiological consequences are presented in Table 15.4, and the resultant doses for each pathway are given in Table 15.1 of this section.

The staff has reviewed the applicant's analysis of the radiological consequences following a postulated control rod ejection accident. The staff concludes that the proposed operation of the Vogtle plant within the limits of the Technical Specifications assumed above provides reasonable assurance that the calculated radiological consequences are well within (less than 25%) the dose guidelines of 10 CFR 100.11.

The staff's conclusion is based on (1) review of the applicant's analysis of the radiological consequences, (2) independent dose calculations using the recommendations of Appendix B of RG 1.77 and the atmospheric dispersion factors as discussed in Section 2 of this SER, and (3) the Westinghouse Standard Technical Specifications for the primary-to-secondary leakage in the steam generators.

15.5 Increases in Reactor Coolant System Inventory

15.5.1 Inadvertent Operation of the Emergency Core Cooling System During Power Operation

ECCS operation could be initiated by a spurious signal or an operator error. An SI signal would ordinarily result in a trip of the reactor followed by a turbine trip. However, operator action can take place to block the SIS signal. The applicant has examined the case in which the reactor trips later in the transient because of low RCS pressure. The DNBR never drops below its initial value.

If the operator fails to turn off the charging pumps, the safety valves will open. Continued operation of these pumps would overfill the pressure relief tank. However, as stated in Table 6.3-1 of the FSAR, the cutoff head of the charging pumps is 6200 ft (2687 psig); so they cannot create 110% of the reactor vessel design pressure (2733 psig) and thus cannot fail the vessel.

15.5.2 CVCS Malfunction That Increases Reactor Coolant Inventory

The evaluation of the consequences of injecting unborated water is included in Section 15.4.6 of this SER. The evaluation of the consequences of injecting borated water is included in Section 15.5.1 of this SER.

15.6 Decreases in Reactor Coolant Inventory

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

In FSAR Section 15.6.1, the applicant provides the results of an analysis for inadvertent opening of a pressurizer safety valve. This event bounds the inadvertent opening of a relief valve. During this event, nuclear power is maintained at the initial value until reactor trip occurs on low pressurizer pressure. The DNBR decreases initially, but increases rapidly following the trip. The minimum DNBR of approximately 1.34 occurred at 33 seconds into the transient. The RCS pressure decreases throughout the transient.

In response to TMI-2 Task Action Plan Item II.K.3.1 (Installation and Testing of Automatic Power Operated Relief Valve Isolation System), the applicant has stated that each of the PORVs has an associated block valve that will automatically close on low pressurizer pressure. The applicant should confirm that a test of the automatic block valve closure system will be conducted following installation. The applicant also stated that the requirements of TMI-2 Task Action Plan Item II.K.3.2 (Report on Overall Safety Effect of PORV

Isolation System) is not applicable to the plant. This results from compliance with Item II.K.3.1. The applicant must demonstrate that the automatic PORV isolation system has been overridden manually in the control room and that this has been taken into account in the plant emergency procedures and in operator training, in order to mitigate the consequences of a steam generator tube rupture (SGTR).

15.6.2 Radiological Consequences of the Failure of a Small Line Carrying Primary Coolant Outside Containment

There are a number of small lines carrying primary coolant outside the containment. The applicant has provided an analysis of an accidental break in the chemical and volume control (CVS) letdown line outside containment, but downstream of the containment isolation valves, as a worst-case failure of one of these lines. This break would release 194 gpm of primary coolant to the auxiliary building before isolation could be expected. The break would cause a low level in the volume control tank, and the operator could diagnose the break and shut the appropriate isolation valve to isolate the leak.

The staff has performed an independent assessment of the potential dose consequences of the release of primary coolant outside the containment. The staff assumed that 20 minutes will elapse before the operator isolates a CVCS line break in response to receipt of the low-level signal. Thus, a total of 3880 gallons of primary coolant could be released. The staff estimated that 39% of the hot reactor coolant would flash into steam upon entering the auxiliary building atmosphere, and assumed that a proportional fraction of the iodine dissolved in the coolant would become airborne in gaseous or particulate form. In the absence of ESFs designed to detect and mitigate the consequences of such a release, the staff assumed that this airborne iodine can escape directly to the environment at ground level, without delay or effective filtration. Other assumptions are given in Table 15.6.

The staff concludes that the consequences of a postulated small-line failure outside the containment, assuming the primary coolant equilibrium iodine concentration permitted by the Standard Technical Specifications, in combination with an accident-generated iodine spike, do not exceed a small fraction of the exposure guidelines of 10 CFR 100.11. The results of the staff's calculations are given in Table 15.1.

The staff's conclusion is based on (1) review of the applicant's classification and identification of small lines in accordance with GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," and RG 1.11, "Instrument Lines Penetrating Containment"; (2) review of the applicant's analysis of radiological consequences of a failure in the CVCS line; (3) independent dose calculations using Position C.1.b. of RG 1.11 and conservative atmospheric dispersion factors as discussed in Section 2 of this report; and (4) the Westinghouse Standard Technical Specifications for the equilibrium iodine concentrations in the primary coolant system. The staff will review the specific Vogtle Technical Specifications to ensure that the coolant activity limits assumed above are not exceeded.

15.6.3 Steam Generator Tube Rupture

The applicant has provided an analysis of the systems response and radiological consequences of a steam generator tube rupture (SGTR) accident. The staff has requested justification for the asserted ability to isolate the affected steam generator within 30 minutes and for the assumed mitigative capability of systems to reduce the radiological consequences of the accident. In response to the staff's request, the applicant states that the Westinghouse Owners Group is investigating several SGTR licensing concerns and will address the staff's concerns through a generic resolution in late 1984. Upon receipt of this additional information, the staff will complete its review of the SGTR event and the radiological consequences thereof. The staff's review and conclusions will be reported in a future supplement to the SER.

15.6.4 Radiological Consequences of a Main Steamline Break Outside Containment (BWR)

This SRP section does not apply to PWRs.

15.6.5 Loss-of-Coolant Accident

In FSAR Section 15.6.5, the applicant has analyzed the double-ended cold-leg guillotine (DECLG) as the most limiting large-break LOCA. The analysis was accomplished using three different flow coefficients. The results of this calculation show that the DECLG with a Moody break discharge coefficient of 0.6 with maximum safety injection is the worst case. In this analysis, peak clad temperature reached is 2172°F. For the small-break LOCA, the applicant has determined that a cold-leg rupture of less than 10-in. diameter is the most limiting. The analysis was performed for 3-in., 4-in. and 6-in.-diameter breaks. The results show that the 4-in.-diameter break results in the highest peak clad temperature (1537°F). The 3-in. break results in the greatest amount of zirconium/steam reaction (0.78%). Both of these accidents are terminated by SIS and ECCS operations. Only safety-grade equipment is used to mitigate the accident.

The applicant has performed analyses of the performance of the ECCS in accordance with the Commission's regulations (10 CFR 50.46 and Appendix K to 10 CFR 50) except as noted in SER Section 6.3.5.1. The analyses considered a spectrum of postulated break sizes and locations. As shown in NUREG-0390, these analyses were performed with an evaluation model that the staff had previously reviewed and approved. The results show that the ECCS satisfies the following criteria:

(1) The calculated maximum fuel rod cladding temperature does not exceed 2200°F.

(2) The calculated maximum local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

The applicant is a member of the Westinghouse Owners Group that is evaluating TMI-2 Task Action Plan Item II.K.2.13 (Thermal Mechanical Report: Effect of High Pressure Injection on Vessel Integrity for Small-Break LOCA With No Auxiliary Feedwater). The staff's review of this item will be covered in NRC Unresolved Safety Issue A-49, "Pressurized Thermal Shock."

In response to TMI-2 Task Action Plan Item II.K.3.30 (Revised Small-Break LOCA Methods To Show Compliance With 10 CFR 50, Appendix K), the applicant stated that Westinghouse has submitted a new small-break evaluation model to NRC. The staff is currently reviewing this submittal. The applicant further stated that after the staff's review of this evaluation model is completed, a plant-specific submittal on this issue will be supplied. This will address TMI-2 Task Action Plan Item II.K.3.31 (Plant-Specific Calculations To Show Compliance With 10 CFR 50.46). These items are confirmatory.

The staff concludes that the calculated performance of the ECCS following postulated LOCAs conform to the Commission's regulations and to applicable regulatory guides and staff technical positions except as noted, and the ECCS performance is considered acceptable for the postulated accidents.

15.6.5.1 Radiological Consequences of a Loss-of-Coolant Accident

The limiting fault postulated as the design basis for the containment and its associated engineered safety features, and as a demonstration of the adequacy of the distances to the EAB and the LPZ is a loss-of-coolant accident (LOCA) in conjunction with the release of a substantial fraction of the fission product inventory of the core, as set forth in 10 CFR 100.11(a). The analysis has included the sources and radioactivity transport assumptions specified in

RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," as well as additional guidance contained in the Standard Review Plan.

This postulated event involves the assumed availability for release from the containment atmosphere of 100% of the core's inventory of noble gas, and 25% of the iodine inventory. Although the containment is assumed to be intact, two pathways for slow leakage of these fission products to the environment were identified:

(1) leakage from containment
(2) leakage from ESF systems outside containment

The contribution of these two leakage pathways to the calculated offsite doses are discussed below.

Containment Leakage Pathway

The safety features of the Vogtle plant include a containment designed to minimize the leakage of fission products from postulated accidents involving the failure of the first two barriers against a release of fission products, i.e., the fuel cladding and primary pressure boundary. The containment consists of a post-tensioned concrete primary containment vessel with a carbon steel liner. Another engineered safety feature is the containment spray system with a sodium hydroxide additive to achieve a slightly basic pH in the water collecting in the containment sump following a LOCA. The staff's calculation of the consequences of the hypothetical LOCA used the conservative assumptions of Positions C.1.a through C.1.e of RG 1.4, Revision 2. The primary containment was assumed to leak at a rate of 0.2% per day for the first 24 hours and 0.1% per day after 24 hours. The analysis took into account radiological decay during holdup in the containment, mixing in the containment, iodine decontamination by the ESF spray system, and conservative estimates of dispersion of the fission products in the environment. A list of assumptions used in the calculation of the LOCA doses is given in Table 15.2.

Post-LOCA Leakage From ESF System Outside Containment

During the recirculation mode of operation of the emergency core cooling system (ECCS) following a LOCA, the sump water is circulated outside containment to the auxiliary building for cooling and re-injection via the ECCS or containment spray system. Normal leakage of equipment in this fluid system, or malfunctions, such as a pump seal failure, would result in an airborne release of any volatile forms of fission products carried by the sump water to the auxiliary building. For Vogtle, the ECCS area in the auxiliary building is served by an ESF air filtration system (the auxiliary building exhaust system) which would be expected to collect and process such releases through filters. Therefore, doses resulting from passive failure of equipment carrying this fluid were not explicitly considered (as specified in SRP Section 15.6.5, Appendix B) in staff calculations.

In FSAR Table 15.6.5-4, the applicant has identified a value of 50 gpm as the maximum amount of leakage from ECCS equipment following an accident. Following the Standard Review Plan, the staff evaluated the potential radiological consequences from this release pathway assuming a leakage rate of twice the applicant's value. The staff's resultant radiological consequence estimates were 71 rem to the thyroid at the EAB and 148 rem to the thyroid at the LPZ boundary. The staff considers 50 gpm to be a larger-than-necessary leakage value, that could be substantially reduced. The applicant has committed to implement a program to minimize such leakage in accordance with the TMI Task Action Plan requirements.

Conclusions

The staff's calculated thyroid and whole-body doses from the hypothetical LOCA are given in Table 15.1. The staff concludes that the distances to the exclusion area and to the LPZ boundaries of the Vogtle site, in conjunction with the ESFs of the Vogtle plant design, are sufficient to provide reasonable assurance that the total radiological consequences of a postulated LOCA will fall within the exposure guidelines set forth in 10 CFR 100.11. This conclusion is based on the staff's review of the applicant's analyses and on the independent analysis performed to verify that the sum of the calculated doses arising from containment leakage and ECCS component leakage outside containment are within the guidelines.

15.7 Radioactive Releases From a Subsystem or Component

15.7.1 Deleted*

15.7.2 Deleted*

15.7.3 Later

15.7.4/15.7.5 Radiological Consequences of a Fuel-Handling Accident

In the evaluation of the fuel-handling accident, the criteria and methodology used by the staff are based on 10 CFR 100, GDC 61(3), Positions C.1.a through C.1.f of RG 1.25, and SRP Section 15.7.4. The staff assumed that a single fuel assembly is dropped into the fuel pool during refueling operations and that all of the fuel rods in the assembly were damaged, releasing radioactive materials in the fuel gaps into the pool.

For the case of a fuel-handling accident in the fuel building, the applicant estimates the time for the radioactive materials that escape from the pool to travel from the detector to the isolation damper to be slightly over 0.6 second. The closure time of the isolation dampers in the normal exhaust system following receipt of an isolation signal is estimated at 6 seconds. Because the travel time is less than the isolation time by more than 5 seconds, the staff assumed that the entire activity release escapes from the fuel building without reduction by the ESF filters.

In the case of a fuel-handling accident occurring inside containment, radioactive release was assumed to mix with 25% of the containment atmosphere; isolation was assumed to occur within 15 seconds. No filtration credit was assumed for containment activity that escapes before containment isolation occurs.

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*Deleted from the July 1981 edition of the Standard Review Plan (NUREG-0800).

The estimated offsite doses for the postulated fuel-handling accidents inside containment and inside the fuel building are shown in Table 15.1. The list of assumptions and parameters used in the analysis are given in Tables 15.5a and 15.5b. The potential doses for the fuel-handling accidents are well within the guideline values given in 10 CFR 100. Therefore, the staff concludes that the applicant has provided a design that satisfies the portion of GDC 61 concerning appropriate containment, confinement, and filtering systems.

With regard to a potential spent-fuel-cask-drop accident, a type 1 single-failure-proof crane (used in handling the cask) designed according to NUREG-0554 is prevented from being moved over the fuel. Pending final review of this crane (see Section 9.1.5), the staff will be able to conclude that the likelihood of a spent-fuel-cask-drop accident is sufficiently small that no radiological consequence analysis is required.

15.8 Anticipated Transients Without Scram

Anticipated transients without scram (ATWS) are events in which the scram system (reactor trip system) is postulated to fail to operate as required. This subject has been under generic review by the staff for several years.

In December 1978, Volume 3 of NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors" was issued describing the proposed type of plant modifications the staff believed necessary to reduce the risk from anticipated transients with failure to scram to an acceptable level. The staff issued requests for the industry to supply generic analyses to confirm the ATWS mitigation capability described in Volume 3 of NUREG-0460 and, subsequently the staff presented recommendations on plant modifications to the Commission in September 1980. The staff recommended to the Commission that rulemaking be used to determine the required modifications to resolve ATWS concerns as well as the required schedule for implementing of such modifications. The Vogtle facility is subject to rulemaking on this matter. Rulemaking was accomplished with publication in the Federal Register (49 FR 26036) on June 26, 1984, of the final rule, "Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants." The staff has not yet completed action to implement the rule in accordance with the Commission directive in the rule.

In the meantime, the following discusses the bases for operation of Vogtle at full power while the schedule for final implementation of the ATWS rule is being developed.

NUREG-0460, Volume 3, states:

> 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 judgement in view of: (a) the estimated arrival rate of anticipated transients with potentially severe consequences in the event of a 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 action to be required by the new rule, the staff concludes that Vogtle can operate because the risk from ATWS events in the time period to implement the rule is acceptably small. As a prudent course, to further reduce the risk from ATWS events during the interim period before completing the plant modifications determined by the Commission to be necessary, the staff requires that emergency procedures be developed to assist operators in recognizing and mitigating an ATWS event.

These procedures shall include 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 electric portion of the reactor scram system. When implemented, these procedures will provide an acceptable basis for interim operation of the Vogtle facility based on the staff's understanding of the plant response to postulated ATWS events.

Conclusions

As noted in Section 13.5.2 of this SER, the applicant has committed to implement EOPs based on the Westinghouse ERGs that have been endorsed by Generic

Letter 83-22. These guidelines include instructions for coping with ATWS. The staff concludes that the applicant's commitment to implement procedures based on these guidelines is acceptable on an interim basis for full-power operation. Future modifications will be needed to implement the ATWS rule; the staff will determine the required schedule for implementing such modifications.

15.9 TMI-2 Task Action Plan Requirements

Later

15.9.1

Later

15.9.2 Item II.F.1 Attachment 1, Noble Gas Effluent Monitor, and Attachment 2, Sampling and Analysis of Plant Effluents

Potential gaseous accident release pathways are the following:

(1) the plant vent, which includes discharges from the containment purge system, the auxiliary building HVAC system (which includes discharges from the gaseous radwaste system and the containment piping penetration area filter and exhaust system), the fuel-handling building HVAC, and the containment electrical penetration area filter and exhaust system in the control building

(2) the condenser air ejector and steam packing exhauster system

(3) the steam generator safety relief valves and atmospheric dump valves

(4) the auxiliary feedwater steam turbine exhaust vent

(5) the steam generator blowdown line break overpressurization relief damper.

Effluent radiogas, iodine, and particulate monitors are provided at the plant vent and at the condenser air ejector and steam packing exhauster. A strap-on monitor is also provided at the main steamline. Rather than using noble gas effluent monitors, the applicant has proposed to use the main steamline monitors to estimate the releases from the actuation of the steam generator safety relief valves, atmospheric pump valves, and the auxiliary feedwater steam turbine exhaust vent.

The applicant also has proposed to use the steam generator liquid monitor and the appropriate steam generator blowdown line flow instrumentation to estimate the activity released from the steam generator blowdown line break overpressurization relief damper. These proposals differ from the SRP provisions for the instrumented monitoring or sampling and analysis of identified gaseous effluent paths in the event of postulated accident releases. The SRP further provides that the design of the systems meets the provisions of NUREG-0737 which states that the use of the main steamline monitors to estimate the releases from the safety relief valves is acceptable. However, since there is no noble gas effluent monitor on the steam generator blowdown line break overpressurization relief damper, this is an open item.

The applicant has stated that design information pertaining to accident range noble gas effluent monitors and continuous sampling of gaseous effluents for postaccident releases of radioactive iodines and particulates has not been finalized and will be available by May 1985. This is a confirmatory item.

15.9.3

Later

15.9.4

Later

15.9.5 **Item III.D.1.1** Integrity of Systems Outside Containment Likely to Contain Radioactive Material

The applicant has committed to a program to reduce leakage from systems outside containment which could contain highly radioactive fluids during a serious transient or accident to "as low as practical" levels and has stated that the specified information on the proposed program will be submitted no later than 4 months before fuel loading. This is a confirmatory item.

15.9.6 - 15.9.14

Later

Table 15.1 Radiological consequences of design-basis accidents as calculated by the staff

Postulated accident	Exclusion-area-boundary dose, rem*		Low-population-zone dose, rem	
	Thyroid	Whole body	Thyroid	Whole body
Loss of coolant:				
Containment leakage				
0-2 hr	98	2.6		
0-8 hr			33	0.9
8-24 hr			12	0.2
24-96 hr			10	0.1
96-720 hr			9	0.1
Total containment leakage	98	2.6	64	1.3
ECCS component leakage	71	0.2	148	0.2
Total	169	2.8	212	1.5
Steamline break outside containment:				
Long-term operation case (DEI-131 at 1 μCi/gm)**	2.7	<1.0	1.6	<1.0
Short-term operation case (DEI-131 at 60 μCi/gm)	3.1	<1.0	1.5	<1.0
Control rod ejection:				
Containment leakage pathway	26	<1.0	43	<1.0
Secondary system release pathway	9.7	<1.0	1.8	<0.1
Fuel-handling accident in fuel-handling area	53	0.7	8	0.2
Fuel-handling accident inside containment	0.3	<0.1	0.1	<0.1
Small-line break	4.6	<0.1	0.8	<0.1

*The short-term diffusion estimate (χ/Q) used in the analysis are those presented and discussed in Section 2.3.4. The meteorological models described in regulatory guides referenced in these analyses are modified by those presented in RG 1.145. See Section 2.3.4 of this SER for further discussion of the meteorological models.

**DEI-131 is the dose-equivalent iodine-131 concentration, as defined in the Standard Technical Specifications.

Table 15.2 Assumptions used in the calculation of loss-of-coolant-accident doses

Parameter and unit of measure	Quantity
Containment leakage contribution	
Power level, MWt	3,565
Operating time, yr	3
Fraction of core inventory available for containment leakage, %	
Iodine	25
Noble gases	100
Initial iodine composition in containment atomsphere, %	
Elemental	91
Organic	4
Particulate	5
Containment leak rate, %/day	
0-24 hr	0.2
After 24 hr	0.1
Containment volume, ft^3	
Sprayed volume	2.15×10^6
Unsprayed volume	6.05×10^5
Containment mixing rate from cooling fan operation, ft^3/min	174,000
Containment spray system	
Maximum elemental iodine decontamination factor	100
Spray removal coefficients/hr	
Elemental iodine	10
Particular iodine	0.45
Organic iodine	0
Relative concentration values, sec/m^3*	
0-2 hr at the exclusion area boundary (EAB)	1.8×10^{-4}
0-8 hr at the low population zone (LPZ) boundary	3.1×10^{-5}
8-24 hr at the LPZ boundary	2.2×10^{-5}
24-96 hr at the LPZ boundary	1.0×10^{-5}
96-720 hr at the LPZ boundary	3.4×10^{-6}
ECCS leakage outside containment	
Power, MWt	3,565
Sump volume, gal	905,080
Fraction of iodine assumed volatilized and/or airborne as aerosol, %	0.1

*Used for all accidents.

Table 15.2 (continued)

Parameter and unit of measure	Quantity
<u>ECCS leakage outside containment</u> (continued)	
Leak rate, gph (twice the maximum operational leakage defined in FSAR Table 15.6.5-4)	6000
Leak duration, hr	720
Delay time, hr	0.50
Filter efficiency for iodine, %	
Elemental and particulate	99
Organic iodine	99

Table 15.3 Assumptions used to evaluate the radiological consequences following a postulated main-steamline-break accident outside containment

Parameter and unit of measure	Quantity
Power, MWt	3565
Preaccident dose equivalent I-131 in primary coolant, μCi/gm	1.0 (case 2)*
Preaccident dose equivalent I-131 in primary coolant, μCi/gm	60.0 (case 3)**
Primary-to-secondary leak rate, as limited by Technical Specifications, gpm	1.0
Leakage in the affected steam generator, gpm	1.0
Fraction of iodine entering shell side of the steam generator released to the environment, %	1.0
Ratio of iodine release rate from fuel during iodine spike to release rate from fuel during steady-state operation	500:1

*Long-term operation case.

**Short-term operation case.

Table 15.4 Assumptions used for estimating the radiological consequences following a postulated control-rod-ejection accident

Parameter and unit of measure	Quantity
Power, MWt	3565
Primary-to-secondary leak rate, gpm	1.0
Fraction of the fuel rods experience cladding failure, %	0.1
Fraction of noble gas and iodine inventory in gap of failed rods, %	0.1
Fraction of the fuel rods experiencing fuel melting, %	0.0025
Fraction of iodine inventory released from rods experiencing melting, %	0.5
Fraction of iodine entering steam generator secondary side released to environs, %	0.1
Time of primary and secondary system pressures equalization, sec	3300
Fraction of iodine plated out in containment, %	0.5
Design leak rate of containment, %/day	0.2
Iodine concentration (DEI-131) in the secondary coolant, μCi/gm	0.1

Table 15.5a Assumptions used for estimating the radiological consequences following a postulated fuel-handling accident in fuel-handling area

Parameter and unit of measure	Quantity
Power level, MWt	3,565
Number of fuel rods damaged	264
Total number of fuel rods in core	50,952
Radial peaking factor of damaged rod	1.65
Shutdown time, hr	100
Inventory released from damaged rods (iodines and noble gases), %	10
Pool decontamination factors	
Iodine	100
Noble gases	1
Iodine forms in atmosphere above pool, %	
Elemental	75
Organic	25
Iodine removal efficiencies for ABGTS (spent fuel pool area), %	
Elemental	No filters assumed
Organic	No filters assumed

Table 15.5b Assumptions used for estimating the radiological consequences following a postulated fuel-handling accident inside containment*

Parameter and unit of measure	Quantity
Power level, MWt	3,565
Number of fuel rods damaged	264
Total number of fuel rods in core	50,952
Radial peaking factor of damaged rod	1.65
Shutdown time, hr	100
Inventory released from damaged rods (iodines and noble gases), %	10
Reactor water cover decontamination factors	
Iodine	100
Noble gases	1
Iodine fractions released from reactor water cover, %	
Elemental	75
Organic	25
Iodine removal efficiencies for containment effluent, %	
Elemental	No filters assumed
Organic	No filters assumed

*25% containment free volume mixing, 15,000 ft^3/min flow rate for 15 sec assumed.

Table 15.6 Assumptions used for estimating accidents involving small-line breaks outside the containment

Parameter and unit of measure	Quantity
Coolant released, lb	23,500
Fraction of coolant released flashed to steam, %	39
Coolant contaminant concentration, μCi/gm	1.0
Spiking factor (iodine release rate multiplier)	500

16 TECHNICAL SPECIFICATIONS

Later

17 QUALITY ASSURANCE

17.1 General

The description of the quality assurance (QA) program for the operations phase of the Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle), is contained in Section 17.2 of the Final Safety Analysis Report (FSAR). The staff evaluated this QA program through FSAR Amendment 9 on the basis of a review of this information and discussions with the applicant. The staff assessed the applicant's QA program for the operations phase to determine if it complies with the requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"; the applicable QA-related ~~R~~regulatory ~~G~~guides listed in Table 17.1; and the Standard Review Plan (SRP), Section 17.2, Revision 2, "Quality Assurance During the Operations Phase."

17.2 Organization

The structure of the organization reponsible for the operation of Vogtle and for the establishment and execution of the operations phase QA program is shown in Figure 17.1.

The Executive Vice President - Power Supply is responsible for establishing the policies, goals, and objectives of the applicant's QA program for Vogtle. This individual is the final management authority responsible for developing, implementing, changing, and reviewing the operations QA program.

The Senior Vice President - Nuclear Power and the Vice President and General Manager - Nuclear Operations are responsible to the Executive Vice President - Power Supply for implementing the QA program for plant operations at all nuclear power generating plants in the applicant's system. The General Manager - Vogtle Nuclear Operations reports to the Vice President and General Manager - Nuclear Operations through the Manager - Nuclear Operations.

The Manager - Nuclear Planning and Control reports to the Vice President and General Manager - Nuclear Operations. This position provides long-range planning and scheduling of maintenance work to be performed at Vogtle and provides the corresponding long-range manpower plan.

The Manager - Nuclear Training reports to the Vice President and General Manager - Nuclear Operations. This position provides training programs to ensure compliance with regulations and standards and to ensure that nuclear operations personnel have the education, training, and skills to safely operate and maintain Vogtle.

The Manager - Nuclear Engineering and Chief Nuclear Engineer is responsible to the Vice President and General Manager - Nuclear Operations for day-to-day monitoring of Vogtle activities, including monitoring those areas of nuclear fuel management, procurement, and processing where the three overlap.

The General Manager - Vogtle Nuclear Operations is responsible to the Vice President and General Manager - Nuclear Operations for all activities at Vogtle, including implementation of the operational QA program requirements (except for controls that are assigned to the QA department). The General Manager - Vogtle Nuclear Operations is also responsible for the safe operation of Vogtle.

The plant staff (as described in SER Chapter 13) performs safety-related activities in accordance with written, approved procedures. QA requirements are included in these procedures. The superintendents and supervisors in the Vogtle organization will be responsible for implementing the operational QA program for activities under their purview.

The Quality Control Supervisor, as a member of the plant staff, is responsible to the General Manager - Vogtle Nuclear Operations for administering and implementing an effective quality control (QC) inspection program at Vogtle. QC specialists report to the Quality Control Supervisor. The QC organization is involved in day-to-day safety-related meetings at Vogtle, such as meetings to plan work, and to review the operation, as well as routine meetings.

The Vice President and Chief Engineer - Power Supply Engineering and Services is responsible for providing engineering and technical support to the various power supply organizations. During Vogtle operation, this person is responsible for managing design engineering support, ascertaining that the suppliers are qualified, and reviewing procurement documents for quality requirements.

The General Manager of Quality Assurance and Radiological Health and Safety (GMQA), located at the general office, is responsible to the Executive Vice President - Power Supply for assuring implementation of the operational QA program and for managing activities of the applicant's QA organization. The GMQA has a staff at the general office and a staff located at the Vogtle site to conduct QA activities. The GMQA keeps the applicant's management personnel informed about the effectiveness and implementation of the operational QA program.

The Vogtle QA Manager reports to the GMQA and is responsible for ensuring that all participants implement the operational QA program for Vogtle. The Vogtle QA Manager ensures that Vogtle plant managers establish and maintain satisfactory QA programs and that activities at the site conform to those QA programs. This individual ensures that contractors (e.g., Southern Company Services, Bechtel Power Corporation) and suppliers of safety-related materials, equipment, and services establish and maintain satisfactory QA programs. The Vogtle QA Manager maintains and controls Vogtle's QA Manual.

Southern Company Services (SCS) is the architect/engineer for Vogtle during operation. In addition, SCS provides QA support; this support includes performing audits, and reviewing engineering procedures and the qualifications of suppliers. SCS also administers the Vogtle contract for engineering services provided by Bechtel. Activities within the SCS work scope are governed by SCS policy and procedures manuals which are reviewed and concurred with by the SCS QA organization. The GMQA performs or initiates audits of these functions.

Bechtel is under contract to SCS to provide architect/engineering services. The work scope includes plant design, development of purchase recommendations for equipment and materials, administration of purchase orders resulting from

SCS-developed purchase recommendations, and support of the SCS supplier surveillance functions by providing procurement surveillance services for selected safety-related items. Activities within the Bechtel work scope are governed by procedures manuals. The GMQA performs or initiates audits of these functions.

The GMQA and the Vogtle QC Supervisor and their staffs have the authority and organizational freedom to (1) identify quality problems; (2) initiate, recommend, or provide solutions to problems through designated channels and verify implementation of satisfactory solutions; and (3) to stop or control further processing, delivery, or installation of nonconforming material.

17.3 Quality Assurance Program

The QA program for the operation of Vogtle describes the QA policies, goals, objectives, and requirements to be implemented at the plant in order to ensure that safety-related activities are performed in a controlled manner and documented to provide objective evidence of compliance with NRC regulations and guidance. The QA program is implemented by the QA Manual that includes the QA policies, procedures, and instructions. This document presents the detailed techniques and methods by which the requirements of Appendix B to 10 CFR Part 50 and the provisions of the NRC regulatory guidance shown in Table 17.1 are satisfied. It is reviewed and concurred in by the Executive Vice President - Power Supply.

The QA program requires that QA documents encompass detailed controls for (1) translating codes, standards, and regulatory requirements into 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) purchasing items and services; (6) identifying materials, parts, and components; (7) performing special processes; (8) inspecting and/or testing material, equipment, processes, or services; (9) calibrating and maintaining measuring and test equipment; (10) handling, storing, and shipping items; (11) identifying the inspection, test, and operating status of items; (12) identifying and dispositioning nonconforming items; (13) correcting conditions adverse to quality; (14) preparing and maintaining QA records; and (15) auditing activities that affect quality.

The QA program requires the establishment and continuous implementation of the QA indoctrination, training, and retraining program to ensure that persons involved in safety-related activities are knowledgeable in QA instructions and implementing procedures and demonstrate a high level of competence and skill in the performance of their quality-related activities.

Quality is verified through surveillance, inspection, testing, checking, and auditing of work activities using procedures, instructions, and/or checklists. Inspections are performed by inspectors who have been qualified and certified in accordance with codes, standards, or company training programs.

The GMQA is responsible for establishing and implementing the audit program. Qualified personnel not having direct responsibility in the areas being audited use written procedures and checklists to perform the audits. The QA program establishes a comprehensive audit system to ensure that the QA program requirements and related supporting procedures are effective and properly implemented during operations. Audits include an objective evaluation of (1) QA practices, procedures, and instructions; (2) work areas, activities, processes, and items; (3) the effectiveness of implementation of the QA program; and (4) conformance with policy directives.

The QA program requires that managers who bear responsibility in the area audited document those results and review them in order to determine and take corrective action that is required. Reaudits determine if nonconformances have been effectively corrected and if the corrective action precludes reported nonconformances. The GMQA reviews audit findings, which indicate quality trends and the effectiveness of the QA program, and reports these findings to the Executive Vice President - Power Supply on a regular basis.

17.4 Conclusions

On the basis of review and evaluation of the QA program description contained in Section 17.2 of the FSAR for Vogtle, the staff concludes, subject to resolution of issues identified in Section 17.5 below, that:

(1) The applicant's QA organization provides sufficient independence from cost and schedule (when opposed to safety considerations), authority to effectively carry out the operations QA program, and access to management at a level necessary to perform its QA functions.

(2) The QA program, including the list of safety-related structures, systems, and components to which it applies, as indicated in FSAR Tables 3.2.2-1 and 17.3-1, describes requirements, procedures, and controls that, when properly implemented, comply with the requirements of Appendix B to 10 CFR Part 50 and with the acceptance criteria contained in SRP Section 17.2, Rev. 2.

Accordingly, the staff concludes, subject to resolution of the issues identified in Section 17.5 below which delineates open items, that the applicant's description of the QA program (QAP) is in compliance with applicable NRC regulations and acceptable for the operations phase of the Vogtle Electric Generating Plant.

17.5 Outstanding QA Issues Through FSAR Amendment 9

(1) The response to NRC Question 260.8 regarding commitments to Regulatory Guides concerning QA, provided in FSAR Amendment 9, says: "See revised Section 1.9." The staff reviewed the revised Section 1.9 and finds the following issues outstanding.

(a) The applicant's first Vogtle position regarding RG 1.33 states: "Conform for the QAP during design and construction. Refer to Sections 13.4, 13.5, and 17.2." Regulatory Guide 1.33 addresses operations and is not generally applicable during design and construction. This position should be deleted or clarified.

(b) Clarification 5 (page 1.9-29) and exception 2 (page 1.9-32) to RG 1.33 both reference paragraph 3.4.2 of ANSI 18.7. The clarification refers to the Technical Specifications and the exception refers to FSAR Chapter 13. In both cases the applicant should identify anything in paragraph 3.4.2 to which it is not committing itself.

(c) Clarification 16 (page 1.9-30) to RG 1.33 deletes words from the first sentence in the second paragraph of part 5.2.7 of ANSI 18.7. The clarification should not delete the words "which conform to applicable codes, standards, specifications, and criteria" that describe the written procedures.

(d) Clarification 24 (page 1.9-31) to RG 1.33 substitutes "an equivalent" for "the same" in paragraph 5.2.13.1 of ANSI 18.7. The applicant should identify who (i.e., by position title) determines equivalency.

(e) Clarification 24 (page 1.9-31) and exception 3 (page 1.9-33) to RG 1.33 are identical. They delete the guidance in paragraph 5.2.15 of ANSI 18.7 to review applicable procedures following an unusual incident unless it is an accident, unexpected transient, significant operator error, or equipment malfunction which results in a reportable event. The clarification should be eliminated and the exception should be deleted or justified.

(f) Clarification 28 (page 1.9-32) to RG 1.33 refers to clarification 18 (page 1.9-31). Reference should have been made to clarification 19. Also, the applicant should identify anything in part 5.2.19(3) of ANSI 18.7 to which it has not committed.

(g) Clarification 2 (page 1.9-37) to RG 1.38 states that "equipment used to measure secondary conditions, such as warehouse temperature, humidity, etc., will...not be maintained under the calibration and control program." Does humidity, etc. pertain only to the warehouse or to other parts of the plant? The applicant should justify this position.

(h) Clarification 4 (bottom of page 1.9-37) to RG 1.38 refers to "the clarifications of paragraph 3.2.1, D and E above...." This reference should be deleted or clarified since there appears to be no clarification of paragraph 3.2.1, D or E.

(i) Clarification 6 (page 1.9-38) to RG 1.38 refers to Regulatory Guide 1.37 (Section 1.9) for the applicant's position "regarding nonhalogenated materials in contact with austenitic stainless steel." This reference should be clarified or deleted as the applicant's position on RG 1.37 does not appear to address this subject.

(j) Clarification 11 (page 1.9-38) to RG 1.38 does not allow storage of hazardous material in close proximity to "installed systems required for safe shutdown" rather than in close proximity to "important nuclear plant areas." This apparent departure from ANSI N45.2.2 should be clarified, deleted, or justified.

(k) Clarification 13 (page 1.9-39) to RG 1.38 addresses rotating equipment. The applicant should identify what is being clarified.

(l) Clarification 14 (page 1.9-39) to RG 1.38 indicates that dynamic load tests for re-rating hoisting equipment will use the lift weight rather than 110% of the lift weight as stated in ANSI N45.2.2. This should be deleted or justified.

(m) Exception 1 (page 1.9-40) to RG 1.38 states that the applicant "may choose to protect weld-end preparation stored." Item 7 of part 3.2.1 of ANSI N45.2.2 states that "weld end preparations shall be protected...." This exception should be clarified and justified or deleted.

(n) The applicant's position (page 1.9-40) regarding RG 1.39 indicates that burning and cutting permits are substituted for fire watches. This appears to be a departure from subdivision 3.2.3 of ANSI N45.2.3. Regulatory Position 2 of RG 1.39 states that subdivision 3.2.3 of

ANSI N45.2.3 is not included as part of RG 1.39; therefore, this position should be deleted or justified. ~~The second part of this position should be deleted or justified.~~ The second part of this position states, "Refer to Chapter 17 for further discussion," and this statement should be deleted or made more specific so that its acceptability can be ascertained.

(o) Clarification 1 (page 1.9-51) of RG 1.58 refers to "other FSAR requirements." This reference should be deleted or made more specific so that its acceptability can be ascertained.

(p) Clarification 3 (page 1.9-51) of RG 1.58 exempts inspectors, examiners, and testing personnel (except NDE* personnel) from the 3-level classification portion of ANSI N45.2.6 with no justification or alternative. The applicant should provide one or the other, or delete the clarification.

(q) Clarification 4 (page 1.9-51) of RG 1.58 addresses qualification of inspectors. The applicant should commit to verify that the training and experience of inspection, examination, and testing personnel will ensure their ability to perform the assigned work.

(r) The applicant's position (page 1.9-57) on RG 1.64 states that the "operations QAP conforms with the intent" of the guide. The words "the intent" should be deleted or clarified.

(s) The applicant's position (pages 1.9-71 and 1.9-74) on RGs 1.88 and 1.94, respectively, should be clarified to show the applicant's commitment to comply with the guides during the operations phase. The commitments are not clear.

(t) Clarification 3 (page 1.9-92 of RG 1.123 refers to "an equivalent level" of review and approval of changes to procurement documents.

*nondestructive examination

The applicant should clarify what "equivalent level" means and who (by position title) determines equivalency.

(u) Clarification 7 (page 1.9-92) of RG 1.123 states that the applicant wishes to replace paragraphs 4.2(a), (b), and (c) of ANSI N45.2.13 with the paragraphs shown on pages 1.9-92, 1.9-93, and 1.9-94. The applicant should identify anything in paragraph 4.2 of the standard that it is not committing to. Replacement paragraph b(5) (page 1.9-93) states: "An inspection (of off-the-shelf items) shall be performed to assure that the correct item was received and no damage exists." The staff notes that a typical "identity and damage" inspection may be inadequate to ensure that an off-the-shelf item will perform its required safety function. This paragraph needs to be expanded to describe how the applicant ensures that off-the-shelf items will perform satisfactorily.

(v) Clarification 8 (page 1.9-94) of RG 1.123 is unclear because of the last word, "intent." This word should be deleted or explained.

(w) Clarification 9 (page 1.9-94) of RG 1.123 describes an exception to paragraph 5.3 of ANSI N45.2.13 regarding preaward evaluations. The applicant should describe controls that prevent the exception from becoming the rule.

(x) Clarification 10 (page 1.9-94) of RG 1.123 gives alternative methods to the requirements of paragraph "6.1" of ANSI N45.2.13 for monitoring and evaluating supplier performance (7.1 in the clarification should be 6.1). The alternatives do not appear to cover the areas of change control and the establishment of an exchange method of document information between supplier and purchaser These apparent omissions should be justified or eliminated.

(y) Clarification 16 (page 1.9-95) of RG 1.123 adds limitations to paragraph 8.2 of ANSI N45.2.13. The limitation concerning supplier submittal of nonconformance information only if the end use is affected presumes the supplier has knowledge of end-use information which he may not have. The definitions at the bottom of page 1.9-96 require inspection and/or test regarding end use, but it is not clear that the supplier will have the required information. The applicant should clarify how it ensures the supplier has such information.

(z) Clarification 17 (page 1.9-96) of RG 1.123 is not clear when it commits to conformance to paragraph 10.2 of ANSI N45.2.13 "to the extent that the certificate of compliance is traceable to the purchase order." Paragraph 10.2 addresses certificates of conformance which are not addressed elsewhere in the standard, and the second sentence in the clarification is also not clear. The applicant should explain the clarification.

(aa) The applicant's position relative to RG 1.144 (page 1.9-110) indicates conformance to draft 11, revision 0, of ANSI N45.2.9-1973. This conflicts with the commitment to Revision 2 of RG 1.88 (page 1.9-71) as noted in item S above. The applicant's position should show a commitment to ANSI N45.2.9-1974.

(bb) Table 17.2 shows where the applicant has committed in FSAR Section 1.9 to do something else rather than follow the specific NRC guidance (RG)/ANSI standard. In each case the applicant should identify anything in the specific NRC guidance/ANSI standard not being committed to that is not included in the commitment to something else.

(2) The response to NRC Questions 260.61 and 260.62 regarding the items controlled by the applicant's QA program was provided in FSAR Amendments 8 and 9, respectively. The staff review of the response has resulted in the following items which require a commitment from the applicant that the pertinent provisions of its operational QA program will be applied during the operations phase.

(a) Site drainage system alterations (Q260.61.A.3)

(b) Accident-related meteorological data collection equipment (Q260.61.A.7)

(c) Radiation protection systems (Q260.61.A.8)

(d) Instrumentation to satisfy the requirement of Item III.D.3.3 of NUREG-0737 (Q260.61.C.18)

(e) Containment building polar bridge crane components that perform a safety function (Q260.62.A.6)

(f) Reactor coolant pump seals (Q260.62.A.8)

(g) Shell side of letdown heat exchanger of the chemical and volume control system (CVCS) (Q260.62.B.1)

(h) Shell side of excess letdown heat exchanger of the CVCS (Q260.62.B.2)

(i) Shell side of seal water heat exchanger of CVCS (Q260.62.B.3)

(j) Auxiliary component cooling water surge tank (Q260.62.B.4)

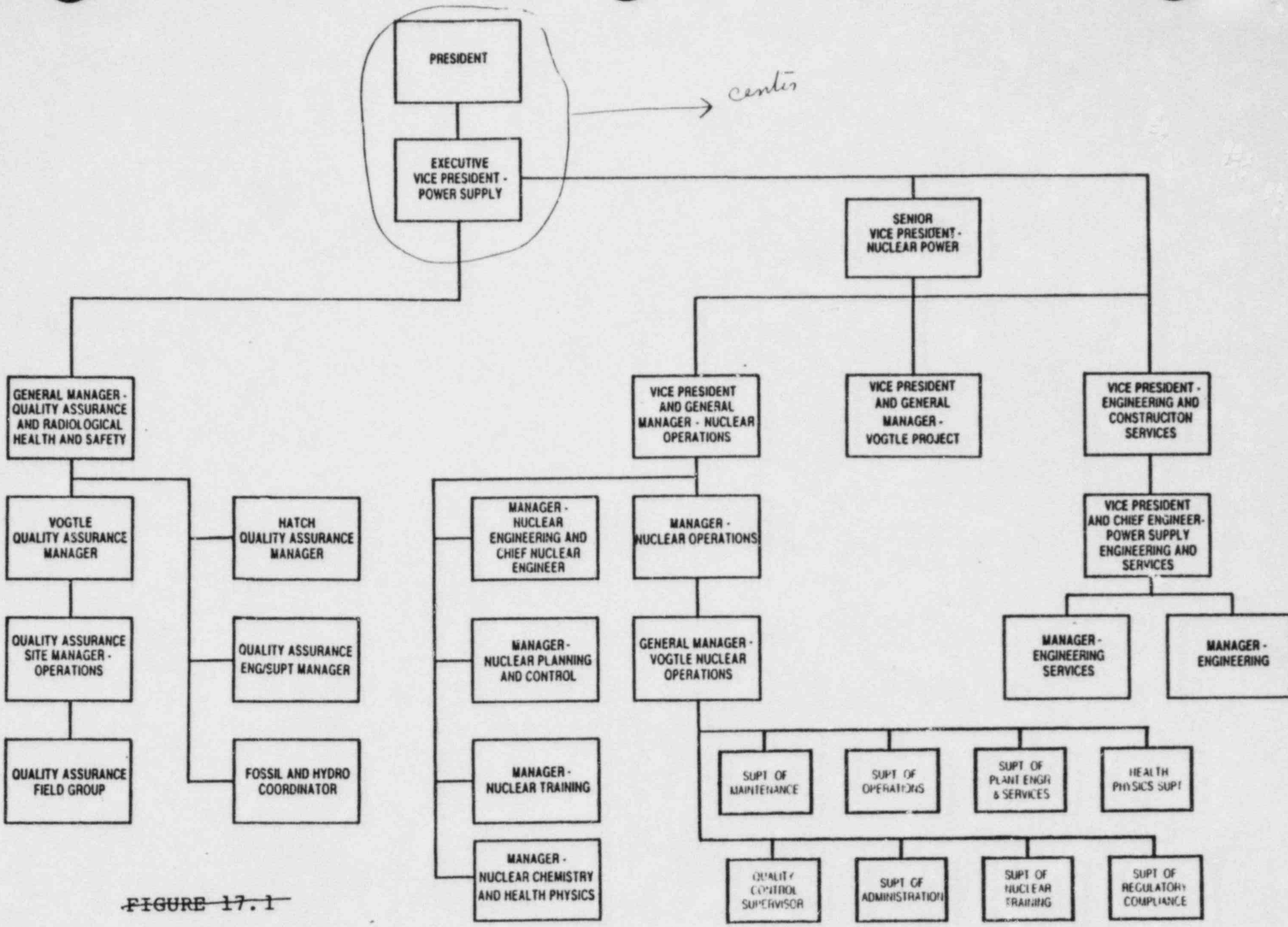
(k) Auxiliary component cooling water pumps (Q260.62.B.5)

(l) Refueling machine (Q260.62.B.6)

(m) Fuel transfer system (Q260.62.B7)

(n) Distribution panels of the 120-V ac power system, Class 1E (Added after transmittal of 260.62, this item was transmitted to the applicant by letter dated September 28, 1984.)

Figure 17.1 Quality assurance program during plant operations



~~FIGURE 17.1~~

~~ORGANIZATIONAL CHART FOR~~ a QUALITY ASSURANCE program DURING PLANT OPERATIONS

Figure 17.1

56

Table 17.1 Regulatory guides applicable to QA program

RG	Rev.	Title	Date
1.8	1-R	"Personnel Selection and Training"	5/77
1.26	3	"Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants"	2/76
1.29	3	"Seismic Design Classification"	9/78
1.30		"Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment"	8/72
1.33	2	"Quality Assurance Program Requirements (Operation)"	2/78
1.37		"Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"	3/73
1.38	2	"Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants"	5/77
1.39	2	"Housekeeping Requirements for Water-Cooled Nuclear Power Plants"	9/77
1.58	1	"Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel"	9/80
1.64	2	"Quality Assurance Requirements for the Design of Nuclear Power Plants"	6/76
1.74		"Quality Assurance Terms and Definitions"	2/74
1.88	2	"Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records"	10/76
1.94	1	"Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants"	4/76
1.116	0-R	"Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems"	6/76
1.123	1	"Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"	7/77
1.144	1	"Auditing of Quality Assurance Programs for Nuclear Power Plants"	9/80
1.146		"Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants"	8/80

Table 17.2 Applicant's alternate commitments

RG/ANSI Std.	FSAR pg/item	No commitment to paragraph	Commitment instead to
1.30/N45.2.4	1.9-26/1	2.2	Appropriate FSAR section
1.33/N18.7	1.9-29/6	4.1	Technical Specifications
1.33/N18.7	1.9-29/7	4.3.1	Technical Specifications
1.33/N18.7	1.9-29/8	4.3.4	Technical Specifications
1.33/N18.7	1.9-29/9	4.4	Technical Specifications
1.33/N18.7	1.9-29/10	4.5	Technical Specifications, FSAR Section 17.2, and RG 1.144
1.33/N18.7	1.9-29/12	5.2.2	Technical Specifications
1.33/N18.7	1.9-29/13	5.2.6	FSAR Section 17.2
1.33/N18.7	1.9-31/19	5.2.8	Technical Specifications
1.33/N18.7	1.9-32/27	5.2.18	FSAR Section 17.2 and RGs 1.37, 1.39, and 1.54
1.33/N18.7	1.9-32/29	5.2.19.1	FSAR Sections 17.2 and 14.2 and RGs 1.30, 1.58, 1.94, and 1.116
1.33/N18.7	1.9-32/30	5.38	Technical Specifications
1.33/N18.7	1.9-32/32	5.2.10	FSAR Section 17.2
1.37/N45.2.1	1.9-35/1	2.5	FSAR Section 17.2
1.37/N45.2.1	1.9-35/3	9	RG 1.88
1.38/N45.2.2	1.9-37/1	2.3	FSAR Section 17.2
1.116/N45.2.8	1.9-86/1	4.4	FSAR Section 17.2
1.123/N45.2.13	1.9-92/6	3.4	FSAR Section 17.2
1.123/N45.2.13	1.9-95/12	6.4	The Operations QAP
1.123/N45.2.13	1.9-95/13	7.4	FSAR Section 17.2
1.123/N45.2.13	1.9-95/14	7.5	RG 1.58
1.123/N45.2.13	1.9-96/18	12	RG 1.144

18 HUMAN FACTORS ENGINEERING

All licensees and applicants for an operating license are required to conduct a detailed control room design review (DCRDR) and to provide a safety parameter display system (SPDS) in response to NRC Task Action Plan Items I.D.1 and I.D.2 (NUREG-0660, May 1980; and NUREG-0737, November 1980, as supplemented by Generic Letter 82-33, December 17, 1982). The purpose of the DCRDR is to identify and correct human engineering discrepancies (HEDs) which might affect the operator's ability to prevent or cope with an accident. DCRDRs should be conducted using the guidance provided in NUREG-0700, "Guidelines for Control Room Design Reviews," dated September 1981, or equivalent guidance. The purpose of the SPDS is to continuously display information from which plant safety status can be readily and reliably assessed. The principal function of the SPDS is to aid control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal condition warrant corrective action by operators to avoid a degraded core. A written SPDS safety analysis shall be prepared describing the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events including symptoms of severe accidents.

In response to Supplement 1 to NUREG-0737, the applicant has negotiated with the staff the following schedules for meeting its DCRDR requirements. A DCRDR Program Plan was submitted, dated September 14, 1984, for Vogtle Units 1 and 2, and is currently under staff review. A Summary Report will be submitted in March 1986 for Unit 1 and September 1987 for Unit 2. After reviewing the applicant's Program Plan, the staff will conduct an in-progress audit of the DCRDR Program Plan. Then, on the basis of an assessment of the applicant's DCRDR Summary Report and the in-progress audit report, the staff will decide if a preimplementation audit is required. The applicant will be informed of the staff's decision, within two weeks after the staff receives the Summary Report. If the preimplementation audit is required, it will be done within one month after receiving the Summary Report. The staff will issue an SER two months after receiving the applicant's Summary Report.

The applicant has committed to submitting an SPDS Implementation Plan and safety analysis for both units in September 1985. The staff will review this plan to confirm: (1) the adequacy of the parameters selected to be displayed to detect critical safety functions; (2) that means are provided to ensure that the data displayed are valid; (3) the adequacy of the design and installation of the system from a human-factors perspective; (4) that the SPDS will be suitably isolated from electrical and electronic interference with equipment and sensors that are used in safety systems; and (5) the adequacy of the verification and validation (V&V) program to ensure a highly reliable SPDS. The projected dates for the SPDS becoming operational and for completion of training is September 1986 for Unit 1 and March 1988 for Unit 2.

19 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Vogtle application for an operating license will be reviewed by the Advisory Committee on Reactor Safeguards. The NRC staff will issue a supplement to this Safety Evaluation Report after the Committee's report to the Commission is available. The supplement will include a copy of the Committee's report, will address comments made by the Committee, and will describe steps taken by the NRC staff to resolve any issues raised as a result of the Committe's review.

20 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and that all of the applicant's directors and principal officers are citizens of the United States. Georgia Power Company, the applicant, is not owned, dominated, or controlld by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but, in accordance with the requirements of 10 CFR 50, the applicant has agreed to safeguard any such data that might become involved. The applicant will rely on obtaining fuel as it is needed from sources of supply available for civilian purposes so that no special nuclear material will be diverted from military purposes. For these reasons, and in the absence of any information to the contrary, the staff finds that the activities to be performed will not be inimical to the common defense and security.

21 FINANCIAL QUALIFICATIONS

On March 31, 1982, the NRC published in the Federal Register (47 FR 13750) amendments to its regulations that entirely eliminate the review related to the financial qualifications of electric utility applicants for construction permits and operating licenses.

On September 12, 1984, the NRC published in the Federal Register (49 FR 35747) a final rule amending its earlier ruling. In response to a remand by the U.S. Court of Appeals for the D.C. Circuit which declared invalid the NRC's March 31, 1982, rule eliminating financial qualification review and findings for electric utilities at all stages of the licensing proceeding, the NRC is amending its regulations to eliminate financial qualification review and findings for electric utilities that are applying for operating licenses for utilization facilities if the utility is a regulated public utility or is authorized to set its own rates. The NRC is reinstating a requirement for financial qualification review and findings for electric utilities that are applying for construction permits.

22 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

Later

23 CONCLUSIONS

On the basis of its evaluation of the application as set forth above, the staff has determined that, upon favorable resolution of the outstanding matters described herein, it will be able to conclude that

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

(2) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

(3) There is reasonable assurance (a) that the activities authorized by the operating licenses 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 I.

(4) The applicant is technically qualified to engage in the activities authorized by the licenses, in accordance with the regulations of the Commission set forth in 10 CFR Chapter I.

(5) 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 operating licenses are issued, the units must be completed in conformity with the construction permits, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power level must be verified by the NRC before the licenses are issued.

Furthermore, before operating licenses are issued, the applicant will be required to satisfy the applicable provisions of 10 CFR 140.

APPENDIX A

CHRONOLOGY OF NRC STAFF RADIOLOGICAL REVIEW OF VOGTLE UNITS 1 AND 2

Later.

APPENDIX B

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APPENDIX C

UNRESOLVED SAFETY ISSUES

Later.

APPENDIX D

ABBREVIATIONS

Later.

APPENDIX E

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