Safety Evaluation Report

NUREG-0101

U. S. Nuclear Regulatory Commission

related to construction of

Phipps Bend Nuclear Plant Units 1 & 2

Office of Nuclear Reactor Regulation

Docket Nos. STN 50-553
STN 50-554

Tennessee Valley Authority

April 1977

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NUREG-0101
APRIL 1977

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION
U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

TENNESSEE VALLEY AUTHORITY
PHIPPS BEND

NUCLEAR PLANT UNITS
No. 1 and No. 2

DOCKET NOS.: STN 50-553 STN 50-554

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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Tennessee Valley Authority (or applicant) filed with the Nuclear Regulatory Commission (or Commission) an application docketed on November 7, 1975 for licenses to construct and operate its proposed Phipps Bend Nuclear Plant (Phipps Bend plant) consisting of two units designated Units No. 1 and No. 2. The Phipps Bend Nuclear Plant proposed site is located in Hawkins County in eastern Tennessee on the Holston River approximately 15 miles southwest of Kingsport, Tennessee. The Phipps Bend plant will utilize the General Electric Company standard nuclear island design. A separate Environmental Report was submitted for the Phipps Bend plant in accordance with the Commission's regulations, 10 CFR Part 51, which implement the National Environmental Policy Act of 1969 (NEPA).

A Preliminary Safety Analysis Report was submitted with the Tennessee Valley Authority application. This report describes the site and the design of the balance-of-plant structures, systems, and components and incorporates by reference the General Electric Company "Standard Safety Analysis Report" (GESSAR-238, Nuclear Island Docket No. STN 50-447, hereinafter referred to as GESSAR). GESSAR describes a standard nuclear island design which incorporates a Mark III Containment and a BWR-6 class boiling water reactor.

The Commission issued WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants" on August 20, 1974. Amendment 1 to WASH-1341, dealing with "options" and "overlaps," was issued January 16, 1975. The regulations governing the submittal and review of standard designs under the "reference system" option are found in Appendix O to Part 50, "Licensing of Production and Utilization Facilities," and Section 2.110 of Part 2, "Rules of Practice," in Title 10 of the Code of Federal Regulations (CFR).

GESSAR was submitted by the General Electric Company in the form of an application for a Preliminary Design Approval from the Commission and was in response to Option 1 of the Nuclear Regulatory Commission's standardization policy. Option 1 allows for the review of a "reference system" that involves an entire facility design or major fraction of a design outside the context of a license application. On July 30, 1973, the application for GESSAR was docketed.

A discussion of our evaluation of GESSAR is presented in our safety evaluation report, dated December 1975, and Supplement No. 1, and Supplement No. 2 thereto dated September 1976 and February 1977 respectively. Copies of the safety evaluation report and its supplements are attached as Appendices A, C and D to this report.

Where we have made use of these evaluations, we have referenced the appropriate sections of the GESSAR safety evaluation report and its supplements in this report.

During our review of the Phipps Bend application, numerous meetings were held with representatives of the applicant, its contractors and its consultants to discuss the design of the facility and the technical material submitted. A chronological listing of the meetings and other significant events in our review of the application is given in Appendix G to this report. We requested that the applicant provide additional information needed for our evaluation and to modify the application to meet certain regulatory requirements. The additional information and modifications to the application were provided in Amendments 1 through 13 to the Preliminary Safety Analysis Report. The Preliminary Safety Analysis Report and copies of these amendments are available for public examination at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555, and at the Kingsport Public Library, Broad and New Streets, Kingsport, Tennessee 37660.

This safety evaluation report summarizes the results of the technical evaluation of the proposed Phipps Bend Nuclear Plant performed by the Commission's staff and delineates the scope of the technical matters considered in evaluating the radiological safety aspects of the facility. Aspects of the environmental impact considered in the review of the facility in accordance with 10 CFR Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection" of the Commission's regulations, which implements the National Environmental Policy Act of 1969, are discussed in the Commission's Final Environmental Statement which was issued in February 1977.

Upon favorable resolution of the outstanding issues discussed herein, and summarized in Section 1.9 of this report, we will be able to conclude that the Phipps Bend Nuclear Plant can be constructed and operated as proposed without endangering the health and safety of the public. Our detailed conclusions are presented in Section 21.0 of this report.

The review and evaluation of the proposed design of the facility reported herein is only the first stage of a continuing review by the Commission's staff of the design, construction and operating features of the Phipps Bend Nuclear Plant. Construction will be accomplished under the surveillance of the Commission's staff. Prior to a decision on the issuance of operating licenses, we will review the final design to determine whether all of the Commission's safety requirements have been met. The facility may then be operated only in accordance with the terms of the operating license, the Commission's regulations, and under the continued surveillance of the Commission's staff.

1.2 General Plant Description

The proposed nuclear island design, as described in the GESSAR Standard Safety Analysis Report, incorporates a single-cycle, forced circulation BWR-6 class boiling water reactor and a Mark III type of vapor suppression containment system, both of which are based on designs with certain new features introduced by the General Electric Company in 1972. The nuclear island scope of design, graphically shown in Figure 1.1 of this report, includes the nuclear steam supply system, the engineered safety features systems, the reactor and auxiliary buildings, the control building, the radwaste building, the fuel building, the diesel-generator buildings, the offgas treatment system (housed in the turbine building), the onsite electrical power system and related systems and structures.

The principal Phipps Bend plant balance-of-plant features that either contain safety-related systems or interface with GESSAR nuclear island safety systems will include the turbine building structure and power conversion system, the instrumentation and control for the turbine building systems, the portions of the reactor protection system from the turbine stop and control valves, the portions of the engineered safety features from the condenser, the turbine steam and power conversion system portion of the Phipps Bend plant balance-of-plant on the turbine side of the main steam and feedwater shut-off valves, the fire protection system for the turbine building and site, including a central carbon dioxide supply system, the essential service water system, including spray ponds as a source of cooling water for normal and emergency plant shutdown of safety systems, the diesel generator fuel oil storage and transfer systems, the condensate storage and transfer system, and the offsite alternating current power system.

1.2.1 Reactor

The reactor core for each of the two BWR-6 class boiling water reactors will contain 732 fuel assemblies. Fuel will consist of slightly enriched uranium dioxide in the form of sintered ceramic pellets. Some of the fuel will contain gadolinium oxide in a mixture with uranium dioxide, also in the form of sintered ceramic pellets. The gadolinium will serve as a "burnable poison" designed to flatten the power distribution and limit the core reactivity variation throughout the core lifetime. The fuel pellets will be enclosed in Zircaloy-2 tubes (cladding) which will be evacuated, backfilled with helium, and sealed by welded Zircaloy-2 end plugs at each end. A Zircaloy-4 fuel channel will enclose a bundle of 63 fuel rods in an 8 x 8 array (one fuel rod position will contain a water filled rod).

The normal reactivity control or rapid shutdown of the reactor will be achieved with bottom-entry cruciform-shaped control rods that will be moved vertically in the spaces between fuel assemblies by a hydraulic mechanism with water as the working fluid. For rapid insertion, nitrogen under pressure in an accumulator will provide the driving force. Each control rod will be independent of the other control rods and will have its own hydraulic control system. A standby liquid control system will also be available for injecting a boron solution into the reactor for emergency long-term reactivity control.

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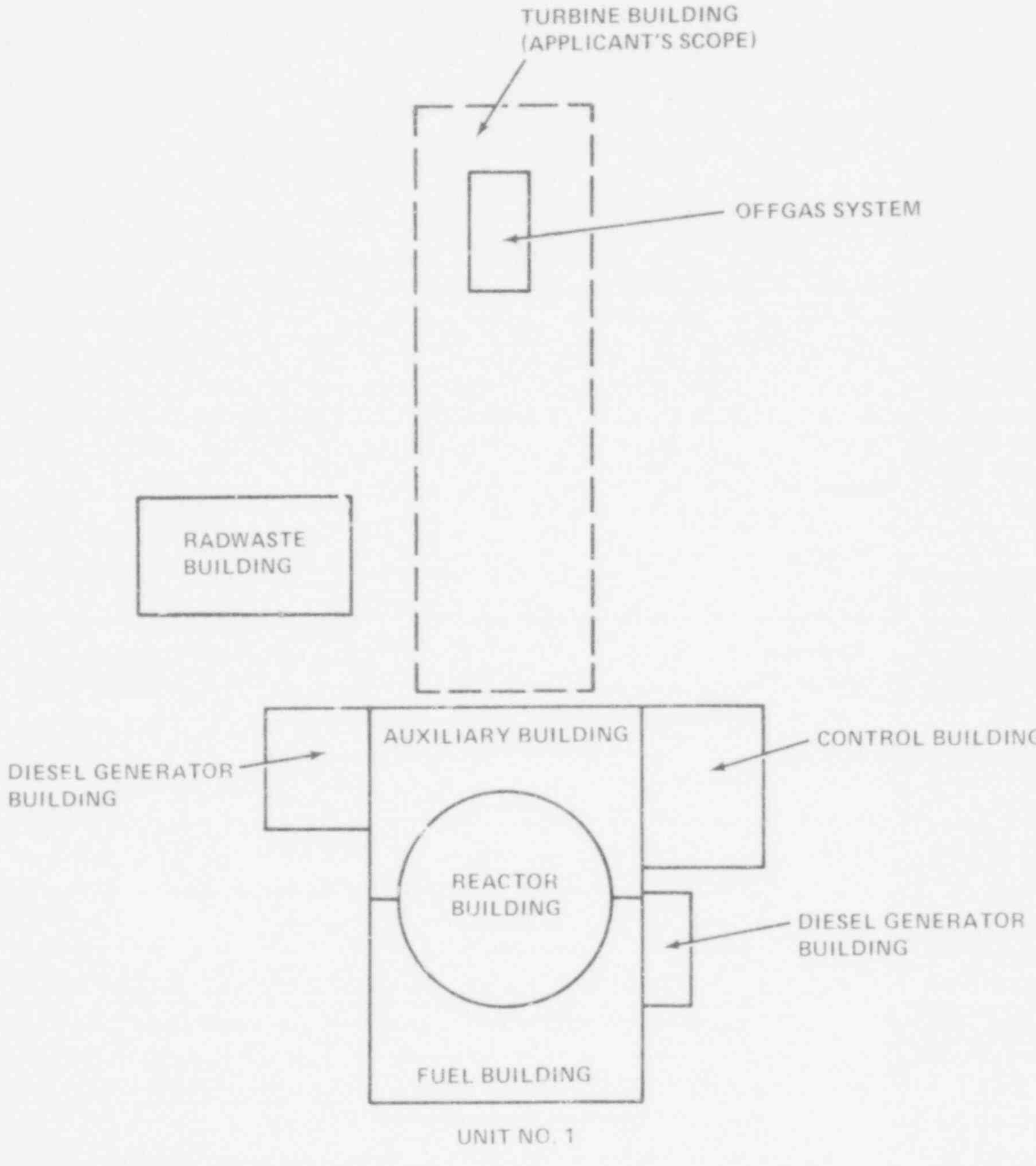

FIGURE 1.1 NUCLEAR ISLAND

1.2.2 Reactor Coolant System

The reactor coolant pressure boundary will include the reactor pressure vessel, the recirculation lines, the main steamlines, the feedwater lines, and branch lines to their outermost containment isolation valves. Water flowing through the core will serve as both moderator and coolant. Movement of water through the core will be accomplished by the driving force from two recirculation pumps supplying water to 20 jet pumps (10 per recirculating pump and associated piping), and by convective forces. Steam from the boiling process in the reactor core will be separated from the recirculating water and dried in the upper region of the vessel, then directed through the four steamlines to the turbine generator system where its energy will be converted into electricity. The steam will then be exhausted to a condenser located beneath the turbine where the condensate will be collected and returned through a cleanup system for recycling through the reactor.

1.2.3 Engineered Safety Features

Engineered safety features will provide the capability to contain fission products assumed to be released during a postulated design basis accident so that radioactive releases will be restricted to acceptable levels, provide for heat removal for emergency short and long-term cooling, and condense steam within the primary containment.

Systems and components designated as engineered safety features will be designed to be capable of performing their function of assuring safe shutdown of the reactor under the adverse conditions of the various postulated design basis accidents described in Section 15 of Appendix A to this report. The Phipps Bend plant's postulated design basis accident information is presented in Section 15 of this report. The engineered safety features will be designed to seismic Category I standards and they will function even with a complete loss of offsite power. Components and systems will be provided with sufficient redundancy so that a single failure of any component or system, will not result in the loss of the plant's capability to achieve and maintain a safe shutdown of the reactor. The instrumentation systems and emergency power systems will be designed to the same seismic, redundancy, and quality requirements as the systems they serve.

1.2.4 Protection Systems

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

The reactor protection system will provide the means to protect against conditions that may cause fuel damage or a loss of integrity of the reactor coolant pressure

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boundary. The reactor protection system will initiate a reactor scram following an abnormal operational transient, pressure pulse, predetermined fuel damage or significant loss of integrity of the reactor coolant pressure boundary.

The design and control for the engineered safety feature systems will consist of adequate instrumentation and controls to sense accident situations and initiate operation of the necessary engineered safety features.

1.2.5 Other Nuclear Island Structures and Systems

The auxiliary building will be located adjacent to the reactor building (Figure 1.1) where the four main steam lines leave the containment structure. The auxiliary building will house the engineered safety features, auxiliary systems and the two trains of electrical equipment which serve these engineered safety features.

The fuel building will be located adjacent to the reactor building diametrically opposite the auxiliary building location and will house the new and spent fuel storage and shipping areas and the standby gas treatment system.

Diesel generator buildings will be located adjacent to the auxiliary and fuel buildings and will house the three diesel generators serving the three electrical load trains.

The control building will be located adjacent to the auxiliary building and will contain the control room and the computer facility. The radwaste building will house the liquid and solid radioactive waste treatment facilities. The gaseous radioactive waste treatment facilities will be located within the turbine building of each unit as shown in Figure 1.1.

1.2.6 Power Sources and Heat Rejection

The Phipps Bend plant will be provided with physically independent offsite electric power sources at the 500 kilovolt and 161 kilovolt levels to supply power for normal startup and shutdown or to operate the engineered safety features in the event of an accident. The normal offsite power source is the 500 kilovolt transmission system which supplies Unit 1 via the 500 kilovolt switchyard and Unit 2 via the 161 kilovolt switchyard. Failure of the normal offsite power source will bring about automatic transfer of the safety and non safety-related buses of both units to an independent reserve 161 kilovolt power source.

Heat rejection from the main condenser and turbine building auxiliary cooling equipment will be provided by a closed circulating water system incorporating one natural-draft cooling tower for each unit and utilizing makeup water from the Holston River. The source of cooling water for normal and emergency shutdown of the two units will be spray ponds.

1.3 Shared Systems

The Phipps Bend Nuclear Plant will be designed such that the sharing of safety-related systems by the two units will be minimized to include only the fire protection system, the liquid and solid radioactive waste treatment system, the offsite electrical power and the two adjacent spray ponds which will provide cooling water for essential equipment during normal and abnormal conditions. Our review has considered sharing of systems and is discussed in appropriate sections of this report.

1.4 Comparison with Similar Facility Designs

Certain features of the GESSAR design are variations of previous General Electric Company designs; however, many aspects of the nuclear island design are very similar to plant designs we have evaluated and previously approved. To the extent feasible and appropriate we have made use of our previous evaluations during our review of these features that are similar to GESSAR.

To assist in better understanding the relationship of the GESSAR (BWR-6/Mark III) design to other BWR designs, a comparative listing of principal parameters and features for the Phipps Bend, Grand Gulf (Docket Nos. 50-416/417), Perry (Docket Nos. 50-441/442), and Riverbend (Docket Nos. 50-458/459) plants are presented in Table 1.1 of this report. In addition the design of the Phipps Bend plant is identical to the Hartsville Nuclear plants design (Docket Nos. STN 50-518/519/520/521). Our safety evaluation reports for the above discussed applications are available for public inspection in the Nuclear Regulatory Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555.

1.5 Identification of Agents and Contractors

The Tennessee Valley Authority's Division of Engineering Design has the responsibility for ensuring that all aspects of the site are compatible with the site envelope specified in GESSAR and that the design of the balance-of-plant is compatible with the site and with GESSAR. The Tennessee Valley Authority's Division of Construction has the responsibility for the construction of the entire facility. The Tennessee Valley Authority's Division of Power Production has the responsibility for operating the facility.

The applicant has contracted with the General Electric Company to supply the nuclear steam supply system and the design of the nuclear island. The General Electric Company has subcontracted with C. F. Braun and Co. to provide the architect-engineering design for the nuclear island buildings. The Brown-Boveri Company will supply the two turbine generators.

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TABLE 1.1

PRINCIPAL PARAMETERS AND FEATURES, PHIPPS BEND AND COMPARABLE PLANTS

Parameter or Feature	Phipps Bend	River Bend	Perry	Grand Gulf
Rated Thermal Power, (megawatts thermal)	3,579	2,894	3,579	3,833
Design Thermal Power, (megawatts thermal)	3,758	3,039	3,758	4,025
Net Electrical Output, (megawatts electrical)	1,233	934	1,205	1,290
Steam Flow Rate, (pounds per hour)	15,396,000	12,431,000	15,396,000	16,488,000
Core Coolant Flow Rate, (pounds per hour)	105,000,000	84,500,000	105,000,000	113,500,000
Feedwater Temperature, (degrees Fahrenheit)	420	420	420	420
Normal Steam Pressure, (pounds per square inch absolute)	1,040	1,040	1,040	1,040
Separator Design Inlet Quality, (percent)	14.88	14.95	14.88	14.74
Number of Fuel Assemblies	732	592	732	784
Number of Movable Control Rods	177	145	177	193
Reactor Vessel Design Pressure (pounds per square inch gauge)	1,250	1,250	1,250	1,250
Reactor Vessel Design Temperature (degrees Fahrenheit)	575	575	575	575
Reactor Vessel Inside Diameter, (inches)	238	218	238	251
Reactor Vessel Inside Height (inches)	850	838	850	876
Number of Recirculation Loops	2	2	2	2
Circulation Pump Flow Rate (gallons per minute)	35,400	32,300	35,400	44,900
Recirculation Loop Inside Diameter, (inches)	22/24	20	22/24	24
Number of Jet Pumps	20	20	20	20
Number of Steamlines	4	4	4	4
Steamline Inside Diameter, (inches)	26	24	26	28
Number of Core Spray Spargers	2	2	2	2

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TABLE 1.1 (Cont'd)

PRINCIPAL PARAMETERS AND FEATURES, PHIPPS BEND AND COMPARABLE PLANTS

Parameter or Feature	Phipps Bend	River Bend	Perry	Grand Gulf
High Pressure Core Spray System, (gallons per minute @ pounds per square inch differential)	1465 @ 1130 6000 @ 200	1325 @ 1130 5010 @ 200	1456 @ 1130 6000 @ 200	1550 @ 1130 7000 @ 200
Pump Motive Type	Motor (Separate diesel generator)	Motor (Separate diesel generator)	Motor (Separate diesel generator)	Motor (Separate diesel generator)
Low Pressure Core Spray System (gallons per minute @ pounds per square inch differential)	6000 @ 122	5010 @ 119	6000 @ 122	7000 @ 122
Low Pressure Coolant Injection System				
Number of Pumps	3	3	3	3
Flow Rate, (gallons per minute @ pounds per square inch differential)	7100 @ 20	5050 @ 20	7100 @ 20	7450 @ 20
Number of Automatic Depressurization Systems	1	1	1	1
Residual Heat Removal System				
Number of Pumps	3	3	3	3
Flow Rate per Pump, (gallons per minute)	7,100	5,050	7,100	[illegible]
Heat Duty (British thermal units per heat exchanger)	45,000,000	35,000,000	45,000,000	50,000,000
Average Power Density, (kilowatts per liter)	56.0	56.0	56.0	56.0
Maximum Design Linear Power (kilowatts per foot)	13.4	13.4	13.4	13.4
Maximum Heat Flux (British thermal units per hour-square foot)	354,000	354,000	354,000	354,000
Maximum Uranium Dioxide Temperature (degrees Fahrenheit)	3,325	3,325	3,325	3,325
Minimum Critical Heat Flux Ratio	$\geq$1.9	$\geq$1.9	$\geq$1.9	$\geq$1.9
Total Peaking Factor	2.22	2.22	2.22	2.22
Fuel Rod Array	8 x 8 (63 fuel rods)	8 x 8 (63 fuel rods)	8 x 8 (63 fuel rods)	8 x 8 (63 fuel rods)
Fuel Rod Diameter (inches)	0.493	0.493	0.493	0.493

1.6 Summary of Principal Review Matters

Our technical review and evaluation of the information submitted by the Tennessee Valley Authority considered the principal matters summarized below:

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

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

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

(4) We evaluated the applicant's engineering and construction 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 planning for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that the applicant will be technically qualified to safely operate the facilities.

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

(6) We evaluated the applicant's quality assurance program for the design and construction of the facilities to assure that the program complies with the intent of the Commission's regulations (10 CFR Part 50) and that the applicant will have proper control over the facility design and construction such that there will be a high degree of assurance that, when completed, the facilities can be operated safely and reliably.

(7) We are evaluating the financial data and information supplied by the applicant as required by the Commission's regulations (Section 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50) to determine that the applicant is financially qualified to design and construct the proposed facilities. We will report the results of our evaluation in a supplement to this report.

1.7 GESSAR Related Matters

1.7.1 GESSAR Preliminary Design Approval

The Preliminary Design Approval for GESSAR was issued to the General Electric Company on December 22, 1975. A copy of the Preliminary Design Approval (PDA-1) and the letter transmitting it to the General Electric Company are attached as Appendix B to this report. The Preliminary Design Approval was subject to four conditions which relate to those features of the GESSAR nuclear island standard design for which we identified requirements that differed from those in GESSAR. In addition, we identified 19 items (Post Preliminary Design Approval items) which require resolution. The resolution of these matters is discussed in the following paragraphs.

1.7.1.1 Preliminary Design Approval (PDA-1) Conditions

Subsequent to the issuance of the Preliminary Design Approval the General Electric Company provided us with acceptable commitments or design alternatives to resolve the four conditions of the Preliminary Design Approval, with the exception of the portion of Condition No. 2 relating to the impact loads in the region 17-19.5 feet above the suppression pool. The applicant by referencing GESSAR through Amendment No. 45 adopted the GESSAR resolutions. In addition the applicant committed to accept our position on the impact loads in the region 17-19.5 feet above the suppression pool for the Phipps Bend plant. We therefore consider these matters resolved. The four conditions are listed below with reference in parentheses to sections in this report and its appendices where our evaluations are discussed.

(1) Tornado missile velocities acceptable to the staff must be adopted (Section 3.5.2 of this report and Section 3.5.3 of Appendix C to this report).

(2) The staff's criteria must be used for establishing dynamic loads on structures located in and above the suppression pool related to (a) dynamic loads generated during the clearing of the safety/relief valve discharge lines, and (b) impact loads on pipes at elevations between 17 and 19.5 feet above the suppression pool (Section 6.2.2 of this report).

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(3) The continuous containment purge system must be designed to meet the design criteria set forth in Branch Technical Position CBS 6-4, "Containment Purging During Normal Operations," and must exhaust through a charcoal filtration system to meet the design dose objectives of Appendix I to 10 CFR Part 50 (Sections 6.2.4 and 11.3.3 of Appendix C to this report).

(4) Interlocks must be provided to preclude operation of any individual main steam line isolation valve leakage collection system if the associated inboard main steam isolation valve in the steam line connected to the leakage collection system is not fully closed (Section 9.3.2 of this report). In addition, the setpoint for the flow element timer in the inboard leakage collection system must be set at 11.5 cubic feet per hour (Section 9.3.2 of this report).

1.7.1.2 GESSAR Post-Preliminary Design Approval Items

Table 1-3 of Appendix A to this report identifies nineteen post-Preliminary Design Approval items. Since the Preliminary Design Approval, the General Electric Company has supplied additional information on these matters. Based on our review of the additional information, we conclude that the General Electric Company has supplied sufficient information on all the post-Preliminary Design Approval items to provide a suitable basis for a decision on issuance of a construction permit to a referencing plant (Section 1.8.2 of Appendix C and D to this report).

In addition, after the Preliminary Design Approval was issued, four new issues were identified, for which the General Electric Company provided acceptable information. These matters are as follows: (1) cracks in the reactor vessel feedwater nozzle blend radii and the nozzle bore surface (Section 5.2.1 of Appendix D to this report), (2) the consideration of asymmetric loadings on the reactor vessel and internals due to postulated pipe ruptures (Section 5.2.1 of Appendix D to this report), (3) cracking in austenitic stainless steel piping (Section 5.2.5 of Appendix D to this report), and (4) the evaluation of a postulated fuel handling accident inside containment (Section 15.3 of Appendix D to this report).

The Tennessee Valley Authority has incorporated the GESSAR resolutions by reference into the Phipps Bend docket. These items are therefore considered resolved for the Phipps Bend application.

1.7.2 Exceptions to GESSAR Proposed by the Applicant

We reviewed the following exceptions to the GESSAR nuclear island design proposed by the Tennessee Valley Authority and concluded that they are acceptable alternatives to the GESSAR design. Our evaluation of these exceptions are presented in detail in the sections of this report indicated in parentheses.

(1) The applicant's specifications for concrete will use the Tennessee Valley Authority specification document identified as "G-2" instead of American National Standards Institute N45.2.5 and American Society of Mechanical Engineers/American Concrete Institute 359 referenced in GESSAR. (Section 3.8.3)

(2) The Phipps Bend radwaste system provides for sharing the liquid and solid radwaste systems between the two units. (Sections 11.2.1 and 11.2.3)

(3) The applicant increased the size of the low conductivity waste tanks and waste filters to accommodate sharing of liquid radwaste system between the two units. (Section 11.2.1)

(4) The applicant added a reserve third charcoal delay train in the gaseous radioactive waste treatment system. (Section 11.2.2)

(5) The applicant provided additional capacity for storing packaged solid radwaste to accommodate sharing of the solid waste processing system between the two units. (Section 11.2.3)

(6) Primary containment leakage tests will be performed during construction in addition to, and not concurrent with, tests required by GESSAR; acceptance criteria for post-construction leakage tests of secondary containment are more stringent than GESSAR requirements. (Section 6.2.5)

(7) The applicant will provide seismic Category I steel lined concrete diesel generator fuel storage tanks rather than the free standing steel tanks described in GESSAR. (Section 9.4.2)

(8) The applicant increased the total containment leak rate (technical specification leak rate) to 0.5 percent per 24 hours from 0.3 percent per 24 hours as described in GESSAR. (Section 6.2.1)

1.7.3 Interfaces with the GESSAR Nuclear Island Design

We reviewed the safety-related interfaces between the GESSAR nuclear island design and the Phipps Bend plant balance-of-plant, and between the GESSAR nuclear island design and the site. Safety-related interfaces are defined as those which are required to mitigate the consequences of postulated design basis accidents, or provide a safe and orderly shutdown, or to monitor process streams and effluent paths for radioactive substances.

The applicant has identified in Tables 1.10-1(T) and 1.10-2(T) of the Preliminary Safety Analysis Report all of the interfaces described in GESSAR and has stated that the balance-of-plant and the site will meet all of the interface requirements presented in GESSAR.

We conclude that the interface information provided in the Phipps Bend Preliminary Safety Analysis Report is acceptable for the construction permit stage of the review.

1.8 Requirements for Future Information

The applicants have identified in Section 1.5 of the Phipps Bend Preliminary Safety Analysis Report, by referencing Section 1.5 of the GESSAR Standard Safety Analysis Report, certain development programs applicable to the Phipps Bend plant. These programs, that are aimed at verifying the nuclear steam supply system design and confirming the design margins, are all being conducted by the General Electric Company. The objectives, schedules for completion, and current results are summarized in GESSAR Standard Safety Analysis Report. Our evaluation of this information is presented in Section 1.8.1 of Appendix A to this report.

In summary, the verification programs have been reviewed and we have concluded that (1) the test programs outlined in GESSAR Standard Safety Analysis Report, if carried out as stated, will provide in a timely manner the necessary information to verify the design and safe operation of GESSAR nuclear island systems, and (2) in the event any of the programs provide unexpected results, appropriate restrictions on operation can be used and/or modification in designs can be made to protect the health and safety of the public.

1.9 Outstanding Issues

We have identified four outstanding issues from our review of the Phipps Bend plant for which additional information is required from the applicant. We require resolution of all these items prior to a decision on issuance of construction permits. The status of these matters will be reported in a supplement to this report.

(1) Emergency core cooling system for GESSAR (Section 6.3)

(2) Water level (flood) design criteria (Section 3.4)

(3) Foundation engineering aspects of the spray ponds (Section 2.5.5)

(4) Evaluation of the applicant's financial qualification (Section 20.0)

1.10 Generic Issues

The Advisory Committee on Reactor Safeguards (Committee) periodically issues a report listing various generic matters applicable to lightwater reactors. Our discussion of these matters is provided in Appendix E to this report which references sections of this report where more specific discussions of the status of generic items concerning the proposed facility is given.

In addition to the generic matters identified by the committees that are listed in Appendix E, we are conducting a generic review of anticipated transients without scram. The results of this study may impact the design of the proposed Phipps Bend plant (Section 15.4 of this report and Section 15.4 in Appendices A and C to this report.)

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Description and Exclusion Area Control

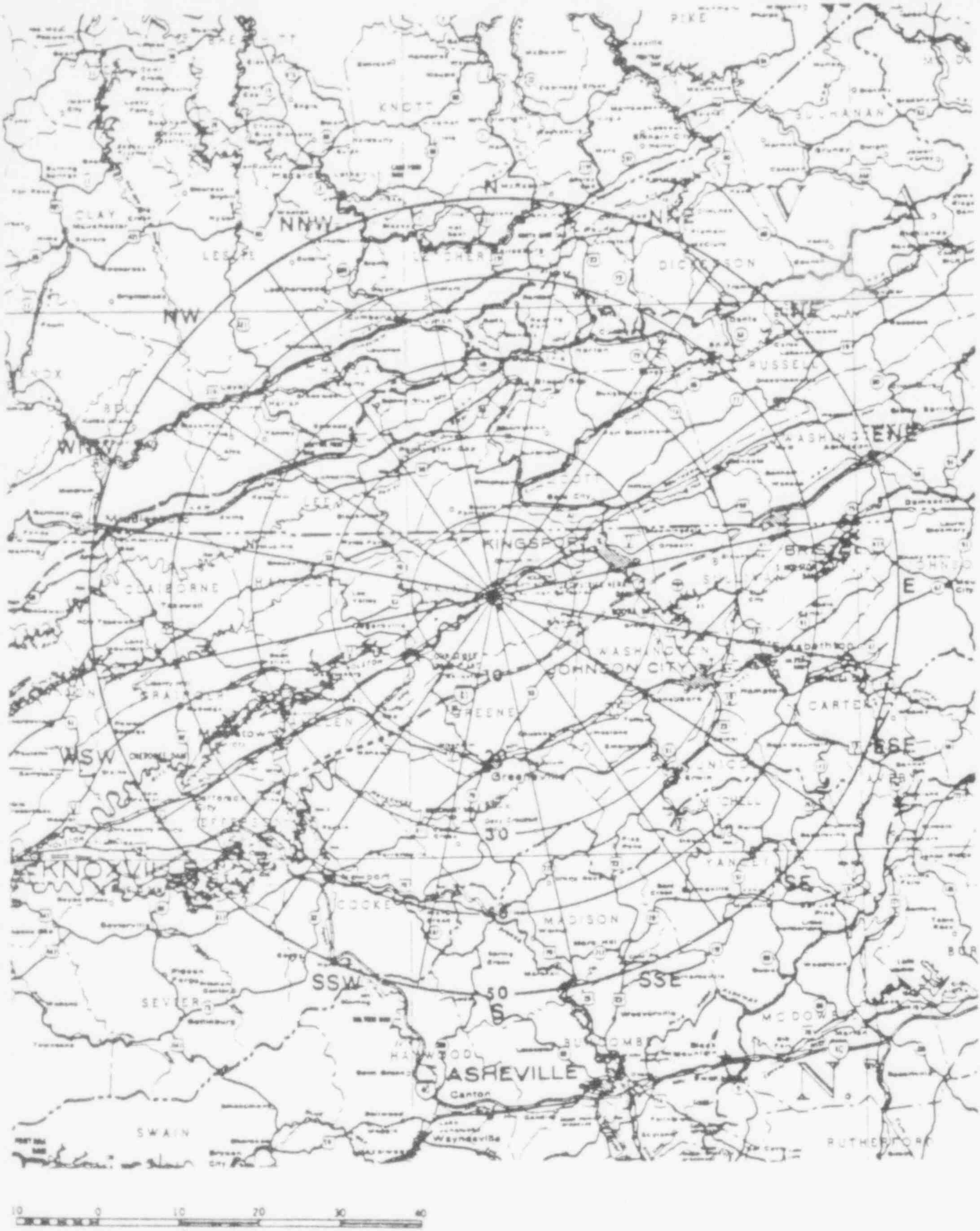
The proposed Phipps Bend site is located in Hawkins County in eastern Tennessee on the Holston River approximately 15 miles southwest of Kingsport, Tennessee. The site is shown on a regional map of the area in Figure 2.1.

The proposed Phipps Bend site consists of approximately 1,270 land acres which lie inside of a bend in the Holston River at river mile 121. The site is bounded on the northeastern through western quandrants by the river. The site topography is characterized by flood plains along the eastern side of the site which rise into a series of ridges on the western side. The exclusion area consists of the site property plus the Holston River where it borders the site property. A map of the site showing the exclusion area boundary lines and the location of the principal plant structures is shown in Figure 2.2. The minimum exclusion area boundary distance is approximately 760 meters (2,490 feet) measured from the center of the Unit No. 1 containment building to the nearest point on the exclusion area boundary.

The exclusion area property is owned by the United States and is in the custody of the Tennessee Valley Authority. No public highways or railroads traverse the exclusion area. Control of the movement of fishermen and boaters on the portion of the Holston River which is within the exclusion area will be initiated by the plant security force in the event of a plant emergency. The applicant plans to arrange with the appropriate state agency for the implementation of additional control procedures on the river as part of the radiological emergency plan. Based on the applicant's custody of the exclusion area property and the commitment to make arrangements with the appropriate state agency for control of the Holston River through the exclusion area in the event of an emergency, we conclude that there is reasonable assurance that the applicant will have the proper authority to determine all activities within the exclusion area, as required by 10 CFR Part 100.

2.1.2 Population Distribution

The region surrounding the proposed Phipps Bend site is not heavily populated. Approximately 18,000 persons resided within 10 miles of the site in 1970. The closest community to the site is Surgoinsville which is located approximately 1.5 miles northwest and had a population of 1,285 in 1970. Two other communities, Mount Carmel and Church Hill, with a combined 1970 population of 5,634 are located between six and 12 miles northeast of the site. The largest urban center within 50 miles of the site is



Reference - Phipps Bend Preliminary Safety Analysis Report

Figure 2.1 Phipps Bend Site Location

POOR ORIGINAL

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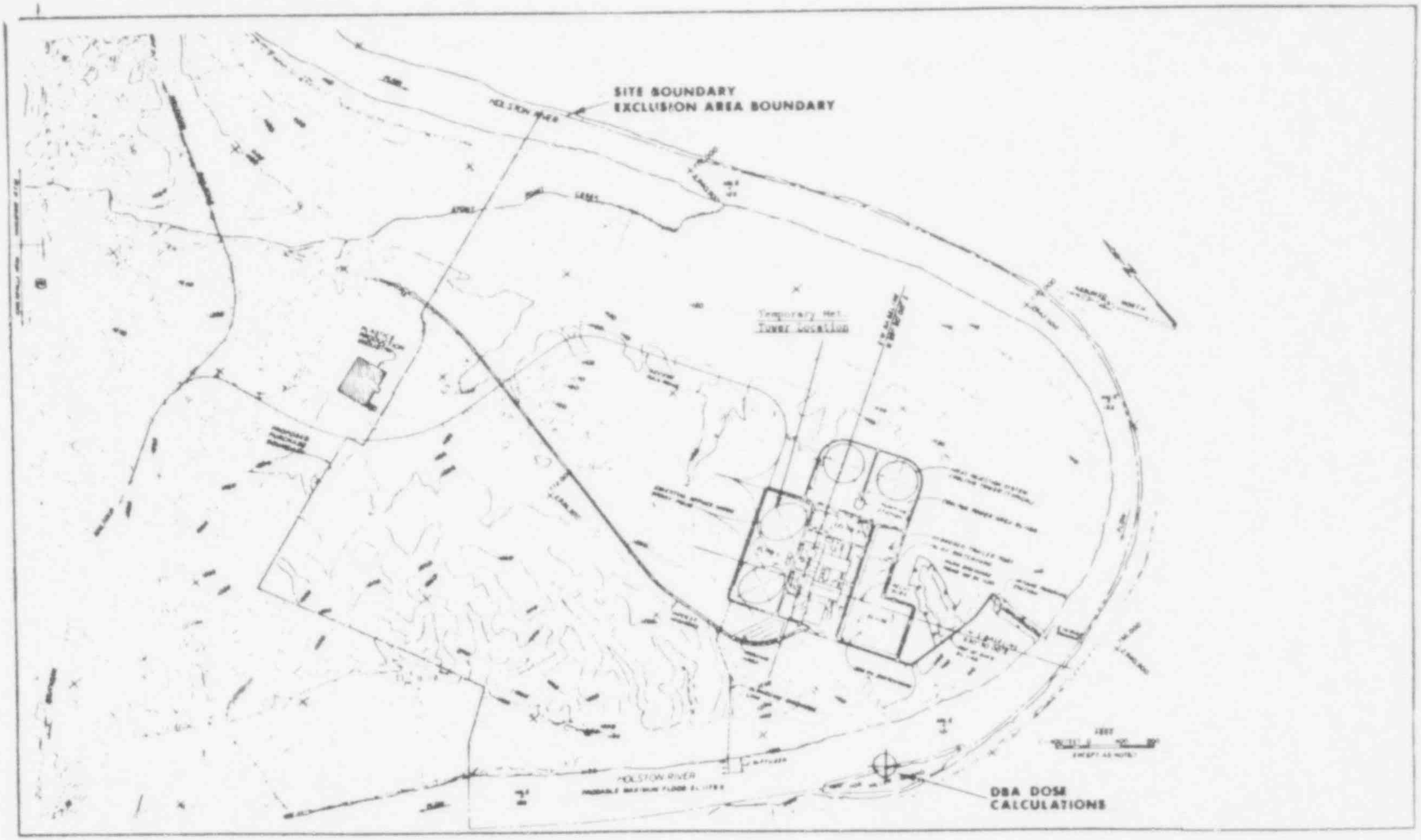

Figure 2.2 Phipps Bend Site Exclusion Area

Reference - Phipps Bend Preliminary
Safety Analysis Report

Kingsport, Tennessee, which is located approximately 15 miles northeast. Kingsport City and the unincorporated suburb of Kingsport North had a combined population of 45,056 in 1970.

The 1970 census and projected cumulative populations for the years 1980 and 2020 are shown in the table below as a function of distance out to 50 miles from the site.

TABLE 2.1

1970 CENSUS AND PROJECTED POPULATIONS

Radius, Miles	1970	1980	2020
0-5	5,265	5,740	10,750
0-10	17,715	23,300	60,645
0-30	306,440	378,575	842,555
0-50	689,600	803,135	1,454,700

The 1980 cumulative resident population, the projected population near the approximate date of the plant operation, is shown in Figure 2.3 as a function of distance from the site. For reference, the cumulative population corresponding to a moderately populated area of 500 people per square mile is also shown. The curves in Figure 2.3 illustrate that the average population density at all distances out to 50 miles from the site is well below 500 people per square mile.

In order to verify the applicant's population data, we obtained an independent estimate of the 1970 population within 50 miles of the site from U. S. Bureau of the Census data and compared this value to the applicant's 50-mile population figure for 1970. We found that the U. S. Bureau of the Census value of 687,265 was in close agreement with the applicant's value of 689,600. We also compared the applicant's projected population growth rate to the year 2020 for the area within 50 miles of the site with the population projections of the U. S. Bureau of Economic Analysis for Economic Areas 50 and 51. These two areas cover the region extending over northeastern Tennessee and southeastern Kentucky and include the 50-mile area surrounding the site. This comparison showed that the applicant's population growth projection of 16 percent per decade for the area within 50 miles of the site exceeded the regional growth projections of five percent per decade and eight percent per decade for Economic Areas 50 and 51, respectively, made by the U. S. Bureau of Economic Analysis.

The applicant has specified a low population zone with an outer boundary distance of 4,830 meters (3 miles). The population of this area was 2,090 in 1970 and the applicant projects that it will increase to 2,395 by 1980. There are no large transient populations within or immediately beyond the low population zone which could significantly alter the population distribution. Based on our review of the population distribution, road network, and land use factors within the low population zone, and by comparison with similar characteristics of previously approved sites,

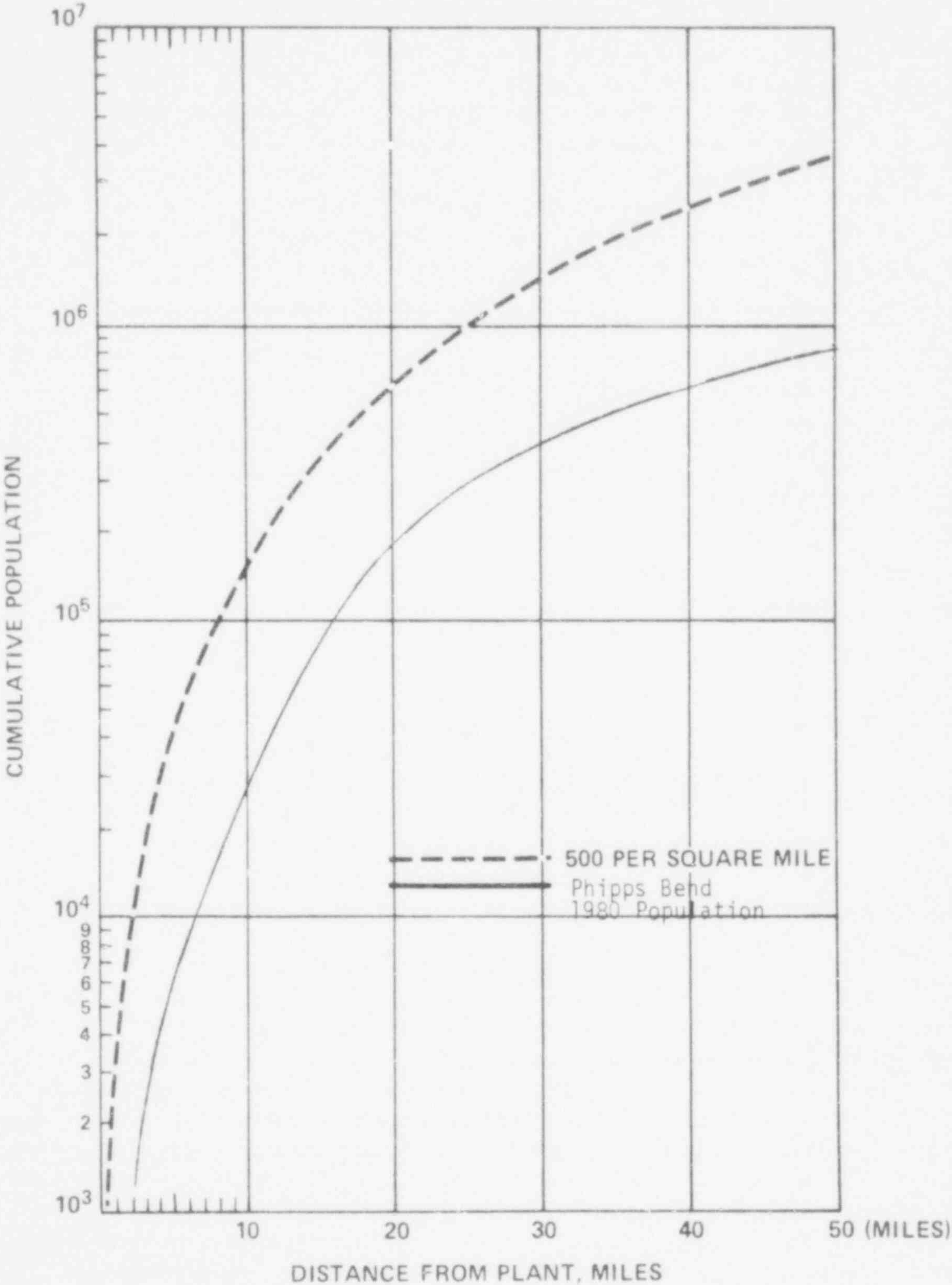

Figure 2.3 Cumulative Population Distribution

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we conclude that there is reasonable assurance that the definition of the low population zone in 10 CFR Part 100 can be satisfied in that we have not identified any factors which would preclude the development of acceptable emergency measures to protect the public within the low population zone (see Section 13.3 of this report).

The nearest population center of 25,000 persons or more, as defined in 10 CFR Part 100, is Kingsport, Tennessee which, in combination with the suburb of Kingsport North, had a 1970 population of 45,056. Kingsport is located approximately 15 miles northeast of the site, a distance which is greater than the minimum population center distance of one and one-third times the distance from the center of the site to the outer boundary of the low population zone, as required by 10 CFR Part 100.

On the basis of the 10 CFR Part 100 definitions of the exclusion area, low population zone, and population center, the applicant's commitment to make arrangements to control traffic on the Holston River through the exclusion area in the event of an emergency; and the calculated radiological consequences of postulated design basis accidents presented in Section 15.2 of this report, we conclude that the exclusion area, low population zone, and population center distances specified for the proposed Phipps Bend site meet the requirements of 10 CFR Part 100 and are acceptable.

2.2 Nearby Industrial, Transportation and Military Facilities

Several industrial plants are located in the Holston River Valley primarily to the northeast of the site location. The closest plant is a plastic manufacturing factory which is located adjacent to the site boundary approximately 5,600 feet north-northeast of the nearest plant safety-related structure. The plant employs about 100 people in the production of injection molded plastic houseware items. On the basis of information supplied by the applicant, the plastics used in the process are styrene and polyethylene of which a maximum inventory of 250,000 pounds each are stored at the plant in pellet form. No explosive materials of sufficient quantity to pose a threat to the safe operation to the proposed Phipps Bend plant have been identified as being stored at the plastic manufacturing factory. The applicant has calculated the potential effects of a fire involving the stored chemicals at the plastics plant and determined the concentrations at the nuclear plant of the toxic products of combustion. The applicant has concluded that the toxic gas concentrations at the Phipps Bend plant would be within the toxicity limits specified in Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Plant Control Room During a Postulated Hazardous Chemical Release." In addition, smoke detectors located in the control building air intakes will alert the plant operators to the presence of combustion products so that protective actions may be taken. We have reviewed the applicant's analysis and assumptions and find them acceptable. We agree with the applicant that a fire at the plastics plant will not adversely affect the safe operation of the Phipps Bend plant. The applicant also states that the officials of the plastics plant do not anticipate any expansion or changes in the plant processes.

Several other manufacturing industries are located between 2.7 and 4.3 miles from the Phipps Bend plant. Based on information on hazardous materials at these locations provided by the applicant, we conclude that the quantities of materials and distances involved are such that potential accidents at those industries will not adversely affect the safe operation of the plant.

The Holston Army Ammunition Plant is located approximately eight miles northeast of the site. This separation distance is adequate to insure that any credible accidental explosion at the ammunition plant will not be a threat to the safety of the nuclear plant. High explosives are shipped to and from the ammunition plant over the Southern Railway line which, at its closest point of approach, passes the site approximately 7,500 feet northwest of the nearest plant safety-related structures. The largest shipment of high explosives which is expected to pass the site is 435,000 pounds. This shipment would be carried in three railroad cars which would be separated to preclude the simultaneous detonation of all three cars. Nevertheless, the applicant has calculated the blast overpressure resulting from the postulated simultaneous detonation of three railroad cars fully loaded with high explosives on the railroad line adjacent to the site and concluded that even this accident would not exceed the plant design criteria. We have also calculated the potential effects of an accidental detonation of three railroad cars fully loaded with high explosives on the railroad and are in agreement with the conclusion reached by the applicant.

The nearest highway to the site is U.S. Highway 11W which passes approximately 1.9 miles to the northwest. Interstate Highway I-81 is located over five miles from the site. There is no commercial barge traffic on the Holston River in the vicinity of the site.

The applicant has identified the toxic chemicals which are stored and transported in the site vicinity and has evaluated postulated accidental releases of toxic chemicals resulting from traffic on the Southern Railway at its closest point of approach to the site. Utilizing conservative assumptions regarding the amount of chemical released to the atmosphere and the meteorological conditions prevailing at the time of a postulated accident, the applicant calculated that concentrations in the control room of chlorine and acetaldehyde from the rupture of a railroad tank car could exceed the toxicity limits specified in Regulatory Guide 1.78. Therefore, the applicant has committed to install detection instrumentation in the control room air intakes and to provide automatic isolation of the control room upon detection of either chlorine or acetaldehyde. Emergency air supply equipment will also be available for the control room operators. Based on our review of the analtyical model and assumptions used in the evaluation, and with the addition of the protective instrumentation for the control room, we concur with the applicant's conclusion that postulated toxic chemical accidents from railroad traffic, or from other transportation routes or storage facilities in the site vicinity, will not interfere with the safe operation of the plant.

The nearest airport to the site is Hawkins County Airport which is located 4.2 miles west of the site. The airport has one asphalt runway 3,500 feet in length and erves only general aviation aircraft operating under visual flight rules. The earest airport with commercial airline traffic is the Tri-Cities Airport which is located approximately 20 miles east of the site. Federal Airway V16N, located approximately seven miles northwest of the site, is the closest civilian airway. At present, the center line of a military low level training route (TR-870) passes approximately 1.2 miles west of the site. By agreement between the Nuclear Regulatory Commission and Department of Defense, military low level training routes are changed at the time an operating license is granted so that at a minimum the perimeter of the training route will clear a nuclear plant site. Based on our review of the aviation activities in the vicinity of the site and on analyses of similar activities at other nuclear power plant sites, we conclude that the probability of an aircraft crashing into the Phipps Bend plant and causing offsite radiological doses in excess of 10 CFR Part 100 guidelines is so low that it need not be provided for in the design of the facility.

A natural gas pipeline 6.25 inches in diameter operating at nominal pressure of 406 pounds per square inch guage lies about 7,500 feet northwest of the site at its closest point of approach. The applicant has evaluated the possible consequences of a postulated rupture of the pipeline with subsequent dispersal and ignition of the gas cloud. We have reviewed the evaluation and agree with the applicant that the natural gas pipeline does not represent a hazard to the safe operation of the plant.

We conclude that the nearby industrial, transportation, and military activities in the vicinity of the Phipps Bend Nuclear Plant site have been properly evaluated and that, with regard to postulated accidents which may occur as a result of these activities, the plant can operate without undue risk to the health and safety of the public.

2.3 Meteorology

Information concerning the atmospheric diffusion characteristics of a proposed nuclear power station site are required in order that a determination may be made that postulated accidental, as well as routine operational, releases of radioactive material are within our guidelines. Further, regional and local climatological information including extremes of climate and severe weather occurrences which may affect the safe design and siting of a nuclear station are required to insure that safety-related station design and operating bases are within our guidelines. We evaluated the meteorological characteristics of a proposed site in accordance with the procedures presented in Section 2.3.1 through 2.3.5 of the Standard Review Plan (November 1975).

2.3.1 Regional Climatology

Although the proposed Phipps Bend site, which is located in northeastern Tennessee, does not lie along any major storm tracks, it is influenced by storms which move along the Gulf Coast and up the Atlantic Coast to the northeast. Mountain influences extract moisture from both typically westerly and easterly flow. Such flow subsequently eaches the site area drier t it would be otherwise. Maximum monthly precipitation occurs in July, most frequently as a result of afternoon or early-night thunderstorms. Warm overrunning moist air associated with low-pressure centers to the south and northeast results in a secondary maximum of late-winter precipitation. Slow moving cold air cells connected with low-pressure systems centered in Pennsylvania and southern New York at times result in extended periods of cold weather. Prolonged periods of hot weather result most frequently from the effects of diurnal heating coincident with subsidence from a high-pressure system dominating the eastern United States.

The average maximum temperature in July is 75 degrees Fahrenheit, while the average January temperature is 36 degrees Fahrenheit. Temperatures may be expected to reach 90 degrees Fahrenheit or higher on about 13 days per year, 32 degrees Fahrenheit or lower on 96 days per year, and equal to or less than zero degrees Fahrenheit on two days per year.

Hurricanes are not common in northeastern Tennessee. Between 1871 and 1974, the centers of two hurricanes (downgraded to tropical storms) passed within 50 miles of the site. Thunderstorms have occurred on approximately 50 to 55 days in an average year. During the period 1955 through 1967, hail greater than three quarters of an inch in diameter has been reported eight times within the one degree square containing the Phipps Bend site. Between 1953 and 1974, 24 tornadoes have occurred in a 100 mile square area which includes the proposed Phipps Bend site. This mean annual frequency of 1.1 tornadoes per year for the 100 square mile area, using methods developed by Thom results in a recurrence interval of about 3200 years for a tornado at the proposed plant site. The design basis tornado characteristics selected by the applicant for the site conform to the recommendations of Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," for this region of the country.

Between 1955 and 1967, an average of 19 windstorms per year with gusts greater than 50 knots (58 miles per hour) occurred in the one degree square containing the proposed Phipps Bend site. The "fastest mile" wind speed reported at the Tri-Cities Airport was 50 miles per hour (May 1951). The design basis wind for the proposed Phipps Bend site is 95 miles per hour at an elevation of 30 feet above grade based on a recurrence interval of 100 years as outlined in Section 3.3.1 of this report. This is within the GESSAR envelope.

In the 35 year period of 1936 through 1970, 70 cases of atmospheric stagnation totalling about 275 days occurred in northeastern Tennessee.

2.3.2 Local Meteorology

Long-term weather records (1941 through 1970) for the site vicinity are available from the Tri-Cities airport, about 22 miles east of the site. At the Tri-Cities airport, the average daily temperature ranges between 85.9 degrees and 64.4 degrees Fahrenheit in July and 46.0 degrees and 26.7 degrees Fahrenheit in January. The

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extreme maximum temperature recorded was 95 degrees Fahrenheit in July 1970. Minus 15 degrees Fahrenheit was the extreme minimum temperature recorded in January 1966.

Annual average precipitation recorded at the Tri-Cities airport is 41.5 inches. With an average of 2.3 inches, October is the driest month, while five inches in July makes it the wettest month. Precipitation is well distributed throughout the year. The maximum 24-hour precipitation recorded was 3.7 inches which fell in October 1974. The maximum 24-hour snowfall was 16.2 inches in November 1952.

Average annual relative humidity in the proposed Phipps Bend area is 70 percent. Heavy fogs (visibility one quarter of a mile or less) are reported on an average of 44 days per year.

Wind frequency by direction at the 150-foot level for the period January 1974 through December 1975 is shown in Figure 2.4. Wind flow at this height was most frequent from the east northeast and west southwest directions with about 22 percent and 16 percent occurrences, respectively. The overall average wind at the 150-foot level reduced to represent the wind at the 33-foot level is 2.7 miles per hour.

2.3.3 Onsite Meteorological Measurements Program

The onsite preoperational onsite measurements began in December 1973 from a 150-foot tower. Wind speed and direction are measured at the 33 and 150-foot levels. Vertical temperature difference is measured between these two levels. Since February 1976 the dew point has been measured at the 33-foot level.

The Tennessee Valley Authority has provided joint frequency distributions of wind speed and direction by atmospheric stability class, based on the vertical temperature gradient measured onsite from January 1974 through December 1975. The wind speed and direction measured at the 150-foot level and reduced to representative effluent release heights provided the bases for our evaluation of atmospheric transport and dispersion characteristics of the site area since these data seem to best represent airflow in the site area. Data recovery, at the 150-foot level, for the period was 88 percent.

2.3.4 Short-Term (Accident) Diffusion Estimates

We have made conservative assessments of post-accident atmospheric diffusion conditions at the proposed Phipps Bend site using onsite meteorological data and the appropriate diffusion model described in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors." In the evaluation of short-term (0-2 hours at the low population zone distance) accidental releases from the station building and vents, the staff assumed a ground-level release with a building wake factor of 1150 square meters.

The relative concentration for the 0-2 hour time period which is exceeded no more than five percent of the time is 1.8×10^{-3} seconds per cubic meter at the exclusion

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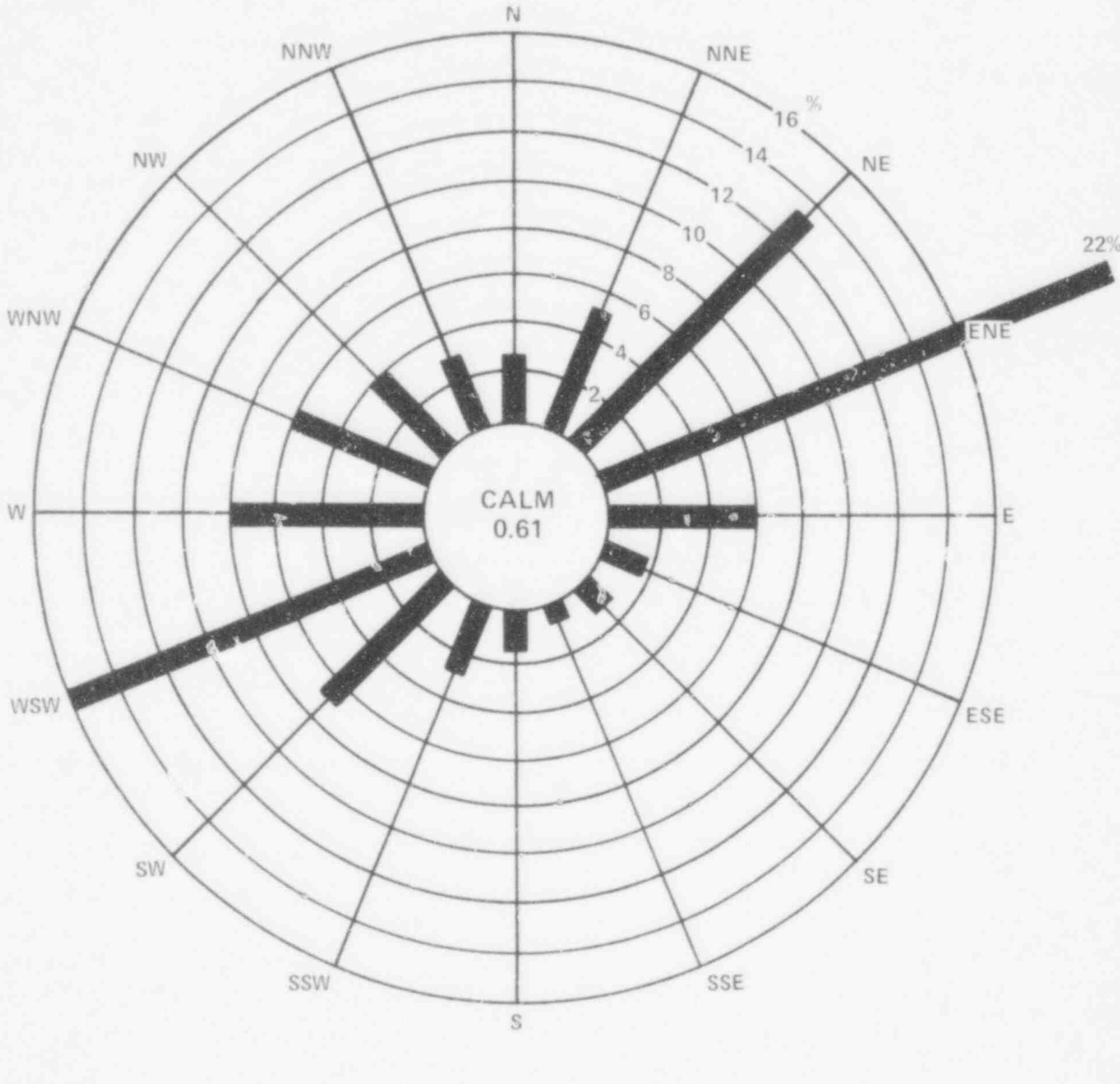

Figure 2.4 Onsite Wind Data, 150-foot Level, January 1974—December 1975

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distance of 740 meters (measured from the outside edge of the containment building). This relative concentration is equivalent to that calculated by assuming Pasquill Type F stability with a wind speed of 0.3 meters per second. The relative concentration values for various time periods at the outer boundary of the low population zone (4830 meters) are shown in Table 2.2:

TABLE 2.2

RELATIVE CONCENTRATION VALUES AT THE OUTER BOUNDARY OF THE LOW POPULATION ZONE

Time Periods	Phipps Bend (3 mile low population zone) Relative concentration (Seconds per cubic meter)	GESSAR (2 mile low population zone) Relative concentration (Seconds per cubic meter)
0-8 hours	1.2×10^{-4}	1.1×10^{-4}
8-24 hours	8.0×10^{-5}	2.3×10^{-5}
1-4 days	3.5×10^{-5}	8.1×10^{-6}
4-30 days	1.1×10^{-5}	1.7×10^{-6}

Although the relative concentration values for the long time periods are not the same as typical values used in our evaluation of the GESSAR nuclear island (Section 15.0 of Appendix A to this report), the results of the calculations shown in Section 15.0 of this report demonstrate the conformance of the Phipps Bend plant to 10 CFR Part 100 guidelines.

2.3.5 Long-Term (Routing) Diffusion Estimates

We have made reasonable estimates of average atmospheric diffusion conditions for the proposed Phipps Bend site using 150-foot wind data adjusted to representative heights of release, since these data seem to represent the regional air flow regimes. To provide relative concentration and deposition values for the site, we used the Straight-Line Trajectory Model, described in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," in evaluating atmospheric transport and dispersion characteristics. Partial elevated releases were considered when the exit velocities and building configurations met the criteria established in Regulatory Guide 1.111. The Tennessee Valley Authority also calculated relative concentrations and deposition values for the site using a Puff Advection Model as described in Regulatory Guide 1.111, with one year (August 1974 through July 1975) of concurrent data from the proposed site and the John Sevier Steam plant about 11 miles down valley from the proposed Phipps Bend site. The results of the Puff Advection Model support the staff's dose estimates (which are summarized in Section 5.7 of the Phipps Bend Final Environmental Statement) based on the Straight-Line Trajectory Model for the controlling receptor locations. Table 2.3 lists the relative concentration and deposition values used in the dose estimates. The Puff Advection Model was used to adjust the diffusion and deposition estimates out to 50 miles.

TABLE 2.3

PHIPPS BEND NUCLEAR PLANT, RELATIVE CONCENTRATION AND DEPOSITION USED IN DOSE ESTIMATE SUMMARY

Source (Type Release)	Type of Receptor	Direction (From Source)	Distance (miles)	Continuous Relative Concentration (seconds per square meter)	Release Deposition (meters^{-2})	Intermittent Relative Concentration (seconds per square meter)	Release Deposition (meters^{-2})
Reactor Vent (Continuous)	Maximum Offsite Point	West-South-West	1.5	3.6×10^{-6}	1.1×10^{-8}		
	Residence/Garden /Meat	East-North-East	1.03	2.4×10^{-6}	9.9×10^{-9}		
	Milk Animal/Farm	West-South-West	2.14	2.2×10^{-6}	6.6×10^{-10}		
Radwaste Vent (Continuous)	Maximum Offsite Point	West-South-West	1.5	7.9×10^{-6}	9.2×10^{-9}		
	Residence/Garden /Meat	East-North-East	1.03	9.6×10^{-6}	2.0×10^{-8}		
	Milk Animal/Farm	West-South-West	2.14	4.6×10^{-6}	3.6×10^{-9}		
Turbine Vent (Continuous and 4 intermittent releases per year, 24 hours each)	Maximum Offsite Point	West-South-West	1.5	4.1×10^{-6}	1.1×10^{-8}	1.1×10^{-5}	3.0×10^{-8}
	Residence/Garden /Meat	East-North-East	1.03	2.6×10^{-6}	9.6×10^{-9}	8.3×10^{-6}	3.2×10^{-8}
	Milk Animal/Farm	West-South-West	2.14	2.4×10^{-6}	7.9×10^{-10}	6.5×10^{-6}	1.6×10^{-8}

2.3.6 Conclusions

Tennessee Valley Authority's onsite meteorological program meets the recommendations and intent of Regulatory Guide 1.23, "Onsite Meteorological Programs." The data from this program, as summarized by Tennessee Valley Authority in the Preliminary Safety Analysis Report, provide an adequate meteorological description of the proposed site and its vicinity for the purpose of making atmospheric diffusion estimates for postulated accidental and routine airborne releases of radioactive releases from the proposed Phipps Bend plant. We have concluded that the onsite meteorological data presented by the applicant for the two-year period, January 1974 through December 1975, provide a representative and conservative basis for estimating atmospheric dispersion at the proposed site.

2.4 Hydrologic Engineering

2.4.1 Hydrologic Description

The proposed Phipps Bend site consists of approximately 1,350 acres inside a bend on the north bank of the Holston River. Surface elevations range from about 1,110 feet along the floodplain to 1,300 feet along the ridge.

The Holston River above the proposed Phipps Bend site drains 2,861 square miles of eastern Tennessee, western North Carolina and southwestern Virginia. The Holston River proper begins just west of Kingsport, Tennessee, at the junction of the North Fork and South Fork Holston Rivers. The river flows generally southwestward joining the French Broad River at Knoxville, Tennessee, to form the Tennessee River.

With the exception of the Watauga River, a major tributary of the South Fork Holston River, the watershed is long and narrow, averaging about 160 miles long and 20 miles wide and lying within the Appalachian Valley Physiographic Province. The major streams flow between parallel ridges running generally northeast-southwest. The terrain is hilly to mountainous except for the comparatively narrow and relatively flat floodplains. In the vicinity of the proposed site, floodplains range from 800 to 4,000 feet wide.

There are three major dams located on the South Fork Holston River: Fort Patrick Henry Dam, Boone Dam and the South Holston Dams. The Watauga River which joins the South Fork Holston River 20 miles above its mouth, heads in the rugged, heavily forested Blue Ridge region of eastern Tennessee and western North Carolina. One major dam, Watauga, is located in the Watauga watershed. The North Fork of the Holston River is not controlled. The Cherokee Dam is located downstream from the site. The reservoir system is operated for flood control, navigation, and power generation with the exception of Fort Patrick Henry which has no reserved flood retention capacity. There are no water control structures proposed or under consideration which would significantly affect river flow conditions at the plant site.

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Flood control above the proposed Phipps Bend site is provided largely by South Holston and Watauga Reservoirs. Elevations in the watershed range from 1110 feet near the plant site up to 6285 feet along the watershed rim. The climate for the watershed is humid temperate with a mean annual precipitation of 44 inches. Major flood producing storms are of two general types, the cool season, winter type and the warm season type including decadent tropical storms or hurricanes. Watershed snowfall averages about 22 inches annually above the proposed plant site, but in the extreme headwaters at higher elevations the annual rainfall average ranges up to near 50 inches. Snowmelt is not a factor in maximum flood determination.

The proposed Phipps Bend plant site is underlain by geologic formations belonging to the Sevier Shale Group of Ordovician age. The formation is a calcareous shale with scattered, thin interbeds of limestone. Locally, the Sevier shale is a fairly productive aquifer, yielding up to 100 gallons per minute to wells. Groundwater will not be used as a water supply for the proposed Phipps Bend site.

2.4.2 Flood Potential

Several flood producing sources were investigated by the applicant and the staff to determine the design basis flood level for the proposed Phipps Bend site. They include a probable maximum flood on the entire Holston River Watershed above the site and on critical subwatersheds including potential consequent dam failures, postulated seismically induced dam failures and the effects of local intense precipitation.

Storm amounts and the time distribution used by the applicant were those determined by the Hydrometeorological Branch of the National Water Service contained in their Report No. 45, "Probable Maximum and TVA Precipitation for Tennessee River Basins up to 3000 Square Miles in Area and Durations to 72 Hours." Loss rate functions and unit hydrograph determinations were reviewed by us and found to be acceptable. Routings were performed using an unsteady flow model developed by the applicant and previously determined to be acceptable by us during the Clinch River Breeder Reactor (Docket No. 50-537) review.

To determine the probable maximum flood at a point when upstream reservoirs would regulate the flow, Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," provides that either the reservoirs be considered full or that an antecedent (or subsequent) storm equal to about one-half the probable maximum precipitation three to five days prior (or after) be used in conjunction with the probable maximum preciptation. Based on extensive studies of the eastern Tennessee area, the applicant and his consultant, the Hydrometeorological Branch of the National Weather Service, have determined that an antecedent storm equal to 30 percent of the probable maximum precipitation is conservative and compatible with the definition of the probable maximum flood as stated in Regulatory Guide 1.59. After extensive review of these studies, we agree that for this particular case a 30 percent of probable maximum precipitation antecedent storm meets the intent of the criteria set forth in Regulatory Guide 1.59 and is therefore acceptable.

Based on the above, the applicant has concluded, and we concur, that the probable maximum flood for the entire watershed above the plant or the probable maximum flood for the watershed above South Holston Dam with concurrent precipitation in the remainder of the basin do not produce the design basis flood level at the plant site.

The applicant has evaluated the effects of a postulated safe shutdown earthquake on upstream dams coincident with the 25-year flood as outlined in Regulatory Guide 1.59. Using conservative assumptions the applicant has determined that the flood produced by the above events would be below plant grade and based on our review of its analysis we concur. The applicant has also evaluated Watauga Dam and determined it to be safe during the postulated occurrence of the operational basis earthquake coincident with one-half of the probable maximum flood from the Watauga basin. For our evaluation of the seismic capability of Watauga Dam, see Section 2.5.5. Accordingly, this event (with the failure of South Holston Dam) would not produce the design basis flood level.

The applicant determined that the probable maximum flood produced by a probable maximum precipitation storm centered above Watauga Dam and South Holston Dam such that Watauga Dam just fails would produce a design basis still water level at the proposed site of 1182.3 feet. The applicant assumed for the residual rainfall a uniform distribution for the remainder of the basin equal to 90 percent of the total basin (2861 square miles) probable maximum precipitation. We expect these conditions will in fact produce the design basis flood, however, we have asked the applicant to verify the conservatism of this last assumption and to verify that a storm centering above and below Watauga Dam such that Watauga Dam just fails would not produce the design basis flood.

The applicant has committed to design and construct the Phipps Bend Nuclear Plant to withstand a maximum still water flood level for an elevation between 1182 feet and 1183 feet plus coincident wind wave and runup effects (about 1.5 feet) and to verify the above assumptions on the residual rainfall and the storm centering. We concur with the applicant's design still water level of between 1182 feet and 1183 feet subject to the verification of his assumptions.

The applicant also evaluated the effects of local probable maximum precipitation on the plant area, including the roofs of safety-related structures. The applicant has concluded that the proposed site grading would preclude flood water due to a local probable maximum precipitation from entering safety-related structures. Based on our review of the applicant's analysis and the applicant's commitment to protect safety-related systems and structures from the design basis flood, we concur.

2.4.3 Water Supply

The makeup water intake will be located on the Holston River to the south of the plant structures. Water will be pumped from a trapezoidal intake channel to the plant at a maximum rate of about 132 cubic feet per second. Condenser cooling will be accomplishe by two natural draft cooling towers located on the northeast side of the plant.

The ultimate heat sink complex will consist of two seismic Category I eliptical spray ponds located to the north of the plant. The ponds are to be identical to the previously approved ponds proposed for the Hartsville Nuclear Plants (Docket Nos. STN 50-518, 519, 520 and 521). The plants (Hartsville and Phipps Bend) are identical and the meteorological conditions are less severe for the Phipps Bend site. Makeup water for the ultimate heat sink beyond the 30-day requirement will be from the Holston River. Accordingly, we have concluded that the meteorological and hydrologic design bases for the ultimate heat sink are conservative and meet the recommendations set forth in Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants" and are therefore acceptable.

2.4.4 Ground Water

Ground water in the region surrounding the Phipps Bend proposed site occurs almost entirely under shallow-system unconfined conditions in consolidated rocks (dolomite, lime-stone, shale, and sandstone) and in the overburden above bedrock. The carbonate rocks are the principal aquifers, of which the Knox Dolomite is the most significant. Few of the shale formations and none of the sandstone formations, yield water to wells in quantites sufficient for other than domestic or stock supplies. The Knox Dolomite underlies about half the surface area of the region and is the principal source of base flow to streams.

Depth to the water table varies with the topography being greater under higher areas and nearer the surface in lower areas. In a few places depth to water table is greater than 100 feet, but in most it is less than 50 feet. Since it is generally shallow, the configuration of the water table conforms fairly closely to surface topography and ground water divides approximate surface water divides.

Locally the Knox Dolomite is the most significant aquifer. The town of Surgoinsville uses less than 40,000 gallons per day from a spring flowing from the Knox. An outcrop belt of the Knox lies about 1.5 miles northwest of the site. Since the dip is to the southeast at about 30 degrees, the Knox is present beneath the proposed plant site depth of more than 1000 feet, far below the zone of active water movement.

Water occurs in the Sevier shale which underlies the proposed plant site in fairly widely spaced, generally small, weathered zones formed along fractures and bedding planes. Most stored water available to the formation is contained in the overburden, which, being mostly made up of clay and silt, is of low permeability but relatively high porosity in comparison to bedrock.

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The Holston River forms a hydraulic barrier to the site on three sides and all ground water discharges from the proposed site will flow to the river. No ground water use is planned for the Phipps Bend plant. The applicant has conservatively selected two feet below final plant grade as the design basis for hydrostatic loading for all safety-related structures.

The applicant and we have independently analyzed the effects of an accidental spill of radioactive liquids at the proposed plant site. For the purposes of this analysis, it was determined that the most critical case would be the failure of the 18,000-gallon high conductivity waste tank in the radwaste building.

Upon a postulated rupture of the high conductivity waste storage tank and the building, the radioactive liquids would mix and travel with the ground water according to the hydraulic gradient of the ground water. The direction of travel would be towards the Holston River. The nearest discharge point would be 2000 feet down-gradient at the river. Any spill would be diluted first by the ground water and further by the Holston River. As outlined in Section 15.3 of this report we have determined that all nuclide concentrations would be small fractions of the limits of 10 CFR Part 20 for unrestricted areas.

2.4.5 Conclusions

Based on our independent review and analyses, we conclude that an adequate water supply can be assured for safety-related purposes, that design basis ground water levels are conservative, and that postulated accidental spills of radioactive liquids will result in radionuclide concentrations that are small fractions of 10 CFR Part 20 limits at public water supplies. We concur with the applicant's still water level. However, we are requiring the applicant to verify the conservatism of his assumptions.

2.5 Geology, Seismology and Foundation Engineering

2.5.1 Introduction

The geology and seismology review of the proposed site performed by us addressed the geologic history of the region including physiographic, lithologic, stratigraphic, and tectonic settings as well as the subregional and site-specific geology and seismology. In addition to reviewing data submitted in the Phipps Bend Preliminary Safety Analysis Report, staff geologists, seismolgists and foundation engineers visited the site and environs on two occasions. During those vists we examined the regional geology, bedrock exposures, and an extensive number of core borings. We have conferred with local geological experts and the applicant's technical personnel concerning geological interpretations within the site region. Information gathered during the review of this site and reviews of other sites in the Southern Valley and Ridge area have aided in the evaluation of this site (Sequoyah Docket

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Nos. 50-327 and 50-328, Watts Bar Docket Nos. 50-340 and 50-341, Bellefonte Docket Nos. 50-438 and 50-439 and the Clinch River Breeder Reactor Docket No. 50-537). Since the regional aspects which apply to this site are reasonably well understood and have also been discussed extensively in these other reviews and safety evaluations, the main effort of this section will be to address specific issues which might have indicated a possible hazard to the safe operation of the Phipps Bend plant at the proposed location.

The particular items of concern at the proposed Phipps Bend site were:

(1) The potential for local surface faulting, and

(2) The determination of the safe shutdown earthquake.

We are satisfied that geologic and seismic investigations performed by the applicant have been sufficient to adequately assess site geologic conditions in accordance with "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Appendix A to 10 CFR Part 100. We conclude as outlined below that the regional tectonic structures do not pose any threat to the safety of the plant. There are also no geologic structures that would cause surface displacement or would tend to localize earthquakes in the site vicinity.

2.5.2 Geology

2.5.2.1 Geologic Setting

The proposed Phipps Bend site is located 15 miles southwest of Kingsport, Tennessee in Hawkins Country, Tennessee. Geologically, it is in the southeastern section of the Valley and Ridge Physiographic Province of eastern Tennessee which is characterized by northeasterly striking ridges and valleys. The geologic structure is predominantly northeast-southwest trending folds and faults which developed during Paleozoic time (greater than 225 million years before present). The faults are extensive thrust faults which dip to the southeast. The applicant in Section 2.5.1.1.4 of the Phipps Bend Preliminary Safety Analysis Report describes the geologic history of the site vicinity as follows:

> "The Phipps Bend area lies near the western border of what was the active portion of the Appalachian geosyncline during most of the Paleozoic Era. During the early portion of the era, in Cambrian time, sands and clays were deposited in shallow, muddy water and these consolidated to form sandstone and shales of the Rome Formation. The syncline gradually depressed and the sea became deeper and broader. At the beginning of the deposition of the Conasauga, the sea received a small amount of sand and much clay. The sediment load gradually changed until at the end of the Conasauga only limy deposits were being laid down. Throughout the succeeding Knox deposition, the sea was deep and still as indicated by the great thickness of limestones and dolomites that were deposited.

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At the close of the Knox, most of the area was uplifted slightly and exposed to erosion. By the Middle Ordovician the land had subsided again and was covered by a shallow and oscillating sea in which a great thickness of limestone and calcareous shale was deposited. These are the formations that will provide the foundation for structures at the Phipps Bend site. At the end of Paleozoic time, during the Allegheny orogenic episode, the rocks at the site were folded and faulted and tilted to the southeast. Since Paleozoic time, weathering and erosion have been the dominant geologic processes at the site with sediment accumulation being restricted to the alluvial and flood plain deposits of the Holston River."

2.5.2.2 Site Geology

The Sevier shale of the Middle Ordovician Chickamauga Group is the foundation bedrock for the plant and is described as a calcareous shale, with layers of gray limestone and siltstone. The bedrock in the site vicinity is part of the structural feature known as the Bays Mountain Synclinorium and in this area, the Sevier shale is known to be severely folded and distorted and contains small faults. At the site the bedrock is overlain in some areas by terrance deposits and shale residum. The thickness of this overburden in the plant area ranges from approximately 13 feet to 64 feet. Most of the plant will be founded on the Sevier shale which locally contains some zones of weathering. Experience gained by the applicant during the construction of the John Sevier steam plant in this same foundation rock indicates that these zones can be adequately treated during excavation and foundation preparation. Specific site foundation conditions are addressed in Section 2.5.3 of the Phipps Bend Preliminary Safety Analysis Report and Section 2.5.4 of this report.

2.5.2.3 Faulting

A major fault, the Carter Valley fault, passes about five miles west of the site. This fault strikes northeasterly and dips to the southeast. This fault is one of many Late Paleozoic thrusts that developed during the Allegheny Orogeny (Permian-Pennsylvanian time, 280-230 million years before present). Studies conducted on the Copper Creek fault for the Clinch River Breeder Reactor plant confirmed the regional geologic history of the region. Radiometric dates of 290 ± 10 million years before present and 280 ± 10 million years before present were obtained for fault gouge material taken from the fault. In addition, the applicant has conducted photogeologic investigations of the Carter Valley fault and has conducted ground surveillance in order to confirm that there has been no later movement on this fault. The results of these investigations, coupled with lack of evidence of any recent offset along this fault and a generally accepted understanding of the tectonic development of the Paleozoic thrust faulting in east Tennessee, indicate that this major fault and other small faults in the site area associated with it are tectonically old. Therefore, we do not consider them hazardous to the safe

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operation of a nuclear plant at this location. These faults are not capable faults within the meaning of "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Appendix A to 10 CFR Part 100.

2.5.2.4 Tectonic Province

Since earthquake activity cannot be reasonably associated with geologic structure in the southern Appalachians, earthquakes are instead identified with the tectonic province in which the site is located. The applicant indicates that the Phipps Bend proposed site is located in the Southern Appalachian Tectonic Province. As defined this province is bounded on the east by the western margin of the Piedmont Province, on the west by the eastern limits of the Cumberland Plateau, on the south by the overlap of the Gulf Coastal Plain Province, and on the north by the re-entrant in the Valley and Ridge Province near Roanoke, Virginia.

In our review, we determined that the proposed site is within the Southern Valley and Ridge Tectonic Province based on provinces which are more in accord with those proposed by King, Eardley, Rogers, and Haldely and Devine for Eastern North America. This province is bounded on the east by the western extent of the Piedmont Tectonic Province; on the west by the Cumberland Plateau; on the south by the Gulf Coastal Plain; and on the north by the northern part of the Valley and Ridge Province.

2.5.2.5 Conclusions

Based on our review of the results of the applicant's site investigation, our site visits, discussions with local geologists and reviews of other sites in the tectonic region we conclude that there is no geologic structure in the vicinity of the proposed site that could cause surface displacement or tend to localize earthquakes at the site. As an additional insurance the applicant has committed to carry out monitoring and geologic mapping of the foundation areas during excavation.

2.5.3 Seismology

2.5.3.1 Vibratory Ground Motion

The historical earthquakes that were considered as having a potential significant affect on the proposed Phipps Bend site are the 1811-1812 intensity Modified Mercalli event near New Madrid, Missouri, the intensity Modified Mercalli IX-X event of 1886 in Charleston, South Carolina, and the intensity Modified Mercalli VIII Giles County, Virginia earthquake. The 1811-1812 New Madrid earthquake lies within the Mississippi Embayment seismic zone. The closest approach to the site of this zone has been established as the eastern limit of the embayment gravity anomalies shown on regional gravity maps in Tennessee Valley Authority, Relationship of Earthquakes and Geology in West Tennessee and Adjacent Areas, and the closest approach of earthquakes as large as the 1811-1812 series is shown to be 380 miles from the proposed site. At

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this distance, the occurrence of a similar event would result in an intensity of Modified Mercalli VII at the proposed Phipps Bend site (Gupta and Nuttli). The Charleston earthquake of 1886 would result in an intensity VI, at the site (Dutton). A repeat of these major earthquakes at New Madrid, Missouri and Charleston, South Carolina would cause less severe ground motion at the proposed Phipps Bend site than events occurring within the Southern Valley and Ridge Tectonic Province. The maximum historic earthquake in the Valley and Ridge Tectonic Province is the 1897 Giles County, Virginia earthquake. In assessing the maximum historic earthquake for the proposed Phipps Bend site, the applicant proposed a re-evaluated intensity of Modified Mercalli VII-VIII for this event. The re-evaluation was based on new information contained in a report entitled "Report on Evaluation of Intensity of Giles County, Virginia Earthquake of May 31, 1897" by Law Engineering Testing Company compiled and submitted as an amendment to the Clinch River Breeder Reactor Preliminary Safety Analysis Report.

After reviewing the new information, the U. S. Geological Survey which has the Federal responsibility for evaluating earthquakes determined that the Giles County earthquake was intensity Modified Mercalli VIII as listed in the internal records of National Oceanographic and Atmospheric Administration (Von Hake). The official National Oceanographic and Atmospheric Administration publication by Cofmann and Von Hake erroneously records an intensity VII for the 1897 Giles County earthquake because of a typographical error (Von Hake). Therefore, following the tectonic province approach described in Appendix A to 10 CFR Part 100, it is assumed that the maximum historic intensity at the Phipps Bend proposed site could equal Modified Mercalli intensity VIII.

2.5.3.2 Safe Shutdown Earthquake

The applicant proposed an effective acceleration for seismic design at the proposed Phipps Bend site of 0.2g based on the intensity-acceleration relationships developed by Coulter, Waldron and Devine, Neumann, Newmark and modified versions of Trifunac and Brady intensity-acceleration relationship. The applicant's modification of the Trifunac and Brady intensity-acceleration relationship is based on changes to the maximum intensity and peak acceleration recorded at Pacoima Dam during the San Fernando earthquake in 1971. We find that the Pacoima Dam record of Modified Mercalli intensity X and 1.25g peak acceleration used by Trifunac and Brady is acceptable and consistent with UNITED STATES EARTHQUAKES and cannot therefore accept the applicant's proposed modifications to the intensity-acceleration relationship. It is our position that the effective acceleration for seismic design associated with intensity correponds to the mean value obtained from the intensity versus acceleration relationship of Trifunac and Brady. Therefore, the Modified Mercalli intensity VIII for the maximum historic earthquake results in an acceleration of .25g. Based on our analysis, we consider .25g an appropriate effective acceleration to be used as the high frequency input to the design spectra recommended in Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants."

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This ground motion is to be put at the finished grade as free field motion. The applicant has stated that he will meet this requirement. The acceleration value of 0.25g is well within the design envelope value of GESSAR which is 0.30g.

2.5.3.3 Operating Basis Earthquake

The operating basis earthquake is defined by Appendix A to 10 CFR Part 100 as an earthquake which could reasonably be expected to affect the plant site during the operating life of the plant. The applicant has proposed to use an acceleration of 0.09g for the operating basis earthquake and to scale the Regulatory Guide 1.60 response spectra to this value. In support of this position the applicant submitted a probabilistic analysis on the basis of which it was concluded that 390 years was the minimum return period corresponding to an earthquake with a maximum ground acceleration of 0.09g (representative of the mean acceleration for Modified Mercalli intensity VI-VII based on the Trifunac and Brady relationship).

We performed an independent analysis using the earthquakes of intensity V or greater which occurred in the Southern Valley and Ridge over a time interval of 117 years. We determined a recurrence relationship for earthquakes in the region. Assuming earthquakes have an equal probability of occurring anywhere in the region, the return period for intensity VI-VII at a particular site in the region was determined to be about 270 years.

The most significant difference between our procedure and that used by the applicant appears to be in the selection of the relationship describing the intensity attenuation as a function of distance from the source. In our procedure we used a relationship developed by Brazee; the applicant used a relationship combining attenuation curves developed by Bollinger with isoseismal data for the Giles County Virginia earthquake of 1897.

We conclude that an earthquake producing intensity VI-VII at the site has a recurrence time on the order of 300 years and an 88 percent probability of not being exceeded during the 40 year operating life of the plant. Based on this conclusion, an acceleration level of 0.09g (representative of the mean acceleration for Modified Mercalli intensity VI-VII based on the Trifunac and Brady relationship) is a conservative acceleration level for the operating basic earthquake at the proposed Phipps Bend site and is consistent with Appendix A to 10 CFR Part 100 and is therefore acceptable.

2.5.4 Foundation Engineering

2.5.4.1 Stability of Subsurface Materials

The proposed Phipps Bend plant site is located in the drainage basin of the Holston River. The plant location is in a bend of the river that has been covered by terrace deposits to an average surface elevation of 1180 feet mean sea level. These terrace deposits are underlain by the folded Sevier shale formation, an interbedded calcareous siltstone and shale. The floodplain adjacent to the river at the bend is about elevation 1115 feet mean sea level and the river elevation is about 1100 mean sea level. West of the plant area the Sevier shale outcrops in ridges up to elevation 1260 feet mean sea level. The plant will be founded on the Sevier shale bedrock. In the plant area the overburden thickness ranges from 13 to 64 feet and averages 26 feet.

The Sevier shale formation is the principal foundation rock found at this site and is at least 1,000 feet thick. Weathered rock, near the top of the rock surface, will be removed during excavation beneath seismic Category I buildings which will be founded on rock. Isolated cavities have been encountered during core drilling of the Sevier shale formation. Cavities noted on the logs of the core holes in the Phipps Bend Preliminary Safety Analysis Report indicate zones where a sudden drop of the drill steel, or a definite increase in the rate of penetration, occurred. It is difficult to determine whether this evidence indicates that the cavity is an open void or whether it contains filling material. The applicant investigated the potential solubility of the rock based on analytical data obtained during its whole rock analyses. These data were compared, by the applicant, with analyses of the solution susceptible Knox and Chicamauga formations obtained at other sites. Potentially soluble components of the Sevier shale make up only 33 percent of the rock, as compared to 95 percent of the Chicamauga and 66 percent of the Knox formations. In addition, the applicant's evaluation of core recovery in zones classified as cavities provides evidence that most of these zones are actually weathered or soft shale, and are not open solution voids. This conclusion is supported by geophysical downhole logging and the applicant's construction inspection experience in the Sevier shale at the John Sevier Steam Plant. We have reviewed the applicant's analysis and evaluation and concur.

Weathering was seen to be structurally controlled in the shales at the John Sevier Steam Plant. Weathered zones extended more deeply into the trough of tight synclinal folds than into rocks beneath anticlinal folds. Bedding plane weathering has been found to be the dominant type of weathering. As noted in the Phipps Bend Preliminary Safety Analysis Report Section 2.5.4.12, measures will be taken in all excavations to assure the adequacy of the foundation. These measures may include overexcavation, grouting and additional exploration. Excavations will be inspected by the construction engineer and project geologist prior to concrete placement. As noted in the response to item 324.9 in our January 28, 1976 request for additional information, we will be advised when excavation work progresses to the point where the placement of backfill or structural foundations is about to occur. At that point we can conduct inspections to assure that the selected foundation elevations for safety-related structures will provide reliable support.

The Sevier shale, where fresh and unweathered, is capable of supporting seismic Category I structural foundations. Competent shale elevations suitable for the support of foundations range from about 1110 to 1155 feet. Compressive strength data presented by the applicant ranged from 1,528 to 11,200 pounds per square inch and averaged 4,973 pounds per square inch. The control and diesel generator

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buildings will be founded on free draining granular fill material placed on rock. Granular fill will be compacted to an average relative density of 85 percent or greater, with a minimum relative density of 80 percent as determined by the American Society for Testing Materials, Standard D 2049. Test fills will be used to confirm the design properties and compaction methods for engineered granular fill. Structures founded on rock and engineered granular fill will have negligible settlement. Settlements will be monitored by the applicant and design provisions will be made to accommodate twice the effective settlement or differential settlement without causing distress to vital pipes and connections.

The evaluation of lateral earth pressure on buried structures for static conditions is based on Coulomb's "wedge of pressure" theory. The coefficient of active earth pressure varies from 0.27 to 0.34 depending on the angle of internal friction of the soil. Similarly, the at rest coefficient varies between 0.4 and 0.3. Dynamic earth pressures will be based on the Westergaard theory and the Mononobe theory.

2.5.5 Slope Stability

The only seismic Category I slopes will be the slopes required for the two ultimate heat sink spray ponds. The ultimate heat sink ponds will be formed by cut and fill operations. The rim of the ponds will be at the plant grade elevation of 1181.5 feet. The ponds will be 16 feet deep with 2.5 feet of freeboard. The ponds will have a nearly impervious liner composed of five feet of clay. Spray headers in the pond will be supported on footings embedded in the clay liner. All pond slopes will be one vertical to three horizontal.

The pond slopes are designed to remain stable under both static conditions and under safe shutdown earthquake of 0.20g conditions. However, as stated in Section 2.5.3.2 of this report, it is our position that the safe shutdown earthquake should be 0.25g. The applicant has agreed with this value. Therefore, the foundation engineering aspects of the spray pond design will have to be reanalyzed. We have asked the applicant to prepare and provide us with the appropriate reanalysis. Our evaluation of the revised analysis will be included in a supplement to the safety evaluation report.

The applicant made a dynamic analysis of Watauga Dam which is upstream from the plant site. Watauga Dam is a 318 foot high rock-fill dam with a central core. The dam was analyzed and found safe against the postulated operating basis earthquake with a peak acceleration of 0.1g. Our consultant the Corps of Engineers reviewed the dynamic analysis and concluded that it is satisfactory.

In summary, subject to the applicant providing an acceptable analysis of the spray pond slopes and based on the information available in the Phipps Bend Preliminary Safety Analysis Report and on our inspection of the excavation, we conclude that the foundation engineering aspects of the proposed power plant will be adequate to meet the requirements of 10 CFR Part 100 and are therefore acceptable.

3.0 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Conformance with General Design Criteria

The applicant has stated that the Phipps Bend plant will be designed, constructed and operated in accordance with the Commission's General Design Criteria as contained in Appendix A to 10 CFR Part 50. On the basis of our review of the documentation supporting this commitment, we have concluded that the proposed facility can be designed, constructed and operated to meet the requirements of the General Design Criteria. Discussions regarding compliance with each criterion are presented in Section 3.1 of the GESSAR Standard Safety Analysis Report and Section 3.1 of the Phipps Bend Preliminary Safety Analysis Report.

3.2 Classification of Structures, Systems and Components

3.2.1 Seismic Classification

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

Structures, systems and components important to safety that will be designed to withstand the effects of a safe shutdown earthquake and remain functional have been identified in an acceptable manner and classified as seismic Category I, in conformance with Regulatory Guide 1.29, "Seismic Design Classification." All other structures, systems and components that may be required for operation of the facility will be designed to other than seismic Category I requirements. Included in this classification are those portions of seismic Category I systems which will not be required to perform a safety function. Our evaluation of the structures, systems and components important to safety within the GESSAR scope of supply that will be designed to withstand the effects of a safe shutdown earthquake and remain functional is presented in Section 3.2.1 of Appendix A to this report.

Three balance-of-plant items classified as seismic Category I structures that do not coincide with the design of the GESSAR are: (1) the emergency service water pumping station, (2) the spray ponds, and (3) the concrete steel lined diesel generator fuel storage tanks. These seismic Category I items replace the cooling water intake structures and the steel diesel generator fuel storage tank design identified in GESSAR.

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

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

3.2.2 System Quality Group Classification

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

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

The applicant has applied the classification system of the American Nuclear Society (Safety Classes 1, 2, 3 and Non-Nuclear Safety), which corresponds to our Quality Groups A, B, C and D identified in Regulatory Guide 1.26, "Quality Group Classifications and Standards," to those fluid containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety.

As noted in Section 3.2.1 of this report, except for the emergency service water pumping station, spray ponds, and diesel generator fuel storage tanks (which the applicant has classified in an acceptable manner in the Phipps Bend Preliminary Safety Analysis Report), all other structures, systems and components important to safety are identified and classified by reference to GESSAR. The fluid system pressure-retaining components important to safety will be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed. These fluid containing components are part of the reactor coolant pressure boundary and other fluid systems to: (1) prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) permit shutdown of the reactor and maintain it in a safe showdown condition, and (3) contain radioactive material. These fluid systems have been classified in an acceptable manner in the Phipps Bend Preliminary Safety Analysis Report in conformance with Regulatory Guide 1.26. Our evaluation of the GESSAR fluid systems classification is presented in Section 3.2.2 of Appendix A to this report.

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

We conclude that fluid systems pressure-retaining components important to safety will be designed, fabricated, erected and tested to quality standards in conformance with the Commission's regulations, the applicable regulatory guide, and industry codes and standards and are acceptable.

3.3 Environmental Loadings

3.3.1 Wind Loadings

Our evaluation of the design for seismic Category I structures that will be exposed to wind forces which are within the scope of the nuclear island design is presented in Section 3.3.1 of Appendix A to this report. The GESSAR design wind speed is 130 miles per hour at 30 feet above grade with a recurrence interval of 100 years. The design wind speed for the proposed Phipps Bend site is 95 miles per hour at 30 feet above plant grade with a recurrence interval of 100 years. This is within the GESSAR envelope. The safety-related balance-of-plant structures will be designed for this 95 miles per hour basic wind determined in accordance with the American Society of Civil Engineers Paper No. 6038, "New Distribution of Extreme Winds in the United States." The design wind loads for the safety-related balance-of-plant structures will be determined by following the same procedures as reviewed for GESSAR which will be in accordance with the American Society of Civil Engineers Paper No. 3269, "Wind Forces on Structure." The design wind loads will be combined with other applicable loads as discussed in Section 3.8 of Appendix A to this report.

3.3.2 Tornado Loadings

All the nuclear island seismic Category I structures will be designed to withstand the effects of the design basis tornado, and all seismic Category I systems and components located within will thereby be protected from its effects. Our evaluation of the design of those structures is presented in Section 3.3.2 of Appendix A to this report.

The design basis tornado specified for GESSAR was selected to meet the most severe tornado conditions listed in Regulatory Guide 1.76, "Design Basis Tornado For Nuclear Power Plants," in that it has a tangential wind velocity of 290 miles per hour and a translational velocity of 70 miles per hour. The pressure drop associated with the tornado is three pounds per square inch in two seconds. Furthermore, an appropriate spectrum of tornado-generated missiles was also postulated.

The procedures that will be used to transform the tornado wind velocity into pressure loadings will be in accordance with the American Society of Civil Engineers Paper No. 3269 except that the pressure will be applied uniformly over the full height of the projected area of the structure and no gust factors will be applied. Those changes result in a more conservative analysis than the American Society of Civil Engineers Paper. Seismic Category I structures will be designed for the pressure drop associated with the design basis tornado which will be treated as a load that varies with time. The tornado missile effects will be determined using procedures discussed in Section 3.5.3 of Appendix A to this report. The total effect of the design basis tornado on seismic Category I structures, systems and components will be determined using an appropriate combination of the effects of wind load, pressure load and missile load. Tornado generated loads will be combined with other loads as discussed in Section 3.8 of Appendix A to this report.

The balance-of-plant seismic Category I items identified in Section 3.2.1 of this report will be designed to the same tornado loadings as utilized for GESSAR and which [illegible]e found acceptable.

All the facility structures that are not to be designed for the tornado effects will be investigated to assure that they will not fail to the extent that they might damage seismic Category I structures and systems. The safety function and structural integrity of seismic Category I structures will thereby be assured.

3.3.3 Snow Loading

The maximum snow and ice load of fifty pounds per square foot selected for the design of safety related structures in the GESSAR nuclear island is adequate for the northern Tennessee area in which the proposed Phipps Bend site is located. Therefore, the design of the nuclear island structures for snow loading will be adequate for the proposed Phipps Bend site.

With regard to the safety-related balance-of-plant structures the applicant has stated that these structures will also be designed to the same snow and ice load criteria as GESSAR. We therefore conclude that the design of these structures for snow loading will be acceptable.

3.3.4 Conclusions

We have concluded that the procedur[illegible] be utilized to determine the loadings on seismic Category I structures in[illegible] the design wind, the design basis tornado and snow specified for the [illegible] acceptable since these procedures provide a conservative basis for eng[illegible]ering [illegible] to assure that the structures will withstand such environmental forces.

The use of these procedures provides reasonable [illegible]e that, in the event of occurrences of the design wind, the design bas[illegible] [illegible]w, the structural integrity of the nuclear island and balance-of-plant seismic Category I structures will not be impaired and, therefore, seismic Category I systems and components located within these structures will be adequately protected and will perform their intended safety functions if needed. Conformance with these procedures is an acceptable basis for satisfying the applicable requirements of Criterion 2 of the General Design Criteria.

3.4 Water Level (Flood) Design Criteria

As discussed in Section 2.4.2 of this report, the applicant has committed to a maximum still water flood level of 1182 to 1183 mean sea level plus coincident wind wave and run up effects. This will increase the design flood elevation approximately five feet above the current design. However, the applicant has also committed to design and construct the Phipps Bend plant to withstand the above stated flood level without loss of safety-related function and is currently evaluating alternate designs for providing the flood protection. We require that the results of the study be submitted for our approval prior to the start of the radiological safety hearing. We will report on our review of the submittal in a supplement to this Safety Evaluation Report.

3.5 Missile Protection

3.5.1 Missile Protection Criteria

The Phipps Bend plant will be designed so that missiles generated by postulated failures of equipment within the facility and from sources outside the facility will not cause or increase the severity of an accident. Our evaluation of the missile protection criteria for GESSAR is presented in Section 3.5 of Appendix A to this report. The following paragraphs therefore discuss only the balance-of-plant criteria.

Safety related systems will be protected against loss of function due to internal missile impact. Rotating components which have the potential of overspeed in excess of design limitations have been considered as potential missile sources. Other potential missile sources include valve bonnets and valve stems. Protection against missiles will be achieved by strategic orientation of components and systems, physical separation, compartmentalization, barriers and equipment design.

We have reviewed the applicant's proposed design criteria and bases for protecting safety-related systems from internal missile impact. We conclude that the design criteria and bases for structures housing essential equipment and for systems being capable of withstanding the effect of internally generated missiles are in conformance with General Design Criterion 4. In addition we conclude that the design criteria and design basis for the central pumping station being capable of withstanding the effects of internally generated missiles are in conformance with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants." Therefore, the design criteria and bases are acceptable.

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3.5.2 Tornado Missiles

The applicant has committed to design the Phipps Bend plant to withstand the impacts of the same spectrum of tornado missiles as presented in GESSAR. We, therefore, conclude that the design of the Phipps Bend plant for protection from postulated tornado generated missiles is acceptable. Our evaluation of the spectrum of tornado generated missiles for GESSAR is presented in Section 3.5.3 of Appendix C to thir report.

3.5.3 Turbine Missiles

Historically, there have been a number of failures in main turbogenerators used in fossil and nuclear power plants. Some have resulted in overspeeds which led to extensive damage where turbine fragments have been ejected for considerable distances. Analysis of the historical experience suggests that a destructive overspeed condition may occur with a probability of the order of 4×10^{-5} per turbine year.

Because turbine missiles from destructive overspeed can be energetic, damage to safety-related structures could occur if such structures were in their path. One means of minimizing the risks associated with a destructive overspeed is to orient the turbine so as to be peninsular relative to safety-related structures.

The Phipps Bend units are oriented in this fashion. They are interactive in that missiles from one unit could strike safety-related structures in another. However, the risks associated with turbine failures are significantly reduced as compared with designs which are non-peninsular. The Phipps Bend turbine generator units also include three-foot thick concrete walls, for radiological shielding purposes, which extend nearly to the top of the low pressure turbine hoods. This feature would tend to reduce the low trajectory missile energies.

The applicant's proposed turbine rotors are different from those analyzed by the staff in past reviews. Instead of the usual disc-shrunk-on-shaft concept, the rotor design for Phipps Bend consists of a series of discs welded together near the rim into a "pancake stack" configuration. The rotor is, in effect, monolithic, and thus the missile fragments could involve more than a single disc. Therefore, we are currently studying the spectrum of potential missiles that could be generated.

Although this study is not completed, based on the inherent protection afforded by peninsular orientation of the turbine and on preliminary evaluations of the probability that any type of missile might cause damage to safety-related structures, we conclude that the current plant orientation is acceptable. The results of the study of potential missiles may confirm the applicant's belief that the probability of potential interactions of turbine missiles from one plant on the adjacent units is so unlikely that these potential interactions need not be considered further. If not, we conclude that other provisions, such as increased testing of turbine valves, can

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be invoked to reduce risks associated with turbine missiles to acceptably low levels. On conclusion of our study, we will reevaluate the impact of turbine missiles on this facility, and will determine whether additional measures are required beyond that already proposed.

3.5.4 Barrier Design Procedures

The analysis of seismic Category I structures, shields and barriers to determine the effect of missile impact will be accomplished as outlined in Section 3.5.4 of GESSAR.

We have reviewed the procedures proposed to be utilized for the nuclear island and the balance-of-plant safety-related structures. The use of these procedures to determine the effects and loadings on seismic Category I structures and missile barriers induced by design basis missiles selected for the plant are acceptable on the following basis. These procedures have previously been reviewed and accepted as outlined in Section 3.5 of Appendix A to this report and provide a conservative basis for engineering design to assure that the structure or barriers are adequately protected against the effects of missile impacts.

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

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

3.6.1 Postulated Rupture of Piping Inside Containment

Our safety evaluation of the criteria and methods for protection against the effects of postulated ruptures of piping inside containment is presented in Section 3.6 of Appendix A to this report.

3.6.2 Postulated Rupture of Piping Outside Containment

Our safety evaluation of the GESSAR criteria and methods for protection against the effects of postulated ruptures of piping outside containment involving piping within the scope of the nuclear island is presented in Section 3.6 of Appendix A to this report. The applicant references GESSAR criteria and methods which we found acceptable for the balance-of-plant design.

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We, therefore, conclude that the design criteria and bases utilized for the protection against the dynamic effects of pipe failures outside the containment are acceptable.

3.7 Seismic Design

3.7.1 Seismic Input

Sections 2.5.3.2 and 2.5.3.3 of this report present our evaluation of the safe shutdown and operating basis earthquake acceleration values for the proposed Phipps Bend site.

The applicant has documented that the seismic design response spectra (for the safe shutdown and operating basis earthquakes) to be applied in the design of seismic Category I structures, systems and components both for the nuclear island and for the balance-of-plant safety related structures will comply with the recommendation of Regulatory Guide 1.60, "Design Response Spectra for Nuclear Power Plants." The specific percentages of critical damping values to be used in the seismic analysis of seismic Category I structures, systems and components are in conformance with Regulatory Guide 1.61, "Damping Values for Seismic Analysis of Nuclear Power Plants." The synthetic time history to be used for seismic design of seismic Category I plant structures, systems and components will be adjusted in amplitude and frequency content to obtain response spectra that envelop the design response spectra specified for the site.

As stated in Section 2.5.3.2 of this report the site seismic design criteria are within the envelope specified in GESSAR for the nuclear island

Conformance with the recommendations of Regulatory Guides 1.60 and 1.61 assures that the seismic inputs to seismic Category I structures, systems, and components will be adequately defined so as to form a conservative basis for the design of such structures systems and components to withstand seismic loadings.

3.7.2 Seismic System Analysis

The applicant refers to GESSAR for seismic analyses. Our evaluation is presented in Section 3.7.3 of Appendix A to this report.

GESSAR specifies six conditions that must be met for each plant utilizing the GESSAR design to ensure the seismic design adequacy of that nuclear power facility. These conditions are as follows:

(1) The maximum ground acceleration at zero period of the site design response spectra must be less than or equal to 0.3g for the safe shutdown earthquake, and 0.15g for the operating basis earthquake.

(2) The site design response spectra must be less than or equal to those given in Regulatory Guide 1.60, normalized to the maximum ground accelerations given in (1) above.

(3) There must be no potential for liquefaction at the plant site due to the safe shutdown or operating basis earthquake.

(4) There must be no potential for fault displacements near or underneath the plant foundation.

(5) The embedment depth of the reactor building must be between 34 to 40 feet (± 0.5 feet excavation tolerance) for soil sites. For sites with shear wave velocities greater than 3500 feet per second, there will be no limitation on embedment depth.

(6) The average shear wave velocity for the top 30 feet of soil must be greater than 500 feet per second.

The Phipps Bend application contains sufficient information that assures that the above six conditions are met.

We have reviewed the information provided and conclude that the seismic system analysis procedures and criteria proposed by the applicant provide an acceptable basis for seismic design.

In addition, we conclude that the seismic analysis methods and procedures proposed by the applicant provide an acceptable basis for system seismic design.

3.7.3 Seismic Instrumentation System

The applicant has documented that it will install the seismic instrumentation specified in GESSAR and evaluated in Subsection 3.7.4 of Appendix A to this report. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with Regulatory Guide 1.12, "Instrumentation for Earthquakes" (Revision 1).

We conclude that the seismic instrumentation program proposed by the applicant is acceptable.

3.8 Design of Seismic Category I Structures

Our evaluation of the design of seismic Category I structures which are part of the nuclear island design scope is presented in Section 3.8 of Appendix A to this report.

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Therefore, the discussion below is limited to those seismic Category I structures, systems, and components which are within the scope of the balance-of-plant.

3.8.1 Essential Service Water System

The essential service water system as evaluated in Section 9.2.1 of Appendix A to this report and modified by Section 9.2.1 of this report will be designed to seismic Category I requirements. The design consists primarily of a seismic Category I pumping station which houses the essential service water system pumps, the high pressure core spray cooling water circulation pumps, and all necessary piping, valves, strainers and electrical switchgear for these systems.

3.8.2 Spray Ponds

Two seismic Category I spray ponds will serve the essential service water systems. Our evaluation of the design of these ponds is presented in Sections 2.4.3 and 9.2.2 of this report.

3.8.3 Concrete Specifications

We require that the construction concrete specifications meet the American National Standards Institute N45.2.5-74, "Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," and the American Society of Mechanical Engineers as well as Boiler and Pressure Code, Section III, Division 2 (ACI-359), "Proposed Standard Code for Concrete Reactor Vessels and Containment," issued for interim trial use and comment, April 1973.

The applicant has proposed to use its concrete construction specification referred to as "G-2". The applicant submitted a comparison of the "G-2" specification with America National Standards Institute N45.2.5-74 and the American Society of Mechanical Engineers/American Concrete Institute 359 in order to support the use of "G-2" during construction of the Phipps Bend plant. We evaluated the comparison and, on the basis that it will provide for an equivalent degree of conservatism, we find the applicant's proposal to use the "G-2" specification for concrete construction acceptable for the Phipps Bend plant.

3.8.4 Conclusions

The proposed use by the applicant of criteria defined in GESSAR for applicable codes, standards and specifications, loads and loading combinations, design and analysis procedures, the structural acceptance criteria, and materials quality control and special construction techniques provides reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within the structures, these structures can be expected to withstand the specified design condition without impairment of their structural integrity or their safety function.

We have concluded that conformance with these criteria, codes, specifications, and standards in designing seismic Category I structures constitutes an acceptable basis for satisfying the requirements of Criteria 2 and 4 of the General Design Criteria.

3.9 Mechanical Systems and Components

The applicant in Section 3.9 of the Phipps Bend Preliminary Safety Analysis Report references GESSAR for all mechanical systems and equipment which includes the balance of plant American Society of Mechanical Engineers Code Class 2 and 3 components. Our evaluation of the criteria, testing procedures and dynamic analysis to be employed to assure structural and functional integrity of piping systems, mechanical equipment and reactor internals is presented in Section 3.9 of Appendix A of this report.

3.10 Seismic Qualification of Seismic Category I Instrumentation and Electric Equipment

Our evaluation of the design of seismic Category I instrumentation and electrical equipment within the scope of the nuclear island design is presented in Section 3.10 of Appendices A and C to this report. The applicant has indicated in Section 3.10.2 of the Phipps Bend Preliminary Safety Analysis Report that its seismic design criteria for the balance-of-plant instrumentation and control and electrical equipment will comply with all applicable design criteria set forth in Section 3.10 of GESSAR Standard Safety Analysis Report. Our evaluation for equipment within the scope of the balance-of-plant is presented in Section 7.8 of this report.

3.11 Environmental Design of Electrical Equipment

Our evaluation for equipment within the scope of the nuclear island design is presented in Section 3.11 of Appendix A and Appendix C to this report. Our evaluation for equipment within the scope of the balance-of-plant is presented in Section 7.8 of this report.

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4.0 REACTOR

Our evaluation of the reactor is presented in Section 4.0 of Appendix A and Appendix C of this report.

5.0 REACTOR COOLANT SYSTEM

5.1 Introduction

Our evaluation of the reactor coolant system is presented in Section 5.0 of Appendix A, Appendix C and Appendix D to this report. Therefore, the discussion below is related to the balance-of-plant.

5.2 Loose Parts Monitoring System

The applicant in its initial design did not provide for installation of a loose parts monitor in the Phipps Bend plant. However, we believe that the state-of-the-art is such that the development of this item for the Phipps Bend plant is practical. We therefore informed the Tennessee Valley Authority that we would require a commitment for installation of a loose parts monitor in each unit prior to full power operation. In Amendment No. 12 to the Phipps Bend Preliminary Safety Analysis Report, the Tennessee Valley Authority provided a commitment to install a loose parts monitor in each unit prior to full power operation.

As a result of the acceptable commitment, we consider this item resolved.

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6.0 ENGINEERED SAFETY FEATURES

6.1 General

Engineered safety features are those systems which will be provided for the protection of the public and station personnel against the postulated release to the environment of radioactive products from the nuclear plant, particularly as the result of the loss-of-coolant accidents. Our evaluation of the GESSAR engineered safety features is presented in Section 6.0 of Appendix A to this report. In the following sections we therefore discuss only items involving the engineered safety features which are in the balance-of-plant scope, items in GESSAR which the applicant took exception to, or items which are conditions in the GESSAR Preliminary Design Approval.

6.2 Containment Systems

6.2.1 Containment Functional Design

Our evaluation of the containment functional design is presented in Section 6.2.1 of Appendix A to this report.

In Section 6.2.1.2 of the Phipps Bend Preliminary Safety Analysis Report, the applicant increased the total containment leak rate (technical specification leak rate) to 0.5 percent per 24 hours from a 0.3 percent per 24 hours as indicated in GESSAR Standard Safety Analysis Report. Since as outlined in Section 15.0 of this report, the site doses for this value are within 10 CFR Part 100 guidelines for various postulated accidents we find this change acceptable.

6.2.2 Dynamic Loads on Structures Located In and Above the Suppression Pool

As is discussed in Section 6.2.1.9 of Appendix A to this report, several phenomena were identified in our GESSAR review of the Mark III containment that could result in dynamic loading of structures located in and above the suppression pool. They are related to (1) pool response to the loss-of-coolant accident, and (2) pool response due to relief valve operation, generally associated with plant transient conditions.

We sent a letter to the General Electric Company dated October 24, 1975 which provided a set of design criteria developed by us which we would find acceptable. The General Electric Company subsequently agreed to adopt our criteria, with two exceptions. The first exception pertains to the dynamic loads generated during the clearing of the safety/relief valve discharge lines. The second exception is related to impact loads on pipes and structures at elevations between 17 and 19.5 feet above the suppression pool. The General Electric Company proposed a load of 30 pounds per square inch in this region which we did not consider adequate. Both of these exceptions remained unresolved at the time of issuance of the GESSAR Safety Evaluation Report and therefore the GESSAR Preliminary Design Approval was conditioned.

With regard to the first exception, we stated in Section 6.2.1.9 of Appendix C to this report that this matter is considered resolved for GESSAR, provided that an in-plant prototype test to verify the relief valve loads is conducted by the utility applicant referencing GESSAR. The Tennessee Valley Authority has adopted the GESSAR relief valve load resolution, and has agreed in Amendment 12 to the Phipps Bend Preliminary Safety Analysis Report to perform an in-plant test and submit the results for our review provided that the test had not been performed in a prototypical containment of the Phipps Bend type prior to full power operation of the Phipps Bend plant. We therefore consider this matter resolved.

With regard to the second exception it is our position that (1) the impact loads on beams should be 115 pounds per square inch up to 18 feet above the suppression pool and then the load should be linearly decreased to 15 pounds per square inch at 19 feet above the suppression pool, and (2) the impact loads on pipe should be 60 pounds per square inch up to 18 feet above the suppression pool and then the load should also be decreased linearly to 15 pounds per square inch at 15 feet above the suppression pool. The Tennessee Valley Authority, in a letter dated March 31, 1977, provided a commitment to design the Phipps Bend plant to the load profile outlined above or to relocate the relevant structures to elevations greater than 19.5 feet above the suppression pool. We find this commitment to be acceptable and we therefore consider this matter resolved for the Phipps Bend plant.

6.2.3 Secondary Containment Functional Design

In our evaluation of the containment bypass sealing systems and the main steam line positive sealing system outlined in Section 6.2.3 of Appendix C to this report, we indicated that General Electric has provided sufficient information on these systems for us to conclude that they are acceptable for the construction permit stage of the review. In addition, as indicated in Section 6.2.3 of Appendix C to this report, the General Electric Company has agreed to provide us with a topical report containing more detailed information on these systems. We will report the results of our review of this report at the final design approval stage of the GESSAR and will require that the results of our review be incorporated into the accident analysis calculations performed at the operating license stage of review for the Phipps Bend plant application. The applicant has committed to accept this resolution. We find this commitment acceptable.

6.2.4 Combustible Gas Control

Our evaluation of the GESSAR combustible gas control provisions is discussed in Section 6.2.5 of Appendix A to this report. We concluded that the design of the combustible gas control systems conforms to the applicable regulations, regulatory guides and staff positions and was acceptable. We stated, however, that we would require the General Electric Company to submit results of their prototype test program for the thermal recombiner as they become available.

Except for the vendor and capacity of the thermal-type combustible gas recombiners the applicant references GESSAR for the combustible gas control systems. The applicant has identified the Westinghouse Electric Corporation and Atomics International as two additional potential suppliers of recombiners.

However, if the applicant selects a supplier for the recombiners other than the General Electric Company which we have not previously approved, we will require the applicant to submit the results of a prototype test program. Also, we would require that the applicant describe the provisions, if any, made to share the hydrogen recombiner between the Phipps Bend plant units.

Our prior review experience for combustible gas control systems is that effective hydrogen control systems can be designed to conform to the requirements of Criteria 41, 42 and 43 of the General Design Criteria and Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident," We therefore conclude that an acceptable system can be provided for combustible gas control following a postulated loss-of-coolant accident. We will review the design of the system during the operating license stage of our review.

6.2.5 Containment Leakage Testing

Our evaluation of the GESSAR containment leakage testing program is described in Section 6.2.6 of Appendix A to this report. We found the GESSAR testing program acceptable.

In Section 6.2.1.4 of the Phipps Bend Preliminary Safety Analysis Report, the applicant took exception to the GESSAR containment leakage testing program. The applicant's exceptions deal primarily with the magnitude of the test pressure to be utilized and the allowable leakage rate for specific tests. In each case where the applicant has taken exception to the GESSAR test program, the revised test pressure is either equal to or more limiting than that specified by GESSAR. In addition, for each case where the allowable leakage rate has been changed by the applicant, the revised value is a more restrictive one than that specified in GESSAR.

We therefore conclude that the Phipps Bend exceptions to the GESSAR containment leakage program are acceptable.

6.3 [illegible]gency Core Cooling System

Our evaluation of the emergency core cooling system is presented in Section 6.3 of Appendix A to this report. Subsequently the General Electric Company informed us that certain calculational errors have been discovered which may effect the performance evaluation of the GESSAR emergency core cooling system (see Section 6.3.2 of

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Appendix D to this report). The resolution of this outstanding item will be reported in a future supplement to the Phipps Bend Safety Evaluation Report.

Although the applicant has referenced GESSAR for the description of the emergency core cooling system test program it is necessary that each applicant referencing GESSAR provide an individual commitment regarding the emergency core cooling system preoperational test program. Based on the information presented in Section 14.1 of the Phipps Bend Preliminary Safety Analysis Report, we conclude that the applicant will comply with Regulatory Guide 1.68, "Preoperational and Startup Program for Water-Cooled Power Reactors," and conclude that the applicant's emergency core cooling system preoperational test program is acceptable at the construction permit stage of review.

6.4 Control Room Habitability Systems

6.4.1 Radiological Protection

The applicant proposes to meet Criterion 19 of the General Design Criteria by the use of concrete shielding and by installing a dual fresh-air inlet system containing redundant deep bed charcoal filters.

As indicated in Section 6.4 of Appendix A to this report, our acceptance of the standard plant control room design for multiple unit sites was potentially sensitive to the location of the dual outside air inlets since a loss-of-coolant accident in one unit can result in the contamination of both inlets of the adjacent unit. We have evaluated control room doses for the Phipps Bend plant assuming a design basis loss-of-coolant accident source term (see Section 15.2.1 of this report for discussion of the loss-of-coolant accident), the Phipps Bend site meteorology, and the exposure of both outside air inlets of one unit to the airborne radioactivity released by the adjacent unit. We find, on the basis of the results of the analysis, that the thyroid and whole body doses to the plant operators for the Phipps Bend plant are within the guidelines of Criterion 19 of the General Design Criteria and are therefore acceptable.

6.4.2 Toxic Gas Protection

The habitability of the control room with respect to potential toxic gas releases has been evaluated by the applicant in accordance with the procedures described in Regulatory Guide 1.78, "Assumptions For Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release." Since a sodium hypochlorite biocide system will be used at the Phipps Bend plant there will be no onsite chlorine hazard. Other chemicals that will be stored onsite are not considered hazardous to control room habitability. A number of toxic chemicals which are shipped on the railroad adjacent to the site have been identified. Analyses of the effects of postulated accidents involving these chemicals indicated that the concentrations of chlorine and acetaldehyde in the control room from the postulated rupture of a railroad tank car could exceed the toxicity limits specified in Regulatory Guide 1.78. Therefore, the applicant has committed to provide detection instrumentation for chlorine and acetaldehyde vapors in the control room air intakes with the capability to automatically isolate the control room upon detection of either chlorine or acetaldehyde. In addition, self-contained breathing apparatus with an air supply sufficient for six hours will be provided. Based on our review of the analytical model and assumptions used in the evaluation, and with the addition of the protection instrumentation for the control room, we conclude that the control room habitability with respect to potential toxic chemical accidents is acceptable.

7.0 INSTRUMENTION AND CONTROLS

7.1 General

The Commission's General Design Criteria, the Institute of Electrical and Electronic Engineers Standards, including the Institute of Electrical and Electronic Engineers Criteria for Protection Systems for Nuclear Power Generating Stations (Institute of Electrical and Electronic Engineers Standard 279-1971), applicable regulatory guides for power reactors, and staff technical positions noted in Table 7-1 of the Standard Review Plan, have been utilized as the bases for evaluating the adequacy of the protection and control systems. This Safety Evaluation Report reflects the results of our review of the Phipps Bend application through Amendment 13.

Since the two unit Phipps Bend plant utilizes the GESSAR design, interface areas must be established at each point where the reference design and the applicant's design meet. This section of the Safety Evaluation Report covers the instrumentation and controls in the balance-of-plant design along with those portions that are in the design and/or supply responsibility of GESSAR but are implemented by the applicant. Therefore, throughout this section when the term "applicant's responsibility" is used, it refers only to the portions in which the applicant is involved in, as discussed above, and not to the entire plant.

7.1.1 Principal Design Criteria

The Phipps Bend plant's principal design criteria were initially found to be incomplete and inadequate. As a result of requesting additional information, the applicant provided a "Codes and Standards Applicability Matrix," Figure 7.1-2(T) in the Phipps Bend Preliminary Safety Analysis Report which identified the principal design criteria to be implemented in those portions of the plant that are the applicant's responsibility. The subject matrix is based on an identical matrix identified as Figure 7.1-2 in the GESSAR Standard Safety Analysis Report. Our evaluation of the GESSAR matrix is contained in Section 7.0 of Appendix A to this report.

The criteria listed in Section 7.1 of the Phipps Bend Preliminary Safety Analysis Report include the applicant's commitment to the General Design Criteria, Institute of Electrical and Electronic Engineers Standards including Standard 279-1971, applicable regulatory guides and branch technical positions.

We believe that this method of referencing the criteria of GESSAR, supplemented by the information in Figure 7.1-2(T) of the Phipps Bend Preliminary Safety Analysis Report, establishes principal design criteria compatibility with GESSAR and at the same time documents the applicant's principal design criteria that satisfy our requirements for safety systems or portions of the safety systems that are in the applicant's area of responsibility.

We have reviewed the principal design criteria documented in the subject matrix and have concluded that the criteria listed are consistent with our requirements for safety systems and are acceptable.

7.1.2 Design Bases

The applicant has referenced GESSAR extensively for the design bases of systems or portions of systems that are its responsibility. Since there was initially a lack of definition of what systems or portions of systems the applicant was responsible for, we were concerned about the acceptability of this referencing approach. Therefore, in response to our request, the applicant provided a list of safety related systems or portions of safety related systems in its scope of supply and the principal design criteria for these systems or portions of systems. The compatibility of these design bases and criteria with GESSAR criteria was demonstrated. As reported in Section 7.1.1 of this report, principal design criteria have been established and accepted. In addition, the applicant modified the Phipps Bend Preliminary Safety Analysis Report to provide more specificity as to which sections of the GESSAR were being referenced for design bases information and committed to adhering to the criteria established in the Phipps Bend Preliminary Safety Analysis Report for integrating these design bases into the design.

We have concluded that this method of referencing design bases information in GESSAR, bounded by the principal design criteria established in the Phipps Bend Preliminary Safety Analysis Report, satisfies our requirements and is therefore acceptable.

7.1.3 Identification of Safety-Related Systems in Applicant's Scope of Responsibility

The applicant identified the following safety systems, or portions of systems, as being in its areas of responsibility:

(1) Reactor protection system,

(2) Essential service water,

(3) Containment and reactor vessel isolation control system,

(4) High pressure core spray,

(5) Reactor core isolation cooling, and

(6) Diesel generator fuel oil storage and transfer system.

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Further discussion describing the scope of responsibility for design and interface information is reported in Sections 7.2 through 7.6 of this report.

7.2 Reactor Protection System

The overall design of the reactor protection system is described in Section 7.2 of the GESSAR Standard Safety Analysis Report. The reactor protection system portions provided by the applicant consist of trip inputs from turbine stop valves, turbine control valves, and turbine first stage pressure instrumentation.

There are four separate and independent devices provided to generate each of these trip inputs. The separation of the instruments and connecting wiring shall be in accordance with the requirements of Electrical and Electronic Engineers Standard 279-1971. The number and method of separation for this instrumentation is consistent with the four-channel design described for the reactor protection system in GESSAR.

In response to our concerns regarding the capability of the Brown Boveri turbine generator instrumentation to provide the reactor protection system inputs within the times required in GESSAR, the design was modified to include the following additional requirements:

(1) The turbine stop valve closure switches will open and generate the signal within 10 milliseconds after the setpoint is reached.

(2) The control valve fast closure signal will be generated within 30 milliseconds of the start of a fast closure.

We have reviewed these additional requirements and have determined that they are identical to the requirements of GESSAR and are therefore acceptable.

The applicant has defined the interface requirements for the reactor protection system as follows:

(1) The instruments and switches to be used to generate the reactor protection system inputs associated with the turbine generator will be provided by the applicant.

(2) The applicant will provide the interconnecting wiring between the reactor protection system input devices noted in (1) and the terminal cabinets in the control building. The applicant's design implementation responsibility ends at these terminal cabinets.

(3) The applicant has identified the channel to sensor to cabinet assignments for all the reactor protection system inputs in his scope of responsibility.

We have reviewed this additional information, and have concluded that the applicant has identified its scope of responsibility for implementing this GESSAR design and the interfaces with the nuclear island. Based on this we conclude that the design meets our requirements and is acceptable.

7.3 Engineered Safety Features Actuation Systems

The overall design of the engineered safety features actuation systems is described in Section 7.3 of GESSAR. The portions of the engineered safety features actuation systems that are the applicant's responsibility are as follows:

(1) Essential Service Water System

The essential service water system in the balance-of-plant scope will consist of instrumentation and controls on that portion of the system that provides water from the ultimate heat sink (spray ponds) for safe shutdown. We have reviewed the information for the portion of the essential service water system in the applicant's scope of responsibility, including the logic diagrams and flow diagrams, and have concluded that the design of instrumentation and controls provided by the applicant for the essential service water is consistent with the three divisional systems established as part of GESSAR.

The portion of the essential service water system in the applicant's scope of responsibility will consist of the essential service water system pumps, associated switchgear, pump instrumentation and controls and spray pond temperature and level instrumentation. The interfaces between this equipment and GESSAR will be as follows:

(a) The applicant's instrumentation and control field wiring interfaces will be in the nuclear island at terminal cabinets in the control room and at the remote shutdown panel. (See Section 7.5 of this report for a discussion of the safety related display instrumentation.)

(b) The controls for the essential service water equipment will be located on the emergency core cooling system benchboard in the nuclear island control room and on local panels in the applicant's essential service water system pumping station.

(c) The controls and the layout of the emergency core cooling system benchboard are the applicant's responsibility, while overall responsibility of the control room complex is in the nuclear island.

(d) The design of the remote shutdown panel is in the nuclear island scope. This is consistent with the description in GESSAR. The applicant will provide the essential service water system instrumentation and controls for Division I in this panel.

The applicant has provided the divisional assignments of instrumentation and indicators for the essential service water system and the termination cabinet assignment for these divisions in the nuclear island.

On the basis of our review of the information, the applicant's commitments to the GESSAR design basis and principal design criteria, and the identification of the applicant's responsibilities for implementing the interfaces with the nuclear island, we conclude that the applicant's portion of the essential system design meets our requirements and is therefore acceptable.

(2) Containment and Reactor Vessel Isolation Control System

The applicant has identified three functions that input to the containment and reactor vessel isolation control systems from portions of the plant that are in its area of responsibility. These three functions are:

(a) Low main condenser vacuum, generated by four pressure transmitters and provided as part of the GESSAR scope of supply.

(b) Main steam line low pressure, generated by four pressure transmitters and provided as part of the GESSAR scope of supply.

(c) Main steam line space-high temperature, generated by 36 temperature sensors, procured by the applicant to GESSAR specifications.

The applicant has identified the interfaces as follows:

(a) The applicant's instrumentation field wiring interfaces will be on local panels in the turbine building and the termination cabinet in the nuclear island.

(b) The wiring of the instruments to the termination cabinets will be the applicant's responsibility while the overall responsibility of the control room complex is within the GESSAR scope.

The applicant has provided the divisional assignment of instrumentation for the containment and reactor vessel isolation control system and the termination cabinet assignment for these divisions in GESSAR.

On the basis of our review of the information on the functional inputs to the containment and the reactor isolation control system (which is in the GESSAR scope), and the identification of the applicant's responsibility for implementing the interfaces with the nuclear island, we conclude that the design of the inputs to the containment and reactor isolation control system meets our requirements and that the design is therefore acceptable.

(3) High Pressure Core Spray and Reactor Core Isolation Cooling Systems

Level instrumentation will be provided for monitoring the condensate storage tank water level. These safety system inputs will monitor the status of the primary water source for high pressure core spray and reactor core isolation cooling systems and will enable the transfer logic (GESSAR scope) for the high pressure core spray and reactor core isolation cooling systems to switch to the suppression pool when the primary water source (condensate storage tank) reaches a low level setting.

The level instrumentation specification and procurement will be within GESSAR scope. The interfaces are as follows:

(a) The applicant's instrumentation field wiring interfaces will be on local panels in the vicinity of the condensate storage tank and termination cabinets in the nuclear island.

(b) The wiring of the instruments to the termination cabinets will be in the applicant's area of responsibility while the overall responsibility of the control room complex is within the GESSAR scope.

The applicant has provided the divisional assignments of instrumentation for the high pressure core spray and reactor core isolation cooling systems and the termination cabinet assignment for these divisions in the nuclear island.

On the basis of our review of the submitted information, the applicant's commitment to the GESSAR design basis and principal design criteria, and the identification of the applicant's responsibilities for implementing the interfaces with GESSAR, we conclude that the design of the applicant's portion of the high pressure core spray system and reactor core isolation system meets our requirements and that the design is therefore acceptable.

(4) Diesel Generator Fuel Oil Storage and Transfer System

The portion of the diesel generator fuel oil storage and transfer system that is safety related and supplied by the applicant consists of the seven day tank, day tank supply pump (transfer pump) and motors, associated switchgear, instruments and controls. The interfaces with the nuclear island are as follows:

(a) The applicant's instrumentation and control field wiring will interface with the nuclear island at terminal cabinets in the control building and at the nuclear island remote shutdown panel. (See Section 7.5 of this report for a discussion of the safety related display instrumentation.)

(b) The controls for the diesel generator fuel oil storage and transfer systems equipment will be located on the long term response panel in the nuclear

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island and on a local panel in the diesel generator building.

(c) The layout of the long term response panel is the applicant's responsibility, as is the design and supply of the diesel generator fuel oil storage and transfer system controls.

(d) Overall responsibility of the control room complex is within the GESSAR scope. The design of the remote shutdown panel is also within the GESSAR scope.

The applicant has provided the divisional assignment of instrumentation and control for the diesel generator fuel oil storage and transfer system and the termination cabinet assignment for these divisions in the nuclear island.

On the basis of our review of the information provided, the applicant's commitment to the GESSAR design basis and the identification of the applicant's responsibility for implementing the interfaces with the nuclear island, we conclude that the design of the applicant's portion of the diesel generator fuel oil storage and transfer system meets our requirements and that the design is therefore acceptable.

7.4 Safe Shutdown Systems

The applicant identified the essential service water system and the reactor core isolation cooling as being the safe shutdown systems in their scope of supply. Our evaluation of these systems is included in Section 7.3 of this report.

7.5 Safety Related and Power Generation Display Instrumentation

The applicant identified the essential service water system and diesel fuel oil storage and transfer system within its scope for which display information will be provided in the control room.

The functions for these systems will be displayed as follows:

(1) Essential Service Water System

The following parameters will be monitored by the Nuclenet 1000 cathode ray tube display system:

(a) Pump head pressure
(b) Pump flow
(c) Pump bearing temperature
(d) Motor bearing temperature
(e) Motor winding temperature
(f) Motor current

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(g) Spray pond level

(h) Differential pressure across traveling screen

The following parameters will have hardwired indicators mounted on the emergency core cooling system panel (H13-P601) in the main control room:

(a) Pump flow

(b) Pump well temperature

(c) Spray pond level

(d) Spray pond sprayer bypass valve position

All parameters monitored by the cathode ray tube will have cathode ray tube alarm displays. In addition to these alarms, the following parameters will have hardwired annunciator windows on the emergency core cooling system panel (H13-P601):

(a) Pump head pressure low

(b) Spray pond level abnormal

(c) Spray pond temperature abnormal

(2) Diesel Generator Fuel Oil Storage and Transfer System

The following parameters will be monitored by the Nuclenet 1000 cathode ray tube display system:

(a) Seven day storage tank level

(b) Transfer pump flow

(c) Day tank level

The following parameters will have hardwired indicators on the long term response panel (H13-P870):

(a) Seven day storage tank level

(b) Transfer pump flow

(c) Day tank level

All of the parameters monitored by the cathode ray tube will have cathode ray tube alarm displays. In addition to the alarms the following parameters will have hardwired annunciator windows on the long term response panel (H13-P570):

(a) Seven day storage tank low level

(b) Transfer pump flow abnormal

(c) Day tank level low

(d) Day tank level high

(3) Remote Shutdown Panel Displays

(a) Essential service water system pump flow A division 1
(b) Essential service water system pump flow B division 1
(c) Spray bypass valve position division 1
(d) Screen wash pump flow division 1
(e) Essential service water system pump A head pressure low division 1

We have reviewed the information for that portion of the safety related and power generation display instrumentation identified by the applicant as being in its scope of responsibility. This included the information discussed in Section 7.3 of this report on a system basis and the descriptive information provided in Section 7.5 of the Phipps Bend Preliminary Safety Analysis Report. On the bas s of our review of this information, the applicant's commitment to the GESSAR design bases, and the principal design criteria, we have concluded that the applicant's portion of the safety related display instrumentation meets our requirements and is acceptable.

In response to our request for interface information, the applicant amended the Phipps Bend Preliminary Safety Analysis Report to further define its scope of responsibility for safety systems or portions of safety systems. We have reviewed this additional information and have concluded that the sensor and indicator divisional assignment and division to the nuclear island terminal cabinet correlation for both the essential service water and the diesel generator fuel oil storage and transfer systems is adequate to define the applicant's scope of responsibility and the interfaces with the nuclear island and is therefore :ceptable.

7.6 Other Instrumentation Systems Required for Safety

During the review process we identified various systems that, though designed by the General Electric Company and described in GESSAR, will be located in areas designed by the applicant. Some of these systems, though not required for automatic reactor protection system or engineered safety facilities functions, will be powered from the reactor protection system power buses in the GESSAR. To date, the following systems (or portions thereof) have been identified as being in this category: the process radiation monitoring, the area radiation monitoring, and the carbon bed vault radiation monitoring subsystem. In cases such as these, we are concerned that separation (as in the case where these devices are located in a non-Category I structure) isolation may be compromised. This concern has been discussed with the applicant and it has stated that it is not possible to identify and describe all the items that fall in this category at this stage of the design. However, the applicant has provided a commitment that the isolation requirements for circuits that are treated as non-Class IE in the balance-of-plant (including all equipment located in non-Category I structures) will be in accordance with the GESSAR Standard Safety Analysis Report Section 8.3.1.1.4 which states that the isolation requirements will conform to Regulatory Guide 1.75, "Physical Independence of Electric Systems" (Revision 1), or the applicant will

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provide an alternate design which is acceptable to us. We consider this approach acceptable at the construction permit stage of the review.

7.7 Control Systems Not Required for Safety

We have reviewed the non-safety control systems that will be provided by the applicant such as the turbine generator pressure regulator and steam bypass control systems. We have found that these control systems are similar to those in other previously accepted boiling water reactor plants. In addition, the Phipps Bend Preliminary Safety Analysis Report description of these systems is consistent with the GESSAR description of these systems. We have concluded that the description of the turbine generator pressure regulator and steam bypass control systems described therein have been acceptably classified as non-safety related systems and, consistent with GESSAR, that failures in these control systems will not adversely affect the safety functions of the plant safety systems or lead to plant conditions more severe than those for which the safety systems are designed to protect against. We conclude that these control and instrumentation systems satisfy our requirements and are acceptable.

7.7.1 Turbine Overspeed Protection

We have reviewed the applicant's proposed turbine overspeed protection system. The overspeed protection system will be comprised of redundant mechanical, hydraulic and electro-hydraulic channels configured to ensure that no single failure would prevent initiation of the turbine overspeed function. The overspeed protection system has provisions for periodic testing while the plant is at power.

We have concluded that the provisions for turbine overspeed protection to perform its function are acceptable.

7.8 Environmental Qualification of Equipment Required for Safety

The applicant referenced GESSAR for information on environmental qualification of equipment required for safety. Due to the lack of information in this reference, we were unable to determine the acceptability of the equipment required for safety that will be supplied by the applicant. In response to our request for additional information, the applicant amended the Phipps Bend Preliminary Safety Analysis Report to include the following additional information.

(1) A list identifying instrumentation, controls and electrical equipment required for safety that will be supplied by the applicant was added to Section 3.0 of the Phipps Bend Preliminary Safety Analysis Report.

(2) A statement was included in Sections 3.10 and 3.11 of the Phipps Bend Preliminary Safety Analysis Report that the applicant will provide the seismic and environmental qualification methods and procedures for instrumentation, controls and

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electrical equipment required for safety that will be supplied by the applicant. This information will be provided for staff review and approval in the post-construction permit period prior to the submittal of the Final Safety Analysis Report:

We have reviewed this additional information and the applicant's commitment and have concluded that it meets our requirements and is acceptable.

7.9 Separation Criteria for Safety Related Electrical Equipment

The applicant amended Section 3.12 of the Phipps Bend Preliminary Safety Analysis Report to include the requirement that it is responsible to satisfy in its design the recommendations of Regulatory Guide 1.75 for separation criteria for instrumentation and electrical equipment.

We find this commitment acceptable for the construction permit stage of the review.

8.0 ELECTRIC POWER

8.1 General

General Design Criteria 17 and 18, Institute of Electrical and Electronic Engineers Criteria for Class IE Electric Systems for Nuclear Power Generating Stations (Institute of Electrical and Electronic Engineers Standard 308-1971) and applicable regulatory guides including Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," Regulatory Guide 1.9, "Selection of Diesel G rator Set Capacity for Standby Power Supplies," Regulatory Guide 1.32, "Use of IEEE Std 308-1971, 'Criteria for Class IE* Electric Systems for Nuclear Power Generating Stations,'" Regulatory Guide 1.41, "Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Assignments," Regulatory Guide 1.75, "Physical Independence of Electric Systems" (Revision 1), and staff technical positions noted in Table 8-1 of th standard Review Plan have been utilized as the bases for evaluating the adequacy of the electric power systems. This section of the safety evaluation report reflects the results of our review of the Phipps Bend Preliminary Safety Analysis Report through Amendment 13.

Since the two unit Phipps Bend plant utilizes the GESSAR design, the interface areas must be established at each point where the reference design and the applicant's design meet. This section of the Safety Evaluation Report covers the portions of the plant that will be designed and supplied by the applicant along with those portions that, while in the GESSAR design and/or supply scope, will be implemented by the applicant. Therefore, throughout this section when the term "applicant's responsibility" is used, it reflects only the portions in which the applicant is involved in as discussed above and not to the entire plant.

8.1.1 Principal Design Criteria

The applicant provided in tabular form the principal design criteria to be implemented in their portions of the electrical design. Tables 8.1.1-(T), 8.1.2-(T), 8.1.3-(T), and 8.1.4-(T) of the Phipps Bend Preliminary Safety Analysis Report are based on identical tables in the GESSAR Standard Safety Analysis Report, and were incorporated into the Phipps Bend Preliminary Safety Analysis Report. In addition, these tables were supplemented to demonstrate applicability of our branch technical positions.

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*Safety classification of the electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling and containment and reactor heat removal.

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The criteria listed include the applicant's commitment to the General Design Criteria, Institute of Electrical and Electronic Engineers Standards including "Institute of Electrical and Electronic Engineers Criteria for Protection Systems for Nuclear Power Generating Stations" (Institute of Electrical and Electrical and Electronic Engineers Standard 279-1971), "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations" (Institute of Electric and Electronic Engineers Standard 308-1971), other applicable Institute of Electrical and Electronic Engineers Standards, regulatory guides and branch technical positions.

We believe that this method of presenting the criteria of GESSAR, supplemented by the branch technical positions, establishes principal design criteria compatibility with the GESSAR and at the same time documents the applicant's principal design criteria that satisfy the Commission's requirements for safety systems that are in the applicant's areas of responsibility (as defined in Section 8.1 of this report).

We have reviewed the principal design criteria documented in these criteria tables and have concluded that the criteria listed are consistent with our requirements for safety systems and are acceptable.

8.1.2 Design Bases

The applicant has referenced GESSAR extensively for the design bases of systems or portions of systems that are the applicant's responsibility (as defined in Section 8.1 of this report). The applicant identified in the Phipps Bend Preliminary Safety Analysis Report the systems and the portions of the systems that are in his area of responsibility. The capability of these design basis and criteria with GESSAR was demonstrated. As outlined in Section 8.1.1 of this report, the principal design criteria for the systems have been established and accepted by us.

We believe that this method of referencing design bases in GESSAR, bounded by the principal design criteria established in the Phipps Bend Preliminary Safety Analysis Report, establishes the applicant's design bases to satisfy our requirements for safety systems or portions of safety systems that are the applicant's responsibility. We have reviewed this information and have concluded it meets our requirements and is acceptable.

8.1.3 Identification of Safety-Related Systems in Applicant's Scope of Responsibility

The applicant amended the Phipps Bend Preliminary Safety Analysis Report to provide additional information defining the responsibility between the applicant's scope and that of GESSAR. This information includes the following:

(1) Figure 8.1-6 (T), "Interface Table for A-C and D-C Auxiliary Power System," was added to provide a correlation to the GESSAR, Standard Safety Analysis Report Figure 8.3.1 Interfaces.

(2) The Phipps Bend Preliminary Safety Analysis Report was amended to clearly specify that the onsite power system, including the diesel generators, is in the GESSAR scope. Selected applicant designed loads will be fed from the onsite power system. These loads have been identified in Figure 8.1-6(T).

We have reviewed this additional information and have concluded it meets our requirements and is acceptable.

8.2 Offsite Power System

The Phipps Bend Nuclear Plant, Units No. 1 and No. 2 will feed power into a 500 kilovolt and a 161 kilovolt transmission system which will ultimately tie into the Tennessee Valley Authority utility grid system. The station's two main generators will provide power at 24 kilovolts through load break switches and isolated phase buses to the main transformers. Individual main transformers for Units 1 and 2 will be connected to 500 kilovolt and 161 kilovolt switchyards, respectively, utilizing a modified breaker and a half bus arrangement. The two switchyards will be connected electrically through an intertie transformer bank. Five 161 kilovolt and three 500 kilovolt lines are proposed for connecting the switchyards to the Tennessee Valley Authority utility grid.

The five 161 kilovolt lines that will terminate at the Phipps Bend plant will consist of one line to the Sullivan 500 kilovolt substation and four lines to the John Sevier Steam Plant. The three 500 kilovolt lines that will terminate at the Phipps Bend plant will consist of two lines to the Sullivan and one line to the Volunteer 500 kilovolt substations. The 161 kilovolt line to the Sullivan substation will be underbuilt on the towers of the Phipps Bend-Sullivan No.2 500 kilovolt line for a short distance at Phipps Bend. The Phipps Bend-John Sevier No.1 and No.2 161 kilovolt lines will be underbuilt on the towers of the Phipps Bend-Volunteer 500 kilovolt line with the terminal line sections at the Phipps Bend plant and John Sevier on double circuit 161 kilovolt towers. Similarly the John Sevier No.3 and No.4 161 kilovolt lines will be underbuilt on the towers of the Phipps Bend-Sullivan 161 kilovolt line No.1. The Phipps Bend-Sullivan 161 kilovolt line will cross under Phipps Bend-Volunteer and Phipps Bend-Sullivan No.1 500 kilovolt lines and the John Sevier No. 1 161 kilovolt line near the switchyard at Phipps Bend. The Phipps Bend-Volunteer and the Phipps Bend-Sullivan No. 1 500 kilovolt lines will be on common right of way but will be separated to minimize the possibility of simultaneous outages of the 161 kilovolt underbuilt circuits.

During the course of our review, we were concerned about the susceptibility of the offsite power system to failures and, as a result of requesting additional information, the applicant stated that detailed information on the separation, crossings and tower heights cannot be supplied at this stage of the design. However, they reaffirmed their commitment that the offsite power lines will be designed to preclude the likelihood that a tower falling or line breaking could simultaneously affect both circuits, and the design will conform to the requirements of General Design Criterion 17. We conclude that this design commitment is acceptable at the construction permit stage of the review.

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The five 161 kilovolt lines and the intertie bank connection to the 500 kilovolt transmission system will provide reliable sources to the two physically separate and independent 161 kilovolt lines supplying offsite power to the Phipps Bend plant.

Two independent 250 volt power house batteries will be provided at the Phipps Bend plant to supply both switchyards. Each transmission line will be protected with primary and back-up relaying.

During normal operating conditions, the main generators will supply electrical power through isolated-phase buses to the main step-up transformers, and the unit station service transformers located adjacent to the turbine building at a point between the load break switch and the low-voltage connection to the main transformers. During normal operation, station auxiliary power will be taken from the main generator through these transformers.

During normal startup and shutdown, the load break switch will be opened, allowing plant auxiliary loads to be fed from the 500 kilovolt grid for Unit No. 1 and the 161 kilovolt grid for Unit No. 2 through the main and unit station service transformers. During normal power operation the load break switch will be closed. Failure of these sources will initiate automatic transfer for both safety and non-safety related buses to the reserve, offsite 161 kilovolt sources which will be relied on for both offsite power paths. The reserve auxiliary power will be supplied by the four reserve transformers which are fed through an independent 161 kilovolt connection from the Tennessee Valley Authority power system. If both of the reserve supplies subsequently fail, the Class IE system will be automatically transferred to the onsite standby power sources (diesel generator sets). See Section 8.3 of this report for the onsite (standby) power system information.

The applicant has stated that the results of steady-state and transient stability studies show that the 161 kilovolt offsite power sources will remain intact as reliable sources of supply to the separate and independent onsite electric power system of each plant for (1) the loss of either or both nuclear units, (2) the loss of the Bull Run unit, and (3) the loss of the Phipps Bend 500 kilovolt line with two underbuilt 161 kilovolt lines.

We have reviewed the offsite power system information including its principal design criteria and the definition of the applicant's area of responsibility (including figures and information tables defining the interfaces and scope). We have concluded that the criteria meet our requirements and are acceptable.

8.3 Onsite Power Systems

The applicant provided the following information with regard to the onsite power system:

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(1) A list identifying and describing each load (whether safety or non-safety) that is in the applicant's scope and that is powered from alternating current and direct current power sources in the onsite power system. A description of the loads as depicted in GESSAR Standard Safety Analysis Report Table 8.3.1 was provided (e.g., division assignment, horsepower, kilowatts, maximum inrush, and kilovolt-amperes).

(2) For loads in the applicant's scope that are sequenced onto the safety buses, the information similar to that depicted in GESSAR Standard Safety Analysis Report Table 8.3.4 was provided (e.g., time to start, number required, and divisional assignment).

(3) The design criteria for the underground cable installations for Class IE systems that are in the balance-of-plant scope of supply for the Phipps Bend plant design was provided.

(4) As part of Section 8.3 of the Phipps Bend Preliminary Safety Analysis Report, the applicant included a statement that all of the onsite power systems, except for those portions identified and discussed in the Phipps Bend Preliminary Safety Analysis Report, are within the GESSAR scope of responsibility.

Initially, we were unable to define the applicant's responsibility for safety systems and portions of safety systems. As a result of requesting additional information the applicant amended the Phipps Bend Preliminary Safety Analysis Report to provide additional information, including figures and tables defining the interfaces and scope of responsibility.

We have reviewed the additional information provided by the applicant and the principal design criteria and have concluded they meet our requirements and are therefore acceptable.

8.4 Fire Stops and Seals for Areas Containing Electrical Equipment

Our evaluation of the fire stops and seals for areas containing electrical equipment is discussed in Section 9.4.1 of this report.

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9.0 AUXILIARY SYSTEMS

9.1 Introduction

The auxiliary systems necessary to assure safe plant shutdown which are in the applicant's balance-of-plant scope include the essential service water system, ultimate heat sink (in conjunction with the essential service water system, and the central pumping station), fire protection system and the diesel generator fuel oil storage and transfer system.

We have reviewed other auxiliary systems, such as the equipment and floor drainage system whose failure would not prevent safe shutdown but could, either directly or indirectly, be a potential source of a radiological release to the environment.

We have also reviewed the design of non-safety related auxiliary systems to ensure that the failure of these systems would neither prevent safe shutdown nor result in potential radioactive releases. These include the component cooling water system, the demineralized water makeup system, the potable water system, the condensate storage facilities, the compressed air system, the process sampling system, the ventilation systems for non-safety related areas and the communication and lighting systems. Failure of the above systems will not affect the capability of safety related systems to effect safe shutdown. We conclude that the design of the above systems are acceptable.

The applicant references GESSAR for the following systems and our evaluation is presented in Section 9 of Appendix A to this report: new fuel storage, spent fuel storage, spent fuel pool cooling and cleanup system, fuel handling system, closed cooling water system, standby liquid control system, control room area ventilation system, spent fuel pool area ventilation system, auxiliary and radwaste area ventilation system, engineered safety feature ventilation system, communications systems, lighting systems, diesel generator cooling water system, diesel generator starting system, diesel engine lubrication system, and diesel generator combustion air intake and exhaust system.

Where systems or portions of systems will be shared by the two units, the applicant has stated that such sharing will not impair their ability to perform their safety functions. We have reviewed those systems and components that will be shared and find that the design meets the requirements of General Design Criterion 5, and are acceptable.

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9.2 Water Systems

9.2.1 Essential Service Water Systems

The essential service water system will provide cooling water to the safety related systems used for normal plant operation, for safe cold shutdown, and for the mitigation of postulated accidents.

Two spray ponds for the facility will serve the entire essential service water system and will be designed to provide an adequate supply of water for all anticipated events assuming single active failures. The GESSAR interface requirements specify that three divisions of essential service water system be provided. Each division consists of one full capacity train for each unit. Since there are two units for this site the applicant has provided a total of six trains.

During normal operation Division I will be in operation. The Division II trains will be utilized only during safe shutdown following a loss-of-coolant accident, or as backup to the reactor core isolation cooling system. Division I or II will be capable of dissipating the total heat load required for a cold shutdown. Division III of each unit will feed one high pressure core spray for each unit. Division I and II for each unit will be powered by independent engineered safety features buses. Two trains will be required to provide cooling for the design basis loss-of-coolant accident. The essential service water system will be designed to seismic Category I requirements.

The essential cooling water central pumping station will be designed (1) to seismic Category I requirements, (2) to protect against unacceptable damage from tornado missiles and (3) for design basis flood protection. Each essential cooling water pump is located in a separate compartment within the central pumping station. The separation walls between compartments will be designed to withstand internal missiles and will have a three-hour fire rating. Two seismic Category I discharge structures will be utilized to return the essential cooling water to the pond.

Based on our review, we conclude that the essential service water system design criteria and bases are in conformance with the (1) requirements of General Design Criterion 44 regarding the ability to transfer heat from safety related components to the ultimate heat sink and to meet the single failure criteria and (2) General Design Criteria and 46 with regard to system design that allows performance of periodic tests and inspections, including functional testing and confirmation of heat transfer capabilities. The essential service water system design criteria and bases also meet the GESSAR system requirements. We therefore conclude that the system essential service water design criteria and basis are acceptable.

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9.2.2 Ultimate Heat Sink

The ultimate heat sink will consist of two seismic Category I spray ponds for the Phipps Bends plant. The ultimate heat sink will dissipate heat from the essential service water system for safe shutdown of the plant under accident conditions.

Our evaluation of the ultimate heat sink's capacity and heat dissipation capability can be found in Section 2.4.3 of this report. In addition, the applicant has demonstrated to our satisfaction in the Phipps Bend Preliminary Safety Analysis Report that the ultimate heat sink is in accordance with Position 2 of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants"; namely, the capability of the ultimate heat sink to withstand (1) the most severe natural phenomena anticipated at the site, taken individually, (2) the site related events that historically have occurred or may occur during plant lifetime, (3) reasonably probable combinations of less severe natural phenomena and/or site related events, and (4) a single failure of manmade structural features and is therefore acceptable.

9.2.3 Potable and Sanitary Water Systems

We have reviewed the description of the potable and sanitary water systems and have determined that the system design criteria preclude connections to systems having the potential for radioactive contamination. Consequently, we find the systems to be acceptable.

9.2.4 Raw Water System

The raw service water system will supply the service water requirements outside the reactor island and will be utilized to provide makeup water to the essential service water system during normal operation. The raw service water system which supplies raw water from the Holston River is not a safety related system and is not required to operate under accident conditions.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

Our evaluation of the design and design basis of the compressed air system which is within the GESSAR scope of supply is presented in Section 9.3.2 of Appendix A to this report.

9.3.2 Main Steam Isolation Valve Positive Seal System

Our evaluation of the design basis main steam isolation valve positive seal system which is within the GESSAR scope of supply is presented in Section 9.3.1 of Appendix C of this report.

9.3.3 Turbine Building Sumps (Drains)

Turbine building drains will be of the following categnries:

(1) Radioactive floor and equipment drains.

(2) Non-radioactive and non-contaminated drains.

(3) Clean oil and oily water drains.

Radioactive floor drainage will be collected in floor drain sumps and transferred to the floor drain neutralizer tanks in the radwaste building for treatment.

Radioactive equipment drainage will be collected in equipment drain sumps and transferred to the waste collector tanks in the radwaste building for treatment.

Non-radioactive and non-contaminated drainage will be collected in the station sump and transferred to the yard drainage pond.

Clean oil will be drained directly to tank trucks for reuse or removal from the plant. Oily water drains will be routed to the oil sump, which will be located at the lowest level of the turbine building. Oil will be accumulated in the sump until a sufficient amount is collected to be pumped into tank trucks for disposal.

Each drain header will be terminated below the minimum water level in the sumps to provide a water seal between the other drain headers.

We have reviewed the turbine building sumps and associated GESSAR interfaces and the interfaces with the liquid radwaste system (see also Section 11.2.1 of this report) and find them acceptable.

9.4 Other Auxiliary Systems

9.4.1 Fire Protection System

The applicant has received our new guidelines as stated in Appe x A of our Branch Technical Position (BTP) APCSB 9.5-1 and has indicated that it will provide a reply by April 18, 1977. We will review the evaluation along with revised design features of the fire protection system and provide the applicant with the results of our evaluation on a timely basis so that they can be effectively incorporated into the final design. The design as presently proposed meets General Design Criterion 3, "Fire Protection" and applicable guidelines in effect prior to issuance of Branch Technical Position APCSB 9.5-1 and for the construction permit stage of the review we find it acceptable. Final approval of the system will depend on the review of the applicant's submittal which will be completed after a decision the issuance of the construction permit; however, based upon our current review of the facility, sufficient flexibility exists in the design to allow implementation of any design changes

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that may be necessary to assure compliance with Appendix A to Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants."

9.4.2 Diesel Generator Fuel Oil Storage and Transfer System

The diesel generator fuel oil storage and transfer system will be designed to provide sufficient storage of fuel oil to allow operation of each emergency diesel generator for a minimum of seven days. The system will be designed to seismic Category I requirements and consist of three independent trains, one for each diesel generator. Each train will include a steel lined reinforced concrete 7 day storage tank, a day tank, and associated pumps. The steel lined reinforced concrete tanks will be located in an underground vault and will be protected from tornado missiles and flooding. Fuel oil will be supplied to each day tank from its corresponding seven day tank, via either of two electric motor driven pumps. Each pump motor will be fed from its associated engineered safety feature systems bus. The seven day fuel oil storage tanks will be served from one of two non-seismically designed 200,000 gallon fuel oil storage tanks.

The interface between the non-seismic and seismic portion of the fuel oil system will be double valved. One of the two valves will be a check valve, thus precluding the possibility of inadvertently draining the seismically designed seven day tanks.

Based on our review, we conclude that the system capacity and design criteria can satisfy their designated safety function and are, therefore, acceptable.

10.0 STEAM AND POWER CONVERSION SYSTEMS

10.1 Summary Description

The steam and power conversion system will be of conventional design, similar to those of previously approved plants. The system will be designed to transfer thermal energy from the reactor and convert it to electrical energy by the turbine generator. The condenser will transfer unused energy in the cycle to the main condenser circulating water. The entire system will be designed for the maximum expected thermal output from the nuclear steam supply system.

In the event of a turbine trip or a large load reduction, the heat generated in the reactor will be dissipated by directing steam through the turbine bypass system to the condenser and through the safety relief valves to the suppression pool if the turbine bypass capacity of 35 percent of power is exceeded.

The review of the steam and power conversion system was conducted to allow us to reach a conclusion as to whether the GESSAR interface requirements were satisfied by the interfacing balance-of-plant system, and to review the turbine generator overspeed protection.

10.2 Turbine-Generator (Control System)

The turbine generator will be a tandem compound type consisting of one double flow high pressure turbine and three low flow pressure turbines. The rotational speed will be 1800 revolutions per minute. The turbine electrohydraulic system will control the steam flow through the turbine by modulating the turbine inlet steam control valves. The turbine protection system and electrohydraulic system will execute their respective functions hydraulically with certain subfunctions being carried out pneumatically or electrically. The turbine control system will be designed to trip the turbine under the following conditions: turbine overspeed, low condenser vacuum, excessive vibration, electrical faults, lube oil pressure and tank level, or manual trip. Overspeed protection will be accomplished by two independent systems, the electrohydraulic system and the mechanical overspeed system. The electrohydraulic system will rapidly close the governor and interceptor valves if 106 percent of rated speed is exceeded. If 110 percent of rated speed is reached, the mechanical overspeed sensor will trip all steam valves (throttle, governor, reheat stop and interceptor valves) to maintain the speed below 112 percent of a rated speed. As a backup, a second completely independent mechanical overspeed sensor will also trip all valves at 112 percent of rated speed.

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Based on our review, we conclude that the turbine-generator overspeed protection design criteria and bases can meet their designated safety functions and are, therefore, acceptable.

The turbine generator control system interfaces with the GESSAR reactor protection system and the main steam isolation valves. Our review of the interfaces are discussed in Section 7.2 of this report.

10.3 Main Steam Supply System

The steam generated in the reactor will be routed to the turbine by four steam lines. Each main steam line will contain safety valves, main steam isolation valves and a shutoff valve.

The portion of the main steam supply system beyond the seismic restraint (main steam shutoff valve) is classified Safety Class Nonnuclear Safety, Quality Group D, and is non-seismically designed.

Based on our review, we conclude that the main steam supply system design criteria and bases are in conformance with the seismic position of Regulatory Guide 1.29, "Seismic Design Classification (Revision 2)," and are, therefore, acceptable.

10.4 Main Condenser

The main condenser will be designed to provide normal heat removal capability of at least 8.1×10^9 British thermal units per hour. This capability will enable the condenser to condense the steam from the turbine exhaust. At maximum expected reactor thermal output the heat removal capability will be at least 8.48×10^9 British thermal units per hour. The hotwell will be designed to provide at least four minutes holdup of the condensate for radioactive decay.

The condenser will be capable of accepting a turbine bypass flow of approximately 35 percent of steam flow. The condenser could become ineffective because of the loss of some or all of its cooling water and/or because of excessive air inleakage. Either of these conditions will cause the loss of condenser vacuum, which will result in turbine trip and a reactor scram.

The main condenser interfaces with the nuclear island through the condenser vacuum trip. These interfaces have been evaluated and found to agree with the requirements identified in GESSAR. They therefore are considered acceptable.

10.5 Main Condenser Evacuation System

The main condenser evacuation system which will consist of mechanical vacuum pumps (hogging pumps) and two 100 percent steam jet air ejectors, will be capable of

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removing all condensible gases (including air inleakage and radiolytic dissociation products originating in the reactor) from the main condenser. The system will be designed so that under operating conditions proper condenser vacuum will be maintained and the gases leaving the evacuation system will have sufficient pressure to be exhausted to atmosphere through the gaseous radwaste system. Under startup conditions the system will have sufficient capacity to guarantee the availability of the turbine bypass system to the reactor within one hour from the time a pressure 125 pounds per square inch absolute is reached in the reactor vessel.

The main condenser evacuation system will be designed to minimize the potential for explosion in the piping upstream of the catalytic recombiner in the offgas system by maintaining sufficient dilution steam in the steam jet air ejector discharge to limit the hydrogen concentration to less than four percent by volume. Additional protection against explosion will be provided by ensuring that the mechanical vacuum pumps (hogging pumps) do not operate when radiolytic hydrogen is present in the main condenser. Furthermore, the steam jet air ejector intercondenser and precooler will be capable of withstanding an explosion in the offgas system.

The evacuation system will be designed to minimize radioactive containment releases to the environment by ensuring delivery of the offgases to the gaseous radwaste system during power operation. The mechanical vacuum pumps, which discharge their effluent through the turbine building exhaust system, will not be permitted to operate if a high radiation level is indicated in the discharge piping.

The hydrogen concentration at the outlet of the second stage air ejector will be maintained below the lower flammable limit of four percent hydrogen in air by the addition of dilution steam. On indication of low steam pressure or low steam flow, the operating steam jet air ejector will be removed from service and the standby steam jet air ejector activated.

The scope of our review included an evaluation of the system's capability to transfer radioactive gases to the gaseous waste or ventilation systems, and the design provisions incorporated to monitor and control releases of radioactive materials in gaseous effluents in accordance with General Design Criteria 60 and 64. We also found the system interfaces as listed in GESSAR to be acceptable. The basis for acceptance in our review has been conformance of the applicant's design, design criteria, and design bases for the main condenser evacuation system to the applicable regulations. Based on our evaluation, we find the proposed main condenser evacuation system to be acceptable.

10.6 Condenser Circulating Water System

The condenser circulating water system will be designed to use water from the cooling towers to remove heat rejected from the main condensers and feedwater pump turbine condensers.

A piping failure in the condenser circulating water system or the condensate system large enough to cause flooding will be detected by high level alarms in the turbine room pumps and condenser pits. Further, should the pumps remain running, there will be no flooding of safety related equipment. The alarm will alert the operator to take action to isolate the equipment or shut down the system.

Based on our review of the design basis and design criteria for the circulating water system, we conclude that this system will perform its intended function and, therefore, is acceptable.

10.7 Condensate Cleanup System

The objective of the condensate cleanup system is to maintain high quality feedwater by removing dissolved solids and corrosion products from the condensate system. The system will be designed to polish the full flow of condensate at a temperature of 140 degrees Fahrenheit and a design pressure of 190 pounds per square inch gauge. The system will also be designed to remove impurities which enter the makeup water or enter the condensate system as a result of condenser tube leakage.

The condensate demineralizer system for each unit will consist of eight mixed-bed demineralizer vessels, seven operating with one spare. The system will also include an external regeneration facility, consisting of a resin separation and cation regeneration vessel, an anion regeneration vessel, three resin storage tanks, two ultrasonic resin cleaners, and a resin receiving tank.

In an effort to reduce the volume of regenerant waste, an extensive reclamation system will be included in the condensate cleanup system. This recycle equipment will consist of acid and caustic reclamation tanks, a high-conductivity waste accumulator tank, a neutralization tank, two low-conductivity waste collection tanks, and a centrifugal filter.

Radioisotopes will concentrate in the demineralizer beds, the concentration will not affect ion exchange capacity but the radioisotope activity level will require shielding of all equipment. The demineralizers will be located in concrete cells. No valves or other equipment with moving parts are to be located inside the cells. Regeneration and reclamation equipment will be located in the central service facility. Liquid waste from the system will be pumped to the waste collector tanks in the radwaste building for treatment. We have reviewed the interfaces between the condensate cleanup system and the radwaste system and find them acceptable.

We have reviewed the design criteria and bases of the condensate cleanup system and conclude that they are in conformance with Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," and are therefore acceptable.

10.8 Condensate and Feedwater Systems

The condensate system will be designed to deliver water from the main condenser hotwell to the suction of the feed pump in sufficient quantity and with sufficient pressure and temperature to ensure that the design bases of the feedwater system can be met.

The reactor feedwater system will be designed to deliver condensate from the suction of the reactor feed pumps to the reactor in sufficient quantity to maintain desired reactor water level during all modes of operation. The system will terminate at the second valve outside containment.

The condensate and feedwater systems are not required to effect or support the safe shutdown of the reactor or engineered safety feature operation. We conclude that failure of these systems will not affect the safety of the Phipps Bend plant.

We have reviewed the condensate and feedwater system designs and the interfaces specified in GESSAR. We conclude that the design and the interface information are acceptable.

10.9 Condensate Storage and Transfer System

The condensate storage facilities will be designed to accommodate the various requirements of the boiling water reactor system and its auxiliaries and to interface with the condensate system to maintain proper condensate inventory in the condenser hotwell. The condensate storage tanks will provide sufficient inventory to add water as required to the condenser hotwell through the condensate makeup valve and will have sufficient capacity to receive water bypassed from the condenser hotwell due to excess water inventory in the hotwell. In addition, the tank will have the capacity required to supply condensate for demineralizer bed regeneration (approximately 35,000 gallons) without initiating makeup to the storage tank from the demineralized water system after regeneration.

The condensate storage facility will be designed to ensure performance of the following functions: (1) supply water to the control rod drive system, (2) supply makeup to the auxiliary boiler system, (3) supply makeup water for the fuel pool and containment pool cooling and cleanup systems, (4) supply water to the reactor water cleanup system for flushing the filter demineralizers, (5) provide an emergency source of water for the high pressure core spray and reactor core isolation cooling pumps, (6) provide water to several hose stations at various locations in the reactor and auxiliary buildings, (7) provide adequate condensate to flood the reactor well during refueling operations, (8) supply water required for initial fill and continuous makeup to the suppression pool, and (9) provide water for backflushing the condensate demineralizer when the unit is out of service.

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The primary components of the storage facility will include a 500,000-gallon steel tank, two 100 percent capacity condensate transfer pumps, a condensate head tank, and all piping and valves required to connect this equipment and interface with other systems. The storage tank will be constructed of carbon steel and lined with a phenolic-epoxy coating for corrosion protection. Two 100 percent capacity condensate transfer pumps (one operating with one on standby) are required to maintain level in the condensate head tank.

We have reviewed the design of the condensate storage and transfer system and the system interfaces with the GESSAR leak detection, reactor core isolation cooling, the high pressure core spray and the liquid radwaste systems. We determined that the interfaces agree with those identified in GESSAR. They are therefore considered acceptable.

10.10 Turbine Gland Sealing System

The turbine gland sealing system will be designed to control radioactive steam leakage and air inleakage from the turbine and large steam valve shaft seal glands. The glands will be sealed with essentially non-radioactive steam from a steam seal evaporator which will evaporate demineralized condensate using extraction steam from the main turbine.

Our review included the source of sealing steam and the provisions incorporated to monitor and control releases of radioactive material in gaseous effluents in accordance with General Design Criteria 60 and 64.

The basis for acceptance in our review has been conformance of the applicant's design criteria, and design bases for the turbine gland sealing system to the applicable criteria referenced above. Based on our evaluation, we find the proposed turbine gland sealing system to be acceptable.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

The Phipps Bend application referenced GESSAR for radioactive waste management within the nuclear island design scope. The two Phipps Bend units will share liquid and solid radwaste systems rather than have independent systems for each unit as evaluated in GESSAR. However each unit will have a separate gaseous waste processing system. Our evaluation of the GESSAR radioactive waste management systems is presented in Section 11.0 of Appendix A to this report.

In our evaluation of the balance-of-plant portion of the waste management systems, the following areas were considered: (1) the capability of the liquid and gaseous waste treatment systems for keeping the levels of radioactive material in effluents "as low as reasonably achievable" based on expected radwaste generated over the life of the plant, (2) the capability of the liquid and gascous systems to maintain releases below the limits specified in 10 CFR Part 20 assuming fission product leakage at design levels from the fuel, (3) the capability of the liquid, gaseous, and solid waste systems to meet the processing demands of the plant during anticipated operational occurrences, (4) the interfaces between the balance-of-plant and radwaste systems within the GESSAR scope, (5) the design features incorporated to control the release of liquids containing radioactive materials due to tank overflows, and (6) the provisions for monitoring and controlling radioactive materials in process and effluent streams and the provisions to detect leakage of radioactive material between systems.

The principal radionuclides associated with the liquid, gaseous and solid wastes are given in our Final Environmental Statement which was issued in February 1977. In making these determinations, we considered waste flows, radionuclide activities, and equipment decontamination factors which are consistent with those expected over the life of the plant, considering both normal operation and anticipated operational occurrences. The liquid and gaseous source terms were calculated using the BWR-GALE Code described in NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors." The principal parameters used in these calculations, along with their bases, are also given in NUREG-0016. Based on our evaluation, we conclude that the liquid and gaseous radwaste treatment systems are capable of reducing releases of radioactive materials in liquid and gaseous effluents to "as low as is reasonably achievable" levels in conformance with 10 CFR Part 50.34a and which meet the dose requirements of Appendix I to 10 CFR Part 50.

Based on our evaluation, as described in detail below, we find the above aspects of the proposed liquid, gaseous and solid radwaste systems and associated process and effluent radiological monitoring systems to be acceptable.

11.2 Radwaste System Description and Evaluation

11.2.1 Liquid Radioactive Waste Treatment System

The liquid radioactive waste treatment system will consist of process equipment and instrumentation necessary to collect, process, monitor, and recycle or release radioactive liquid wastes. Liquid radioactive waste will be processed on a batch basis to permit optimum control of releases. Prior to being released, samples will be analyzed to determine the types and amounts of radioactivity present. Based on the results of the analysis, the waste will be retained for further processing, recycled for eventual use in the plant, or released under controlled conditions.

The liquid radwaste systems will be as described in GESSAR with the following exceptions. The applicant has proposed to install two 78,000-gallon low conductivity tanks and two 54-square foot waste filters to accommodate the additional waste associated with sharing the liquid radwaste systems between the two units. The GESSAR design for one unit indicates two 40-square foot waste filters. These provisions are proposed in the more recent GESSAR-251 (Docket No. STN 50-531) design for shared liquid radwaste systems. We find these exceptions acceptable taking into account the peak and average capacity requirements of the two unit plant.

Our calculated releases of radioactive materials for the liquid radwaste system for normal operation are given in our Final Environmental Statement, issued February 1977. In our Final Environmental Statement, we have determined that the proposed liquid radwaste systems will be capable of reducing the release of radioactive materials in liquid effluents to approximately 0.17 curie per year per reactor, excluding tritium and dissolved gases, and 15 curies per year per reactor for tritium. An isotopic listing of our calculated liquid source term is given in Table 3.4 of our Final Environmental Statement.

The liquid radwaste systems will be located in a structure with a seismic Category I foundation. The seismic design and quality group classifications of the equipment are consistent with our guidelines and with GESSAR criteria. A listing of major liquid radwaste equipment is given in Table 11.1 of this report.

The design capacities of the floor drain neutralizer subsystem and waste collection subsystem are approximately 57,000 gallons per day and 230,000 gallons per day, respectively. We calculate the expected waste flows to the floor drain neutralizer subsystem and waste collection subsystem to be approximately 20,000 gallons per day

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TABLE 11.1

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS CONSIDERED IN RADWASTE EVALUATION

Liquid Systems

Component	Number of Each Component	Capacity (Each)	Quality Group	Seismic Design
Low Conductivity Tank	2	78,000 gallons	Note 1	Note 2
Waste Filter	2	54 square feet	Note 1	Note 2
Filtrate Tank	2	3,000 gallons	Note 1	Note 2
Low Conductivity Demineralizer	1	180 cubic feet	Note 1	Note 2
Backup Demineralizer	1	180 cubic feet	Note 1	Note 2
High Conductivity Tank	3	18,000 gallons	Note 1	Note 2
Waste Evaporator	2	40 gallons per minute	Note 1	Note 2
Concentrated Waste Tank	1	25,000 gallons	Note 1	Note 2
Distillate Tank	1	3,000 gallons	Note 1	Note 2
Distillate Demineralizer	1	180 cubic feet	Note 1	Note 2
Spent Resin Tank	1	10,000 gallons	Note 1	Note 2
Excess Water Tank	2	50,000 gallons	Note 1	Note 2
Detergent Waste Tank	2	1,500 gallons	Note 1	Note 2
Detergent Filter	1	25 gallons per minute	Note 1	Note 2
Detergent Evaporator	1	5 gallons per minute	Note 1	Note 2

Gaseous Systems

Component	Number of Each Component	Quality Group	Seismic Design
Offgas Preheaters	2	Note 1	Note 2
Catalytic Recombiners	2	Note 1	Note 2
Offgas Condenser	1	Note 1	Note 2
Water Separator	1	Note 1	Note 2
10-minute Holdup Piping	1	Note 1	Note 2
Cooler Condenser	2	Note 1	Note 2
Moisture Separator	2	Note 1	Note 2
Dessicant Dryer	4	Note 1	Note 2
Dessicant Regenerative System	2	-	-
Gas Cooler	3	Note 1	Note 2
Glycol Cooler Skid	1	-	-
Delay Beds (12 tons each)	3	Note 1	Note 2

Note 1 - Design Classification per Branch Technical Position, ETSB 11-1.

Note 2 - Seismic Design Criteria given in Branch Technical Position, ETSB 11-1.

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flows and the design flows will provide adequate reserve for processing surge flows. All major processing components in the floor drain neutralizer subsystem and waste collection subsystem will be redundant with the exception of the detergent waste subsystem evaporator which will not be redundant. The failure of this component and the subsequent release of untreated detergent waste could increase the plant releases by approximately 0.01 curie for each month of unavailability, which is an acceptably small fraction of the total liquid releases.

We consider the system capacity and system design to be adequate to meet the demands of the plant during normal operation, including anticipated operational occurrences. The system will be designed to accept wastes from the turbine building floor drain collection subsystem which is a balance-of-plant system, and we find this acceptable.

The system will also be designed to control potential radioactive releases due to overflows from tanks outside containment containing radioactive materials by providing level instrumentation which will alarm in the control room, and by routing overflow lines to collection tanks capable of collecting liquid spillage and retaining it for processing. We consider these provisions to be adequate for preventing the uncontrolled release of radioactive materials to the environment. We find the applicant's proposed system design to be acceptable and in accordance with Branch Technical Position ETSB 11-1, "Design Guidance for Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants" (Revision 1), included as Appendix B to the GESSAR Safety Evaluation Report (Appendix A to this report).

11.2.2 Gaseous Radioactive Waste Treatment System

The gaseous radioactive waste treatment systems will consist of a low temperature charcoal delay system for treating the offgas from the main condenser air ejector, and iodine and particulate control systems for certain building ventilation systems. These systems will be as described in GESSAR with the following exception; in addition to having two charcoal delay trains, each containing 12 tons of charcoal on-line during normal operation as provided in the GESSAR design, the applicant has proposed to install a third 12-ton charcoal delay train which will be held in reserve for contingencies. We conclude that the addition of a third 12-ton charcoal delay train is suitable because it adds capacity above and beyond our requirements.

Our calculated releases of radioactive materials from the gaseous radwaste treatment system for normal operation are given in our Final Environmental Statement. In our Final Environmental Statement, we have determined that the proposed gaseous radwaste treatment systems and plant ventilation system will be capable of reducing the release of radioactive materials in gaseous effluents to approximately 7,300 curies per reactor for noble gases, 0.46 curies per reactor for iodine-131, 0.064 curies per reactor for particulates, 9.5 curies per reactor for carbon-14 and 79 curies per reactor for tritium. The applicant calculated the estimated annual release to be approximately 9500 curies per reactor for noble gases, 0.085 curie per year per reactor for iodine-131 and 9.5 curies per year per reactor from tritium. An isotopic listing of the calculated releases of radioactive material in gaseous effluent is given in Table 3.5 of our Final Environmental Statement.

The offgas system design classification and the system support seismic design criteria will be in conformance with Branch Technical Position ETSB 11-1. The main condenser air ejectors (a balance-of-plant system) will be designed to meet the GESSAR interface requirements for the offgas treatment system. The offgas treatment system will exhaust to the turbine building vent thereby satisfying the GESSAR interface requirements.

Mechanical vacuum pumps will be used to establish main condenser vacuum during plant startups and to maintain an air sweep through the main condenser during plant shutdowns. The effluent from the mechanical vacuum pumps, containing noble gases and iodine, will be released without treatment. Non-radioactive steam from an auxiliary boiler will be used to seal the turbine glands and, therefore, the release of radioactive materials in gaseous effluents from this source will be negligible.

Building ventilation systems that are sources of radioactive gaseous effluents are described in GESSAR except for the turbine ventilation system which is a balance-of-plant system. The turbine building ventilation air will contain noble gases, iodine, and particulates. The turbine building ventilation air effluent will be monitored for radioactivity and released without treatment, except for the sump area. The ventilation exhaust air from the sump area will be processed through a high efficiency particulate air filter-charcoal adsorber system prior to release.

The ventilation system exhaust will be isolated if the activity in the exhaust air exceeds a predetermined level. The ventilation system will be designed to induce airflows from potentially less contaminated areas to areas having a higher potential for radioactive contamination.

11.2.3 Solid Radwaste Treatment System

The solid radwaste treatment system will be designed to collect and process wastes based on their physical form and the solidification processing required prior to packaging. The applicant has proposed to adopt the GESSAR solid waste system design with additional capacity provided for storing packaged wastes. "Wet" solid wastes, consisting of spent demineralizer resins, evaporator bottoms, filter sludges, and chemical drain tank residues, will be combined with cement to form a solid matrix and sealed in shipping containers with a capacity of 170 cubic feet each. Dry solid wastes, consisting of ventilation air filters, contaminated clothing, paper, and miscellaneous items such as tools and glassware, will be compacted into 55-gallon steel drums. Miscellaneous solid wastes, such as irradiated primary system components, will be handled on a case-by-case basis, based on their size and activity. Expected solid waste volumes and activities shipped offsite for each reactor will be 43,000

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cubic feet per year of "wet" solid waste containing 4100 curies of activity and 450 drums per year of "dry" solid waste containing less than 5 curies total, which is acceptable.

The applicant's design provides adequate capacity for approximately 30 days onsite storage of packaged wastes prior to shipping. Our estimate considers the applicant's proposal to provide storage space for 12 high-level waste containers in the radwaste building and to provide reserve space for 25 low-level waste containers in controlled yard areas. The controlled yard areas will include a concrete pad with a six-inch curb to mitigate the effects of potential container leakage or surface contamination washoff and will be fenced and controlled to maintain adequate separation between the pad and facility personnel. In all, the applicant's proposed design provides 1.5 times more storage capacity per unit than does GESSAR.

11.3 Process and Effluent Radiological Monitoring Systems

The GESSAR process and effluent radiological monitoring system was evaluated in Section 11.5 of Appendix A to this report. For the balance-of-plant scope, monitors on effluent release lines from the turbine building will alarm should radiation levels exceed a predetermined value (see Table 11.2 of this report). Systems which are not amenable to continuous monitoring or for which detailed isotopic analyses are required will be periodically sampled and analyzed in the plant laboratory. The streams that will be monitored including the station water discharge lines are presented in Table 11.2.

All normal and potential release pathways will be monitored. The exact location, type of instrument, range, set point, sensitivity, and calibration frequency will be provided at the Final Safety Analysis Report review stage. Similarly, more specific information regarding sampling locations and analyses will be provided at the Final Safety Analysis Report review stage. Our detailed review of the final design adopted by the applicant at the Final Safety Analysis Report review stage will assure his compliance with General Design Criteria 13, 60 and 64, which govern radiological sampling and monitoring. The information provided by the applicant in the Phipps Bend Preliminary Safety Analysis Report along with the monitoring commitment is acceptable for the construction permit stage of the review.

11.4 Evaluation Findings

We have evaluated the applicant's proposed design of the radioactive waste management systems and the exceptions taken to the GESSAR design which the applicant references. We conclude that all exceptions to GESSAR have been suitably identified and justified and are acceptable. The GESSAR balance-of-plant interfaces for these systems are compatible and therefore we conclude that the applicant's design is acceptable.

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TABLE 11.2

PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEM

Stream Monitored

Liquid*

- Component Cooling Water Loops
- Service Water Discharge
- Liquid Radwaste Discharge+
- Cooling Tower Blowdown Discharge

Gas**

- Offgas Discharge+
- Containment and Drywell Ventilation Exhaust+
- Radwaste Building Vent
- Auxiliary Building Vent+
- Fuel Building Vent+
- Turbine Building Vent
- Plant Exhaust Duct
- Clean Steam (auxiliary steam-reboilers)

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*All liquid streams will be monitored for gross gamma activity.

**All gas streams will be monitored for noble gas (beta or gamma); other forms of radioactivity will be sampled for laboratory analysis.

+These monitors will provide annunciation and automatic closure of isolation valves terminating releases or diversion to alternate systems when the radiation level exceeds a predetermined value.

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We have considered the capabilities of the proposed radwaste systems to meet the anticipated demands of the plant due to normal operation and to anticipated operational occurrences and have concluded that the liquid, gaseous, and solid waste system capacities and design flexibilities are adequate to meet the anticipated needs of the plant.

Based on our evaluation, the proposed liquid radwaste system will be capable of maintaining releases of radioactive materials in liquid effluents from the Phipps Bend nuclear plant such that the total body doses to an individual in an unrestricted area will be less than 3 millirems per year or any organ dose less than 10 millirems per year from either reactor, in accordance with Section II.A of Appendix I to 10 CFR Part 50. We conclude that the liquid radwaste treatment system will reduce liquid radioactive effluents to as low as is reasonably achievable levels in accordance with 10 CFR Part 50.34a, and Appendix I to 10 CFR Part 50, and therefore is acceptable.

Based on the staff's evaluation, the proposed gaseous radwaste systems will be capable of maintaining releases of radioactive materials in gaseous effluents such that the annual air dose due to gamma and beta radiation will be less than 10 millirem per unit and 20 millirad per unit respectively at or beyond the site boundary in accordance with Section II.B of Appendix I to 10 CFR Part 50.

Based on our evaluation, the proposed gaseous radwaste systems will be capable of maintaining releases of radioiodine and radioactive material in particulate form such that the dose to any organ of an individual will be less than 15 millirems per year from either reactor in accordance with Section II.C of Appendix I to 10 CFR Part 50. We have considered the potential effectiveness of augmenting the proposed liquid and gaseous radwaste treatment systems using items of reasonably demonstrated technology in accordance with the requirements of Section II.D of Appendix I to 10 CFR Part 50. A detailed discussion of the cost-benefit analysis is contained in Section 3.5 of our Final Environmental Statement, dated February 1977. We have determined that additional augmentation will not effect reductions in the cumulative population dose reasonably expected within a 50 mile radius of the reactor at a cost of less than $1,000 per total body man-rem or $1,000 per man-thyroid-rem.

The above findings are in conformance with Sections II.A, II.B, II.C and II.D of Appendix I and, therefore, we conclude that the liquid and gaseous radwaste treatment systems are capable of reducing releases of radioactive materials in liquid and gaseous effluents to "as low as is reasonably achievable" levels in conformance with 10 CFR Part 50.34a and which meet the requirements of Appendix I to 10 CFR Part 50.

We have considered the potential consequences resulting from reactor operation with a fission product rate release consistent with a noble gas release rate to the reactor coolant of 100 microcuries per thermal megawatt-second after 30 minutes decay and determined that under these conditions, the concentrations of radioactive materials in liquid and gaseous effluents in unrestricted areas will be a small fraction of the limits specified in 10 CFR Part 20.

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We have reviewed the applicant's quality assurance provisions for the radwaste systems, the quality group classifications used for system components, the seismic classification applied to the design of the gaseous waste processing system, and the seismic classification applied to the design of structures housing the radwaste systems. The design of the radwaste systems and structures housing these systems meet the acceptance criteria as set forth in Branch Technical Position ETSB 11-1.

We have reviewed the provisions incorporated in the applicant's design to control the release of radioactive materials in liquids due to inadvertent tank overflows and to prevent uncontrolled releases due to hydrogen explosions in gaseous systems and conclude the measures proposed by the applicant are consistent with our acceptance criteria as set forth in Branch Technical Position ETSB 11-1.

Our review of the radiological process and effluent monitoring system verified that (1) the provisions proposed for sampling and monitoring all normal and potential effluent discharge paths are in conformance with General Design Criterion 64, (2) the automatic termination of effluent releases and the control over releases of radioactive material in effluents are in conformance with General Design Criterion 60 and Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants" (Revision 1), (3) the sampling and monitoring of plant waste process streams for process control are in conformance with General Design Criterion 63, (4) the sampling and analytical programs are in conformance with the guidelines in Regulatory Guide 1.21, and (5) for monitoring process and effluent streams during postulated accidents. The review included piping and instrument diagrams and process flow diagrams for the liquid, gaseous, and solid radwaste systems and ventilation systems, and the location of monitoring points relative to effluent release points. We conclude that the applicant's radiological process and effluent monitoring systems are acceptable.

Based on the foregoing evaluation, we conclude that the proposed radwaste treatment and effluent monitoring systems are acceptable. The basis for acceptance has been conformance of the applicant's designs, design criteria, and design bases for the radwaste treatment and monitoring system to the applicable regulations and regulatory guides referenced above, as well as to staff technical positions and industry standards.

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12.0 RADIATION PROTECTION

12.1 Introduction

The Phipps Bend Preliminary Safety Analysis Report provides information on the methods for radiation protection in the balance-of-plant, including the facility design and layout, equipment design, and a description of the health physics program.

An estimate of occupational radiation exposure to plant personnel is included. Shielding will be provided to reduce radiation levels. Ventilation will be arranged to control the flow of potentially contaminated air. Radiation monitoring will be employed to measure levels of radiation in potentially occupied areas and to measure airborne radioactivity throughout the plant. A description of the health physics program to be provided for plant personnel and visitors during reactor operation, maintenance, refueling, radwaste handling description and analysis of the radiation protection program is included in the Phipps Bend Preliminary Safety Analysis Report and in responses to our requests for additional information.

The criterion used to determine acceptability of the program is that doses to personnel will be maintained within the established limits of 10 CFR Part 20, "Standards for Protection Against Radiation," and that the design and program features are consistent with the guidelines of Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable (Nuclear Power Reactors)." In Section 12.4 of the Phipps Bend Preliminary Safety Analysis Report, the applicant commits to maintain occupational radiation exposures as low as is reasonably achievable and subscribes to the philosophy set forth in Regulatory Guides 8.8 and 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable (Revision 1)," in the design and operation of all facilities utilizing radioactive materials or radiation sources.

On the basis of our review, we have concluded that the applicant's radiation protection program will provide reasonable assurance that doses to personnel will be less than the limits established by 10 CFR Part 20 and maintained as low as is reasonably achievable, consistent with the guidelines of Regulatory Guide 8.8. Therefore, the Phipps Bend radiation protection program is acceptable.

12.2 Shielding

Shielding will be designed as outlined in Section 12 of Appendix A to this report. In meeting the objectives, the applicant has the benefit of prior experience with design and operation of Browns Ferry Units No. 1 and No. 2 (Docket Nos. 50-259 and 50-260), and design of Sequoyah Units No. 1 and No. 2 (Docket Nos. 50-327 and 50-328) and Watts Bar Units No. 1 and No. 2 (Docket Nos. 50-390 and 50-391

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The applicant has provided for five radiation zones, identical with those designated in GESSAR, as a basis for classifying occupancy and access restrictions in various areas within the balance-of-plant. On this basis, maximum design dose rates are established for each zone and used as input for shielding of the respective zones in the turbine building and the central service facility. For example, design radiation levels in operating areas where personnel are expected to be working for a 40-hour week will be less than one millirem per hour.

Radiation protection concepts directed to keeping personnel exposures below regulatory limits have been used throughout in the design of the balance-of-plant. Shielding design and radiation zoning were generally based on operating conditions. To the extent practicable, major sources will be in individually labyrinthed, shielded cubicles. Pipes and ducts will be routed through high-zoned, low access areas when practicable; shielding will be provided for pipe trenches and penetrations.

Designers participated in inspections, dose mappings, and discussions with Browns Ferry personnel. They used these experiences in verifying and modifying design criteria and calculational techniques. Radiation zone maps will be used to identify areas with high radiation potential and thus assure that adequate shielding is maintained. Final design drawings will be reviewed by personnel familiar both with the system design and radiation protection.

Changes or deviations in field run piping which may contain radioactive materials must be reviewed by Tennessee Valley Authority engineering designers with expertise in radiation protection prior to approval and implementation of the change. Procedures will be formulated to assure that deviations from paragraph 12.1.2.1 of the GESSAR Standard Safety Analysis Report will be adequately reviewed by General Electric Company personnel qualified in radiation protection.

The estimates of annual man-rem exposure are based on conservatively assumed radiation sources, design shielding, and calculated average dose rates, using manpower levels similar to current staffing of the applicant's Browns Ferry Nuclear Plant. Occupancy times are based on type of operation or work assumed to be performed by each individual, working 2000 hours per year. The applicant's estimate of 234 man-rems average annual occupational radiation exposure per unit is somewhat lower than recent experience at operating boiling water reactors at Nine Mile Point, Vermont Yankee, and Oyster Creek. The applicant attributes this to improved shielding practices and maintenance procedures and the Tennessee Valley Authority's design philosophy. The applicant points out that there is little data available to extrapolate total occupational doses after long periods of operation and states that total doses might then reach 400 to 500 man-rems per year per unit due to buildup of radioactive material in piping and components. The bases for the applicant's exposure estimates are reasonable, and consistent with the acceptance criteria in our Standard Review Plan.

On the basis of the applicant's design criteria, shield models and operating philosophy, we conclude that adequate consideration has been given in the Phipps Bend Preliminary Safety Analysis Report to the shielding and layout of facilities and components to keep exposures to operating personnel within the applicable limits of 10 CFR Part 20 and, to reduce unnecessary exposure during normal operation of the facility, including the considerations stated in Regulatory Guides 8.8 and 8.10.

12.3 Airborne Radioactivity Monitoring

The turbine building radioactivity monitoring system will be designed to monitor continuously, and to indicate and record levels of radioactive particulates, radioiodine, and radioactive gases in the building exhaust. Two self-contained portable monitoring units will be provided in the turbine building for local sampling of particulate and iodine concentrations, normally at the front standard of the high pressure turbine and at the steam jet air ejector. Visible and audible alarms will be provided locally for high radiation levels and power failure; high radiation levels and monitor malfunctions will be alarmed in the control room. We conclude that this monitoring system is consistent with the provisions of Regulatory Guide 8.8 and is acceptable.

12.4 Health Physics Program

The health physics program objectives are to provide reasonable assurance that the exposure limits of 10 CFR Part 20 are not exceeded, to ensure that occupational radiation exposures are maintained as low as is reasonably achievable, and to further reduce unavoidable exposures. The plant health physicist will be the onsite supervisor of the radiological hygiene branch. He will be responsible for the direction of the health physics program, which covers all plant operations involving potential radiation hazards. He will provide advice and assistance to the plant superintendent and keep him informed at all times on radiation hazards and conditions.

The radiological hygiene branch responsibilities will include: (1) routine and special radiological surveys to monitor all relevant plant areas for levels of radiation, contamination, and airborne radioactive materials; (2) the development of plant radiological procedures and training; (3) provision of personnel dosimetry and whole body counting as required to determine personnel exposure; (4) maintenance and calibration of instrumentation required to monitor radiological conditions; (5) retention of records to demonstrate compliance with regulations and for evaluation of program performance and trends; and (6) review of proposed new or modified plant operations.

The health physics facilities will include an office in the office building and a laboratory and an office in the central service facility, as well as lockers, change rooms, laundry storage, and personnel decontamination rooms. The laboratory will be furnished with necessary hoods and storage space; it will be located at the boundary between the clean and potentially contaminated areas to control access and occupancy

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of the controlled area. Instruments and decontamination supplies will be kept in the laboratory. We conclude that these facilities are sufficient to maintain occupational exposures as low as is reasonably achievable and are consistent with the provisions of Regulatory Guide 8.8.

Protective clothing and equipment to be provided will include: coveralls, laboratory coats, gloves, head covers, foot covers, plastic suits, and respiratory protection, including half-face and full-face masks with filters, full-face masks with constant air flow supplies, and self-contained pressure-demand breathing apparatus. We conclude that the applicant will be able to maintain occupational radiation exposures as low as is reasonably achievable with this equipment.

Monthly film badge service will be provided for personnel entering the plant. For personnel subject to neutron exposure, neutron film will be provided. The dose data will be processed by the Tennessee Valley Authority computer, recorded and tabulated according to: individual's identification, monitoring period, dose in rems, and cumulative dose to date for the current month, current quarter, current year, and entire exposure history.

A mobile whole body counter will be available at the plant periodically, and for additional visits on short notice as necessary. The frequency of checks on an individual will depend upon the likelihood of radioactive contamination of the individual's work environment.

Based on the information provided in the application, and the responses to our requests for additional information, we conclude that the applicant intends to implement a radiation protection program that meets the objectives of Regulatory Guide 8.8 and maintain occupational exposures "as low as is reasonably achievable" as required by 10 CFR Part 20.

13.0 CONDUCT OF OPERATION

13.1 Organizational Structure of Applicant

The Tennessee Valley Authority has ultimate responsibility for the design, construction, quality assurance and operation of the Phipps Bend plant. The General Electric Company will be responsible for the nuclear steam supply systems and the design of the standard nuclear island. The Tennessee Valley Authority will be responsible for the design of the balance-of-plant including the turbine building. The Tennessee Valley Authority's Office of Engineering Design and Construction has responsibility for design and construction activities. The Tennessee Valley Authority's Office of Power has responsibility for the operation and maintenance of the plant. Quality assurance aspects of the project are discussed in Section 17.0 of this report.

The proposed station organization will consist of approximately 195 persons under the direction of the Power Plant Superintendent and the Assistant Power Plant Superintendent. Reporting to the Assistant Plant Superintendent will be: a Power Plant Results Supervisor with a staff of approximately 24 persons responsible for plant performance, reactor engineering, chemistry control and instrument maintenance; a Power Plant Operations Supervisor, with a staff of approximately 48 persons responsible for plant operations; a Power Plant Maintenance Supervisor, with a staff of approximately 58 persons responsible for mechanical and electrical maintenance; a Health Physicist, with a staff of approximately eleven persons, responsible for plant health physics activities; an Engineering Unit with approximately five engineers and additional supporting groups such as administrative services, public safety officers and storekeepers. A Nuclear Plant Quality Assurance Staff Supervisor, with a staff of approximately eight persons, reports directly to the Power Plant Superintendent. This is a customary type of organizational arrangement for a two unit operation the same site. The shift crew for the operation of one unit will consist of sev persons including one senior licensed operator and four licensed operators.

The applicant has stated that the minimum qualification requirements for plant personnel will, at least, meet those described in Regulatory Guide 1.8, "Personnel Selection and Training."

Technical support for the plant staff will be provided primarily by the Tennessee Valley Authority's Division of Power Production.

We conclude that the applicant has established an acceptable organization to implement its responsibilities for the design and [illegible]ion of the Phipps Bend plant

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and that the proposed plant organization, the proposed qualifications of personnel, and the proposed plan for offsite technical support are acceptable.

13.2 Training Program

Responsibility for administration and supervision of the overall training program rests with the Assistant Power Plant Superintendent. He is responsible for ensuring that initial onsite training, retraining, replacement training and general employee training is maintained on an educational level that is adequate for safe and efficient operation of the plant.

The training program for the initial plant staff will be arranged and implemented to meet the needs of each man, depending upon his background, previous training and job assignment. All formal training for the plant staff should be completed well in advance of fuel loading to allow the plant staff sufficient time to aid in the preparation of operating and startup procedures and checkout systems.

Personnel to be used to man the Phipps Bend plant will be trained and experienced personnel and will be drawn from the Tennessee Valley Authority's nuclear plants at Browns Ferry, Sequoyah, Watts Bar, Bellefonte and Hartsville Nuclear Plants. Since there is a possibility that personnel may be transferred to Phipps Bend from both boiling water reactor and pressurized water reactor plants, training schedules have been established based on experience and will be further refined in the Final Safety Analysis Report. Training for persons to be licensed will include: plant system, reactor technology, simulator training at the Tennessee Valley Authority's training center and onsite work study programs. Specialist training for plant engineers, technicians and maintenance personnel will be individually tailored to meet the experience and background of the individuals selected. Browns Ferry Nuclear Plant will be utilized for special training. Station personnel will receive training in radiation control and safety, the emergency plan and the plant security plan.

The information submitted relative to these subjects is acceptable for the construction permit stage of review, for the preoperational test program, for operator licensing and for fuel loading.

13.3 Emergency Planning

We have reviewed the applicant's preliminary plans for coping with emergencies, as detailed in the Phipps Bend Preliminary Safety Analysis Report, Section 13.3, and Amendments 4, 5, 7, 9, 10, 12, and 13 to the Preliminary Safety Evaluation Report. The emergencies listed included fire, personnel injury, tornadoes and high winds, and incidents that could result in the release of significant amounts of radioactivity. The normal shift operating crew will provide the nucleus of the plant's emergency organization. The shift engineer will be responsible for declaring an emergency and acting as emergency director until relieved by the plant superintendent. During a nuclear emergency at the plant site, a central emergency control center located in

Chattanooga will be staffed with personnel to provide assistance to the site as necessary and provide a point of contact with various Tennessee Valley Authority divisions and offices and with offsite organizations. The applicant has identified the notification responsibilities within the organization to ensure prompt and effective communications in the event of an emergency.

Initial contacts and arrangements have been made with the following agencies: Tennessee State Department of Public Health, Tennessee State Department of Civil Defense, Tennessee State Department of Agriculture, Tennessee State Department of Public Safety, Tennessee State Department of Conservation, Energy Research and Development Administration's Savannah River Operations Office, Energy Research and Development Administration's Oak Ridge Operations Office, and Oak Ridge Associated Universities Hospital. Additional participants in the Phipps Bend Radiological Emergency Plan will include

(1) Sheriff's Department of Hawkins County Tennessee
(2) Civil Defense Coordinators of Hawkins County Tennessee
(3) Local Police Department
(4) Local Fire Department
(5) Local Ambulance Service
(6) Holston Valley Community Hospital
(7) Environmental Protection Agency, Region IV, Atlanta, Georgia
(8) Eastern Environmental Radiation Facility, Montgomery, Alabama

The Tennessee State Department of Public Health has the primary responsibility for planning for radiological emergency response in the environs of the plant. This agency will coordinate assistance from other State agencies as required, inform them of actions to be taken under their respective statutory authority, and assist them in developing emergency procedures.

The applicant has performed analyses to confirm the practicability of taking protective measures, including evacuation, within and beyond the site boundary during the expected lifetime of the plant. These protective measures will be based on previously determined dose rates, population distributions, meteorological conditions, and plant conditions that could result in conditions at the site boundary requiring action. The measures will include preplanned evacuation routes, reassembly points, traffic control, and public announcements.

The emergency plans include a description of the medical facilities at the plant and the arrangements made with other facilities to provide additional support. Plant medical facilities will include a treatment area consisting of an emergency room, treatment room, bedroom, physiotherapy room, and waiting room.

A full-time nurse will be on duty during the day shift. A complete stock of medical supplies and first aid equipment will be available. One ambulance will be maintained at the site. Arrangements will be made with a local private ambulance service to provide emergency service to the plant and affected areas in the event that more than one ambulance is required.

Each person having unescorted access to the plant will receive training in emergency procedures. Specific training will be conducted for individuals assigned to the plant emergency organization. This will include first aid training, radiological hygiene training, decontamination training, and training in the emergency procedures. Tennessee Valley Authority will assist in providing training in decontamination and treatment of contaminated patients to the staff of the local hospital and the commercial ambulance service.

The plant will be designed and will incorporate features to assure the capability of plant evacuation and of re-entry to mitigate the consequences of an accident, including radiation emergency alarms, communications system, and evacuation routes. The plant control room will be designed for continuous occupancy during and following the most severe postulated accidents as analyzed in Section 15 of the Phipps Bend Preliminary Safety Analysis Report.

We have reviewed the applicant's preliminary plans for coping with emergencies. We conclude that they meet the requirement of 10 CFR Part 50, Appendix E, Part II, are consistent with facility design features, analyses of postulated accidents, and characteristics of the proposed site location, and provide reasonable assurance that appropriate protective measures can be taken within and beyond the site boundary in the event of a serious accident.

13.4 Review and Audit

The applicant has described his plans for review and audit of plant operations in Section 16.6 of the Phipps Bend Preliminary Safety Analysis Report and committed to a review and audit program generally in accordance with American National Standards Institute N18.7, "Administrative Controls for Nuclear Power Plants." We conclude that these plans are acceptable for the construction permit review.

13.5 Procedures

Actions concerning structures, systems or components of the Phipps Bend plant that are safety related will be conducted according to approved written procedures. In accordance with Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," the plant procedures will be divided into the following categories:

(1) Administrative Procedures
(2) Operations Procedures
(3) Maintenance Procedures
(4) Surveillance Procedures
(5) Abnormal Procedures
(6) Emergency Procedures

(7) Refueling Procedures
(8) Tech 'cal Procedures
(9) Radiological Procedures

The information submitted on these subjects is acceptable for the construction permit phase of review.

13.6 Industrial Security

On February 24, 1977 the Commission published new requirements for the physical protection of nuclear power plants against acts of sabotage (10 CFR 73.55). This new rule does not require applicants for construction permits to demonstrate compliance at this stage but does require such at the operating license stage. As a result of our review of the applicant's preliminary plans for physical security, the staff concludes that a satisfactory planning base has been described by the applicant upon which a complete security program can be developed to demonstrate compliance with the new regulations at the appropriate time. We will continue to work with and provide guidance to the applicant to assure this end.

14.0 INITIAL TESTS AND OPERATION

We have completed our review of the information provided by the Tennessee Valley Authority on the initial test program for the Phipps Bend plant. The review included:

(1) Evaluation of the scope of the applicant's test program including the responsibilities and qualifications of participating organizations, the general testing objectives, the divisions between major phases of the test program, the administrative controls governing the test program, and the extent to which the test program would verify the functional adequacy of the facility.

(2) Evaluation of the testing proposed for unique or first-of-a-kind design features for the facility.

(3) Evaluation of the applicant's plans relative to utilization of Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water Cooled Power Reactors," and other regulatory guides applicable to testing in the formation of its test program.

(4) Evaluation of the applicant's plans for review and utilization, where applicable, of operating experiences from other reactors in the development of its test program.

(5) Evaluation of the applicant's test program schedule to establish that sufficient time for testing is planned and that the schedule is compatible with the schedules for the hiring and training of plant personnel.

(6) Evaluation of the applicant's plans to utilize plant operating and emergency procedures to the extent practicable during preoperational testing.

(7) Evaluation of the applicant's plans to augment the station staff, as necessary, during the test program.

On the basis of this review we have concluded that the Tennessee Valley Authority has made acceptable preliminary plans for the staffing, development and conducting of the initial test program.

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15.0 ACCIDENT ANALYSIS

15.1 Introduction

Our evaluation of the capability of GESSAR to withstand abnormal operational transient and postulated accidents is presented in Section 15.0 of Appendices A, C and D to this report. Therefore, the discussion below is limited to assessing the radiological consequences of accidents postulated as design basis accidents for the Phipps Bend plant.

15.2 Radiological Consequences of Design Basis Accidents

The radiological consequences resulting from the design basis loss-of-coolant, fuel handling, and control rod drop accidents have been evaluated in order to demonstrate the effectiveness of the GESSAR engineered safety features for the Phipps Bend site. Our acceptance criteria are that the doses from the design basis accidents, as evaluated by us, must be within the exposure guidelines of 10 CFR Part 100. As indicated in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," the exposure guidelines considered appropriate at the construction permit stage are 150 rem to the thyroid and 20 rem to the whole body.

On the basis of our experience with the evaluations of the steam line break accident for boiling water reactor plants of similar design, we have concluded that the consequences of this accident can be controlled by limiting the permissible radioactivity concentrations in the reactor coolant so that potential offsite doses will be small. We will include limits in the technical specifications on the coolant activity concentrations such that the potential two-hour doses at the minimum exclusion distance, as calculated by us, will be appropriately small fractions of the guideline values of 10 CFR Part 100.

15.2.1 Loss-of-Coolant Accident

In evaluating the design basis loss-of-coolant accident for the proposed Phipps Bend site, the containment model used to calculate the GESSAR loss-of-coolant accident doses was modified to account for (1) the difference in the primary containment design leak rate for Phipps Bend (0.5 percent per day) versus the leak rate used in the GESSAR analysis (0.3 percent per day) and (2) the decision by the General Electric Company to replace the original main steam line isolation valve leakage control system with a new positive sealing system which is designed to prevent the leakage of fission products through the closed main steam isolation valves.

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The primary containment was assumed to leak at the design leak rate of 0.5 percent by volume per day throughout the course of the accident. The majority of this leakage (92 percent) was assumed to go to the shield building annulus and the remaining leakage (eight percent) to the auxiliary and fuel building areas of the secondary containment. The leakage which goes to the shield building annulus was assumed to go directly without mixing to the intake of the shield building annulus recirculation and exhaust system. A fraction of the leakage which entered the shield building annulus recirculation and exhaust system was assumed to be exhausted to the atmosphere through the standby gas treatment system with the remainder of the leakage which entered the shield building annulus recirculation and exhaust system recirculated to the shield building annulus where credit was given for mixing in 50 percent of the annulus free volume. The split of the leakage between the exhaust and recirculation fractions was assumed to be in the same proportion as the flow rates in the exhaust and recirculation paths of the shield building annulus recirculation and exhaust system.

The eight percent of the primary containment leakage which goes to the auxiliary and fuel building areas of the secondary containment, with the exception of the first 60 seconds, was assumed to be transported directly to the standby gas treatment system with no credit for holdup or mixing. For the first 60 seconds following the accident, the primary containment leakage to the auxiliary and fuel buildings was assumed to be released untreated to the atmosphere as the pressure in these areas is not sufficiently negative to insure filtration through the standby gas treatment system.

The Tennessee Valley Authority has committed to incorporate design provisions to eliminate containment bypass leakage and to provide a positive sealing system for the main steam line isolation valves. The main steam positive leakage control system will be designed so that the inward pressure on the isolation valves will always exceed the outward pressure following an accident thereby precluding the leakage of fission products through the valves. Our evaluation of the design basis for the main steam isolation valve positive seal system is presented in Sections 6.2.3 and 9.3.1 of Appendix C to this report.

The doses resulting from the design basis loss-of-coolant accident are given in Table 15.1 and the principal assumptions used in the analysis are listed in Table 15.2. Based on our evaluation of the postulated loss-of-coolant accident and the commitment of the Tennessee Valley Authority to eliminate containment bypass and main steam line leakage, we find that the postulated loss-of-coolant accident doses for the Phipps Bend plant are well within the exposure guidelines of Regulatory Guide 1.3 for the evaluation of a plant at the construction permit stage. As outlined in Section 6.2.3 of this report, we will incorporate the results of our review of the General Electric Company topical report on the containment bypass and main steam line positive sealing systems into our evaluation of the radiological consequences of the loss-of-coolant accident during the Phipps Bend operating license review.

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TABLE 15.1

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

Accident	Exclusion Area Boundary[1] 2-Hour Dose, Rem		Low Population Zone[2] 30-Day Dose, Rem	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss-of-Coolant	44	12	70	3.2
Fuel Handling	3.4	2.2	0.2	0.1
Control Rod Drop	16	3.2	6.2	0.5

(1) Minimum exclusion area boundary distance is 760 meters.

(2) Low population zone outer radius is 4,830 meters (3 miles).

TABLE 15.2

ASSUMPTIONS USED TO CALCULATE LOSS-OF-COOLANT ACCIDENT DOSES FROM CONTAINMENT LEAKAGE

Power level	3758 megawatts thermal
Operating Time	3 years
Containment Leak Rate	0.5 percent per day
Bypass Fraction	0 percent
Core Fraction Available for Leakage from Containment:	
Iodines	25 percent
Noble Gases	100 percent
Standby Gas Treatment System Filter Efficiency For All Iodines	99 percent
Primary Containment Volume	1.68×10^6 cubic feet
Shield Building Annulus Volume	5.04×10^5 cubic feet
Mixing Fraction in Annulus Volume	50 percent
Shield Building Reci culation System Flow Rates, cubic :eet per minute	

Time Period	Exhaust	Recirculation
0-220 seconds	0	0
220 seconds - 2 hours	320	4680
2-8 hours	110	4890
8 hours - 30 days	70	4930

Primary Containment Leak Paths
(Percent of Primary Containment Leakage)

Time Period	To Annulus	Exhausted to Atmosphere Thru Standby Gas Treatment System	To Atmosphere Untreated
0-60 seconds	92	0	8
60-220 seconds	92	8	0
220 seconds - 2 hours	86.1	13.9	0
2-8 hours	90	10	0
8 hours - 30 days	90.7	9.3	0

Relative Concentration Values (seconds per cubic meter)

0-2 hours at 760 meters	1.8×10^{-3}
0-8 hours at 4830 meters	1.2×10^{-4}
8-24 hours at 4830 meters	8.0×10^{-5}
1-4 days at 4830 meters	3.5×10^{-5}
4-30 days at 4830 meters	1.1×10^{-5}

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15.2.2 Fuel Handling Accident

The fuel handling accident was evaluated by employing an analytical model identical to that used during the GESSAR review in conjunction with the Phipps Bend atmospheric diffusion factors. The calculated doses, shown in Table 15.1, are well within the exposure guidelines of 10 CFR Part 100. The principal assumptions used in the analysis of the fuel handling accident are shown in Table 15.3.

15.2.3 Control Rod Drop Accident

The radiological consequences of the control rod drop accident were evaluated for the proposed Phipps Bend site. The model used in the analysis was updated to include the revisions which have been made to the control rod drop accident model since the GESSAR review. These revisions concern the amount of activity assumed to be available for release to the environment and the condenser leak rate. As these revisions tend to offset each other, the total impact on the doses is not significant. The control rod drop accident doses, shown in Table 15.1, are well within the exposure guidelines of 10 CFR Part 100. The assumptions employed in the evaluation of the control rod drop accident are shown in Table 15.4.

15.3 Postulated Radioactivity Releases Due to Liquid Tank Failures Outside Containment

The consequences of component failures which could result in contaminated liquid releases to the environs were evaluated for components containing liquid radioactive materials located outside containment. The scope of the review included the calculation of radionuclide inventories in station components at design basis fission product levels, the mitigating effects of the plant design, and the effect of site geology and hydrology.

Based on our evaluation, a rupture of the high conductivity waste tank will result in the highest radionuclide concentrations in the nearest unrestricted surface or potable water supply. This tank will have a volume of 18,000 gallons. In our evaluation, we assumed a tank failure to occur with the tank 80 percent full and with a liquid activity of approximately 3.4 microcuries per cubic centimeter (consistent with an offgas rate of 100 microcuries per thermal megawatt-second after 30 minutes delay). We assumed the radioactive liquid to enter the ground water and migrate towards the Holston River. As stated in Section 2.4.4 of this report, there are no ground water users between the site and the river. The estimated travel time is 40 days and dilution of the released water by the ground water and surface water is approximately 36,000 to one.

Considering radioactive decay over the 40 day transit time and a dilution factor of 36,000 to one, a postulated rupture of the high conductivity waste tank would result in concentrations at the nearest postulated recipient which are a small fraction of the limits of 10 CFR Part 20 for unrestricted areas.

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TABLE 15.3

ASSUMPTIONS USED TO CALCULATE CONSEQUENCES

OF FUEL HANDLING ACCIDENT

Power Level	3758 megawatts thermal
Shutdown Time	24 hours
Total Number of Fuel Rods in the Core	46,116
Number of Fuel Rods Involved in Accident	98
Power Peaking Factor of Damaged Rods	1.5
Inventory Released From Damaged Rods (Iodines and Noble Gases)	10 percent
Pool Decontamination Factors	
Iodines	100
Nobles Gases	1
Iodine Fractions Released from Pool	
Elemental	75 percent
Organic	25 percent
Filter Efficiencies	
Elemental	99 percent
Organic	99 percent
Relative Concentration Value	
0-2 hours	1.8×10^{-3} seconds per cubic meter

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TABLE 15.4

ASSUMPTIONS USED TO CALCULATE CONSEQUENCES

OF CONTROL ROD DROP ACCIDENT

Power Level	3758 megawatts thermal
Total Number of Fuel Rods in Core	46,116
Number of Fuel Rods Involved in Accident	770
Power Peaking Factor of Damaged Rods	1.5
Fraction of Fission Product Inventory Released to Coolant	
Noble Gases	100 percent
Iodines	50 percent
Iodine Fraction Released to Condenser	10 percent
Iodine Fraction Plated Out in Condenser	90 percent
Condenser Leak Rate	1.0 percent per day
Relative Concentration Values (seconds per cubic meter)	
0-2 hours at 760 meters	1.8×10^{-3}
0-8 hours at 4830 meters	1.2×10^{-4}
0-24 hours at 4830 meters	8.0×10^{-5}

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Based on the foregoing evaluation, we conclude that the postulated failure would not result in radionuclide concentration in excess of 10 CFR Part 20 limits at the nearest potable or surface water supply. Therefore, we will not require the applicant to incorporate additional provisions in the design to mitigate the effects of component failures involving contaminated liquids.

15.4 Anticipated Transients Without Scram

Our evaluation of anticipated transients without scram for GESSAR is presented in Section 15.4 in Appendix A and Appendix C to this report. The applicant by letter dated August 23, 1976 (J. E. Gilleland of the Tennessee Valley Authority to Ben C. Rusche of the Nuclear Regulatory Commission) committed to adopt the GESSAR resolution.

15.5 Conclusions

On the basis of our evaluation, we conclude that the Phipps Bend loss-of-coolant accident doses meet the exposure guidelines of Regulatory Guide 1.3. In addition, the doses for the fuel handling accident and the control rod drop accident are well below the exposure guidelines of 10 CFR Part 100. We have also concluded that radioactivity released due to postulated liquid tank failures outside of the containment would not result in radionuclide concentrations in excess of 10 CFR Part 20 limits at the nearest potable or surface water supply.

16.0 TECHNICAL SPECIFICATIONS

The technical specifications in an operating license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the Commission. Final technical specifications will be developed and evaluated at the operating license stage. However, in accordance with Section 50.34 of 10 CFR Part 50, an application for a construction permit is required to include preliminary technical specifications. The regulations require an identification and justification for the selection of those variables, conditions or other items which are determined as a result of the preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, with special attention given for those items which may significantly influence the final design.

We have reviewed the proposed technical specifications presented in Section 16.0 of the Phipps Bend Preliminary Safety Analysis Report with the objective of identifying those items that would require special attention at the construction permit stage, to preclude the necessity for any significant change in design to support the final technical specifications. The proposed technical specifications are similar to those being developed or in use for plants of a similar design to the Phipps Bend Nuclear Plant. We have not identified any items which require special attention at this stage of our review.

On this basis we have concluded that the proposed technical specifications are acceptable.

17.0 QUALITY ASSURANCE

17.1 General

The description of the quality assurance program for the Phipps Bend plant is contained in Section 17 of the Phipps Bend Preliminary Safety Analysis Report. Section 17 describes the quality assurance program of the applicant, the Tennessee Valley Authority; the supplier of the nuclear steam supply system, and the overall designer of the nuclear island the General Electric Company; and the designer of the nuclear island buildings the C. F. Braun and Company.

The Tennessee Valley Authority's Division of Engineering Design is the principal designer, architect engineer, and contractor for the balance-of-plant. The Tennessee Valley Authority's Division of Construction is the contractor responsible for constructing the entire facility in accordance with design specifications furnished by the Division of Engineering Design for the balance-of-plant and for the General Electric Company for the nuclear island. The C. F. Braun and Company, as a subcontractor of the General Electric Company, will provide the architect-engineering design of the nuclear island buildings which will include the reactor building, the fuel building, the auxiliary building, the radwaste building, the control building and the diesel generator building.

The applicant has contracted with the General Electric Company to supply the nuclear steam supply systems and design the nuclear island. The General Electric Company will be responsible for establishing and implementing a satisfactory quality assurance program for its scope of work. The Tennessee Valley Authority will be responsible for the total Phipps Bend plant's quality assurance program and will be organized to control and verify the quality assurance programs of contractors furnishing safety related equipment.

Our evaluation of the description of the Phipps Bend plant's quality assurance program is based on a review of this information and detailed discussions with the applicant to determine the qualification and capability of the applicant and its principal contractor, the General Electric Company, and the General Electric Company's principal subcontractor, the C. F. Braun and Company, to comply with the requirements of Appendix B to 10 CFR Part 50, applicable regulatory guides, and industry standards.

17.2 Tennessee Valley Authority

17.2.1 Organization

The Tennessee Valley Authority's corporate organization is shown in Figure 17.1. The Board of Directors establishes general policies and programs, reviews and appraises progress and results, and approves projects and programs. Reporting to the Board of

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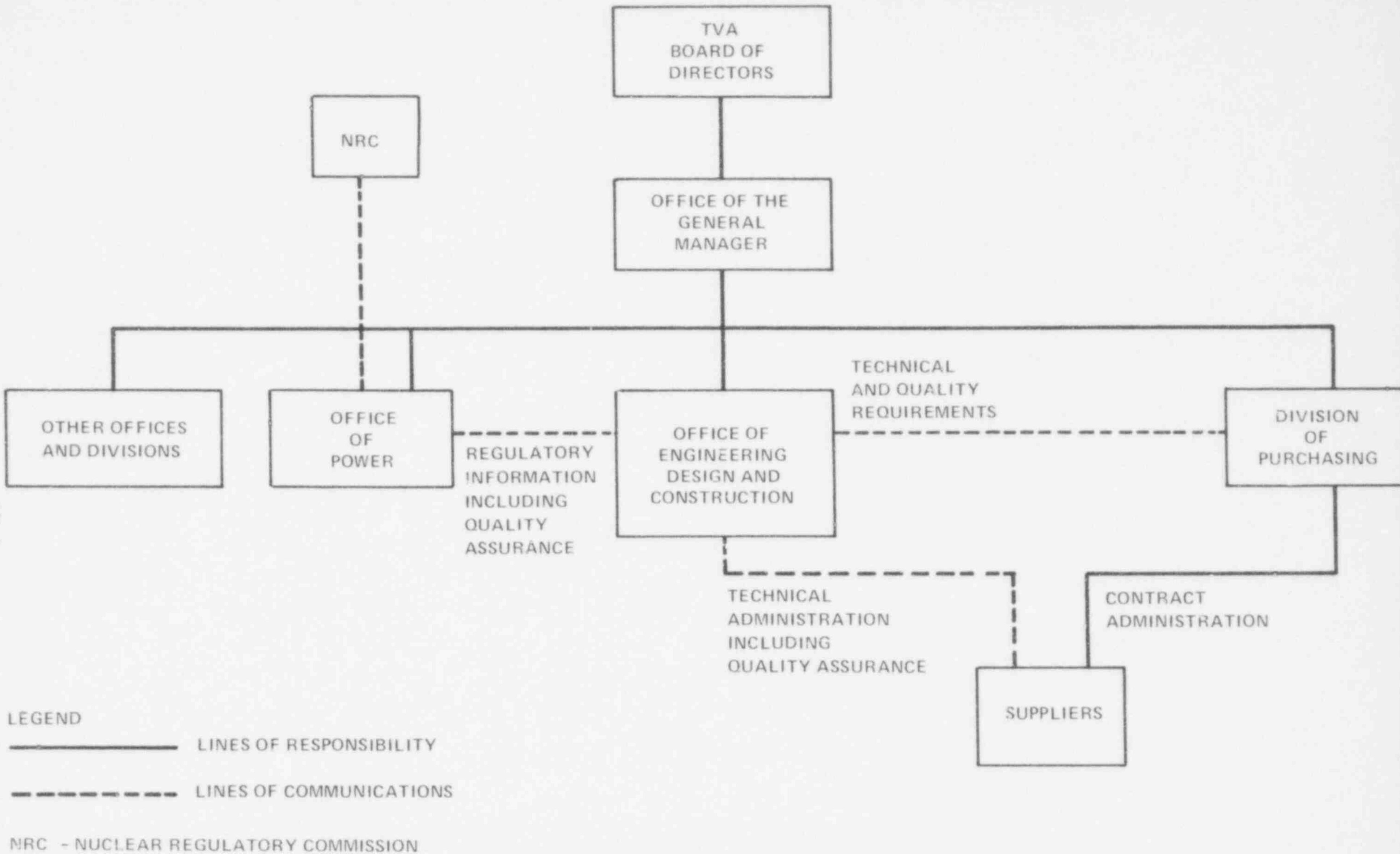

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FIGURE 17.1 TENNESSEE VALLEY AUTHORITY ORGANIZATION

Directors is the General Manager who is the principal Tennessee Valley Authority administrative officer. Reporting to the General Manager are the three major organizational elements of the Tennessee Valley Authority involved in the Phipps Bend Nuclear Plant; these are (1) Office of Power, (2) Division of Purchasing, and (3) Office of Engineering Design and Construction. The Office of Power has overall responsibility for coordinating the safety analyses and licensing arrangements, and for providing an overview of the quality assurance program. The Division of Purchasing is responsible for administering all procurements in accordance with the technical and quality assurance requirements of the Office of Engineering Design and Construction. The Office of Engineering Design and Construction is responsible for all design and construction activities.

The Office of Engineering Design and Construction organization is shown in Figure 17.2. The Office of Engineering Design and Construction is divided into two major divisions: the Division of Engineering Design and the Division of Construction. The Manager of the Office of Engineering Design and Construction has overall responsibility for quality assurance during design and construction. He has delegated the quality assurance responsibilities as follows: (1) quality assurance in design and procurement to the Director of Engineering Design, (2) quality assurance in construction to the Director of Construction, and (3) determination that the overall quality assurance program for the Office of Engineering Design and Construction meets the applicable requirements to the Quality Assurance Manager.

The Division of Engineering Design has the Quality Assurance Staff and an Inspection and Testing Branch. The Division of Engineering Design Quality Assurance Staff is managed by a chief who is responsible to the Division of Engineering Design for reviewing and coordinating the quality assurance program within the department and for its interface with others. The Division of Engineering Design staff is responsible for: (1) internal audits, (2) supplier audits, (3) review and signature of purchase orders and (4) approval of procedures and surveillance plans of the Inspection and Testing Branch. The Inspection and Testing Branch is responsible to the Division of Engineering Design for assuring that suppliers fulfill the technical and quality assurance requirements specified in the procurement documents.

The Division of Construction has a Quality Assurance Staff and a Construction Engineer's organization to verify that its quality assurance functions have been correctly performed. The Division of Construction Quality Assurance Staff is managed by a chief who is responsible to the Division of Construction for reviewing and coordinating the quality assurance program within the department and for its interface with others. The Division of Construction Quality Assurance Staff is responsible for: (1) internal audits, (2) review and approval of all quality control procedures issued by the Division of Construction, and (3) review and approval of site procurement documents for quality assurance/quality control requirements. The Construction Engineer's organization is responsible to the Division of Construction through the

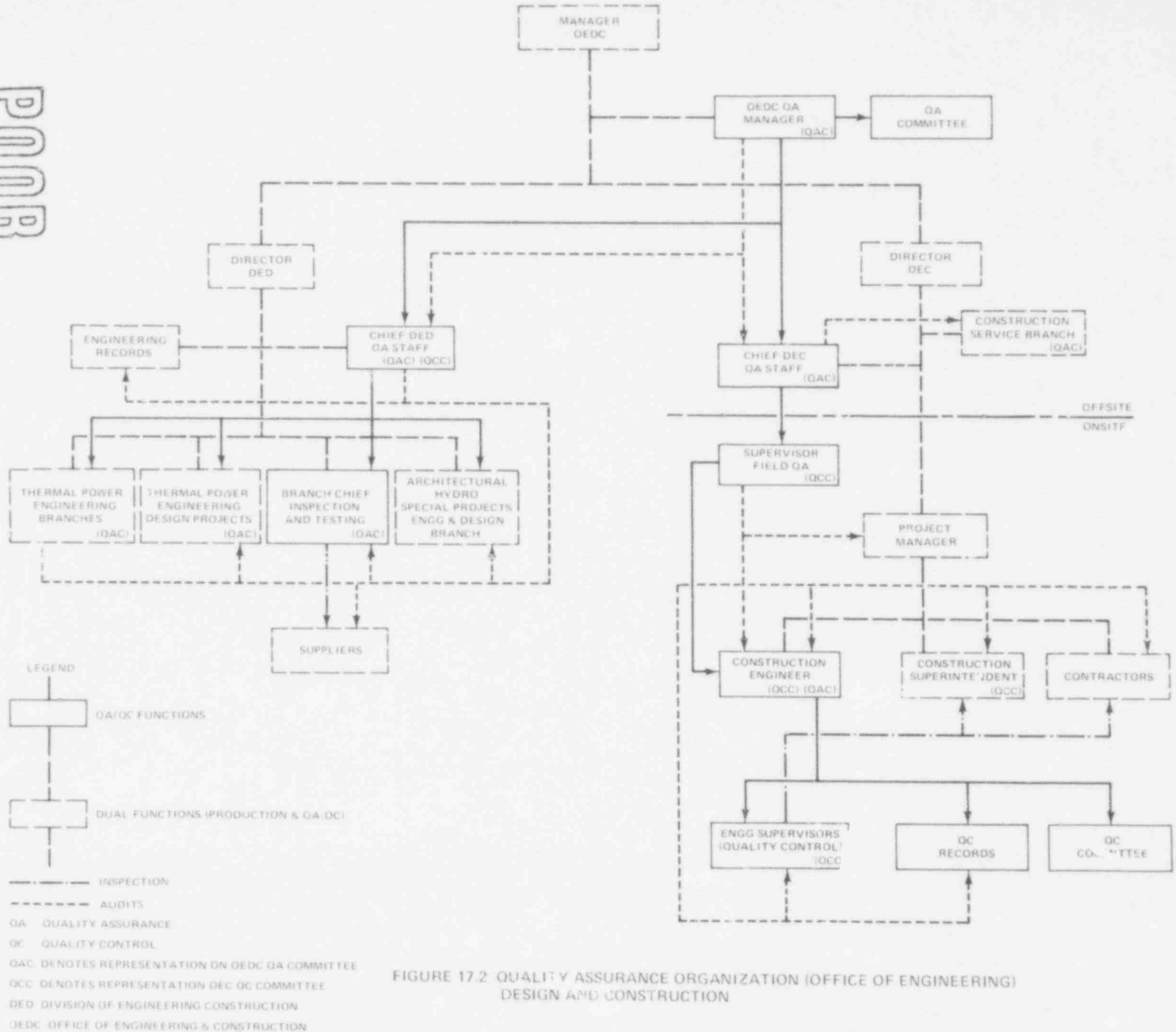

FIGURE 17.2 QUALITY ASSURANCE ORGANIZATION (OFFICE OF ENGINEERING) DESIGN AND CONSTRUCTION

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Project Manager for performing quality control activities in accordance with the Division of Construction Quality Assurance Staff's approved procedures for work performed on site by the Tennessee Valley Authority work forces and contractors.

Stop work authority has been delegated to inspectors in the Inspection and Testing Branch and to the Construction Engineer.

The Quality Assurance Manager is chairman of the Office of Engineering Design and Construction Quality Assurance Committee established to obtain better coordination and agreement among quality affecting groups. The Committee is composed of representatives from these groups. The Quality Assurance Manager is on the same organizational level as those whose work he verifies and he has direct access to the Manager of Office of Engineering Design and Construction to obtain satisfactory resolution of problems.

Management audits are performed under the direction of the Quality Assurance Manager and are reported to the Manager of Office of Engineering Design and Construction. They consist of periodic audits of the design and procurement phase activities and records; and of construction phase procedures, activities, and records.

We find that the Tennessee Valley Authority's description for implementing its quality assurance program (with corporate level management involvement, authority from management to enforce quality assurance requirements, and stop work authority) is acceptable.

Our evaluation of the Tenneseee Valley Authority's quality assurance/quality control organizations is that they are sufficiently independent of the organizations whose work they verify; they have clearly defined responsibilities and authorities; they have adequately defined the qualification requirements for their supervisory personnel; they are organized so that they can identify quality assurance problems in organizations performing quality related work; they can initiate, recommend, or provide solutions; and they can verify implementation of solutions. We therefore conclude that the Tennessee Valley Authority's organization complies with Appendix B to 10 CFR Part 50 and is acceptable.

17.2.2 Quality Assurance Program

Chapter 17 of the Phipps Bend Preliminary Safety Analysis Report provides a listing of the quality assurance documents and quality related procedures used to administer the quality assurance program with a brief description of the contents of each. A matrix of these procedures cross referenced to the related criteria of Appendix B to 10 CFR Part 50 is also given. All quality assurance policy procedures are approved by the Manager of the Office of Engineering Design and Construction, the Directors of the Division of Engineering Design and the Division of Construction, and the Quality Assurance Manager. Based on our review, we conclude that each criterion of Appendix B

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to 10 CFR Part 50 has been specifically included in written procedures within the Tennessee Valley Authority's quality assurance program. The structures, systems, and components that are subject to this program have been identified in the Phipps Bend Preliminary Safety Analysis Report.

The applicant has committed to comply with the requirements of the Gray Book, "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," WASH-1283 (Revision 1) issued May 24, 1974, and the Green Book, "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," WASH-1309 issued May 10, 1974, or has described alternatives acceptable to us. The Gray and Green Books provide guidance on quality assurance requirements during the design, procurement, and construction phases of nuclear power plants. We find, with the commitment and our review of the description of the quality assurance policies and procedures, that the Tennessee Valley Authority's quality assurance program is acceptable.

The Tennessee Valley Authority has established a program for the indoctrination and training of its quality assurance/quality control personnel and technical personnel in quality related activities. This includes training as to the purpose, scope, and implementation of quality related manuals, procedures, and instructions and both training and qualification in the principles and techniques of activities being performed.

Basic design criteria are developed by responsible Thermal Power Engineering Branches in the Division of Engineering Design; these design criteria form the basis for translating such requirements into detailed designs which are developed by the Thermal Power Engineering Design Projects (See Figure 17.2). All design documents are reviewed in accordance with written procedures which require that the document be reviewed by qualified engineers other than those preparing the original document. All designs are reviewed and analyzed in accordance with applicable codes, standards, and regulatory requirements. We find the Tennessee Valley Authority's description of its design control satisfactory.

To provide control of purchased safety related structures, systems and components, each prospective supplier's quality assurance program must be reviewed and approved by the Division of Engineering Design Quality Assurance Staff. The Division of Engineering Design Quality Assurance Staff reviews purchase requisitions to assure that quality assurance requirements are adequately specified. The Inspection and Testing Branch is responsible for performing inspections or surveillance at suppliers' plants in accordance with procedures and surveillance plans approved by the Division of Engineering Design Quality Assurance Staff. Audits and feedback of nonconformance data are used by the Division of Engineering Design Quality Assurance Staff to measure supplier performance. We find the Tennessee Valley Authority's description of its control of purchased material, equipment and services satisfactory.

The applicant has established program requirements for the Tennessee Valley Authority and its contractors which assure there will be a documented system of records attesting to quality.

A comprehensive system of planned and documented audits is described by the Tennessee Valley Authority in the Phipps Bend Preliminary Safety Analysis Report. The audit program includes management audits, the Division of Engineering Design and the Division of Construction internal audits, and supplier and contractor audits. The Tennessee Valley Authority audits include evaluation of work areas, activities, processes and items.

Audits are preformed in accordance with written procedures or checklists by appropriately trained personnel having no direct responsibilities in the area audited. Audit results are documented and reported to appropriate levels of management for corrective action. Responses to the Tennessee Valley Authority's audit findings are verified for implementation and effectiveness during followup audits. We find the Tennessee Valley Authority's description of the audit activities for Phipps Bend plant acceptable.

Based on our review of the description of the Tennessee Valley Authority's quality assurance program contained in Chapter 17 of the Phipps Bend Preliminary Safety Analysis Report, we find that there are adequate and well defined procedures, a commitment to our quality assurance guidance, assurance of an independent inspection program, an adequate personnel training program, a documented system of records attesting to quality, an audit system to inform management of the effectiveness of the quality assurance program, and satisfactory management assessment of the status and adequacy of the quality assurance program.

We conclude that the Tennessee Valley Authority's quality assurance program for the Phipps Bend plant includes an acceptable quality assurance organization with adequate policies, procedures, and instructions to implement a program that will satisfy the requirements of Appendix B to 10 CFR Part 50.

17.3 General Electric Company

The General Electric Company is responsible for providing the nuclear steam supply systems for the Phipps Bend plant. Figure 17.3 shows the organization of the Boiling Water Reactor Operations of the General Electric Company which provides nuclear plant services and equipment. The Boiling Water Reactor Operation operates under a Deputy Division General Manager of the Nuclear Energy Division. Reporting to this Deputy Division General Manager are Department General Managers and the Manager, Product and Quality Assurance Operation.

Each Boiling Water Reactor Operations Department and the Product and Quality Assurance Operation contain an organization specifically responsible for quality assurance which reports at a management level sufficient to assure independence consistent with Criterion 1 of Appendix B to 10 CFR Part 50.

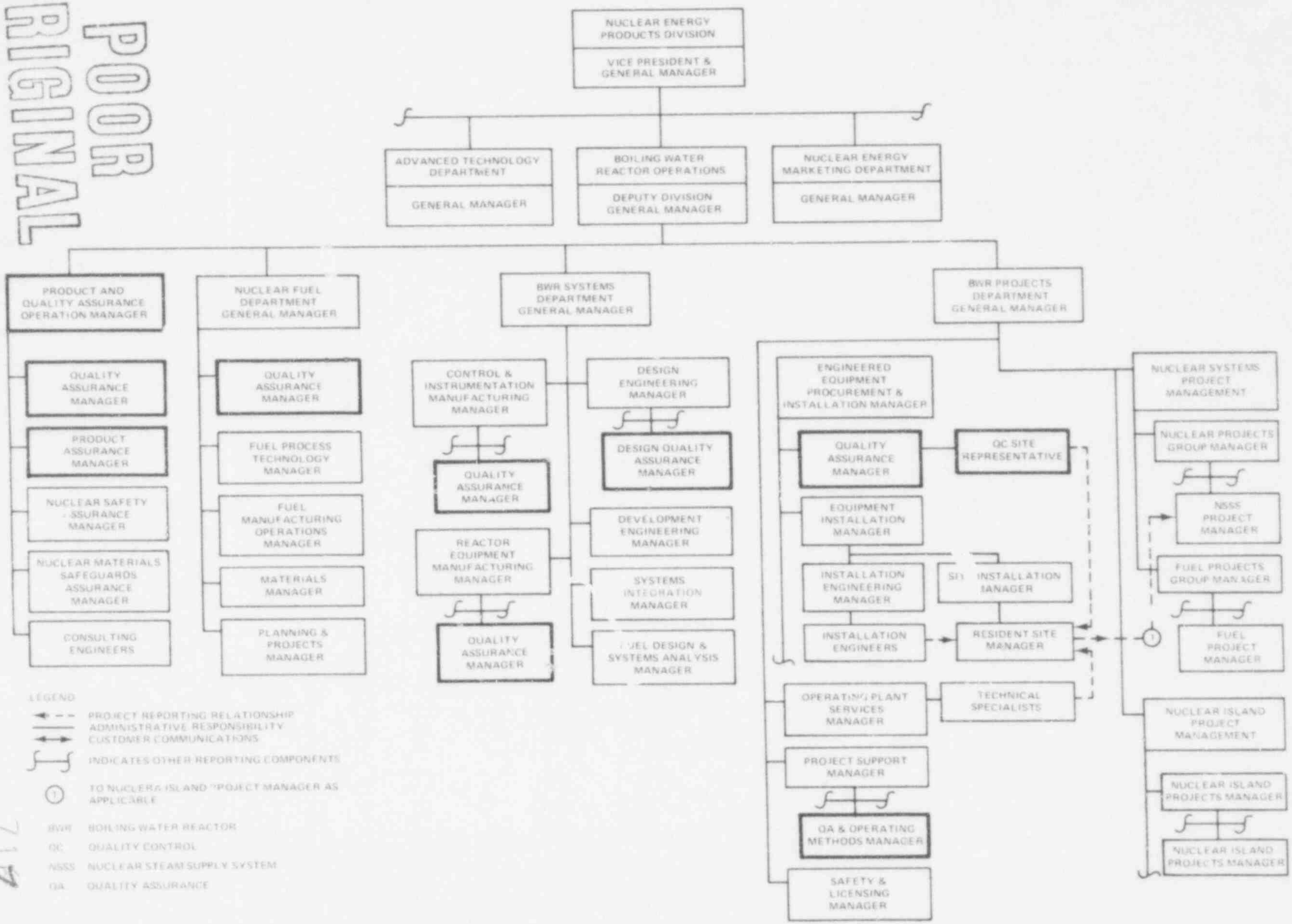

FIGURE 17.3 BOILING WATER REACTOR OPERATIONS QUALITY ASSURANCE ORGANIZATION

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Quality management in each Boiling Water Reactor Operations Department and the Product and Quality Assurance Operation is free of prime responsibility for schedule or cost, has the authority to stop work pending resolution of quality related matters, and has the freedom to: (1) identify quality related problems; (2) initiate, recommend, or provide solutions to quality related problems; (3) verify implementation of solutions, and (4) prevent further processing, shipment, installation, or utilization of nonconforming items until proper dispositioning has occurred. We find that the General Electric Company has adequately defined the responsibilities of the organizations performing quality assurance activities and that they are acceptable.

The Deputy Division General Manager of the Boiling Water Reactor Operations has established a Quality Council which includes the managers of the major quality assurance organizations in the division. The Manager, Product and Quality Assurance Operation is Chairman of the Quality Council. The Quality Council, which meets quarterly, permits development of solutions to common quality related problems and provides a separate line of communications to top Boiling Water Reactor Operations management. In addition, the Manager, Product and Quality Assurance Operation audits the Boiling Water Reactor Operations engineering, manufacturing, procurement, and construction organizations to assess the scope, implementation, and effectiveness of the quality assurance program.

The quality assurance program applies to the safety related systems and components within the General Electric Company's scope of work. The quality assurance program is compatible with the quality assurance guidance provided by us in the Gray and Green Books.

Though the basic scope of the quality assurance system used by the various Boiling Water Reactor Operations organizations is the same, each functional organization has its own system of guides, procedures, instructions, and manuals that prescribes the methods for accomplishing its portion of the quality assurance program. Division instructions, prepared by the Product and Quality Assurance Operation and issued by authority of the General Manager of the Nuclear Energy Division, establish procedures and practices where a standardized, uniform approach is necessary for control.

A matrix which relates the procedures of the various manuals to the applicable quality assurance criteria of Appendix B to 10 CFR 50 is given in the Phipps Bend Preliminary Safety Analysis Report. Based on our review of this matrix, we conclude that each criterion has been specifically covered in written procedures within the General Electric Company's quality assurance program.

The Quality Assurance program includes provisions for the control of design information. Design inputs are reviewed, and analyses are accomplished in accordance with applicable codes, standards, and regulatory requirements. Knowledgeable groups within the General Electric Company, including quality assurance personnel, independently review drawings and equipment specifications prior to issuance.

To provide control of purchased items and services, the General Electric Company evaluates the quality assurance programs of each prospective supplier of safety related items. Quality assurance engineers review purchase requisitions, purchase orders, and subsequent change notices. The General Electric Company reviews and retains supplier documentation which demonstrates acceptable quality, audits, and feedback of supplier performance.

The General Electric Company executes a comprehensive audit program which provides the Boiling Water Reactor Operations management with information on the effectiveness of the quality assurance program. The General Electric Company audits activities affecting quality at the General Electric Company and at supplier facilities. Audit areas include all quality related procedures and operations. Trained personnel not having direct responsibilities in the area being audited conduct the quality assurance audits in accordance with defined procedures and checklists.

In our review, we have evaluated the General Electric Company's quality assurance program for compliance with our regulations and applicable regulatory guides and industry standards. Based on this review, we conclude that the General Electric Company quality assurance program includes an acceptable organization and contains the necessary quality assurance provisions, requirements, and controls for compliance with Appendix B to 10 CFR Part 50 and applicable guides and standards and is acceptable for the nuclear steam supply systems for the Phipps Bend plant.

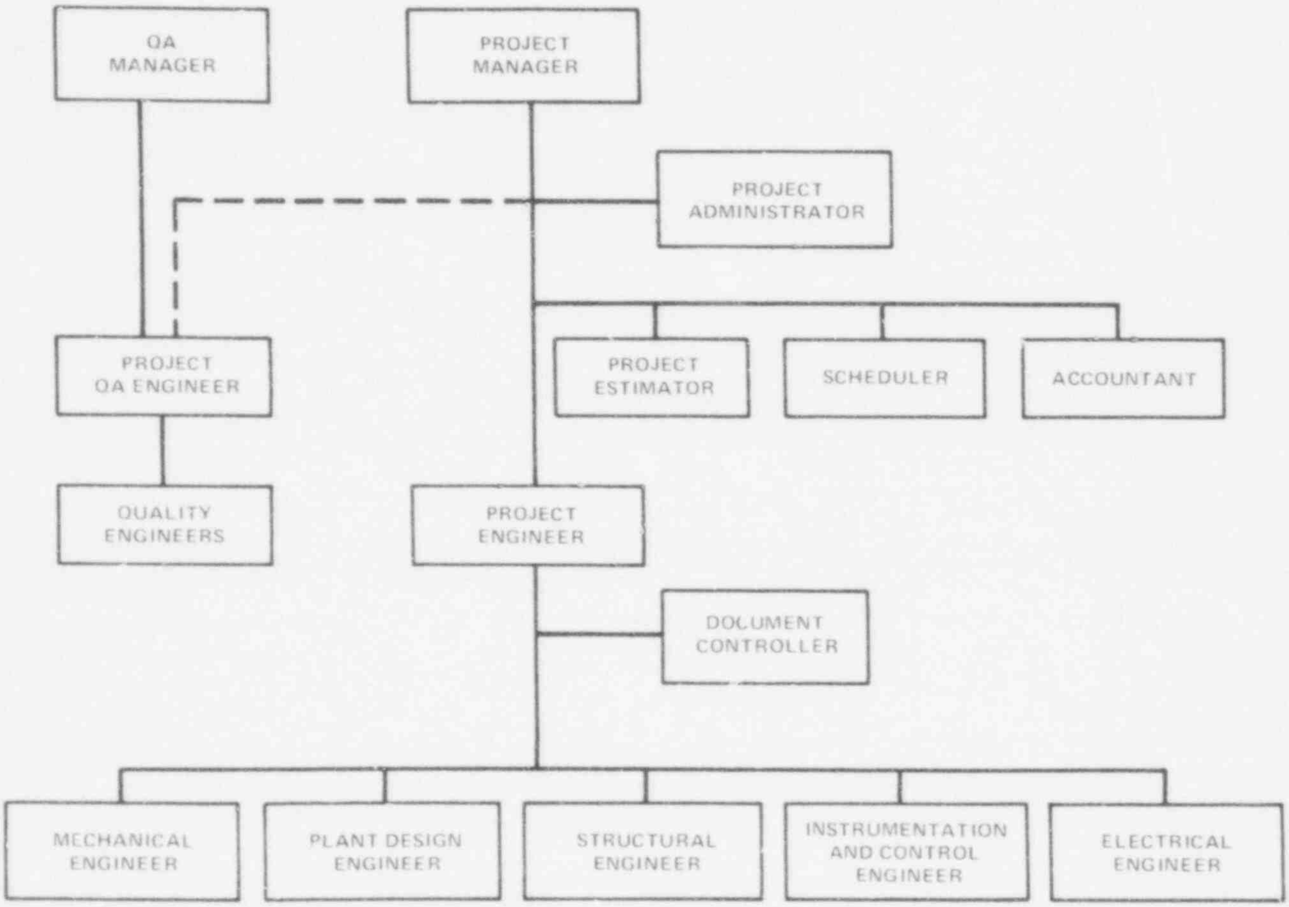
17.4 C. F. Braun and Company

The C. F. Braun and Company organization responsible for design activities is shown in Figure 17.4. The quality assurance organization is directed by the Vice President of Project Management who coordinates all the C. F. Braun and Company quality assurance/quality control activities, and directs these activities with respect to engineering.

Figure 17.5 shows the C. F. Braun and Company project organization. The Project Manager has the overall responsibility for planning and managing the project. The project quality assurance organization is headed by a Project Quality Assurance Engineer, who is responsible to the Quality Assurance Manager for coordinating all project quality assurance activities, and is the quality assurance representative on the Project Manager's staff. He is also responsible for assuring that quality assurance activities are integrated into project planning.

The Quality Assurance Manager has the responsibility, authority, and organizational freedom to establish quality assurance standards and procedures. He assures that the C. F. Braun and Company quality assurance program is properly established and that the quality assurance procedures are implemented.

Quality assurance personnel have the authority and organizational freedom to identify quality related problems; to initiate, recommend, or provide solutions; and to control further processing, delivery, or installation of a nonconf[illegible]ing item until proper



LEGEND

— — — COMMUNICATION AND COORDINATION

QA - QUALITY ASSURANCE

FIGURE 17.4 C. F. BRAUN AND COMPANY PROJECT ORGANIZATION

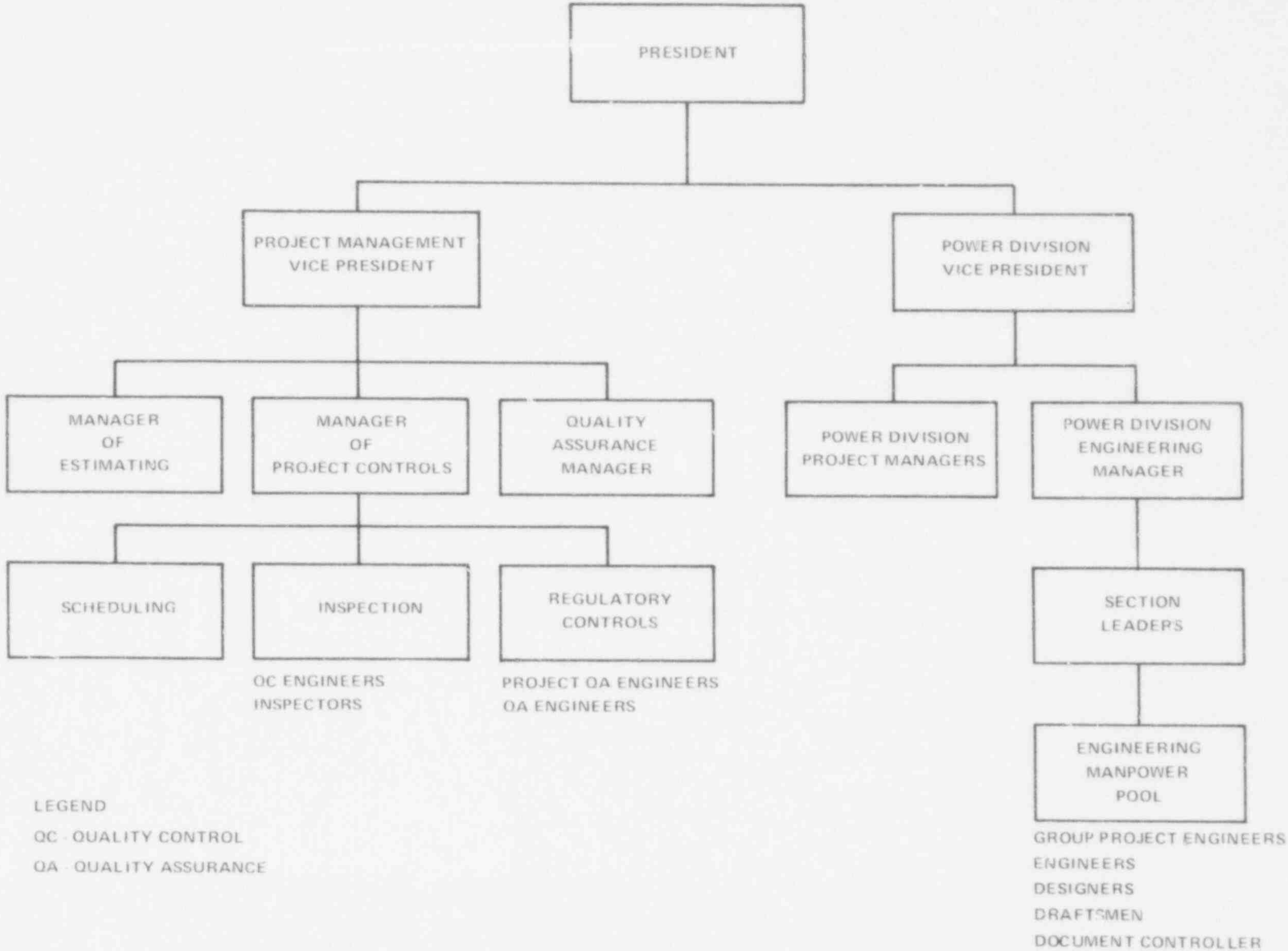


FIGURE 17.5 C. F. BRAUN AND COMPANY CORPORATE QUALITY ASSURANCE AND ENGINEERING ORGANIZATION

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disposition of the nonconformance has been approved. We conclude that the C. F. Braun and Company's organization provides sufficient direction and control of the quality assurance organization and have sufficient authority and organizational freedom to perform their functions effectively and without reservation.

Major quality assurance activities for the C. F. Braun and Company design of the nuclear island which are carried out by its quality assurance organization are: (1) review of design documents and (2) audits.

The quality assurance program applies to all safety related structures, systems, and components within the C. F. Braun and Company scope of work. C. F. Braun and Company has committed to comply with the Commission guidance on quality assurance provided in the Gray Book during the design of the nuclear island.

The quality assurance program for engineering includes review of applicable engineering instructions, procedures, specifications, and drawings to assure the quality requirements are clearly, accurately, and adequately stated. The program requires that design work be verified or reviewed by individuals within the engineering organization not responsible for originating the design and that a determination is made that the engineering specifications, procedures, instructions, and drawings comply with regulatory requirements and design bases.

The quality assurance program also provides for a comprehensive system of detailed audits to be performed by the C. F. Braun and Company quality assurance organization. The audits encompass the review and evaluation of all quality related activities associated with the quality assurance program and involve procedures, work areas, activities, and records. The program requires that the audits be conducted in accordance with preestablished procedures by qualified personnel not having direct responsibilities in the area being audited. The results are documented and distributed to the appropriate levels of management.

In our review, we have evaluated the C. F. Braun and Company's quality assurance program for compliance with our regulations and applicable Regulatory Guides and industry standards. Based on this review, we conclude that its organizational arrangement and its quality assurance program description comply with Appendix B to 10 CFR Part 50 and applicable guides and standards and are acceptable for the design of the reactor island.

17.5 Implementation of Quality Assurance Program

The Office of Inspection and Enforcement has conducted inspections to examine the implementation of Phipps Bend plant's quality assurance program. Based on its inspections and assessment, the Office of Inspection and Enforcement concludes that the implementation of the Phipps Bend Preliminary Safety Analysis Report commitments in the Phipps Bend plant's quality assurance program is consistent with the status of the nuclear project.

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17.6 Conclusion

In our review, we have evaluated the quality assurance program of the Tennessee Valley Authority, the General Electric Company, and the C. F. Braun and Company for the Phipps Bend Nuclear Plant to determine compliance with our regulations, applicable regulatory guides and industry standards. Based on this review, we conclude that the Tennessee Valley Authority and its principal contractor and subcontractor have described acceptable organizations, and that their quality assurance programs comply with Appendix B to 10 CFR Part 50, applicable regulatory guides and industry standards, and are acceptable for the design, procurement, and construction of the Phipps Bend plant.

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Phipps Bend application for a two unit facility is being reviewed by the Advisory Committee on Reactor Safeguards (Committee). We intend to issue a supplement to this Safety Evaluation Report after the Committee's report to the Commission, relative to its review, is available. The supplement will append a copy of the Committee's report and will address comments made by the Committee, and will also describe steps taken by the staff to resolve any issues raised as a result of the Committee's review.

19.0 COMMON DEFENSE AND SECURITY

The applicant states that the activities to be conducted will be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are citizens of the United States.

The applicant is not owned, dominated, or controlled by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

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20.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to the data and information required to establish financial qualifications for an applicant for a facility construction permit are Section 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50. To assure that we have the latest information to make a determination of the financial qualifications of an applicant, it is our current practice to review this information during the later stages of our review of an application. We are continuing our review of the financial qualifications of the applicant and will report the results of our evaluation in a supplement to this report.

21.0 CONCLUSIONS

Based on our analysis of the proposed design of the Phipps Bend plant, consisting of two units designated Units No. 1 and No. 2, and upon favorable resolution of the outstanding matters set forth in Section 1.9 of this report and discussed in appropriate sections of this report, we will be able to conclude that, in accordance with the provisions of Paragraph 50.35(a) of 10 CFR Part 50:

(1) The applicant has described the proposed design of the facility including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public;

(2) Such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration will be supplied in the Phipps Bend Final Safety Analysis Report;

(3) Safety features or components which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components;

(4) On the basis of the foregoing, there is reasonable assurance that (a) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility and (b) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public;

(5) The applicant is technically qualified to design and construct the proposed facility;

(6) The applicant has reasonably estimated the costs and is financially qualified to design and construct the proposed facility; and,

(7) The issuance of permits for the construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

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APPENDIX A

SAFETY EVALUATION REPORT
BY THE

OFFICE OF NUCLEAR REACTOR REGULATION
U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF
GENERAL ELECTRIC
STANDARD SAFETY ANALYSIS REPORT
(GESSAR-238 NI)

DOCKET NO. STN 50-447

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ABBREVIATIONS

a-c	Alternating current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AEC	United States Atomic Energy Commission
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
Btu/hr	British thermal units per hour
BWR	Boiling Water Reactor
cal/gm	Calories per gram
cfs	Cubic feet per second
CHF	Critical Heat Flux
Ci/yr	Curies per year
CP	Construction Permit
DBA	Design Basis Accident
d-c	Direct current
DG	Diesel Generator
ΔT	Differential Temperature
DOT	Department of Transportation
kW	kilowatt
kW/ft	kilowatt per foot
kW/l	kilowatt per liter
lb	Pound
lb/hr	Pounds per hour
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPZ	Low Population Zone
m	Meter
m^2	Square meters
MM	Modified Mercalli (earthquake intensity)
MCHFR	Minimum Critical Heat Flux Ratio
MCPR	Minimum Critical Power Ratio
mhos/cm	Reciprocal ohms per centimeter
mph	Miles per hour
mrem	Millirem
mrem/yr	Millirem per year

MSL	Mean Sea Level datum
MSLIV	Main Steam Line Isolation Valve
MWe	Megawatts electrical
MWt	Megawatts thermal
ECCS	Emergency Core Cooling System
ESF	Engineered Safety Feature
°F	Degrees Fahrenheit
FPS	Fire Protection System
FSAR	Final Safety Analysis Report
ft	Feet
GDC	AEC General Design Criteria
GE	General Electric Company
gpm	Gallons per minute
Gd_2O_3	Gadolinium oxide
HEPA	High Efficiency Particulate Air
hr	Hour
HPCS	High Pressure Core Spray
ID	Inside Diameter
IEEE	Institute of Electrical and Electronics Engineers
in	inch
kV	Kilovolt
k_{eff}	Effective multiplication factor
NDT	Nil Ductility Transition
NFPA	National Fire Protection Association
NOAA	National Oceanic and Atmospheric Administration
NPSH	Net Positive Suction Head
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OD	Outside Diameter
PCA	Primary Coolant Activity
PDA	Preliminary Design Approval
PMF	Probable Maximum Flood
PMH	Probable Maximum Hurricane
PMS	Probable Maximum Surge
PMP	Probable Maximum Precipitation
PSAR	Preliminary Safety Analysis Report
psi	Pounds per square inch
psid	Pounds per square inch differential
psig	Pounds per square inch gauge (above atmoshperic)
QA	Quality Assurance
QC	Quality Control
RBCCWS	Reactor Building Closed Cooling Water System
RCPB	Reactor Coolant Pressure Boundary
Rem	Roentgen equivalent man
RHR	Residual Heat Removal

ABBREVIATIONS (Cont'd)

rpm	Revolutions per minute
RPCS	Rod Pattern Control System
RWCS	Reactor Water Cleanup System
scfm	Standard Cubic Feet per Minute
SER	Safety Evaluation Report
sec/m^3	Seconds per cubic meter
SGTS	Standby Gas Treatment System
SSE	Safe Shutdown Earthquake
SWS	Service Water System
TLD	Thermal Luminescent Dosimeter
UO_2	Uranium Dioxide
$\mu Ci/sec$	Microcuries per second
X/Q	Relative Concentration

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Appendix A

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 Introduction

The initial AEC (now the Nuclear Regulatory Commission or NRC) policy statement on standardization of nuclear power plants was issued on April 28, 1972. It provided the impetus for both industry and the NRC to initiate active planning in their respective areas in order to realize the benefits of standardization while maintaining protection for the health and safety of the public and for the environment. In a subsequent statement issued on March 5, 1973, the NRC announced its intent to implement a standardization policy for nuclear power plants. The standardization policy presents three procedural options for standardization applications. Option 1 is a "reference system" concept that involves the review of an entire facility design or major fractions of a facility design outside of the context of a license application. The standard design would be referenced in subsequent license applications. Option 2 is a "duplicate plant" concept in which a limited number of duplicate plants are to be constructed within a limited time span. Option 3 is a "License to Manufacture" concept in which a number of identical plants would be manufactured at one location and moved to a different location for operation.

On April 30, 1973, General Electric Company (GE or applicant) filed the General Electric Standard Safety Analysis Report (GESSAR) for the nuclear island scope in response to Option 1 of the Commission's policy statement. On July 30, 1973, the GESSAR application was docketed. (Docket No. STN 50-447). The information in GESSAR has been supplemented by Amendments 1 through 39. GESSAR and amendments thereto are available for public inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C.

The review of GESSAR is being carried out by the staff pursuant to Appendix O to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," using a similar procedural sequence to that used for custom plant reviews. The initial phases, that is, preliminary review, question rounds, etc., are analogous to the normal construction permit stages of review; however, the conclusion of the review will not result in the granting of a construction permit. Instead, a Preliminary Design Approval (PDA) will be issued by the Regulatory staff following satisfactory completion of the staff and ACRS reviews. Guidance regarding the staff review, change procedures, supplementary Safety Evaluation Reports (SERs), post-PDA review, Final Design Approval (FDA), duration of the PDA/FDA, and PDA/FDA changes is provided in WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants" issued on August 20, 1974 and in the letter to the applicant granting the PDA. GESSAR contains safety information for a BWR-6/Mark III nuclear power plant, including the nuclear steam supply system (NSSS), the engineered safety feature systems, the containment and auxiliary buildings, the control room, radioactive waste system and related systems and structures. This complex is referred to as the nuclear island. See Figure 1.1 for the scope of the nuclear island.

The scope of our review for GESSAR is more extensive than that for other BWR-6 plants under review during the same period that GESSAR has been under review. In addition to the GESSAR-238 nuclear island discussed in this SER, we are currently reviewing a GESSAR-238 NSSS and a GESSAR-251 NSSS. The rated core thermal power of the plant is 3579 MWt (1220 MWe net). The ECCS design basis power is 3758 MWt (105% of rated power). Because it is not related to any particular site, GESSAR contains site envelope parameters to which its design is applicable. These site envelope parameters have been choosen to permit utilization of the standard design in much of the United States.

Not included in the GESSAR nuclear island scope are the turbine-generator and auxiliaries, the turbine building, portions of the main steam system (beyond the main steam shutoff valves located in the auxiliary building), the main condenser, the circulating water system and intake structure, condensate storage facilities. offsite electrical power, and the ultimate heat sink, raw and potable water systems, parts of the service and instrument air systems outside the nuclear island, the auxiliary steam system and, of course, the site.

Since GESSAR does not cover the entire facility, it is necessary to specifically and extensively describe the safety-related interfaces between the nuclear island and its mating portion of the plant and site. The interface information should address the pertinent safety-related design requirements including the dimensional, structural and operating environmental inputs to transient and accident analysis, and the testing and performance requirements necessary to assure the compatibility of the nuclear island to its mating portion of the plant.

For GESSAR, the interfaces were reviewed between the nuclear island and the site (meteorology, hydrology and seismology) in Sections 2 and 15 of this SER, and between the nuclear island and the mating balance of plant (BOP) described in Section 1.10 of GESSAR. A discussion of GESSAR interfaces is presented in Section 1.10.2 of this SER, except for the instrumentation and control area interfaces review which we have not completed (see Section 7.8).

General Electric also requested the staff to review the interfaces between the GESSAR-238 nuclear steam supply system (NSSS) and the BOP in their letter dated November 21, 1974. This review is currently being conducted on a separate docket (Docket No. STN 50-550) and a separate safety evaluation report will be prepared for this review.

Specific responsibilities of the General Electric Company are delineated in GESSAR. These responsibilities include such things as special studies (such as the soil-structure interaction studies mentioned in Section 3.7.3 of this SER), qualification testing of equipment and design of piping and equipment within the nuclear island. An example of shared responsibility is in Section 5.2.2 which discusses the surveillance program related to the operation of safety/relief valves where GE and utility applicants will be required to report on the performance of those valves.

An applicant referencing GESSAR is required to adopt all designs, tests, operating limitations, and inspections identified throughout GESSAR. This means that any commitments made by GE in GESSAR will be automatically included in the utility's PSAR that references GESSAR.

This Safety Evaluation Report (SER, summarizes the results of the technical evaluation of GESSAR as performed by the Commission's Regulatory staff and delineates the scope of the technical matters considered in evaluating the radiological safety aspects of the proposed facility necessary to provide the standard design a Preliminary Design Authorization (PDA) which will make it an acceptable "reference system" under Option 1 of our "Standardization Policy."

This SER is a composite of the original SER issued in November 1974 and three supplements dated December 1974, February 1975, and March 4, 1975 and replaces these previously issued reports.

In the course of the safety review of the material submitted, we held a number of meetings with representatives of the General Electric Company and their consultants to discuss the plant design and performance under postulated accident conditions. During our review, we requested the applicant to provide additional information that we needed for our evaluation. This additional information was provided in amendments to the application. As a result of our review, a number of changes were made in the facility design. These changes are described in the applicant's amendments. Section 1.7 provides a listing of some of the principal design changes which were made. Section 1.8 provides a summary of the development and verification test programs planned to support the design described in GESSAR and a summary of the principal post-PDA review areas that have been identified in this SER. A chronology of the principal actions relating to the processing of the application is attached as Appendix A to this Safety Evaluation Report (SER). A copy of the report of the Advisory Committee on Reactor Safeguards (ACRS) is attached as Appendix F to this SER, and we have addressed their comments in Section 18 of this SER.

In applying Regulatory Guides in our review of GESSAR, we have taken the following positions to permit GE to freeze their design. First, we have established March 1, 1974 as the cutoff date for GE to demonstrate compliance with published regulatory guidance such as Regulatory Guides published through 1.75. Second, for items of important safety significance discussed in Regulatory Guides issued after that date, implementation for GESSAR plants will be in accordance with the implementation guidance in the guide.

The review and evaluation of the proposed design of the facility reported herein is only the first stage of a continuing review by the Nuclear Regulatory Commission's staff of the design, construction, and operating features of the GESSAR facility. Utilities may reference GESSAR and match it with BOP designs, which include the turbine building and site, in their construction per t applications. Construction

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activities will be accomplished by utility applicants under the surveillance of the Commission's staff. We intend to review the final design of GESSAR when it becomes available to determine that the Commission's safety requirements have been satisfied prior to providing a Final Design Authorization (FDA). Then a utility would again be able to reference GESSAR in its operating license application. After an operating license is issued, the facility may then be operated only in accordance with the terms of the operating license and the Commission's regulations under the continued surveillance of the Commission's staff.

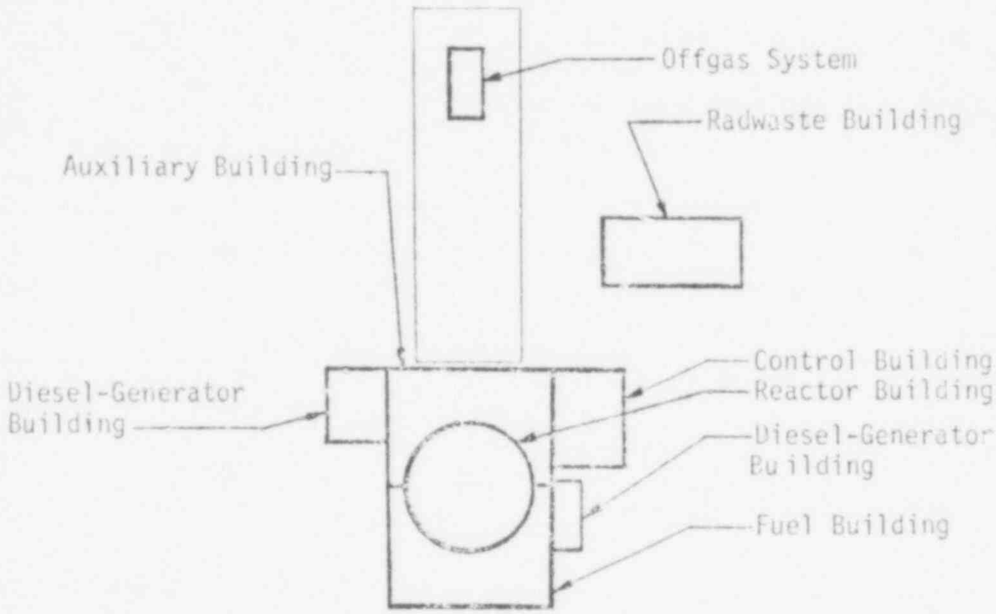
1.2 General Plant Description

The nuclear steam supply system (NSSS) for GESSAR utilizes the BWR-6 class of boiling water reactor. Each NSSS will have 20 jet pumps supplied by two recirculating water lines, four main steamlines, and two feedwater lines. Fuel rods for the reactor will contain slightly enriched uranium-dioxide (UO_2) in sintered ceramic pellets. Some of the fuel rods will have ceramic fuel pellets that contain gadolinium-oxide (Gd_2O_3) in a mixture with the uranium-dioxide. The gadolinium will serve as a "burnable poison" designed for power pattern and reactivity control. The fuel pellets will be enclosed in Zircaloy-2 cladding tubes which will be evacuated, backfilled with helium, and sealed by welding Zircaloy end plugs in each end. A fuel channel will enclose a bundle of 63 fuel rods in an 8 x 8 array (one fuel rod position will contain a water filled rod). Water flowing through the core will serve as both a neutron moderator and as a coolant. Movement of water and a two phase water-steam mixture through the core will be accomplished by the driving force from the 20 jet pumps (10 per recirculation line) and two recirculation pumps and from convective forces. Steam from the boiling process in the reactor core will be demoisturized and dried, then vented through the four main steamlines to the turbine-generator system (which is outside the GESSAR scope) where its energy will be converted into electricity. The condenser and condenser cooling system are not a part of the GESSAR scope.

An off-gas treatment system consisting of a recombiner, condenser, moisture separator, and deep-bed charcoal filters will provide for retention of noble gases for decay to acceptable concentration levels prior to release from the plant's vent.

The reactor coolant pressure boundary will include the reactor vessel, the two recirculation lines, main steamlines, feedwater lines, and branch lines to their outermost isolation valves.

Enclosing the reactor system will be a reinforced concrete cylindrical structure (called the "drywell"). Enclosing this "drywell" structure is the steel containment structure. The drywell's function is to force most of the steam, released in a postulated accidental pressure boundary break, through the suppression pool located at the bottom of the containment, thus, condensing the steam (vapor suppression) and limiting the pressure buildup below the containment's maximum design pressure of 15 psig. Piping restraints have been designed and will be installed within the containment to limit the movement of piping during its postulated post-rupture movement (pipe whip) so that safety-related components are appropriately protected. A hydrogen control system is included which will limit the concentration of hydrogen evolved



Nuclear Island

Figure 1-1

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during a postulated loss-of-coolant accident, which over a relatively short time period could build up to unacceptable levels in the drywell, by mixing it with the air in the larger containment volume. Long term hydrogen buildup will be controlled by oxygen-hydrogen recombiners located within the auxiliary building. Isolation of the primary containment will occur automatically whenever there exists a potential for the release of radioactivity due to high activity levels in containment. For instance, the primary containment and the nuclear steam supply system will be isolated and shut off, respectively, for the unusual conditions of low water level in the reactor vessel, high radiation level in the main steamline, main steamline high flow or low pressure, drywell high pressure, and other conditions described in GESSAR.

The reactor protection system (RPS) will provide the means to protect against conditions that may cause fuel failures or a breaching of the nuclear system process barrier, thereby limiting uncontrolled releases of radioactivity to unrestricted areas within 10 CFR Part 100 requirements. The RPS will initiate a reactor scram following an abnormal operational transient or pressure pulse, or following a gross failure of fuel or of the nuclear system process barrier. The RPS will be a reliable system designed to meet the standards specified in IEEE-279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

Normal reactivity control or rapid scram (shutdown) of the reactor will be achieved by the bottom-entry cruciform-shaped control rods (neutron absorbers) that will be moved vertically in the spaces between fuel assembly channels by a hydraulic drive mechanism; water is the hydraulic fluid. For rapid insertion, nitrogen under pressure in an accumulator provides the driving force. Each control rod will be independent of the other rods and have its own hydraulic control system. A standby liquid control system will also be available for use in injecting a boron solution into the reactor for emergency, long-term reactivity control.

Engineered safety features will provide the capability to contain fission products assumed to be released during a hypothesized design-basis accident to restrict radioactivity releases to acceptable levels, provide for heat removel for emergency core cooling, and condense steam within the containment. Details on these engineered safety features are presented elsewhere in this Safety Evaluation.

This containment will house the reactor and its pressure suppression type primary containment system. The auxiliary building will house the engineered safety features and their auxiliaries and Division 1 and 2 switchgear.

Operation of the standby gas treatment system (SGTS) will produce a negative internal pressure, after building isolation, such that the atmospheres within the building enclosing the containment and within the auxiliary building will be filtered and discharged to the environment via the SGTS.

Other safety-related structures in the nuclear island such as the fuel building, the control building, the radwaste building, and the diesel generator buildings are also described in this SER.

1.3 Shared Systems

GESSAR is a design for a single unit. Shared systems have not been discussed or reviewed for GESSAR. Structures, systems and components within the nuclear island that are important to safety will not be shared. Some BOP items that are related to safety systems may be shared. Included in this area are such things as offsite power and the ultimate heat sink. The detailed designs and interface requirements for the shared systems will be reviewed in the context of a utility's application for a construction permit.

1.4 Comparison with Similar Facilities

Many features of GESSAR are new GE designs, however, many aspects of the plant are similar to those we have evaluated and previously approved for other nuclear power plants. To the extent feasible and appropriate, we have made use of our previous evaluations during our review of those features that are similar to GESSAR. The results of our review of GESSAR, as listed in this SER will be somewhat different than previous SER's. The GESSAR SER will be different in that it is intended to stand on its own with a minimum use of references to previous designs and reviews. To the extent possible, we have presented, in this SER, our technical bases for each safety conclusion reached.

To assist in better understanding the relationship of the GESSAR (BWR-6/Mark III) design to other BWR designs, a comparative listing of principal parameters and features for GESSAR, Grand Gulf, Perry and LaSalle is presented in Table 1-1 of this SER. Our SER's for these other applications are available for public inspection in the PDR at 1717 H Street, N.W., Washington, D.C. 20545.

1.5 Identification of Agents and Contractors

As stated previously, General Electric Company is responsible for the "nuclear island" scope of supply. Future utility applicants referencing GESSAR will retain their own architect-engineers, constructors, turbine-generator vendor, and consultants as needed. For each future application, we will review the technical qualifications of the applicant, along with his contractors, to manage, design, construct and operate a specific reactor plant prior to the issuance of a CP.

1.6 Summary of Principal Review Matters

Our technical review and evaluation of the GESSAR information submitted by the applicant included and excluded the principal matters discussed below.

a. We evaluated the site design envelope parameters including the wind loadings, design bases tornado, design bases flood evaluation, the design bases earthquake, the snow loading, and maximum precipitation. Items in our site evaluation that are not within the GESSAR scope and that will be covered in future applications for plants referencing GESSAR are the exclusion area determination and control, the population distribution, use of adjacent lands and waters, the effects of the presence of nearby military facilities, effects of industrial or

TABLE 1-1

Principal Parameters and Design Features
of GESSAR, Perry, Grand Gulf and LaSalle

Parameter or Feature	GESSAR	Perry	Grand Gulf	LaSalle
Rated Power Level (MWt)	3579	3579	3833	3323
Design Power (MWt) (ECCS design bases)	3758	3758	4025	3489
Net Electrical Output (MWe)	1220	1220	1290	1086
No. Fuel Assemblies	732	732	784	764
Fuel Rod Array	8x8*(63 rods)	8x8*(63 rods)	8x8*(63 rods)	7x7(49 rods)
No. of Control Rods	177	177	193	185
*Fuel assembly contains 1 water rod.				
Maximum Design Linear Power (kw/ft)	13.4	13.4	13.4	18.5
Reactor Vessel, ID (in.)	238	238	251	251
Vessel Height, inside (ft. in.)	70-10	70-10	73-0	72-11
Vessel Wall Thickness (in.)	5.7	5.7	6.14	6.14
Clad Thickness (in.)	1/8	1/8	1/8	1/8
No. Recirculation Loops	2	2	2	2
Recirc. Pump Flow Rate (gpm)	35,400	35,400	44,9[illegible]	47,250
No. Jet Pumps	20	20	24	24
No. Steam Lines	4	4	4	4
Steam Line ID (in.)	26	26	28	26
Core Flow (lb/hr) x 10^6	105	105	113.5	108.5
Steam Flow (lb/hr) x 10^6	15.396	15.396	16.488	14.95
ECCS				
No. LPCS pumps	1	1	1	1
Flow (gpm) at 122 psid	6000	6000	7000	6350
No. HPCS Pumps	1	1	1	1
Flow (gpm) at 1130 psid	1465	1465	1550	1550
No. RHR Pumps (LPCI mode)	3	3	3	3
Flow Rate at 30 psid per pump	7100	7100	7450	7450
No. ADS Systems	1	1	1	1
Containment Design				
Type	Mark III	Mark III	Mark III	Mark II
Drywell	Concrete Cylinder	Concrete Cylinder	Concrete Cylinder	Concrete Cone Section
Design Pressure (psig)	30	30	[illegible]	45
Containment	freestanding steel cylinder	freestanding steel cylinder	reinforced concrete	N/A
Design Pressure (psig)	15	15	15	N/A

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transportation accidents close to the plant, the consequences of an aircraft crash, the evaluation of the meteorological measurements program, the effects of toxic chemical or gaseous releases on plant operation, and the effects of explosions near the facility.

Our site evaluation was performed to determine whether the proposed design could meet the Commission's siting criteria (10 CFR Part 100) at typical sites that have received approval in the past.

b. We evaluated the design and expected performance of the nuclear island's structures, systems, and components important to safety to determine whether they are in accord with the Commission's General Design Criteria (GDC), the Commission's Quality Assurance Criteria, and other applicable guides, codes and standards, and whether any departures from criteria, codes and standards have been identified and justified.

c. We evaluated the expected response of the facility to various anticipated operating transients and to a broad spectrum of postulated accidents and determined that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered. We performed conservative analyses of these design basis accidents and determined that the calculated potential offsite doses that might result in the very unlikely event of their occurrence would be well within the Commission's guidelines for site acceptability, as given in 10 CFR Part 100, for typical sites.

d. We did not evaluate the plans for the conduct of plant operation, the organizational structure, the technical qualifications of the operating and technical support personnel, or the measures taken in the unlikely event of an accident that might affect the general public. These items are outside GE's scope of supply in GESSAR and will be addressed in future specific utility applications referencing GESSAR. This is consistent with the requirements of Appendix O to 10 CFR Part 50.

e. We evaluated the design of the systems provided for control of the radiological effluents from the plant and determined that these systems, in conjunction with an acceptable BOP design, reasonable meteorology and site boundaries, will be able to control the release of radioactive wastes within the limits of the Commission's regulations in 10 CFR Part 20 and that the plant will be operated in such a manner as to reduce radioactive releases to levels that are as low as practicable in accordance with the Commission's regulations in 10 CFR Part 50.

f. We did not evaluate the financial qualifications of any utility applicant to determine whether its financial position is adequate to design and construct the facility, since this is a matter for review in individual CP applications. This is also consistent with the requirements of Appendix O to 10 CFR Part 50.

The GESSAR application contained new and significantly modified features that are different than previous BWR-6/Mark III containment designs that have been evaluated by the staff. These items are noted below.

a. GE has proposed new designs in five instrument and control areas. These changes are a new control rod position detection system; a new method of increasing the negative reactivity during a scram to cope with changes to scram-reactivity during core life; the use of ganged control rods; a revised rod pattern control system; and finally, a solid state, 2-out-of-4 protection system. This is discussed in Section 7 of this SER.

b. Based on further boiling transition tests, GE has developed a new figure of merit for expressing the BWR thermal margin. The result is a new GE Thermal Analysis Basis (GETAB), which is discussed in Section 4.4 of this SER.

c. We worked with GE in obtaining resolution to some previously outstanding problems including post-LOCA H_2 generation and control, a main steam line sealing system suppression pool bypass and testing, drywell structural and leakage testing, quality classification of main steam radwaste and auxiliary systems and interface definition and quantification.

1.7 Facility Modifications as a Result of Regulatory Staff Review

1.7.1 Facility Modifications Made by GE

As a consequence of our review, a number of changes were made to the GESSAR design. These modifications are discussed in greater detail in appropriate sections of this report. Examples of changes which were made are as follows. References to sections of this SER where further discussion is provided are in parentheses.

(1) An increase in the wind loading, snow and ice loadings and elevation of ground water with respect to the foundation mat will permit the plant to be used on more sites (3.3.1, 3.3.3, 2.4.3).

(2) The seismic instrumentation program has been augmented (3.7.4).

(3) A main steam line leakage control system has been added (9.3.1).

(4) The RCIC system has been upgraded to an engineered safety feature (5.4.4).

(5) The operability of active components will be verified by testing (3.9.2.4).

(6) Design measures have been taken to protect against the dynamic effects associated with pipe breaks (3.6)

(7) A finite element method will be used to analyze various soil conditions to evaluate soil structure interactions (3.7.3).

(8) Fuel building has been upgraded to withstand tornado missiles (9.1.2).

(9) Methods for Seismic Analysis comply with requirements of Regulatory Guides 1.60 and 1.61 (3.7.1).

(10) Main Steam Line and Feedwater Piping Reclassification (3.7.2).

(11) Mark III containment changes to accomodate pool swell testing results (6.2.1.9).

(12) Increased drywell design pressure margin (6.2)

(13) Tests to verify that controls on stainless steel are adequate to prevent sensitization (5.2).

(14) GE agreed to preoperational vibration tests on Class 1 and 2 piping systems (3.9.1.1).

(15) Reduced containment leakage design criteria.

1.7.2 Facility Modifications Required by the Staff

During our review, GE did not agree with several staff requirements. As a result, these requirements will become conditions of the PDA. Further discussion is provided in the section of the SER listed in the parentheses:

(1) GE must use acceptable tornado missile velocities (3.5.).

(2) We will require GE to adopt the staff's criteria for establishing dynamic loads on structures located in and above the suppression pool (6.2.1.9).

(3) If GE wishes to purge the containment continuously, they must meet the requirements concerning filtration and purge line size (6.2.4).

(4) GE must provide measures to preclude operation of any individual MSIV-LCS if the associated inboard MSIV in the steam line connected to the LCS is not fully closed (9.3.1). In addition, the setpoint for the flow element timer in the inboard LCS must be set at 11.5 cfh (15.3.1).

1.8 Requirements for Future Technical Information

1.8.1 Development Of BWR Technology

The applicant has identified in GESSAR Section 1.5 and we have listed in Table 1-2 of this SER, the research and development programs applicable to the GESSAR plant. These programs are aimed at verifying the nuclear steam supply system and containment designs and confirming the design margins. The objectives, schedules for completion, and current results are summarized throughout this SER.

General Electric Company has recently informed us that as a result of the normal design process, certain modifications are expected to be made in the BWR-6 8x8 fuel design. These changes are being made to improve the performance of the fuel during normal operations. An added effect is that the average power density would be reduced. These changes will be reviewed on a post-PDA basis. The present schedule calls for GE to submit a topical report discussing the design changes in Ocotober 1975. This is to be followed by a lead test assembly program scheduled to start in January 1976. The staff expects to complete their review and publish a SER on the new fuel in November 1976. In the event the proposed changes are not acceptable, the present fuel design described in Section 4 of GESSAR and our SER would be acceptable. We consider this acceptable for the issuance of a PDA for GE.

We conclude that the applicant has identified and will perform the research and development necessary for the design and safe operation of the plant on a timely

TABLE 1-2

DEVELOPMENT AND VERIFICATION TEST PROGRAMS AND MAJOR POST-PDA EVALUATION EFFORT

PROGRAM	PURPOSE	STATUS	DESCRIBED IN
I. Fuel Surveillance Program	To verify performance of select 8 x 8 fuel assemblies	Ongoing	SER Section 4.2.1 Response to Question 4.40
II. Critical Heat Flux Testing	Develop more accurate heat flux correlation	Complete	SER Section 4.4 SER Appendix E - GETAB NEDO-10958
III. Instrumentation for Vibration and Loose Parts Detection	Develop a system which utilizes sensors exterior to RPV to provide continual monitoring for impact and vibration of loose parts during operation	Under Development	SER Section 5.2.1.4 GESSAR Section 1.5
IV. Safety Relief Valve Surveillance Program	Verify reliability of new valve design	Ongoing	SER Section 5.2.2
V. Verification of Pressure Suppression Design	To develop data necessary to substantiate analytical model describing performance of Mark III containment design	Ongoing	SER Section 6.2.1.6 GESSAR Section 1.5
VI. Suppression Pool Dynamics	To investigate pool swell phenomenon and its impact on Mark III design	Ongoing	SER Section 6.2.1.9 Table 6.2-2
VII. Core Spray Distribution	To verify that core spray headers will provide adequate cooling water to all assemblies	Results due late '75	SER Section 6.3.1 GESSAR Section 1.5
VIII. Study of Effects of Relief Valve Blowdown During Various Operating States	To demonstrate that inadvertent ADS actuation has acceptable safety consequences	Ongoing	SER Section 7.3.2.2 GESSAR Question 7.26

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TABLE 1-3

POST PDA ITEMS

ITEM	DISCUSSED IN SER IN SECTION	SCHEDULE STATUS
1) Leakage characteristics of primary coolant pump seals.	3.2.1	Staff questions to be issued in early 1976. GE response prior to FDA
2) Description of combined effects of SSE and Steam line break.	3.9.1.4	Topical response due Mid-1976
3) a. List of specific equipment to be seismically qualified. b. The qualification procedures to be used.	3.10	Under staff review; estimated for completion mid-1976.
4) a. List of specific equipment to be environmentally qualified. b. The qualification procedures to be used.	3.11	Under staff review; estimated for completion mid-1976.
5) Preliminary design of drywell penetrations.	3.11.1	Under staff review; estimated for completion by mid-1976.
6) Procedures and methods to be used to qualify the shield building, containment and drywell penetrations.	3.11.1	Complete staff review mid-1976.
7) Implementation methods of Separation Criteria for Safety Related Electrical Equipment.	3.12	Staff review complete mid-1976.
8) Detailed Information on: a. Lattice Physics Methods b. BWR Simulation Code c. Verification of Core Calculation Methods	4.3.7	The following Topical Reports have been or will be submitted (a) Draft 8/6/75 (b) Draft 12/10/75 (c) Drafts 7/22/75 and 7/1/75 Final 1/30/76
9) Confirming data from large scale Mark III test for short term containment response.	6.2.1.6	Liquid blowdown, small break, elevated temperature, and multivent tests and verification of Pressure Suppression design - submitted by Nov. 1977.
10) Assumptions used to size containment vacuum breakers.	6.2.1.5	Response by Febuary 1976.
11) Environmental design criteria for isolation valves and other safety-related equipment in drywell and containment.	6.2.4	Response by mid-1976.

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TABLE 1-3 (Contd.)

POST PDA ITEMS

ITEM	DISCUSSED IN SER IN SECTION	SCHEDULE STATUS
12) Address Question 6.125 (manual operation action on ECCS following a LOCA).	6.3.1	Response by October 1976.
13) Proprietary version of 8x8 Z_r Spray Cooling Test.	6.3.1	Topical Report due Dec. 1975.
14) New Modified Instrumentation anc Control Systems Preliminary Design Review for: a. Reactor Trip System b. Engineered Safety Features Actuation Systems c. Safe Shutdown System d. Safety Related Display Instrumentation e. All other instrumentation required for safety. f. Control System-Reactor Manual Control System, Recirculation Flow Control, Gaseous and Liquid Radwaste Controls, Feedwater Flow Control, and interaction between safety and non-safety control systems.	7.0	Currently under staff review; estimated for completion by Spring 1976.
15) Scope of Onsite Electrical System for GESSAR.	8.0	Response by early 1976.
16) Review of HPCS-(onsite electrical system).	8.0	Questions from staff review of Topical NEDO-10805 to [illegible] issued late 1975. GE response prior to FDA review.
17) Fast Scram System and Chapter 15 Transients.	4.2.3 and 15.2	GE information on criteria, preliminary design, confirmatory test program, and transient evaluation by late 1975.
18) Preliminary Design of System to Control Bypass Containment Leakage.	Section 6.2.3	Due April 1976.
19) Anticipated Transient Without Scram (ATWS)	15.4	Resolution prior to FDA review.

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schedule, and that, in the event that results of any of this work are not successful, appropriate restrictions on operation can be imposed or proven alternate designs can be utilized to protect health and safety of the public.

1.8.2 Post-PDA Review

As discussed in WASH-1341, part of the evaluation of new or unusual design features may result in our review continuing into the post-PDA phase. Table 1-3 lists areas scheduled for continued staff review after the PDA issuance, but prior to GE's submittal of the GESSAR design for an FDA review, together with a schedule estimate for the submittal of post-PDA information or completion of staff review. The major item of this type in GESSAR is the instrumentation and control (I&C) areas. We are currently reviewing this area, and as stated in Chapter 7 of this SER, we will report the results of our review in a supplement to this SER and present our evaluation to the ACRS for thier consideration. We have concluded in this SER that the proposed I&C system designs are within the state-of-the-art and the conceptual designs, design bases and criteria given in GESSAR for these areas are acceptable to issue a PDA.

1.9 Technical Qualifications

The General Electric Company is the applicant for a PDA for GESSAR. General Electric is responsible for the design of the nuclear island. They have subcontracted to the C. F. Braun and Company to provide engineering services related to the design of the nuclear island structures. Construction and operation responsibilities belong to the utility referencing GESSAR.

General Electric Company has been engaged in the design, development, construction and operation of boiling water, test and research reactors for 20 years. They have also gained experience by conducting nuclear research and development programs for the utility industry and government. At present, GE has about 20 reactors licensed to operate throughout the world and has many reactor years of successful operating experience behind it.

C. F. Braun & Co., have been performing engineering and construction services throughout the world since 1909. They have provided these services to the chemical, mining and utility industries and in recent years have designed major facilties for the AEC.

We have reviewed the qualifications of the General Electric Company and its subcontractor (C. F. Braun & Co.) and have concluded that they are technically qualified to design the GESSAR facilities, and to provide the equipment within their scope of supply.

1.10 Conformance of GESSAR to 10 CFR 50 Appendix O

1.10.1 General

On January 17, 1975, about 18 months after docketing GESSAR for review, the NRC published in the Federal Register (40FR2974) amendments to 10 CFR 50 related to the standardization of nuclear reactors. Appendix O to 10 CFR 50 is related to the "Reference System" concept which applies to GESSAR. This appendix sets out the procedures for filing, staff review and refereal to the ACRS of reference designs we have reviewed GESSAR against the information requirements specified in Appendix O. In general, the information required for a reference system is the same as that needed for a custom plant review. Most of the requirements of 10 CFR 50.24 need to be met for the reference design. Other information related to the special circumstances associated with a standard design is also needed. This includes interface information (discussed in Section 1.10.2 of this SER), site envelope parameters (see Sections 2 and 15 of this SER), quality assurance of the applicant (Section 17 of the SER) and qualifications of the applicant (Section 1.9).

Appendix O also requires a discussion of the features that affect plans for coping with emergencies in the facility. Features such as the post-accident monitoring instrumentation including inplant activity monitors (Sections 7.6, 11.5.1 and 12.2) have been included in the standard design application.

All of the above items have been reviewed and our evaluations are presented elsewhere in this SER.

1.10.2 Interfaces

Interface information was provided in GESSAR between the nuclear island and the site which included meteorology (Section 2.3 of GESSAR), hydrology (Section 2.4 of GESSAR) and seismology (Section 2.5 of GESSAR); and between the nuclear island and the mating balance of plant (BOP described in Section 1.10 of GESSAR. GE has identified all systems and components which interface with the BOP, and for each such system and component has established the fundamental requirements that must be satisfied by the mating portion of the BOP. Section 2 of this SER contains the site related interfaces. In Table 1.10.1 of GESSAR, GE identifies the functional interface and, to the extent possible, the quantitative parameters (such as temperature, pressure and flow) which must be met in order to meet the nuclear island safety-related requirements provided at the interface between the nuclear island and about 14 BOP systems.

In some cases, not all of the numerical values are available at this time since the final design of nuclear island equipment and systems is not yet available. This is particularly the case for instrumentation interfaces with the BOP discussed in Section 7.8 of this SER. The detailed quantitative information is not available at this time and is not needed for the PDA, since identification of the interfaces as well as the criteria and bases for the interfaces are all that is needed for the PDA.

As we continue our review of interfaces during the evaluation of the mating BOP designs, new and outstanding interface parameters will be addressed. It is the responsibility of the utility-applicant to satisfactorily resolve these interfaces for their application. At the FDA stage, we will review the interface details and complete the interface evaluation of GESSAR.

We conclude that the nuclear island interface information provided in GESSAR is acceptable for the PDA.

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2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

These will be reviewed for each application referencing GESSAR.

2.2 Nearby Industrial, Transportation and Military Facilities

These will be reviewed for each application referencing GESSAR.

2.3 Meteorology

Details on the atmospheric diffusion characteristics of a proposed nuclear power plant site are required in order that a determination may be made that postulated accidental as well as routine operational releases of radioactive materials are well within NRC guidelines. The meteorological characteristics of a proposed site are determined by staff evaluation of meteorological data collected at the site in accordance with Regulatory Guide 1.23, "Onsite Meteorological Programs." This procedure will be followed in review of any application for a site which will contain the GESSAR standard plant. The discussion which follows relates to the "envelope" of site meteorological conditions proposed by the applicant for the standard plant which provides an indication, in advance of the examination of a particular site, of the type of site for which the standard plant as proposed is suitable.

2.3.1 Regional Climatology

The General Electric Standard Plant is designed for an environmental temperature range of from -40°F to +115°F (-40°C to +46°C), which is adequate for over 90 percent of the contiguous United States. The design basis wind velocity of 130 mph at a height of 30 feet above grade with a recurrence interval of 100 years would be adequate for greater than 90% of the sites although some east and gulf coast sites may be excluded since the winds associated with the Probable Maximum Hurricane could exceed 130 mph. The design basis maximum tornadic wind velocity of 360 mph and maximum pressure drop of 3.0 psi is adequate for all areas of the United States. The design basis snow and ice load of 50 pounds/square foot is adequate for many areas within the contiguous United States, although such loads may be exceeded in several regions, especially in some northern portions of the country. Further discussion of the snow load limitation is provided in Section 3.3.3 of the SER. Sites for which the extreme temperatures are outside the range presented above, for which the design basis wind of 130 mph may be exceeded, or for which the design basis snow and ice load may be inadequate will be evaluated on a case-by-case basis.

2.3.2 Local Meteorology

The design basis atmospheric dispersion conditions specified for GESSAR are equivalent to those presented in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors." Larger exclusion and low population zone distances can compensate for sites with less favorable meteorology. Case-by-case analysis will be done for proposed nuclear plant sites within the United States. A meteorological data collection program, conforming to the provisions of Regulatory Guide 1.23, "Onsite Meteorological Programs," can verify that atmospheric dispersion characteristics of the site are within the design envelope of the plant.

2.3.3 Onsite Meteorological Measurements Program

The applicant has stated that the dispersion conditions at the site will be verified. Since the onsite meteorological measurements program is assumed to be site specific, GE does not include a description of an onsite meteorological measurements program. This is the responsibility of the utility applicant and will be reviewed on a case-by-case basis.

2.3.4 Short-Term (Accident) Diffusion Estimates

The procedures used by the applicant (as presented in Regulatory Guide 1.3) to determine accident diffusion estimates are generally acceptable although the assumption of an elevated release from building vents if wind speeds are less than 7 mph is not acceptable to the staff. Of those sites previously evaluated by the staff, only 30% had higher X/Q values (more restrictive dispersion conditions) than the value of 1.0 x 10^{-3} sec/m^3 used in the calculations for the 0-2 hour period in Section 15.0.

2.3.5 Long-Term (Routine) Diffusion Estimates

The applicant does not provide any meteorological assumptions or procedures to be used in evaluating the long-term atmospheric dispersion characteristics of a site. Diffusion estimates for routine releases from plant buildings and vents are to be based on meteorological data collected onsite. An evaluation of annual average atmospheric dispersion values for routine releases will have to be made on a case-by-case basis at each site using meteorological data collected onsite in an acceptable manner (see Regulatory Guide 1.23), over at least an annual cycle.

2.3.6 Conclusions

Justification at sites where any of the meteorological parameters fall outside the envelope listed in Section 2.3 of GESSAR will have to be presented by the specific utility applicants. In addition, the final determination of the acceptability of any

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proposed site, with respect to meteorological conditions as they affect Part 100 guideline dose calculations, will be based upon evaluation of meteorological conditions applicable to that site in combination with the applicable exclusion area and low population zone distances.

2.4 Hydrology

2.4.1 Floods

The applicant proposed that grade levels for all safety-related facilities be located at least one foot above the design bases flood level, including an allowance for coincident wind generated waves, and has referenced Regulatory Guide 1.59, "Design Bases Floods for Nuclear Power Plants," for criteria. We conclude that these bases are acceptable criteria.

Protection of safety-related structures against locally heavy precipitation has been identified as roof drains with 4 inches per hour capacity and overflow capability to limit standing water to 9.5 inches (about 49 pounds per square foot). Specific site drainage will be reviewed with individual applications to assure that ponding above plant grade adjacent to safety-related buildings has been precluded.

2.4.2 Safety-Related Water Supply - Ultimate Heat Sink

This is a subject that is the responsibility of and will be discussed by the specific utility applicant. Interface requirements that need to be met by the ultimate heat sink are discussed in Sections 1.10 and 9.2.1 and Table 9.2.8 of GESSAR.

2.4.3 Groundwater

The GESSAR design envelope includes provisions for groundwater levels up to within 2 feet of plant grade in the design of the facility.

2.4.4 Conclusions

The proposed flood and groundwater criteria are acceptable to the staff.

2.5 Geology and Seismology

The geology, seismology and foundation engineering investigations required by Appendix A to 10 CFR Part 100 and the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Revision 2 will be reviewed by the staff for each individual site for which an application is made. The staff believes that the proposed seismic design of 0.3g will be adequate for 70% of the nation east of the Rocky Mountains. The limiting condition associated with 0.3g is the potential for foundation liquefaction. This can be overcome by compaction or other means. This condition as well as the potential for subsidence will have to be evaluated on a case-by-case basis.

3.0 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Conformance with NRC General Design Criteria

General Electric Company presented in Section 3 of GESSAR, their evaluation of the design bases for GESSAR, with respect to the NRC's General Design Criteria (GDC) as contained in Appendix A to 10 CFR Part 50. Based on our evaluation of the preliminary design and of the proposed design criteria, we conclude that subject to the applicant's adoption of the additional requirements made by us, as discussed in this report, they are in conformance with the GDC.

3.2 Classification of Structures, Systems and Components

3.2.1 Seismic Classification

Except for those items identified below, structures, systems and components important to safety that are required to withstand the effects of a Safe Shutdown Earthquake and remain functional have been properly classified as seismic Category I items. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

All other structures, systems and components that may be required for operation of the facility are designed to other than seismic Category I requirements. Included in this classification are those portions of Category I systems which are not required to perform a safety function. Structures, systems and components important to safety that are designed to withstand the effects of a Safe Shutdown Earthquake and remain functional have been identified in an acceptable manner in Table 3.2-1 of GESSAR.

GE classified the discharge piping from the relief valve to the anchor point as seismic Category I and the remainder of the piping from the anchor point to the suppression pool as non-seismic Category I. The staff did not originally agree with this non-seismic Category I classification since GE had not shown that failure of the discharge piping would not damage or degrade other safety-related equipment.

In Amendment 20 (Section 5.2.2.2.3 of GESSAR), GE noted that the loads for which this portion of piping were to be designed resulted in frequent supports anchored to the drywell wall. Failures of this piping between supports would not adversely affect the functioning of any safety-related equipment. We have reviewed this information and agree with GE that this section of piping may be excluded from meeting the seismic Category I requirements per Part C.2 of Regulatory Guide 1.29.

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GE proposed a Quality Group D and non-seismic classification for those lines of the component cooling water system that are used to provide cooling to the recirculation pump motors. The staff was concerned that the loss of cooling water from these non-seismic lines could result in a motor seizure that could result in exceeding reactor thermal limits. Analyses of reactor recirculation pump motor behavior following complete loss of cooling water were provided to the staff. The GE analyses indicate that if the initial cooling water loss alarm and the subsequent bearing temperature alarm (about 6 minutes later) are both ignored, the bearings will continue to operate another 6 to 10 minutes before melting. The response further states that such melting will not cause motor seizure and assuming the worst possible steel to steel friction, the motor will trip on overload caused by the added friction. Under these conditions, there would be only a slightly more rapid flow coastdown than that following a normal pump trip and will not result in unacceptable fuel damage. Review of these results indicate that the consequences of the cooling water failure are acceptable. We therefore conclude that the proposed Quality Group D and non-seismic classification of those lines of the component cooling water system that are used to provide cooling to the recirculation pump motors is acceptable.

The staff plans to further review with the applicant, during the post-PDA review, the leakage characteristics of primary coolant through the pump seals as a result of assumed seal water or cooling water failure to assure that the resulting leakage can be made up through plant cooling systems. As this study progresses, further information may be required and certain system changes could be indicated. We conclude that the potential nature of the system changes involved permit this study and evaluation to be completed as a post-PDA review item.

GE classified the offgas system for GESSAR as non-seismic Category I. The staff did not agree with this non-seismic classification, and this was identified as an area of concern by the Advisory Committee on Reactor Safeguards (ACRS) requiring resolution acceptable to the NRC staff prior to issuance of a PDA. It is the staff's position that the charcoal decay tank supports should be designed to an intermediate level of seismic design and be capable of meeting OBE requirements as shown by a simplified seismic analysis. With this done, the resultant relatively low probability of failure of these tanks allows us to judge the radiological consequences of their failure against 10 CFR Part 100 exposure guidelines instead of the 10 CFR Part 20 limits or the limits used in Regulatory Guide 1.29.

The applicant has now proposed to design the offgas system delay tank supports to the seismic design criteria listed in Branch Technical Position ETSB 11-1 which is included as Appendix B to this SER. The conservatively calculated doses resulting from the postulated failure of the delay tank supports are small fractions (typical values range between 0.5 rem and 10 rem depending on the site) of the 10 CFR Part 100 guideline exposures. On this basis, we conclude that this deviation from Regulatory Guide 1.29 is acceptable.

The basis for acceptance in the staff's review has been conformance of the applicant's designs, design criteria and design bases for structures, systems and components important to safety with the Commission's regulations as set forth in General Design Criterion 2, and to Regulatory Guide 1.29, "Seismic Design Classification," technical staff positions (such as Appendix B to this SER), and industry standards such as the ASME code.

The staff concludes that structures, systems and components important to safety that are designed in accordance with seismic Category I requirements provide reasonable assurance that the plant will perform in a manner providing adequate safeguards of the health and safety of the public.

3.2.2 System Quality Group Classification

Except for those items identified below, fluid system pressure-retaining components important to safety will be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed. The applicant has applied the classification system identified in Regulatory Guide 1.26, "Quality Group Classifications and Standards" to those fluid containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety where reliance is placed on these systems: (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintenance in the safe shutdown condition, and (3) to contain radioactive material. These fluid systems have been classified in an acceptable manner in Tables 3.2-1 and 3.2-3, Figures 3.2-1 and 3.2-2 of GESSAR and on system Piping and Instrumentation Diagrams.

The applicant has classified the discharge piping from the relief valve to the first seismic anchor point as Quality Group C and the remainder of the piping from the anchor point to the suppression pool as Quality Group D. The staff did not agree that Quality Group D was an acceptable classification for the major portion of this piping since failure of the discharge piping may damage or degrade other safety-related equipment. We required the piping from the downstream side of the relief valves to the suppression pool be classified Quality Group C unless GE could justify that there are no unacceptable safety consequences resulting from failure of the Quality Group D portion of the piping. As discussed in Section 3.2.1 of this report, GE has justified their classifications to our satisfaction in Amendment 20 and their design is acceptable.

The applicant has classified the Liquid Radwaste and Offgas systems as Quality Group D augmented in accordance with Branch Technical Position ETSB 11-1, "Design Guidance for Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants," which is attached as Appendix B to this SER.

GE has stated that components of the main steam and feedwater systems important to the safety of the plant will be designed, fabricated, erected and tested in accordance with the quality and seismic design requirements described in the letter of April 19, 1974, from J. M. Hendrie of the NRC to J. A. Hinds of General Electric. A copy of this letter is enclosed as Appendix C. These requirements are acceptable as an alternate to the guidelines currently specified in Regulatory Guide 1.26 (March 27, 1972) and 1.29 (August 1973).

The basis for acceptance in the staff's review has been conformance of the applicant's design, design criteria. and design bases for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping and valves in fluid systems important to safety with the Commission's Regulations as set forth in General Design Criterion I, the requirements of the Codes specified in Section 50.55a of 10 CFR Part 50, and to Regulatory Guides 1.26, technical staff positions, and industry standards.

GESSAR's pressure-retaining components of the reactor coolant pressure boundary are being designed to the Codes in effect now. CP applicants who reference GESSAR will need to meet the provisions of 10 CFR 50.55a which are in effect at the time of docketing. We will be periodically updating GESSAR to include code revisions and new Regulatory Guides as stated in Section 1.2 of this SER.

The staff concludes that fluid system pressure-retaining components important to safety that are designed, fabricated, erected and tested to quality standards in conformance with these requirements provide reasonable assurance that the plant will perform in a manner providing adequate safeguards to the health and safety of the public.

3.3 Environmental Loadings

3.3.1 Wind Loadings

All the Nuclear Island Category I structures listed in Table 3.2-1 of GESSAR will be designed to withstand the effects of the design wind, and all Category I systems and components located within will thereby be protected from its effects. Category I systems and components located outside the structures and thus exposed to the wind, will be designed to withstand its effects.

The design wind specified for GESSAR has a velocity of 130 mph at an elevation of 30 feet above grade based on a recurrence interval of 100 years. In applications where the plant is located in an area where a higner wind velocity is expected, the plant Category I structures, systems and components will be reevaluated.

The procedures that will be used to transform the wind velocity into pressure loadings on structures, systems or components, and the associated distribution of wind pressures and drag coefficients will be in accordance with the American Society of Civil Engineers Paper No. 3269, "Wind Forces on Structures." This paper has been widely used and recognized and has been accepted for use by the Regulatory staff.

The design wind loads will be combined with other applicable loads as will be discussed in Section 3.8 of this report.

3.3.2 Tornado Loadings

All the Nuclear Island Category I structures listed in Table 3.2.1 of GESSAR will be designed to withstand the effects of the Design Basis Tornado, and all Category I systems and components located within will thereby be protected from its effects. Category I systems and components located outside the structures and thus exposed to the tornado, will be designed to withstand its effects.

The Design Basis Tornado specified for the plant is selected to meet the worst tornado conditions listed in Regulatory Guide 1.76 in that it has a tangential wind velocity of 290 mph and a translational velocity of 70 mph. The pressure drop associated with the tornado is 3 psi in 2 seconds. Furthermore, an appropriate spectrum of tornado-generated missiles is also postulated as will be discussed in Section 3.5 of this report.

The procedures that will be used to transform the tornado wind velocity into pressure loadings will be in accordance with ASCE Paper No. 3269 except that the pressure will be applied uniformly over the full height of the projected area of the structure and no gust factors will be applied. The changes result in a more conservative analysis than the ASCE paper. Category I structures will be designed for the pressure drop associated with the Design Basis Tornado which will be treated as a load that varies with time. The tornado missile effects will be determined using procedures discussed in Section 3.5 of this report. The total effect of the Design Basis Tornado on Category I structures, systems and components will be determined using an appropriate combination of the effects of wind load, pressure load and missile load.

Tornado-generated loads will be combined with other applicable loads as will be discussed in Section 3.8 of this report.

It is an interface requirement that all the Nuclear Island Category I structures will be interfaced with non-Category I structures in a manner such that the failure of a non-Category I structure due to tornado effects will not compromise the integrity of a Category I structure. The safety functions and structural integrity of Category I equipment and structures will thereby be assured.

3.3.3 Snow Loading

As discussed in Section 2.3.1, the design basis snow and ice load of 50 pounds/square foot is adequate for many areas within the contiguous United States, although such loads may be exceeded in several regions, especially in some northern portions of the country. In these regions, some of the other loads (wind and seismic) are typically lower. Therefore, the total loads on the structure, even with the higher snow and ice loads, will be less than the total loads for other regions. In other words, even where the snow and ice loads are greater than 50 pounds/square foot, that may not be

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the limiting design loading for the structures. Sites for which the design basis snow and ice load may be inadequate will be evaluated on a case-by-case basis. Snow and ice loadings will be combined with other applicable loads as discussed in Section 3.8.1 of this SER.

3.3.4 Conclusions

We conclude that the procedures that will be utilized to determine the loadings on seismic Category I structures induced by the design wind, the Design Basis Tornado and snow loading specified for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that, in the event of a design wind, a Design Basis Tornado or snow, the structural integrity of the plant seismic Category I structures will not be impaired. Seismic Category I systems and components located within these structures are thereby adequately protected and will be expected to perform their intended safety functions if needed. Conformance with these procedures is an acceptable basis for satisfying in part the requirements of General Design Criterion 2.

3.4 Water Level (Flood) Design

3.4.1 Flood Protection

The design basis flood elevation for GESSAR is approximately one foot below the plant finished grade elevation including allowance for coincident waves and resultant runup. We will verify this flood protection on a case-by-case basis.

With this as a design basis for the standard plant, we conclude that the flood protection for seismic Category I structures, systems and components for those plants that reference GESSAR will be adequate.

3.4.2 Design Procedures

With the proposed plant grade one foot above the elevation of the design basis flood, Category I structures, systems and components will be protected from the hydrodynamic phenomena associated with the flood. The hydrostatic effect of the flood, however, will be considered in the design of all Category I structures exposed to the water head. All seismic Category I structures will be designed to remain stable when subjected to either overturning moments or uplift forces of the flood. We have reviewed the proposed loading combinations discussed in Section 3.8.4 of GESSAR and conclude that the proposed design procedures to be used for GESSAR are acceptable.

3.4.3 Conclusions

We conclude that the procedures that will be utilized to determine the loadings on seismic Category I structures induced by the design flood level specified for the plant are acceptable since these procedures provide a conservative basis for

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engineering design to assure that the structures will withstand such environmental forces. Conformance with these design procedures is an acceptable basis for satisfying in part the requirements of General Design Criterion 2.

3.5 Missile Protection Criteria

Structures, shields and barriers that will be designed to withstand the effects of the various postulated missiles, are listed in Section 3.5.1 of GESSAR. The missiles that can potentially impact each structure, shield or barrier are identified in Section 3.5.2.

3.5.1 Internal Missiles

Missile protection will be provided to ensure the safe shutdown capability of the GESSAR plant. The seismic Category I structures, equipment and nuclear safety related systems will be protected from postulated missiles generated by internal rotating or pressurized equipment through basic plant arrangement such that, if equipment failure should occur, the resulting missile will not cause the failure of the seismic Category I structures or nuclear safety related equipment. Physical barriers will be provided, when required, to isolate the missile source or to shield essential components. Redundant engineered (safety) equipment will be protected such that one missile cannot simultaneously damage both trains.

Based on the review of the missile protection criteria given in GESSAR Section 3.5, we conclude that internally generated missile protection for the GESSAR plant is acceptable.

3.5.2 Turbine-Generator Missiles

We did not include turbine missiles in our review since the orientation of the turbine with respect to safety-related features of the site, which is important in assessing the effects of such missiles, is not within the GESSAR scope. The effects of turbine missiles will be evaluated on a case-by-case basis.

3.5.3 Tornado Missiles

We have reviewed the information in GESSAR Sections 3.5.3, 3.5.4, and 3. .5 concerning tornado missiles, their impact velocities and trajectories, and the tornado missile protection provided for the GESSAR plant. The original velocities proposed by GE for the various tornado missiles were unacceptable to the staff. In Amendment 25 to GESSAR, GE demonstrated to our satisfaction that adequate penetration resistance was provided for GESSAR structures against missiles with velocities listed in Column B, Table II, below and in Table 3.5.1 of GESSAR. However, the staff still finds the GE missile velocities (Table 3.5.4 of GESSAR) used as design bases for structural stability and spalling evaluations unacceptable, as they are unconservatively low and are not adequately supported by data. Based on a survey of tornado generated missiles from previous applications, the staff established guidance for such missile velocities noted in Column A, Table II. We will require as a condition of the PDA that GE adopt, as their design basis, either the NRC velocities

TABLE I

Missile*	Dimensions	Weight
A - Wood plank	4" x 12" x 12'	200 lb
B - Steel pipe	3" ϕ, 10' long, schedule 40	78 lb
C - Steel rod	1" ϕ x 3' long	8 lb
D - Steel pipe	6" ϕ, 15' long, schedule 40	285 lb
E - Steel pipe	12" ϕ, 15' long schedule 40	743 lb
F - Utility pile	13.5" ϕ x 35' long	1490 lb
G - Automobile	20 ft^2 frontal area	4000 lb

* These missiles are to be considered as striking on both horizontal and vertical surfaces. Missiles A, B, C, D, and E are to be considered at all altitudes and missiles F and G at altitudes less than 30 feet above all grade levels within 1/2 mile of the facility structures. Any sites with elevations higher than plant grade within 1/2 mile will have to be examined on a case basis with respect to design against missiles F and G.

TABLE II

Missile	Column A NRC[1] Velocity	Column B GE Penetration[2] Velocity	Column C TVA[3] Horizontal Velocity
A	423 fps	423	368
B	211	211	268
C	317	275	259
D	211	211	230
E	211	211	205
F	211	211	241
G	100	74	100

1 These missiles are to be capable of striking in all directions.

2 GE has committed to using these velocities as the design basis for penetration only. They have not committed to their use as a basis for structural stability and spalling evaluations. For these conditions, GE proposed much lower impact velocities.

3 TVA vertical velocities are equal to 80% of their horizontal velocities.

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from Column A, Table II, below or the no tumbling horizontal velocities from TVA Topical Report TVA-TR74-1 (Column C, Table II, below) which the staff has reviewed and found acceptable for licensing use. We will make the use of acceptable velocities a condition of our PDA.

The analysis of structures, shields and barriers to determine the effects of missile impact will be accomplished in two steps.

In the first step, the potential damage that could be done by the missile in the immediate vicinity of impact on concrete targets will be determined by the use of the Modified Petry formula. Furthermore, secondary missiles, that could potentially be generated by spalling of the target, will be prevented by fixing the target thickness well above that determined by penetration. In the case of steel targets, formulas developed by the Stanford Research Institute for estimation of penetration of missiles will be used.

In the second step of the analysis, the overall structural response of the target when impacted by a missile will be determined using established methods of impactive analysis where the momentum of the missile is transferred to the target to determine the energy that has to be absorbed by the target.

The load of the missile impact, whether the missile is environmentally generated or accidentally generated within the plant, will be combined with other applicable loads as will be discussed in Section 3.8 of this report.

We conclude that the procedures that will be utilized to determine the effects and loadings on seismic Category I structures and missile barriers induced by design basis missiles selected for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures or barriers are adequately protected against the effects of missile impacts.

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

3.6 <u>Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping</u>

With respect to systems located inside containment, the criteria to be employed for determination of the systems which are evaluated, the locations and types of piping breaks which are postulated and the protection measures against pipe whip to be provided will be consistent with the provisions of Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

GE has informed us that the design of the fast scram system requires the use of 1-1/4" piping in the inlet scram piping in lieu of 1" piping. We have analyzed the effects of postulated breaks in this piping and conclude that measures need not be taken to protect against the effects of pipe whip since:

a) The hydraulic control units (HCU's) and piping to the control rod drives are located in containment away from other safety related equipment; therefore should a line fail, it would not affect any safety related equipment but only impact on other HCU lines. As discussed in Regulatory Guide 1.46, a whipping pipe will only rupture an impacted pipe of smaller nominal pipe size and lighter wall thickness.
b) The total amount of energy contained in the 1-1/4" piping between the normally closed scram insert valve on the HCU module and the ball-check valve in the control rod housing is small. In the event of a rupture of this line, the ball-check valve will close to prevent reactor vessel flow out of the break.
c) Even if a number of the HCU lines ruptured, the scram function would not be impaired since reactor vessel pressure would scram the control rods.

The methods of analysis described in GESSAR will adequately account for the dynamic loadings on systems, structures and components that are associated with pipe rupture assumptions and will provide adequate assurance that the containment structure, unaffected system components, and those systems important to safety which are in close proximity to the systems in which postulated pipe failures are assumed to occur, will be protected.

The dynamic analyses described in GESSAR for determination of restraint loading resulting from postulated pipe ruptures will yield conservative results for the large clearance, large deformation restraints described in GESSAR (i.e., a gap size of approximately six inches) when used with the thrust forces calculated in accordance with the relationship given in Section 3.6 of GESSAR. Design limits proposed by GE in Section 3.6.3.1.5.1 of GESSAR for use in the design of the pipe whip restraints will result in deformation limits as conservative as ours for all methods and all material employed. The methods used for formulating the hydrodynamic forcing functions induced by pipe rupture and the dynamic analysis for the pipe whip motion provide an acceptable basis for restraint design. The criteria used for the identification, design, and analysis of piping systems where postulated breaks may occur constitute an acceptable design basis in meeting the applicable requirements of the General Design Criteria 1, 2, 4, 14, 15, 31, and 32.

The provisions for protection against the dynamic effects associated with pipe ruptures and the resulting discharging coolant provide adequate assurance that, in the event of the combined loadings imposed by an earthquake of the magnitude specified for the Safe

Shutdown Earthquake and a concurrent single pipe break at one of the design basis break locations, the following conditions and safety functions will be accommodated and assured:

(1) the magnitude of a design basis loss-of-coolant accident cannot be aggravated by potential multiple failures of piping,
(2) the reactor emergency core cooling systems can be expected to perform their intended function assuming a single failure.

The applicant has described in GESSAR Section 3.6.1.4 the criteria which will be used in the analysis for high and moderate energy line breaks outside containment. During the course of piping design for these high energy systems, consideration will be given to physical separation of pipes from safety-related equipment and instrumentation, as the primary protection against the effects of a postulated pipe break and single failure criteria.

In implementing these criteria, the applicant will designate design basis break locations throughout all high energy piping systems. These postulated break locations will be chosen on the basis of highest relative stress, or significant changes in flexibility of the piping. The protection provided against the dynamic effects of postulated pipe breaks and discharging fluids in piping systems containing high energy fluids and located outside the containment is adequate to prevent damage to structures, systems and components to the extent considered necessary to assure the maintenance of their structural integrity. Such protection provides reasonable assurance that the safe shutdown of the reactor can be accomplished and maintained.

In addition, for those piping systems not considered as high energy systems, GE will postulate leakage cracks in accordance with the above referenced section of GESSAR to assure that essential equipment and components are protected from fluid spraying, flooding and consequent environmental conditions developed. The criteria used for the identification, design and analysis of high and moderate energy fluid lines outside containment where postulated breaks and cracks may occur constitutes an acceptable design basis for satisfying the applicable requirements of NRC General Design Criterion 4.

3.7 Seismic Design

3.7.1 Seismic Input

The input seismic design response spectra (OBE and SSE) and the damping values applied in the design of seismic Category I structures, systems and components comply with the provisions of NRC Regulatory Guides 1.60, "Design Response Spectra for Nuclear Power Plants" and 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," respectively.

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The synthetic time history used for seismic design of Category I plant structures, systems and components is adjusted in amplitude and frequency to obtain response spectra that envelope the GESSAR design response spectra defined by Regulatory Guide 1.60, normalized to 0.3g for the SSE.

Conformance with provisions of NRC Regulatory Guides 1.60 and 1.61 provides reasonable assurance that for, an earthquake whose intensity is 0.15g for OBE and 0.30g for SSE, the resulting accelerations and displacements imposed on Category I structures, systems and components are adequately defined to assure a conservative basis for the design of such structures, systems and components to withstand the consequent seismic loadings. Compliance with the provisions of these Guides constitutes an acceptable basis for satisfying the requirements of NRC General Design Criterion 2.

3.7.2 Seismic System Analysis

3.7.3 Seismic Subsystem Analysis

Modal response spectrum and time history methods for multi-degree-of-freedom systems form the bases for the analyses of all major Category I structures, systems and components. Governing response parameters are combined by the square root of the sum of the squares of each modal response to obtain the modal maxima when the modal response spectrum method is used. The absolute sum of the modal responses are used for closely-spaced modal frequencies. The square root of the sum of the squares of the maximum co-directional responses are used in accounting for the three components of the earthquake motion. Floor response spectra inputs used for design and test verification of structures, systems and components are generated from the time history method. Dynamic analyses of vertical seismic-systems are employed for all structures, systems and components where analyses show significant structural amplifications in the vertical direction. The system and subsystem analyses are performed based on elastic theory. The finite element approach is used to evaluate the soil-structure interaction effects for deeply embedded Category I structures. Deeply embedded has been defined by GE as being the case where the embedment is greater than 15% of the smaller horizontal dimensions of the foundation mat. Soil spring and other equivalent methods are used to determine the soil-structure interaction effects of cases other than deeply embedded Category I structures. Non-linear stress-strain and damping relationships for soil are incorporated in the finite element analysis of soil-structure interaction.

The applicant performed eighteen cases of soil-structure interaction studies using finite element methods for generating the GESSAR seismic design envelope. These cases constitute a parametric study to verify their design envelope. The cases cover a soil depth range of 75 ft. to 300 ft., a shear wave velocity range of 500 fps to 3500 fps, a Poisson's ratio range of 0.30 to 0.38, a groundwater level at the elevation of the foundation mat for 17 cases, at a level 2 feet below grade for the 18th case, and a constant embedment depth of 40 ft., and in two directions 90° apart

For each of the cases mentioned above, the sequence of analytical operations involved was, (a) deconvolution analysis to obtain motion at base of soil profile, (b) one dimensional finite element analysis to verify free field motion and to establish maximum element size, (c) establishment of a full finite element mesh to represent the soil surrounding and underlying major Category I structures, (d) establishment of an appropriate model of the structure and combination with the finite element mesh to obtain the required soil-structure representation, and (e) evaluation of appropriate response spectra and seismic design envelope spectra.

The applicant proposes six conditions to be satisfied by all plants referencing GESSAR. Satisfying these six conditions ensures the seismic adequacy of the nuclear power plant referencing GESSAR. The six conditions are as follows:

1. The maximum ground acceleration at zero period of the site design response spectra is less than or equal to 0.3g SSE, 0.15g OBE.
2. The site design response spectra are less than or equal to those given in Regulatory Guide 1.60 normalized to the maximum ground accelerations given in 1. above.
3. There is no potential for liquefaction at the plant site due to the SSE or OBE.
4. There is no potential for fault displacement near or underneath the plant foundation.
5. The embedment depth of the reactor building is between 34 to 40 feet (± 0.5 feet excavation tolerance) for soil sites. For sites with shear wave velocities greater than 3500 fps, there is no limitation on embedment depth.
6. The average shear wave velocity for the top 30 feet of soil is greater than 500 fps.

We find these six conditions to be acceptable.

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

3.7.4 Seismic Instrumentation Program

The installation of the specified seismic instrumentation in the reactor containment structure and at other Category I structures, systems, and components discussed in GESSAR Section 3.7.4 constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with Regulatory Guide 1.12.

We conclude that the seismic instrumentation program proposed by the applicant is acceptable.

3.8 Design of Seismic Category I Structures

3.8.1 Concrete Containment

This section is not applicable to the GESSAR application.

3.8.2 Steel Containment

The reactor coolant system will be housed within a free-standing steel cylindrical shell topped with a hemi-ellipsoidal dome and fixed at its bottom into a concrete mat covered with a liner plate. The steel containment will be enclosed by a reinforced concrete shield building. The containment will employ the Mark III pressure suppression system which will be utilized to limit the post-LOCA containment pressure and temperature transients.

The steel containment including all its penetrations will be designed, analyzed, fabricated, constructed, inspected and tested in strict accordance with the rules of Subsection NE of the ASME Boiler and Pressure Vessel Code Section III, Division 1, as augmented by Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," with the single exception to the Regulatory Guide noted below.

The containment will be designed for all the various load combinations that are considered credible, including appropriate combinations of accident and seismic loads. In addition, the containment has the capability to withstand a post-LOCA flooded condition to approximately 7'-0" above the top of the reactor core and in conjunction with an Operating Basis Earthquake. Such a flooding condition may be required to recover the fuel in the reactor after a LOCA.

Regulatory Guide 1.57 states that normal design limits should be used whenever the containment is subjected to the concurrent loadings that result from flooding of containment for accident recovery and the vibratory motion of 50 percent of the SSE. General Electric's design does not meet this criterion since the containment is not designed to meet normal limits for this loading combination but rather they meet a limit that is between the emergency and faulted limits. We have determined that this deviation from Regulatory Guide 1.57 is acceptable. Our basis for this determination is that this loading combination will result in stresses that are well below the yield condition of the containment material (less than about 80% of yield) and therefore the containment would maintain its integrity with adequate margin in the unlikely event that an OBE should occur during post-accident recovery.

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The materials that will be used in the construction of the containment will meet the requirements of Article NE-2000 of Subsection NE of the ASME Section III Code. The bottom region of the containment that will be submerged in the suppression pool will not be coated except for a narrow band at the water line. At the staff's request, GE modified GESSAR to include an inservice inspection program to detect any corrosion of the steel shell, particularly pitting, and execute appropriate corrective measures. As discussed in Section 1.1 of this SER, any utility applicant referencing GESSAR agrees to perform all tests and inspections stipulated therein as a part of their PSAR.

After the completion of construction and prior to operation the containment will be subjected to a structural proof test at 1.15 times the design pressure in accordance with ASME Section III, Subsection NE-6321. We conclude that the criteria that will be used in the analysis, design and construction of the steel containment structure to account for the loadings and conditions that are anticipated to be experienced by the structure during its service lifetime are in conformance with established criteria, and with codes, standards and specifications listed in GESSAR Chapter 3 that are acceptable to the Regulatory staff.

The use of these criteria as defined by applicable codes, standards, and specifications, the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control and special construction techniques; and the testing and inservice surveillance requirements, provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within and outside the containment, the containment structure will withstand the specified conditions without impairment of its structural integrity or safety function. A Category I concrete shield building will protect the containment from the effects of wind and tornadoes and various postulated accidents occurring outside the shield building. Conformance with these criteria constitutes an acceptable basis for satisfying in part the requirements of General Design Criteria 2, 4, 16, and 50.

Several phenomena have been identified in our reviews of the Mark III containment that could result in dynamic loading of structures located in or above the suppression pool as a result of a LOCA or relief valve operation. These phenomena, and the design provision to prevent damage resulting from the phenomena, are discussed in more detail in Section 6.2.1.9 of this SER.

3.8.3 Containment Interior Structures

The major containment interior structures include the drywell, the reactor pedestal and shield wall, the refueling pool and operating floor, and various other intermediate floors. The codes applicable to these interior structures are listed in Section 3.8.3 of GESSAR.

The drywell will be a reinforced concrete cylindrical structure with a flat roof that is stiffened by two deep girders forming the refueling pool. It will completely enclose the reactor vessel and the recirculation system. Its primary function is to divert the steam released during a LOCA to the suppression pool. Because of this important function on which the proper functioning of the pressure suppression system depends, the Regulatory staff has requested that the drywell be treated to a certain extent as a containment structure. Accordingly, the design and analysis procedures and the loads and load combinations will be similar to what is normally used and accepted for concrete containments.

The lower portion of the drywell wall, which forms the suppression pool wall with the horizontal vents emplaced in it, is to be constructed in a fairly novel form. The large number of 2-foot diameter vent pipes passing through this section of the wall makes the normal reinforced concrete construction method impractical. Instead, this portion of the pressure-retaining wall is proposed as an unreinforced composite section, with external steel plates carrying part of the circumferential and longitudinal stresses.

We are requiring that the drywell be subjected to a pre-operational leak test and a structural proof test using air at or above the design pressure to verify the capability of the completed vessel to withstand the maximum design pressure. Details of the drywell leak test are provided in Section 6.2.1 of this SER.

A description of the pre-operational structural proof test for the drywell is given in GESSAR Section 3.8.3.7. Strain and deflection measurements will be taken during the structural proof testing of the prototype* drywell and compared with the pretest predictions. The data from this test will provide a measure of the margin available to sustain the thermal and seismic loads.

Guard pipes will be provided for the main steam lines and other high pressure lines listed in Table 3.8.5 and Figure 6.2-15 of GESSAR which traverse the boundaries between the inside of the drywell and the outside of the shield building. In the event of a rupture of a high energy line between the shield building and the drywell, the high pressure fluid from the line would be retained within the guard pipe and exhausted into the drywell instead of leaking into and potentially overpressurizing the containment. The proposed design of the guard pipes is documented in Section 3.8.6.1.1 and illustrated in Figure 3.8-25 of GESSAR and further discussed in Section 6.2.1 of this SER. These guard pipes will also be designed, constructed, and tested in accordance with Subsection NE of the ASME Section III Code.

The other interior structures will also be designed for appropriate load combinations listed in Section 3.8.3.1.3 of GESSAR and are acceptable to the Regulatory staff.

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*A prototype drywell is defined as one that incorporates a new or unusual design feature, in part or in full, that has not yet been confirmed by a test on a prototype drywell with appropriate instrumentation. For example, drywells for plants of the same size, but not replicated, and designed by different Architect Engineers, will be considered separate prototype designs.

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These loads include appropriate combinations of normal operating loads, seismic loads, the loss-of-coolant accident (and other accidents involving high energy pipe ruptures) including temperature, pressure, jet impingement, pipe whip, pipe rupture reaction forces and pool dynamics effects (Section 6.2.1.9) as discussed in Section 3.8.3.1.3 of GESSAR.

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

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

3.8.4 Other Category I Structures

Category I structures other than containment and its internal structures that are included in the Nuclear Island will be built from structural steel and concrete. The structural components consist of slabs, walls, beams and columns. The design methods for concrete will follow those specified in the ACI-318 Code, and for steel will follow the AISC specifications with appropriate modifications requested by the staff (as described in Attachment II to Section 3 of our November 1, 1973 request for additional information to GE) and incorporated by GE in Section 3.8 of GESSAR to account for loading conditions peculiar to nuclear power plants.

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

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

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either the structural integrity or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying in part the requirements of General Design Criteria 2 and 4.

3.8.5 Foundations

The foundation of the containment will be a concrete mat. It will be analyzed to determine the effects of the various combinations of loads expected during the life of the plant. The analysis described in GESSAR Section 3.8.5 will take into account bending moments, shear, and soil pressure for a plate on an elastic foundation. The containment foundation will be designed and constructed in accordance with the rules of the proposed ACI/ASME Code for "Concrete Vessels and Containments" with certain modifications requested by the staff. Foundations of other Category I structures, likewise, are reinforced concrete mats. Such foundations will be designed in accordance with the ACI-318 Code. These concrete foundations will be designed to resist various combinations of dead loads, live loads, environmental loads, including winds, tornadoes, OBE, and SSE, and loads generated by postulated ruptures of high energy pipes.

We conclude that the criteria described in Section 3.8.5 of GESSAR, that will be used in the analysis, design and construction of all the Nuclear Island Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime, are in conformance with established criteria, industry codes, standards, and specifications that are acceptable to the Regulatory staff.

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

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

3.9.1.1 Piping Vibration Operational Test Program

Preoperational piping vibration tests will be conducted on the main steam line and the recirculation system. In response to our concern, GE has committed to perform preoperational vibration tests on all ASME Class 1 and 2 piping. With this requirement met, the preoperational vibration test program which will be conducted during startup and initial operating conditions on all safety-related ASME Class 1 and 2 systems, restraints, components and supports is an acceptable program for issuance of

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the PDA. The tests will provide additional verification that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design operational transients. The formal detailed description of the planned tests will be included in the FSAR. The planned tests will develop loads similar to those experienced during reactor operation. Compliance with this test program constitutes an acceptable basis for verifying the existence of adequate design margins as specified in NRC General Design Criterion 1, and is suitable for the PDA licensing stage.

3.9.1.2 Seismic Qualification of Mechanical Equipment

Proper functioning of safety related mechanical equipment is essential to assure the capability of such equipment to perform protective actions in the event of a Safe Shutdown Earthquake (SSE). The dynamic testing and analysis procedures, described in Section 3.9.1.2 of GESSAR will be implemented to confirm that all Category I mechanical equipment will function during and after an earthquake of magnitude up to and including the SSE, and that all equipment support structures are adequately designed to withstand seismic disturbances are acceptable.

Subjecting the equipment and its supports to these dynamic testing and analysis procedures confirms the existence of adequate design safety margins such that in the event of an earthquake at the site, the Category I mechanical equipment, as identified in GESSAR, will continue to function during and after a seismic event, and the combined loading imposed on the equipment and its supports will not exceed applicable code allowable design stress and strain limits. Limiting the stresses of the supports under such loading combinations provides an acceptable basis for the design associated with seismic events, as well as operational vibratory loading conditions without gross loss of structural integrity.

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

3.9.1.3 Preoperational Vibration Assurance Program for Reactor Internals

With regard to flow-induced vibration testing of reactor internals, the applicant has stated in GESSAR that the first BWR/6 plant of each size will be considered a prototype design and will be instrumented and subjected to both cold and hot two-phase flow testing to demonstrate that flow-induced vibrations similar to those expected during operation will not cause damage. The BWR/6 plants currently scheduled for prototype testing are River Bend (size 218), Perry (size 238) and Grand Gulf (size 251). Specific predictions and acceptance criteria will be supplied at the operating license review (FSAR) stage of each of the cited plants.

The preoperational vibration assurance program as planned for the reactor internals provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions that will be comparable to those experienced during operation. The combination of tests, predictive analysis and post-test inspection provide adequate assurance that the reactor internals may be expected, during their service lifetime, to withstand the flow-induced vibrations of reactor operations without loss of structural integrity. The continued integrity of the reactor internals in service is essential to assure the retention of all reactor fuel assemblies in their place as well as to permit unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown.

The conduct of the preoperational vibration tests constitutes an acceptable basis for demonstrating design adequacy of the reactor internals in partially fulfilling the requirements of NRC General Design Criteria 1 and 4 and in conforming with the provisions of Regulatory Guide 1.20, "Vibration Measurements on Reactor Internals."

3.9.1.4 Analysis Methods for LOCA Loadings

To confirm the structural design adequacy of the reactor internals, including the control rod assemblies, the applicant has described the dynamic analysis of the reactor internals, together with the unbroken piping loops, which will be performed under the combined effects of the postulated occurrence of a loss-of-coolant accident and a safe shutdown earthquake as well as an SSE and a steam line break. The dynamic system analysis which will be performed, provides an acceptable basis for confirming the structural design adequacy of the reactor internals and the unbroken piping loops to withstand the combined dynamic effects of the postulated occurrence of a recirculation line break plus an SSE and an SSE plus a steam line break. The analysis will provide adequate assurance that the combined stresses and strains in the components of the reactor coolant systems and reactor internals, for these faulted conditions, will not exceed the allowable design stress and strain limits of ASME Section III, Appendix F (faulted limits) for the materials of construction, and that the resulting deflections or displacements of any structural element of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The assurance of structural integrity of the reactor internals under a recirculation line break or a steam line rupture concurrent with the most adverse loading event (SSE) provides added confidence that the design may be expected to withstand a spectrum of lesser pipe breaks and seismic loading events. Compliance with the dynamic system analysis and acceptance criteria listed above, constitutes an acceptable basis for satisfying the requirements of NRC General Design Criteria 2 and 4.

3.9.2 ASME Code Class 2 and 3 Components

All safety-related ASME Code Class 2 and 3 systems, components and equipment will be designed to sustain normal loads, anticipated transients, the Operating Basis Earthquake and the Safe Shutdown Earthquake within design limits which are consistent with those outlined in NRC Regulatory Guide 1.48, "Design Limits and Loading Conditions." The specified design basis loading combinations, as applied to the design of the safety-related ASME Code Class 2 and 3 pressure-retaining components in systems classified as Category I, provide reasonable assurance that in the

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event an earthquake should occur at the site or other upset, emergency or faulted plant transients should occur during normal plant operation, the resulting combined stresses imposed on the system components are expected to remain within the allowable design stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without gross loss of structural integrity. The applicant's design load combinations and associated stress and deformation limits specified for all ASME Code Class 2 and 3 components constitute an acceptable basis for design in satisfying NRC General Design Criteria 1, 2 and 4 and are consistent with recent Regulatory staff positions.

The criteria used in developing the design and mounting of ASME Class 2 and 3 safety and relief valves provides adequate assurance that, under discharging conditions, the resulting stresses will be within the allowable design stress and strain limits of ASME Section III for the materials of construction as discussed in Section 3.9.2.5 of GESSAR. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function.

The criteria used for the design and installation of ASME Class 2 and 3 overpressure relief devices constitute an acceptable design basis in meeting the applicable requirements of NRC General Design Criteria 1, 2, 4, 14, and 15 and are consistent with the provisions of Regulatory Guide 1.67, "Installation of Overpressure Protection Devices."

3.9.3 Component Operability Assurance Program

The applicant has provided in GESSAR Section 3.9.4 a program to assure the operability of active components which is acceptable to the staff. Active components are defined as those pumps required to function and valves required to open or close during or following the specified plant condition. The applicants Operability Assurance Program will include the testing of all active pumps and valves prior to installation, qualification after installation, and periodic testing during their plant service life to verify adequacy of performance.

The conduct of the applicant's proposed operability assurance program will provide adequate assurance of capability of active pumps and valves in seismic Category I systems including those which may be classified as ASME Code Class 1, 2 and 3, to withstand postulated seismic loads in combination with other significant loads without loss of structural integrity, and to perform the "active" function (i.e., pump operation, valve closure or opening) when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated. The specified component operability assurance procedures constitute an acceptable basis for implementing the requirements of General Design Criteria 1, 2 and 4 as related to operability of ASME Code Class 1, 2 and 3 active pumps and valves.

3.9.4 Components Not Covered by the ASME Code

Table 3.2.1 of GESSAR lists the safety-related mechanical components in the GESSAR scope which are not covered by the ASME code, along with design requirements. The design procedures that will be applied to the control rod assemblies and control rod drives are based upon a) acceptable methods of analysis, b) specification of appropriate allowable stresses, and c) a proper definition of the applied loads. These design procedures are supported by specialized tests, including an endurance test of a prototype of a control rod drive mechanism. The use of these design criteria provides reasonable assurance that the control rods and the control rod drives can be expected to withstand the imposed loads associated with normal reactor operation, anticipated operational transients, postulated accidents, and seismic events, without loss of their structural integrity or impairment of their function. Compliance with these design criteria satisfies the applicable requirements of Criteria 2 and 14 of the General Design Criteria.

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

General Design Criterion 2 (GDC-2) requires, in part, that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Regulatory Guide 1.29 specifies the structures, systems, and components which are designated as Category I and which should be designed to remain functional during and following the occurrence of a safe shutdown earthquake. As part of our evaluation for the PDA, we reviewed the material presented in GESSAR to determine whether the requirements of GDC-2 will be met and whether the scope of the seismic qualification program will conform to the staff position set forth in Regulatory Guide 1.29. The specific areas covered by the staff review are:

1. The criteria for seismic qualification,
2. The methods and procedures used to implement the criteria by tests or a combination of tests and analyses, and
3. The scope of the seismic qualification program, i.e., the specific equipment to be seismically qualified.

GE has stated in Section 3.10.1.3 of GESSAR that all Category I electrical equipment will be seismically qualified in accordance with the staff position, "Electrical and Mechanical Equipment Seismic Qualification Program," dated December 1973. We conclude that the criteria set forth in the staff position which was adopted by GE are acceptable as a basis for seismic qualification of Category I instrumentation and electrical equipment.

We have not completed our review of the methods and procedures to be used in implementing the seismic design criteria nor of the scope of the seismic qualification program. GE has agreed to submit for staff review: (1) a list of the specific equipment which will be seismically qualified, and (2) the qualification procedures to be used in implementing the seismic design qualification criteria. These two aspects of the review will be conducted post-PDA.

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We have concluded that the commitment by GE to comply with the seismic design set forth in the staff position, "Electrical and Mechanical Equipment Seismic Qualification Program," dated December 5, 1973, is an acceptable basis for the PDA. It is expected that the review of the qualification methods and procedures and the scope of the seismic qualification program can be completed during our review of the preliminary designs for the instrumentation systems which will also be conducted as a post-PDA item as discussed in Section 7.1 of this SER. We will report the results of our evaluation of these two items when our post-PDA review is complete.

3.11 Environmental Design of Mechanical and Electrical Equipment

General Design Criterion 4 (GDC-4) requires, in part, that structures, systems, and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. As part of our review for the PDA, we reviewed the material presented in GESSAR to determine whether the requirements of GDC 4 with respect to the environmental design of safety-related mechanical and electrical equipment will be met. The specific areas covered by the staff review are:

1) The criteria for environmental qualification of safety-related equipment,
2) The methods and procedures to be used to implement the criteria by tests or by a combination of tests and analyses, and
3) The scope of the environmental qualification program, i.e., the specific equipment to be environmentally qualified.

In response to our requests for additional information, GE stated that a program is underway to qualify all Class IE equipment to the requirements of IEEE Std 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." In Amendment No. 24 to GESSAR, GE has stated that it will develop an instrumentation and control aging program as a method of implementing the aging requirements of IEEE Std 323-1974 and that such a program to obtain a qualified life of 40 years may require participation by utility-users. GE has also stated that it will comply with the requirements of IEEE Std 382-1972, "Trial-Use Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations," as modified Regulatory Guide 1.73. With respect to compliance with the requirements of IEEE Std 334-1971, "Trial-Use Guide for Type Tests of Continuous Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations," as modified by Regulatory Guide 1.40, GESSAR does not use any continuous duty Class I motors inside the containment.

We have not completed our review of the methods and procedures to be used to implement the environmental qualification criteria nor of the scope of the environmental qualification program. During a meeting with GE on October 29, 1974, it was agreed that GE would submit for staff review: 1) a list of the specific equipment that will be environmentally qualified, and 2) the qualification procedures to be used in implementing the environmental qualification criteria. These two

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aspects of the review will be conducted as post-PDA items requiring resolution prior to the final design approval review. Some of the methods that may be used to comply with the requirements of IEEE Std 323-1974 involve design changes such as installing additional equipment. Resolution of the methods to be used to implement IEEE Std 323-1974 must occur before the design is established.

We have concluded that the commitments by GE to comply with the requirements of IEEE Std 323-1974 and IEEE Std 382-1972 as modified by Regulatory Guide 1.73 are an acceptable basis for the PDA.

3.11.1 Electric Penetrations in the Shield Building, Containment and Drywell Walls

We have reviewed the information presented in Sections 3.8.6.2 and 7.1.2.4 of GESSAR pertaining to the electric penetrations for the shield building, containment and drywell. The preliminary design for the drywell penetrations will be submitted and reviewed as a post-PDA item.

GE has stated in Section 7.1.2.4 of GESSAR that they will meet the requirements of IEEE Std 317-1972, "Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations."

We have concluded that the review of the preliminary design for the drywell penetrations and the procedures and methods to be used to qualify the shield building, containment, and drywell penetrations can be conducted during the post-PDA review phase. We will report the results of this review in a supplement to the SER.

3.12 Separation Criteria for Safety-Related Mechanical and Electrical Equipment

We have reviewed the proposed design criteria for the separation of redundant safety equipment as set forth in Section 3.12 of GESSAR. We have concluded that these criteria meet the requirements of General Design Criteria 3, 17 and 21 pertaining to physical independence of Class IE circuits and the regulatory position of Regulatory Guide 1.75 and are acceptable.

We will review the implementation of these criteria after receipt of a preliminary design for the instrumentation systems. Particular emphasis will be placed on the review of the design of the isolation devices used where signals are transmitted between redundant divisions of equipment. We will report the results of our review of the implementation methods in a supplement to the SER. We have concluded that these criteria are acceptable for the PDA. Our review of the implementation methods will be conducted as a post-PDA item in conjunction with our review of the preliminary design of the instrumentation systems.

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4.0 REACTOR

4.1 General

The nuclear steam supply system includes a General Electric Company (GE) boiling water reactor (BWR) which generates steam for direct use in the steam-driven turbine generator. The design of the General Electric Standard Safety Analysis Report (GESSAR) reactor is similar to the Grand Gulf Nuclear Station Units 1 and 2 and to the Perry Nuclear Power Plant, Units 1 and 2 that have been reviewed by the Regulatory staff at the construction permit stage.

The fuel and heat source consists of slightly enriched uranium dioxide pellets contained in sealed zirconium alloy tubes about one-half inch in diameter. These fuel rods, which are over twelve feet long, are assembled into fuel assemblies each consisting of 63 fuel rods plus one spacer-capture water rod in an 8 x 8 array within a square open-ended zirconium channel box. Seven hundred and thirty-two of these fuel assemblies form a roughly cylindrical core.

The core is supported in a domed cylindrical shroud inside the reactor vessel. Steam separators and dryers are mounted on the shroud dome. Two external, motor-driven, constant speed recirculating pumps inject high-velocity water into 20 jet pumps which are located in the annulus between the shroud and the reactor vessel. The high velocity water from the jet nozzles entrains and imparts energy to additional water from the annular region. The combined flow enters the bottom of the reactor core and boils as it passes upward through the fuel assemblies.

The steam is separated from the steam-water mixture which emerges from the core by the steam separators and dryers. The steam flows to the turbine-generator through four 26-inch diameter main steam lines. The heated condensate returns to the reactor through two 24-inch feedwater lines and is injected into the annulus between the shroud and the vessel.

Control of the fission reaction within the core is achieved by the movement of neutron absorbing cruciform-shaped control rods, and by variation of the flow rate through the core, thereby changing the steam fraction and moderator density. Individual hydraulic drives permit the control rods to be axially inserted to any degree desired or to be inserted fully and swiftly upon receipt of a trip signal (scram). Core flow rate is varied by the flow control valves in the recirculation lines.

4.2 Mechanical Design

4.2.1 Fuel Mechanical Design

The core of the GESSAR reactor will contain 46,116 fuel rods manufactured by encasing uranium dioxide fuel pellets, of approximately 94% theoretical density, within Zircaloy-2 cladding. The fuel pellets are chamfered and undished and have the shape of of a right circular cylinder with a height to diameter ratio of nearly one. Several fuel rod and fuel assembly design parameters are listed in Table 4.2-1 of this SER.

Local hydriding of the cladding ID surfaces has occurred in several operating reactors and caused clad failures and higher than desired offgas activity. Water vapor present in the rods after manufacture was assumed to be the cause. The fuel rod manufacturing process has since been modified, and a hydrogen-getter has been added in the rod as a means of assuring that moisture is not present or will not cause internal hydriding of the clad. The getter material consists of zirconium alloy chips, loosely packed in a stainless steel tube and placed in the plenum area of the fuel rod. The hydrogen-getter is located in a relatively low temperature environment, due to its position in the core, so that it will not react with or reduce the integrity of the cladding during normal, abnormal, or accident conditions.

The staff requires that the applicant assume that densification of uranium dioxide fuel pellets will occur during irradiation. The methods used by GE to calculate the effects of fuel pellet densification have been submitted by GE in Topical Report NEDM-10735. The methods in this report have been reviewed and accepted by the staff, and are applicable to the GESSAR design.

Fuel assembly design concerns are directed to maintenance of basic assembly geometry for adequate coolant passage and preservation of cladding integrity to contain the fission products within the fuel rod.

In Section 4.2.1.3.5 of GESSAR, GE describes the loadings and design limits of the fuel assembly and cladding. They discussed the engineering design limits in terms of stress, strain, deflection, fatigue life and creep rupture. In addition, analytical methods to be used to demonstrate design adequacy are described. Such material properties as cladding yield and ultimate stresses, and other thermal properties are given. We reviewed those design bases in detail and found that they provide an acceptable description of design bases for the 8 x 8 fuel assembly. Details of our evaluation of the 8 x 8 fuel design are included in Appendix D of this report which deals with the 8 x 8 reloads. The only difference between the BWR-6 8 x 8 fuel and the reload 8 x 8 fuel is that the total active fuel length is 4 in. greater in the BWR-6 fuel and the fission gas plenum length is 0.75 in. greater for the BWR-6 rods. These changes are not significant enough to change our general conclusions regarding 8 x 8 fuel given in Appendix D, since: (1) the design limits for both 8 x 8 designs are the same, (2) the service life and peak linear heat generation rate are the same, and (3) the same analytical methodology applies and the accident consequences are unchanged for both designs.

GE has conducted a test of the 8 x 8 design spacer grid and spacer-water rod locking arrangement. In addition, they now have in progress a fuel surveillance program on preselected 8 x 8 reload fuel assemblies during refueling outages. A calculation of cladding strain for the 8 x 8 fuel based on an empirical formula together with gross diameter measurement of irradiated burst tests was submitted by GE as Topical Report

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TABLE 4.2.1

Fuel Assembly Data	
Overall length, in.	176
Nominal active fuel length, in.	148
Fuel rod pitch, in.	0.640
Space between fuel rods, in.	0.147
Channel wall thickness, in.	0.120
Fuel bundle heat transfer area, ft^2	100.3
Fuel Rod Data	
Outside diameter, in.	0.493
Cladding thickness, in.	0.034
Pellet outside diameter, in.	0.416
Fission gas plenum length, in.	12.00
Pellet immersion density, gm/cc	10.42

NEDO-10505, May 1972. We will require an adequate update of that report once GE obtains additional information from the previously mentioned programs. We will also require that the results of the fuel surveillance program, and an adequate stress report for each component of the 8 x 8 fuel assembly, together with safety margins and updated information on cladding strain, be fully documented in the applicant's FSAR.

Prior to our final approval of the 8 x 8 fuel assembly design, at the FDA stage of our review process, we will need to review the following information:

(1) Pressure and temperature capabilities listed on pages 4.2-1 and 4.2-9 of GESSAR should be given in terms of specific values or curves as a function of time.
(2) The analysis method that will be used to predict the combined effects of the LOCA and seismic events on the fuel assembly should be submitted. Specific stress, strain and deflection criteria should be given for this load combination.
(3) The analytical methods of creep buckling should be submitted and should include the creep rupture or creep-fatigue-rupture interaction curve.
(4) The design limit for instability such as instantaneous static or dynamic buckling and creep buckling should be given.
(5) An analysis method for predicting the deformation of the channel box should be submitted.
(6) A stress limit should be given for the peak stress which deals with stress concentrations and transient non-linear thermal stresses.
(7) A justification for setting a 0.060 inch fuel rod deflection limit should be given.

Although the design of the unfueled spacer-capture rods is new, it is based on experience with similar designs. Fuel assemblies with eccentrically located spacer-capture rods have been successfully operated in the Humboldt Bay reactor. General Electric has accumulated extensive fuel operating experiences with fuel whose range of design parameters envelopes the 8 x 8 fuel. Based on this experience, the above items are expected to be similar to the previous design and will be within acceptable limits.

We conclude that the mechanical design criteria of the fuel are acceptable for a PDA.

4.2.2 Reactivity Control Systems

Reactor power can be controlled or regulated by:

(1) movement of the control rods,
(2) inserting liquid poison (boron) into the core via the Standby Liquid Control System, or
(3) variation of the reactor coolant recirculation system flow rate.

In addition, a burnable poison, in the form of Gadolinium Oxide (Gd_2O_3), is added, during the fuel manufacturing process, into certain of the high enrichment assemblies. The addition of this burnable gadolinium poison has the tendency to levelize the

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reactivity of the high enrichment rods throughout their residence time in the core. In addition, a more uniform power distribution is achieved in the reactor core.

The use of gadolinium, which has a high absorption cross section for neutrons, results in a decrease in the neutron flux, and hence the generated power, in the area adjacent to the UO_2-Gd_2O_3 rods. In this manner, the neutron flux of these high enrichment rods can be lowered at the beginning of fuel life, when the rods would have the highest reactivity. With increasing residence times in the core, the quantity of gadolinium in the UO_2-Gd_2O_3 rods is gradually depleted while, at the same time, the quantity of active uranium in these UO_2-Gd_2O_3 rods is also being depleted via fission reaction. Thus, the reactivity of gadolinium bearing fuel rods will not vary with increasing burnup as much as non-gadolinium bearing rods. The thermal conductivity of such rods is slightly lower than that of UO_2 rods. However, the rods are expected to operate at relatively lower power than a UO_2 rod. A different end plug design is used to distinguish them from other fuel rods.

Gadolinia-urania fuel rods have been used by General Electric since 1965 and a substantial number of these rods have successfully achieved high burnups. Failure statistics for gadolinia-urania rods show a slight decrease in failure rate for the burnable poison rods compared with normal UO_2 rods. Preliminary post-irradiation examinations have shown no evidence of unusual rod behavior. Fission gas releases were found to be typical of UO_2, and no significant variations in Gd_2O_3 concentration were discovered. Recent measurements of thermal conductivity and linear thermal expansion show no degradation of UO_2 values for the small additions of gadolinia used. General Electric will provide a detailed description of the performance experience and physical property measurements for gadolinia-urania rods in a topical report prior to the FDA review of GESSAR. Prior to that time, the favorable experience and existing properties data, coupled with the fact that gadolinia rods are never operated at greater than 85% of the maximum UO_2 linear heat ratings, provide assurance that these fuel rods will be acceptable for use in GESSAR.

Control rods (177 in number) are used to bring the reactor through the full range of power (from shutdown to full power operation), to shape the reactor power distribution, and to compensate for changes in reactivity resulting from fuel brunup. Each control rod drive has separate control and rapid insertion (scram) devices. The drives have a common supply pump (and one parallel spare pump) as the hydraulic pressure source for normal operation and a common discharge volume for scram operation. In Amendment 26 to GESSAR, GE committed to modifying the scram system in order to achieve faster control rods insertion rates. These changes are discussed in Section 4.2.3 of this SER.

A control-rod-ejection accident, to be distinguished from the rod drop accident, is precluded by a control rod housing support structure located below the reactor pressure vessel, similar to that installed on the other large General Electric reactors. This structure limits the distance that a ruptured control rod drive housing could be displaced, so that any resulting nuclear transient would not be sufficient to cause fuel rod failure.

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Reactor power can also be controlled through changes in the primary coolant recirculation flow rate. This system, first introduced on the Zimmer plant, was evaluated by the staff during the Zimmer review for a construction permit and was found satisfactory.

The Standby Liquid Control System (SLCS) is available to pump sodium pentaborate into the reactor vessel. This system is designed to bring the reactor to a cold shutdown condition from the full power steady-state operating condition at any time in core life, independent of the control rod system capabilities. The injection rate of the system is adequate to compensate for the effects of xenon decay. The SLCS is further discussed in Section 4.3.3 of this SER.

On the basis of our review of the control rod, flow control and standby liquid control systems design, and the supporting evidence accumulated from operation of similar systems in other General Electric reactors, we conclude that these systems will meet the functional performance requirements, as described in GESSAR Sections 4.2 and 4.3, and are acceptable. The details of the proposed design of the new Rod Pattern Control System, which will allow use of ganged rod motion, have not yet been submitted by GE for Regulatory staff review. We will review the design details of this system post-PDA and report on the review in a supplement to this SER.

4.2.3 Fast Scram

The rapid closure of the turbine stop valves when the reactor is at full power near the end of cycle has the potential for causing a significant reactor coolant pressure boundary (RCPB) pressure transient. The source of this problem is that as the reactor nears the end of cycle the control rods approach a fully removed position below the core while the neutron flux and hence the power flux is skewed toward the top of the reactor. Therefore, when a scram occurs, the control rods must travel a greater distance before they will have a significant effect on terminating the transient. To cope with this problem in GESSAR, GE originally proposed a prompt relief trip (PRT) system. The PRT system utilizes sensors connected to the turbine stop valves to trip the reactor and open the relief valves once the turbine stop valves begin to close. The additional steam voids created by the opening of the relief valves adds negative reactivity to the reactor resulting in a more rapid reactor shutdown and lowering the pressure peak of the transient.

We had not completed our review of the PRT system when, in Amendment 26, GE committed to substitute a "fast scram" system for the PRT. The "fast scram" consists of modifications to the present control rod drive (CRD) hydraulic system. These changes will enable 75% rod insertion to occur in about 1-1/2 seconds as compared to about 2.78 seconds with unmodified rod drives. These system modifications (enlarging the CRD insert lines to 1.25" as compared to 1.0" and increasing the accumulator pressure to 1800 psi as compared to 1500 psi) will eliminate the need for PRT since the "fast scram" system will be able to insert negative reactivity at a much higher rate than the unmodified scram system and at a rate comparable to the combined negative reactivity insertion rates of the unmodified scram system and the PRT system.

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Although the design details of these modifications have not been provided to the NRC staff, GE stated at the 179th ACRS meeting on March 7, 1975 that results have been obtained at GE's San Jose test facility that indicate that GE can achieve the scram time design goals stated above.

On page 256 of the transcript of that meeting, GE states that "The design will include a rather ambitious test program." The NRC staff finds this acceptable, but requires that GE provide, on the GESSAR docket, a detailed description of (and a commitment to) the planned test program.

While the staff believes that the proposed design changes are not a significant departure from the present design, we will complete our review of GE's transient evaluation to be submitted as part of the post-PDA effort (as listed in Section 1.8 of this SER).

4.3 Nuclear Design

The GESSAR BWR/6 reactor core is comprised of fuel lattice cells (four fuel assemblies and one cruciform shaped control rod) as shown in GESSAR Figures 4.1-1 and 4.2-3. The control rods contain 76 stainless steel tubes (19 in each wing) filled with vibration compacted boron carbide powder. Water, which serves as both moderator and coolant, occupies all space not taken by fuel, control rods or structural material. All of the water gaps between fuel assemblies are of the same size. Some of the water gaps, which do not include a control rod, are provided with guide tubes for both fixed and moveable neutron flux detectors. Guide tubes are located in the space near the corners of two adjacent fuel assemblies.

There are a number of noteworthy features of the fuel lattice cell which are applicable to the first fuel cycle. These are: (1) the fuel rods are of four different uranium-235 enrichments, (2) the average enrichment of the uranium-235 isotope in a fuel bundle is 2.07% by weight, and (3) a number of fuel rods will incorporate axially distributed gadolina.

We have reviewed and evaluated General Electric's design bases for the GESSAR reactor. The design bases consist of both safety design bases and power generation design bases. The general requirements of the safety design bases are (1) that sufficient negative reactivity feedback be provided to prevent fuel damage as a result of abnormal operational transients, (2) that nuclear characteristics as required be exhibited to assure that the reactor has no inherent tendency toward divergent or limit cycle operation, and (3) that the excess reactivity of the core be limited sufficiently to assure that the reactivity control systems are capable of making the reactor subcritical with the highest worth control rod fully withdrawn. The general requirements of the power generation design bases are (1) that sufficient reactivity be provided to reach the desired burnup for full power operation, (2) that continuous, stable regulation of core excess reactivity be allowed, and (3) that sufficient negative reactivity feedback be provided to facilitate normal maneuvering and control.

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In addition to the general safety and power generation bases, the GESSAR reactor is designed to meet a number of specific GE design bases. These are listed below:

1. The power reactivity coefficient must always be negative.
2. The moderator void reactivity coefficient must be negative.
3. The Doppler reactivity coefficient must be negative.
4. Control rod operating patterns and withdrawal sequences must be specified so that control rod worths are sufficiently low to prevent damage to the reactor system in the event of a rod drop.
5. The maximum control rod withdrawal speed must not be greater than 3.6 in/sec.
6. Control rod withdrawal increments must be limited so that a rod movement of one increment does not result in a reactor period which cannot be controlled by the operator.
7. The power generation rate must be controlled so that the linear heat generation rate of 13.4 kW/ft is not exceeded and so that MCPR is not less than the operating limit for the plant.
8. The control rod system must be capable of shutting down the reactor ($K_{eff}<1.0$) at any time with the highest worth rod stuck out.
9. Sufficient burnable poison must be included in the nuclear design to ensure that the shutdown requirements can be met throughout the core life.
10. The backup shutdown system must be capable of making the reactor subcritical at a temperature of 20°C. It must be capable of inserting at least 600 ppm natural boron between a minimum rate of 6 ppm/min and a maximum rate of 25 ppm/min.

Based on our review, we conclude that the nuclear design bases for the GESSAR reactor are acceptable, since they are in conformance with GDC 10, 11, 12, 26, and 27.

4.3.1 Power Distribution

The reactor power distribution is a function of both the nuclear design and the reactor operating state. Constraints are placed on the power distribution which preclude exceeding fuel design safety limits either during normal operation or during transients. Target peaking factors for the GESSAR reactor are given in Table 4.3.1

A study of power distributions in boiling water reactors is given in Appendix 4A of GESSAR. Appendix 4A indicates that the GE design methods are capable of adequately representing operating reactor states. The design methods are compared with measured data for both gross and local power distributions. The effect on power distributions of rod patterns, fuel burnup, flow variations, void distribution, xenon, hot and cold reactor conditions, and load following are discussed. The errors and uncertainties associated with the analytical methods are also discussed and have been accounted for in the evaluation of fuel performance with the process computer.

The discussions on power distribution in GESSAR Section 4.3 and in Appendix 4A are acceptable for a PDA.

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TABLE 4.3.1

NUCLEAR DATA SUMMARY

	GESSAR
Design Peaking Factor	
Maximum Fuel Bundle to Average Fuel Bundle	1.40
Axial Peak-to-Average	1.40
Local Peak-to-Average	1.13
Total Peak-to-Average	2.22
Water-to-Fuel Volume Ratio	2.50
Uranium Weight per Bundle (lb)	415
Maximum Core Reactivity, All Rods in (K_{eff})	<0.965
Maximum Core Reactivity, Strongest Rod Out (K_{eff})	<0.99
Reactivity of Movable Control Rods, Cold (ΔK)	0.17
Range of Reactivity Coefficients	
Fuel Doppler Coefficient ($\Delta K/k/°F$)	-1.2 to -1.3×10^{-5}
Moderator Void Coefficient ($\Delta K/k/\%$ void)	-1.0 to -1.6×10^{-3}

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GE is currently addressing staff concerns related to statistical analysis of reactor data in establishing and accounting for uncertainties in GE Topical Report NEDO-20340. We conclude that the resolution of our concerns regarding GESSAR Appendix 4A can be accomplished as part of our review of that topical report post-PDA.

GESSAR incorporates monitoring systems which will permit the operating conditions of the reactor core to be monitored periodically to ensure compliance with design safety limits.

The in-core neutron monitoring system is composed of the Source Range Monitoring (SRM) subsystem, the Intermediate Range Monitoring (IRM) subsystem, the Local Power Range Monitoring (LPRM) subsystem, the Average Power Range Monitoring (APRM) subsystem, and the Traversing In-core Probe (TIP) subsystem. The SRM range varies from the source range to about 10^{-3}% of full power. The IRM's cover from 10^{-4} to 20% of full power. The LPRM range varies from a few percent to 150% of full power. The APRM's provide a continuous indication of average reactor power from a few percent to 150% of rated reactor power. The APRM subsystem is based on a subset of the LPRM detectors. The TIP subsystem is used to calibrate the LPRMs and to provide detailed axial flux distributions.

We conclude that the information presented concerning the monitoring of power distributions is acceptable.

4.3.2 Reactivity Coefficients

A discussion on the reactivity coefficients of the GESSAR reactor is presented in GESSAR Section 4.3.2.3. The most important reactivity coefficients which determine the stability and dynamic behavior of the GESSAR reactor are the Doppler reactivity coefficient, the moderator void reactivity coefficient, and the moderator temperature reactivity coefficient. The power reactivity coefficient, which is associated with stability to power oscillations due to xenon and other causes, is a function primarily of the Doppler and moderator void reactivity coefficients.

The Doppler reactivity coefficient is a reactivity change associated with the Doppler broadening of absorption resonances of a material and is caused by changes in temperature. The Doppler reactivity coefficient is negative for the GESSAR reactor. The absolute magnitude of the coefficient increases with both increasing moderator temperature and increasing void fraction because the resonance escape probability is inversely proportional to the water to fuel ratio. The Doppler reactivity coefficient also becomes more negative as a function of fuel burnup due to the buildup of plutonium isotopes. Values of the Doppler reactivity coefficient are given in Table 4.3.1 of this report. In various transient analyses, the Doppler reactivity coefficient is taken to be -0.126¢/$^{\circ}$F and is multiplied by a design conservatism factor of 0.9.

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The GESSAR reactor has a large negative moderator void coefficient of reactivity and a moderator temperature coefficient of reactivity which is much smaller in magnitude. These coefficients are obtained from partial derivatives of the infinite multiplication factor and neutron leakage as a function of control fraction* with respect to the variables of temperature or void content with the reactor near critical. Of the two, the moderator temperature coefficient is less significant and plays a role only near the inlet region of a hot operating reactor where the void content is smallest. This coefficient may become slightly positive near the end of the fuel cycle. The strong moderator void coefficient of reactivity, on the other hand, gives the GESSAR reactor a number of important characteristics such as (1) the capability of using coolant flow control for load following, (2) inherent ability to self-flatten the radial power distribution, and (3) stability to xenon induced spatial power oscaillations. Values of the void reactivity coefficient are given in Table 4.3.1 of this report. In various transient analyses, the moderator void coefficient is taken to be -11.5¢/% $\Delta v/v$ and multiplied by a design conservatism factor ranging in value from 0.9 to 1.25.

We have reviewed this information and conclude that the discussion in GESSAR of the reactivity coefficients is acceptable. We find that the important prompt (Doppler) and void reactivity coefficients are negative throughout the fuel cycle. We further conclude that the absolute magnitudes of these coefficients are sufficiently large to ensure the stability of the GESSAR reactor during power operation.

4.3.3 Control Requirements and Control

We have reviewed and evaluated the information presented for the GESSAR Section 4.3.2.5 on control requirements and control. Due to the excess reactivity in the initial core, the GESSAR reactor is capable of energy extraction of 12,000 to 19,000 MWd/T averaged over the initial core loading, depending on the initial uranium enrichment. The excess reactivity is needed to compensate for reactivity losses due to moderator heating and boiling, fuel temperature increases, equilibrium and peak xenon, samarium poisoning, fuel depletion, and other low cross section fission product poisons.

The control rods provide a number of important operating functions. They are a means for (1) rapidly decreasing the core reactivity during a reactor trip by being driven into the core, (2) bringing the reactor into the power operating range from either cold or hot shutdown conditions by planned rod withdrawal, (3) compensating for fuel depletion by planned rod withdrawal, and (4) shaping the power distribution by selective movement. The control rods are capable of shutting down the reactor ($K_{eff} < 1.0$) throughout the entire first fuel cycle for the most limiting condition, that is, for the reactor at 20°C and for the highest worth control rod stuck out. The uncertainty associated with the calculation of the shutdown margin was estimated by GE to be about 0.005 ΔK.

*The control fraction is defined as the ratio of the length of control rods inserted into the reactor to the total inserted length of all of the control rods.

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Control rod withdrawal sequences are selected prior to operation in order to optimize core performance and to achieve low individual rod worths. The maximum controlled rate of reactivity addition during startup is 0.0011 ΔK/sec. This value is based on the withdrawal of an in-sequence rod assuming a total rod worth of 0.0100 ΔK, a peak incremental rod worth of 0.00033 ΔK/in, and a maximum rod speed of 3.6 in/sec. Reactivity addition rates are considerably reduced at hot operating conditions from those under startup conditions due to the effects of void formation and redistribution as a rod is withdrawn.

The control of the reactor is not only dependent upon the movement of control rods but also upon changes which occur in various system parameters. Because the pressure changes caused by a turbine throttle operation bring about reactor power changes in a direction opposite to changes in reactor pressure, the reactor is operated as a constant pressure device. The plant output is increased or decreased by changing the reactor circulating water flow and/or moving the control rods. As indicated previously, reactor startup from cold or hot conditions is accomplished by withdrawing control rods and keeping the recirculating water flow at a fixed value. The reactivity differences between the hot standby condition (5% power, 30% flow), as defined by GE, and the cold critical condition are 0.069 ΔK and 0.041 ΔK for beginning and end of cycle, respectively. These reactivity differences include the effects of temperature, void fraction, and xenon changes. By adjusting the recirculating water flow, the reactor power can be varied over approximately 35% of the power range. The power change produced by varying the recirculating water flow is nearly uniform and is based on curves developed during the reactor startup phase which correlate reactor power and flow for various control rod patterns. Control rod changes may also be made in the power range in conjunction with changes in the recirculating water flow; however, load following is usually accomplished by varying recirculating water flow. Spatial power disturbances, such as those caused by xenon redistribution, present no special control problem to the GESSAR reactor. The large negative power coefficient provides strong inherent damping of such disturbances or oscillations.

The GESSAR BWR/6 incorporates a standby liquid control system to satisfy the requirements of GDC 26. This system is capable of injecting a natural boron solution at the rate of 6 to 25 ppm/min and can bring the system coolant to a concentration of at least 600 ppm. Based on the reactivity worth of the boron, this liquid control system, independently of any control rod action, is capable of shutting down the reactor to 20°C from full power throughout the fuel cycle.

We conclude that the discussion in GESSAR of the control requirements and control is acceptable. We find that there is sufficient shutdown margin throughout the fuel cycle. We conclude that spatial power disturbances will be strongly damped by the large negative power coefficient and that power changes by control rod movement and/or changes in recirculating water flow can be made in an acceptable manner with respect to effects on the power distribution. We further conclude that adequate control of the excess reactivity exists throughout the fuel cycle. Finally, we conclude that a second shutdown control system requirement is met by the standby liquid control system.

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4.3.4 Control Rod Patterns and Reactivity Worths

The postulated control rod drop accident discussed in Section 15.3.2 of this SER assumes that a bottom entry control rod has been fully inserted and becomes stuck in this position, unknown to the reactor operator. The drive is then assumed to become uncoupled and fully withdrawn. The rod subsequently falls from the core inserting an amount of reactivity corresponding to its reactivity worth.

To preclude damage from such an event, the GESSAR BWR-6 reactor is equipped with a Rod Pattern Control System (RPCS). During the startup power range, the RPCS restricts operator movement of control rods to a preselected program of rod patterns. Using these preplanned rod patterns restricts the incremental worth of any control rod to approximately 0.01 ΔK or below. With this restriction on the maximum incremental control rod worth, the fuel design safety limit (a peak fuel enthalpy limit of 280 cal/gm) will not be exceeded even if the rod drop velocity were to reach its maximum value of 2.79 ft/sec.

The RPCS controls the withdrawal of individual control rods until the reactor reaches the state of 50% rod density (rods withdrawn by 50%). From 50% rod density to 25% power, the RPCS controls the movement of individual control rod gangs (or groups). In this power range the function of the RPCS is the same as before - to limit the worth of individual control rods (or gangs in this case) to preclude undesirable effects resulting from a control rod drop accident.

The RPCS is not required above 25% power since the rods are then withdrawn to such an extent that the postulated drop of any one rod or rod gang would not add enough reactivity to the core to result in fuel damage from a reactor excursion. The RPCS is further discussed in Section 7.7.1 of this SER.

In the startup range the maximum in-sequence and out-of-sequence control rod worths are computed by means of full core, three group, two-dimensional XY diffusion calculations. Homogenized cross sections are used for each fuel bundle. These cross sections are generated by using the GE standard lattice design methods for the controlled or uncontrolled fuel bundle. The effects of the axially distributed gadolinia are included in the XY diffusion calculations by using average cross sections and axial bucklings obtained from one-dimensional, three group, axial diffusion calculation.

In the power range, the control rod calculations are affected by the formation of steam voids in the moderator. The maximum control rod worth is calculated by means of three-dimensional XYZ diffusion theory for a control rod fully inserted or fully withdrawn for a constant void distribution. The initial void distribution is obtained from a three-dimensional coupled nuclear-thermal hydraulic calculation with the maximum worth out-of-sequence control rod fully inserted.

We conclude that the information presented on control rod patterns and reactivity worths is acceptable. Although the control rod patterns and withdrawal schemes are quite complex, we find that the Rod Pattern Control System and the nuclear instrumentation can limit the worth of a control rod and the power peaking factor. Finally, we conclude

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that the restrictions on the rod patterns will limit the control rod worth to approximately 0.01 ΔK and that no dropped rod would produce a peak enthalpy of 280 cal/g even if the rod were dropped at 2.79 ft/sec.

4.3.5 Criticality of Fuel Assemblies in Storage

We have reviewed and evaluated the information presented for the GESSAR BWR-6 on the criticality of fuel assemblies. The criticality analyses are performed assuming a higher-than-normal average fuel enrichment and also assuming that there are no control rods or gadolinia. For the dry condition, the multiplication factor, K_{eff} is <0.50. In the fuel handling facilities two fuel bundles give $K_{eff} \sim 0.74$, four bundles give $K_{eff} \sim 0.90$. Sixteen to twenty fuel bundles represent a critical array. During fuel handling and storage, procedural controls will be established to preclude the possibility of personnel arranging four fuel bundles in a square array outside the confine of the fuel racks. See Section 9.1 of this report for further discussion of fuel criticality.

We conclude that the discussion of criticality of fuel assemblies as presented in GESSAR is acceptable. We find that the procedural controls outlined are sufficient to prevent K_{eff} from exceeding 0.90 under normal conditions of fuel handling and storage and 0.95 for abnormal conditions.

4.3.6 Vessel Irradiation

We have reviewed and evaluated the information presented for the GESSAR BWR-6 on vessel irradiation. A one-dimensional, discrete ordinates transport code is used to calculate the neutron fluence at the pressure vessel assuming continuous reactor condition at rated power for 40 years. A radial power distribution representative of conditions throughout the life of the plant was used. Axial power distributions were calculated. The calculated fluence at the pressure vessel for neutrons of energies above one MeV is about 2.4×10^{18} neutrons/cm^2.

Based on our review of the methods employed, we conclude that the neutron fluence at the pressure vessel wall has been conservatively estimated.

4.3.7 Analytical Methods

We have reviewed and evaluated the information presented for the GESSAR BWR-6 on the analytical methods. The basic calculational procedures used by GE for generating neutron cross sections are part of its so-called Lattice Physics Model. In this model the many-group fast and resonance energy cross sections are computed by a GAM-type of program. The fast energies are treated by multigroup integral collision probabilities to account for geometrical effects in fast fission. Resonance energy cross sections are calculated by using the intermediate resonance approximation with energy-and-position-dependent Dancoff factors included. The thermal cross sections are computed by a THERMOS-type of program. This program accounts for the spatially varying thermal spectrum throughout a fuel bundle. These calculations are performed for an extensive combinacion of parameters including fuel enrichment and distribution, fuel and moderator temperatures, burnup, voids, void history, the presence or absence of adjacent control rods, and gadolinia concentration and distribution in the fuel rods. As part of the

Lattice Physics Model, three-group, two-dimensional XY diffusion calculations for one or four fuel bundles are performed. In this way, local fuel rod powers can be calculated, as well as single-bundle or four-bundle (with or without a control rod present) average cross sections.

The single or four-bundle averaged neutron cross sections which are obtained from the Lattice Physics Model are used in either two- or three-dimensional diffusion calculations. Two-dimensional, XY calculations are usually performed in three-groups at a given axial location to obtain gross power distributions, reactivities, and average three-group neutron cross sections for use in one-dimensional axial calculations. The three-dimensional diffusion calculations use 1.5 energy groups and can couple neutron and thermal hydraulic phenomena. These three-dimensional calculations are performed using 24 axial nodes and 1 radial node per fuel bundle resulting in about 14,000 to 20,000 spatial nodes; however, at the design stage geometrical symmetry is used to reduce the size of the calculation. This three-dimensional calculation provides the best simulation of the GESSAR BWR-6 and yields gross three-dimensional power distributions, void distributions, control rod positions, reactivities, eigenvalues, and also average cross sections for use in the one-dimensional axial calculations.

The one-dimensional axial calculations are space-time diffusion calculations which are coupled to a single channel thermal-hydraulic model. This axial calculation is used to generate the scram reactivity function for various core operating states. This one-dimensional space-time code has been compared by GE with results obtained using the industry standard code, WIGLE.

The Doppler, moderator void, and moderator temperature reactivity coefficients are generated in a rudimentary manner from data obtained from the Lattice Physics Model. The effective delayed neutron fraction and the prompt mode neutron lifetime are computed using the one-dimensional space-time code. The power coefficient is obtained by appropriately combining the void, Doppler and moderator temperature reactivity coefficients.

The behavior of the GESSAR BWR-6 to any induced power oscillations is discussed in GE Topical Report APED-5652. The effect of spatially varying xenon concentrations on the stability of the GESSAR BWR-6 is specifically discussed in GE Topical Report APED-5640. These studies show that the GESSAR BWR-6 is stable to any xenon-induced power oscillations because of the damping effect of the large, negative, spatially varying void coefficient.

Section 4.3 of GESSAR does not provide any comparisons of calculations of K_{eff} with measured data for hot and cold conditions and with and without equilibrium xenon and samarium present. Comparison with experimental data of calculated control rod worths in the cold condition, shutdown margins for various conditions, the reactivity worths of the distributed gadolinia, and reactivity coefficients for various conditions is similarly lacking.

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We conclude that the discussion of the analytical methods in GESSAR indicates that they are state-of-the-art. However, the analytical methods need to be more fully described and documented in terms of the equations, numerical techniques, and methods of solution. The neutron cross section data base needs to be fully described and documented. Furthermore, the analytical methods need more experimental verification and documentation over as wide a range of boiling water reactor parameters and operating states as possible and must include a discussion and evaluation of the uncertainties involved.

In GESSAR Amendment 23, GE proposed to the staff a schedule for the submittal of topical reports addressing lattice physics methods, the BWR simulation code, the verification of lattice physics methods and the verification of the core calculation methods. We conclude that resolution of our concerns related to the core physics analytical methods can be accomplished as a part of our review of these topical reports. Therefore, we consider the information contained in GESSAR Section 4.3 and the GE commitments to provide additional information in this area by submitting topical reports acceptable for the PDA stage.

Final resolution of this item can be postponed until after PDA issuance.

4.3.8 Summary of Evaluation of Nuclear Design

Our review has established that sufficient information has been presented in GESSAR to conclude that the nuclear design and operational boundaries are understood and that the reactor can be expected to meet required limitations over the appropriate range of operation. In particular, we conclude that sufficient information has been presented on such reactor characteristics as power distributions, reactivity coefficients, and control for use in steady state limits and in the transient and accident analyses described in Chapter 15.

4.4 Thermal and Hydraulic Design

The core thermal and hydraulic design bases are formulated to limit the local power density and coolant flow within the core to values such that the fuel damage limits are not exceeded during normal operation or operational transients. One damage limit is the critical heat flux. Previously the critical heat flux limits were determined by the Hench-Levy correlation. Now, the Regulatory staff has approved for licensing a new GE proposed critical heat flux correlation called GEXL (General Electric Critical Quality Xc-Boiling Length). The occurrence of boiling transition is predicted with greater accuracy by use of the GEXL correlation than by that of the Hench-Levy critical heat flux correlation. The GEXL correlation was formulated on the basis of best fit of data rather than a lower limit line to the then existing rod bundle critical heat flux data of the Hench-Levy correlation. The GEXL correlation is based on much more extensive data and a wider range of axial heat flux profiles than the Hench-Levy correlation.

The Regulatory staff has also approved for license a new GE proposed thermal design method used in meeting the design objective of preventing fuel cladding damage due to overheating. This method is called GETAB (General Electric Thermal Analysis Basis). The results of our review on GETAB are given in Appendix E of the GESSAR SER.

The GETAB method uses the uncertainties of the reactor operating variables and the GEXL correlation together with the most limiting reactor power distribution to determine the thermal limit. The uncertainties associated with the parameters affecting steady state bundle power are treated statistically in order to satisfy the criterion that, during a core wide transient, 99.9% of the rods in the core are not expected to experience boiling transition. The thermal limit for the 8 x 8 fuel used in the BWR-6 design is stated in terms of the critical power ratio (CPR), where the critical power ratio chosen for the reactor safety design and operation is defined as the ratio of the critical bundle power to the operating bundle power. (Transition boiling begins at the critical power.) This CPR term is more representative of the available thermal margin than the previously used CHFR. The GESSAR reactor thermal margin is stated in terms of the minimum value of the critical power ratio, MCPR, corresponding to the most limiting fuel assembly in the core.

The fuel cladding integrity Safety Limit MCPR established by the GETAB method for the GESSAR reactor is 1.07. This Safety Limit MCPR has been determined to meet the criterion that more than 99.9% of the rods would not be expected to experience boiling transition.

The Operating Limit MCPR is 1.21 which is the limiting condition for steady state power operation at rated power and flow. The Operating Limit MCPR is based on the most severe anticipated abnormal operating transient yielding maximum decrease in MCPR. The GESSAR accident analysis shows that the load rejection without bypass transient results in the maximum reduction of the MCPR. The Δ MCPR is 0.14. Addition of this Δ MCPR to the Safety Limit MCPR of 1.07 gives the minimum Operating Limit MCPR of 1.21.

In conjunction to the limiting conditions of operation by use of the MCPR thermal margin, the maximum linear heat generation rate (LHGR) of the fuel should not be expected to exceed 13.4 kw/ft during normal power operation. For a short-term transient operation the LHGR is expected to rise to 15.6 kw/ft. Center line melting begins at 21 to 22 kw/ft.

The scope of our thermal-hydraulic design review included the design criteria and thermal-hydraulic performance. The applicant's thermal-hydraulic analyses were performed using analytical methods and correlations that have been previously reviewed by the staff and found acceptable. Differences between the proposed core design (and criteria) and those designs and criteria that have been previously reviewed and found acceptable by the staff were reviewed. We found that all such differences were satisfactorily justified by the applicant.

The staff concludes that the GETAB thermal design methods and the thermal-hydraulic design of the core will conform to the Commission's regulations and to applicable Regulatory Guides and are considered acceptable for the PDA.

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In Section 4.4.3.5 of GESSAR, GE presents typical values of stability and hydrodynamic performance and references calculations that predate introduction of the BWR-6 design. We will require GE to update the stability analysis prior to the submittal of the first BWR-6 FSAR.

5.0 REACTOR COOLANT SYSTEM

5.1 Summary Description

The principal components of the reactor coolant system are the reactor pressure vessel, the reactor recirculation system, the main steam and feedwater lines, and the pressure relief system. These items form the major components of the reactor coolant pressure boundary (RCPB). The pressure boundary also contains portions of the Reactor Core Isolation Cooling System, the Residual Heat Removal System and the Reactor Water Cleanup System. Portions of these systems as well as other piping that extend from the reactor vessel out to the second outermost isolation valve are considered within the reactor coolant pressure boundary.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary Components

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

5.2.1.1 Compliance with 10 CFR Part 50, Section 50.55a

Components of the reactor coolant pressure boundary, as defined in 10 CFR Part 50, Section 50.55a, have been properly identified and classified as ASME Section III, Code Class 1 components. These components within the reactor coolant pressure boundary will be constructed in accordance with the requirements of the applicable codes and addenda as specified by 10 CFR Part 50, Section 50.55a, Codes and Standards, and discussed previously in Section 3.2.2 of this SER.

The staff concludes that construction of the components of the reactor coolant pressure boundary, in conformance with the Commission's regulations, provides reasonable assurance that the resulting quality standards are commensurate with the importance of the safety function of the reactor coolant pressure boundary and is considered acceptable.

5.2.1.2 Applicable Code Cases

The specified ASME Code Cases, whose requirements will be applied in the construction of pressure-retaining ASME Section III, Code Class 1, components within the reactor coolant pressure boundary (Quality Group Classification A), are acceptable to the staff. We conclude that compliance with the requirements of these Code Cases in conformance with the Commission's regulations is expected to result in a component quality level commensurate with the importance of the safety function of the reactor coolant pressure boundary and is considered acceptable.

5.2.1.3 Design of Active Pumps and Valves

GE has described, in GESSAR Section 5.2, the design requirements which apply to active pumps and valves in the GESSAR design. We have reviewed this information and find it acceptable. Section 3.9.4 of GESSAR also provides a program which will be used to assure the operability of all active components. Our conclusions regarding this program are given in Section 3.9.3 of this SER.

5.2.1.4 Loose Parts Monitor

Occasionally, miscellaneous items such as nuts, bolts, and other small items have become loose parts within reactor coolant systems. In addition to causing operational inconvenience, such loose parts can damage other components within the system or be an indication of undue wear or vibration. For the past few years we have required many applicants to initiate a program, or to participate in an ongoing program, the objective of which was the development of a functional, loose parts monitoring system within a reasonable period of time. General Electric Company has stated that they have initiated a long-term program for the purpose of development of a vibration monitoring system for light water reactors. The objective of the program is the development of a system requiring sensors on only the outside surface of the reactor pressure vessel to provide continual monitoring for the impact and vibration of loose parts during reactor operation. We have concluded that this is acceptable for this stage of the GESSAR review.

5.2.2 Overpressure Protection

The pressure relief system prevents overpressurization of the reactor coolant boundary under the most severe operational transients and limits the reactor pressure during normal plant isolation and load rejections so as to prevent opening of the spring safety valves. The valves of the pressure relief system also are part of the Automatic Depressurization System, which is a subsystem of the emergency core cooling system described in Section 6.3.

The pressure relief system consists of 19 dual purpose safety/relief valves. All are mounted on the main steam lines within the primary containment drywell between the reactor vessel and the isolation valves. All discharge through piping directly to the suppression pool. The valves are all spring-loaded with the set pressures in the range from 1165 to 1205 psig. At the set pressure of the highest set valve, the valves have a combined capacity equal to 112% of rated steam flow. The valves are also actuated at

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relie, g set pressures within the range of 1105 to 1145 psig. These valves contain auxiliary pnuematic actuators and can be operated either by automatic or remote manual controls at any pressure above atmospheric. For overpressure relief, a pressure switch on each valve initiates the pneumatic actuator at the relieving set pressure. Eight of the valves can be pneumatically actuated by a signal from the Automatic Depressurization System (ADS). All of the safety relief valves are equipped with a safety grade pneumatic accumulator and a check valve in the supply line so that the valve can be actuated even if the pneumatic supply fails.

The ability of the pressure relief system to prevent overpressurization of the reactor coolant pressure boundary is evaluated assuming that: (1) the plant is operating at design conditions (105% of rated steam flow and a reactor vessel dome pressure of 1045 psig), (2) the most severe operational transient occurs (closure of the main steam line isolation valves), (3) the direct scram signal from the valve position switches fails and scram is effected by the fastest indirect scram signal (high neutron flux), and (4) at least one valve is inoperative.

The method that we require be used to determine total valve capacity is described below.

(a) Whenever system pressure increases to the relief pressure set point of a group of valves having the same set point, half of those valves are assumed to operate in the relief mode, opened by the pneumatic actuator. The relieving capacity of these valves used in the analysis will be the capacity of the valves at the lower relieving pressure. The capacity at this pressure will be determined in accordance with the procedure specified in NB 7800 of ASME Section III. This will specifically include the 90% of average tested capacity inherent in the code certification procedure.

(b) For valves considered to operate in the relief mode, the overpressure protection analysis will take into account the time delay and the resulting system pressure increase associated with the response characteristics of the pressure sensing system and the valve assembly. The relief rate will be considered to increase in a straight line relationship proportional to the system pressure as the transient progresses, until the system reaches 103% of the <u>spring</u> set pressure for a particular group of valves. At that pressure and at higher pressures, the valves will be assumed to relieve at their ASME stamped capacity.

(c) When the system pressure increases to the valve spring set pressure of a group of valves, those valves not already considered open are assumed to begin opening and to reach full open at 103% of the valve <u>spring</u> set pressure. At 103% of the <u>spring</u> set point pressure and higher, all these valves are assumed to relieve at their ASME stamped capacity.

The peak vessel pressure resulting from an analysis performed essentially as described above is more than 50 psig within the ASME allowable pressure of 1375 psig, and is therefore acceptable. The description, logic, and design criteria for the overpressure protection control and instrumentation system (I&C) needed to satisfy the requirements

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of the ASME code, Section III, Paragraph NB 7630 is currently under review and the staff evaluation will be reported in a supplement to our SER, as part of our Chapter 7 review. However, as stated in Section 7.1 of this report, we conclude that this part of the I&C system is acceptable based on our evaluation of the criteria and conceptual design.

A small fraction of the pressure relief valves at some BWR plants have inadvertently opened during certain transients. An evaluation of these inadvertent openings indicates that the potential exists for the same mechanism to prevent these valves from opening when required. Even though these failures have not resulted in overpressurization or compromised the integrity of the reactor containment system, they do represent a deviation from the anticipated performance of an essential safety system (overpressure relief system) that has safety implications such as excessive vessel cooldown rate, increased probability of fuel coolant loss and potential for a common mechanism causing failure to open. Changes in design, equipment, inspection and testing can be made to improve the safety and safety/relief valves' performance on GESSAR plants. GE has proposed a new valve design. They stated that the safety/relief valves to be used on GESSAR plants will be "...balanced type, spring loaded safety valves provided with an auxiliary power actuated device which allows opening of the valve even when pressure is less than the safety-set pressure of the valve." GE further states that valve problems on operating BWR's were "...associated principally with multiple stage pilot operated safety/relief valves. These newer, power operated safety valves employ significantly fewer moving parts wetted by the steam, and are therefore considered an improvement of the previously used valves."

Design details and drawings of the valves have been provided to the Regulatory staff. In addition, appropriate "bench" test data has been provided that verifies improved performance. GE has agreed to work with the staff and their utility customers to maintain a surveillance program once the new valves become operational on any BWR. The purpose of the program is to accumulate information in a convenient format and in sufficient detail to allow identification of generic problems and be able in addition to determine such things as exactly what the problem is, what caused it, possibly how to alleviate it, and what the effects on the plant are. Information that is to be reported will include all abnormalities ranging from minor wear observed during normal inspection (even if repairs were made) to complete failures of the valve including failure to open or close and inadvertent operation. Details of the problem, the valve type and operating conditions, failure modes and reasons and remedial action should all be reported. Details of the surveillance program will be provided during the FDA review. Utility-applicants referencing GESSAR will be required to participate in this program as was previously discussed in Section 1.1 of this SER.

We conclude that the pressure relief system, in conjunction with the reactor protection system, will provide adequate protection against overpressurization of the primary coolant boundary and unnecessary operation of the valve. The instrumentation and controls for the overpressure protection system will be reviewed along with the I&C system

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mentioned in Chapter 7 of this report and the staff will report on the review in a supplement to this SER.

5.2.3 Fracture Toughness

5.2.3.1 Compliance with Code Requirements

We have reviewed the materials selection, toughness requirements, and extent of materials testing proposed by General Electric to provide assurance that the ferritic materials used for pressure retaining components of the reactor coolant boundary will have adequate toughness under test, normal operation, and transient conditions. The ferritic materials will meet the toughness requirements of the ASME Boiler and Pressure Vessel Code, Section III.

The fracture toughness tests and procedures required by Section III of the ASME Code for the reactor vessel and other ferritic components provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for pressure retaining components of the reactor coolant pressure boundary.

5.2.3.2 Operating Limitations

We have reviewed the operating limitations that will be imposed on the plant and included in the Tech Specs, and conclude that the reactor will be operated in a manner that will minimize the possibility of rapidly propagating failure, in accordance with Appendix G, 10 CFR 50.

The use of Appendix G as a guide in establishing safe operating limitations, using results of the fracture toughness tests performed in accordance with the Code and NRC regulations, will ensure adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and NRC regulations, constitute an acceptable basis for satisfying the requirements of General Design Criterion 31, Appendix A of 10 CFR Part 50.

5.2.3.3 Reactor Vessel Material Surveillance Program

The toughness properties of the reactor vessel beltline material will be monitored throughout its service life with a material surveillance program that will comply with the Appendix H, 10 CFR 50, and is consistent with programs that have been found acceptable for other BWR plants. The program will meet the requirements of ASTM E-185-73. The predicted neutron fluence for this reactor vessel is only 3.3×10^{18} nvt. The program is acceptable with respect to the number of capsules, number and type of specimens and retention of archive material.

The surveillance program constitutes an acceptable basis for monitoring radiation induced changes in the fracture toughness of the reactor vessel material, and will satisfy the requirements of General Design Criterion 31, Appendix A of 10 CFR Part 50. As stated previously in Section 1.1, utility applicants referencing GESSAR will be required to meet commitments made in GESSAR related to tests and inspections.

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5.2.4 General Material Considerations

The materials used for construction of the reactor coolant pressure boundary (RCPB) have been identified by specifications and found to be in conformance with the requirements of Section III of the ASME Code.

The RCPB materials of construction that will be exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment as proven by extensive testing and satisfactory performance. General corrosion on all materials except carbon and low alloy steel will be negligible. For those materials, conservative corrosion allowances have been provided for all exposed surfaces of carbon and low alloy steel in accordance with requirements of the ASME Code, Section III. Further protection against corrosion problems will be provided by control of the chemical environment and composition of the reactor coolant. The thermal insulation used in these areas is compatible with the materials of construction and is in conformance with the recommendations of Regulatory Guide 1.36, "Non-metallic Thermal Insulation for Austenitic Stainless Steel."

5.2.5 Austenitic Stainless Steel

Controls upon the welding of components constructed of austenitic stainless steel will be exercised to minimize sensitization and to prevent the occurrence of microfissures. The applicant has agreed to demonstrate the adequacy of current welding controls by conducting tests to determine the ferrite content of production welds and to evaluate the degree of sensitization in welded type 304 and 316 stainless steel.

Cleaning and cleanliness control are in accordance with the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" and ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants."

Material selection, fabrication practices, cleaning procedures and protection procedures in accordance with these recommendations provide reasonable assurance that the austenitic stainless steel in the reactor coolant pressure boundary will be free from hot cracking (microfissuring) and in a metallurgical condition which minimizes the susceptibility to stress corrosion cracking during service.

In late 1974, small cracks in the reactor coolant pressure boundary (RCPB) were observed at several operating BWR's. The causes of the cracks were reviewed and evaluated by the staff and our conclusions are given in the staff report, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping in Boiling Water Reactor Plants," NUREG 75/067, December 1975.

5.2.6 Inservice Inspection Program

To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be inspected periodically.

The design of the reactor coolant system incorporates provisions for access for inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Code. Methods will be developed to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel. The conduct of periodic inspections and hydrostatic testing of pressure retaining components in the reactor coolant pressure boundary in accordance with the requirements of ASME Section XI Code provides reasonable assurance that evidence of structural degradation or loss of leaktight integrity occurring during service will be detected in time to permit corrective action before the safety functions of a component is compromised. The inservice inspection program for Class 2 and 3 components will fully satisfy the provisions of Regulatory Guide 1.51, "Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components." Compliance with the inservice inspections required by ASME Section XI Code constitutes an acceptable basis for satisfying the requirements of General Design Criterion 32, Appendix A of 10 CFR Part 50.

5.2.7 RCPB Leakage Detection System

Coolant leakage within the primary containment may be an indication of a small through-wall flaw in the reactor coolant pressure boundary. The leakage detection system proposed for leakage to the containment will include diverse leak detection methods, will have sufficient sensitivity to measure small leaks, will identify the leakage source to the extent practical, and will be provided with suitable control room alarms and readouts. The major components of the system are the containment atmosphere particulate, iodine, and radiogas monitors, the drywell floor drain sump system, and the drywell cooler drain system. Indirect indication of gross leakage will be obtained from the containment pressure and temperature indicators. The leakage detection systems proposed to detect leakage from components and piping of the reactor coolant pressure boundary are in accordance with Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" and provide reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions. Compliance with the recommendations of Regulatory Guide 1.45 constitutes an acceptable basis for satisfying the requirements of General Design Criterion 30, Appendix A of 10 CFR Part 50.

5.2.8 Reactor Vessel and Appurtenances

The design, stress analysis, fabrication, inspection, and quality assurance requirements of GESSAR reactor vessels will conform to the rules of the ASME Boiler and Pressure Vessel Code, Section III, and all applicable Code Cases.

The stringent fracture toughness requirements of the ASME Code, Section III, will be met. Also, operating limitations on temperature and pressure will be established for this plant in accordance with Appendix G, "Protection Against

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Non-Ductile Failure," of the ASME Boiler and Pressure Vessel Code, Section III, and Appendix G, 10 CFR 50.

We conclude that GESSAR reactor vessels:

(1) Will be designed, analyzed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and pertinent Code Case listed above.
(2) Will be made from materials of controlled and demonstrated high quality.
(3) Will be inspected and tested to provide substantial assurance that the vessels will not fail because of material or fabrication deficiencies.
(4) Will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation or during most upsets in operation, and that the vessel will not fail under the conditions of any of the postulated accidents.
(5) Will be subjected to monitoring and periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under the service conditions.

5.3 Thermal Hydraulic System Design

5.3.1 Analytical Methods and Data

The analytical methods, thermodynamic data and hydrodynamics data used are similar to those used in the Grand Gulf, LaSalle, Bailly and Zimmer designs and are acceptable to the staff. These are also presented in Section 4.4.

5.3.2 Load Following Characteristics

The load following characteristics of the reactor coolant system provide the capability for one of the principal modes of BWR operation. The design of the BWR includes the ability to follow load demands over a reasonable range without requiring operator action. The power can be controlled over approximately a 35% power range by flow control. Because of the negative void coefficient, load following is accomplished by varying the reactor recirculation flow. To increase power, the recirculation flow rate is increased thus sweeping voids from the moderator and increasing core reactivity. As reactor power increases, more steam is formed and the reactor stabilizes at a new power level with the transient excess reactivity balanced by the new void formation. Conversely, when less power is required the recirculation flow rate is reduced. The resultant formation of more voids in the moderator automatically decreases the reactor power to that commensurate with the new recirculation rate.

The effects of transient events such as loss of full or partial coolant flow, load changes, coolant pump speed changes, and startup of an inactive loop are discussed in Section 15.1 of GESSAR and our evaluation is discussed in Section 15.2 of this SER.

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5.4 Component and Subsystem Design

5.4.1 Reactor Recirculation System

The reactor recirculation system consists of two loops external to the reactor vessel, within the drywell, that provide automatic load following capability over the range of 65 to 100 percent of rated power. The loops provide the piping for the driving flow of water to the 20 reactor vessel jet pumps. Each loop contains one high capacity (constant speed) motor-driven pump, a flow control valve, two motor operated gate valves (for pump maintenance), and a bypass around the discharge gate and flow control valves. In each loop, subcooled water leaves the vessel in a 22-inch suction line and enters the suction of the recirculation pump (which is below the vessel water level).

The water is discharged at a head of 865 feet (at a flow rate of 35,400 gpm). The flow control valve varies the flow rate over a 35% power range normally from 65 to 100 percent power. The water from the recirculation pumps flows to 20 (10 per loop) jet pumps which are located in the reactor vessel and accelerates a portion of the flow in the annulus. Water not accelerated by the jet pumps returns to the recirculation pump through the suction lines. There are various system interlocks on the flow control valves and bypass valves that provide assurance that adequate pump NPSH will be available and protect the pump from bearing or cavitation damage.

During their review of GESSAR, the ACRS listed the potential for missiles resulting from recirculation pump motor overspeed as a generic concern requiring resolution satisfactory to the NRC staff. A decoupling device will be installed in the shaft between the pump and the motor such that in the event of a LOCA, the "turbining" of the pump could not result in destructive motor overspeed that in turn could generate missiles which could cause the loss of any engineered safety feature. The staff has reviewed the information contained in GESSAR on the reactor recirculation system and finds it acceptable.

5.4.2 Main Steam Line Flow Restrictors

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

5.4.3 Main Steam Line Isolation Valves (MSLIV)

Rapid acting isolation valves are located on each steam line on each side of the primary containment. On various signals from the plant protection system these valves close and isolate the reactor coolant from other portions of the plant. At the same time isolation occurs, the same signals from the plant protection system are sent to various backup and emergency systems so that they automatically function as described in Section 6.3.

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The closure time for the MSLIV's is established in GESSAR based on the Chapter 15 analyses of two events. The valve closure time is adjustable and is set to close no quicker than 3 seconds and no slower than 5.5 seconds. The three second closure time is based on the consequences of the closure of a main steam line isolation valve as discussed in Section 15.1.4 of GESSAR. The three second limit prevents overpressurization of the reactor coolant pressure boundary. The 5.5 second limit is based on the steam line break accident outside of containment. The analysis of a sudden, complete steam line break outside the drywell shows the fuel clad is protected if the valve closes in 5.5 seconds or less. We have reviewed the analyses by GE used to establish these values and conclude that they are acceptable and we will make them a part of the technical specifications for the facility. GE has committed to providing a main steam line isolation valve sealing system to control the release of fission products via valve leakage following a LOCA. Our evaluation of this system is given in Section 9.3.1 of this SER.

5.4.4 Reactor Core Isolation Coolant System (RCIC)

The RCIC system is a backup, high pressure source of reactor coolant that will operate independently of the normal plant a-c power supply. Its operational purpose is to provide an alternate source of reactor coolant to the vessel and to provide sufficient coolant to remove residual heat following a reactor shutdown and loss of feedwater flow without requiring depressurization of the reactor. The RCIC consists of a pump driven by a steam turbine taking steam from one of the main steam lines upstream of the isolation valve and adjacent to the reactor. The pump takes suction from either the condensate tank or the suppression pool and discharges it to the reactor vessel through a head spray nozzle. The system is designed to Class I standards and is capable of being tested while the reactor is in operation. It has also been classified as an Engineered Safety Feature since it functions as an ECC system during certain postulated events such as the control rod drop accident, where it serves as the primary backup system to the HPCS.

The Reactor Core Isolation Cooling (RCIC) system includes the piping, valves, pump, turbine, instrumentation, and controls used to maintain water inventory in the reactor vessel whenever it is isolated from the main feedwater system. The HPCS provides a redundant backup for this function. The scope of review of the RCIC system for the GESSAR plant includes piping and instrumentation diagrams, equipment layout drawings, and functional specifications for essential components.

The drawings, component descriptions, design criteria, and supporting analysis have been reviewed and have been found to conform to Commission Regulations as set forth in the General Design Criteria. The RCIC system has been found to conform to Regulatory Guide 1.29, "Seismic Design Classification." The RCIC system and HPCS system jointly conform to General Design Criteria 2, 4, and 34. The two systems have been found capable of transferring core decay heat following a feedwater isolation and reactor shutdown, from the reactor to the suppression pool, so that the core Minimum Critical Power Ratio does not decrease below 1.07 and the pressure within the reactor coolant pressure boundary does not exceed 110% of design pressure. This capability has been found to be available even with a loss of offsite power and with a single active failure. The staff concludes that the design of the Reactor Core Isolation Cooling System conforms to the Commission's regulations and to applicable Regulatory Guides and is considered acceptable.

5.4.5 Residual Heat Removal System

The Residual Heat Removal (RHR) system is designed for two principal normal modes of operation besides the safety-related modes. For normal usage, the RHR system functions to remove reactor decay and residual heat during either a normal shutdown or following isolation of the reactor. In one safety-related mode of operation, the RHR system (as LPCI) restores and maintains coolant inventory in the reactor vessel after a loss-of-coolant accident. In the other safety-related mode of operation, the RHR System provides heat removal capability in the containment during the post-LOCA period. These safety-related modes of operation are further discussed in Sections 6.2 and 6.3 of this report.

The RHR system consists of two heat exchangers, three main system pumps, and associated valves, piping, controls and instrumentation. All functional components are designed to satisfy seismic Category I design requirements. The main system pumps are sized on the basis of flow required during the LPCI mode of operation which is the mode requiring the maximum flow rate. The heat exchangers are sized on the basis of their heat removal duty following a LOCA.

Two loops, each consisting of one heat exchanger and one RHR pump and auxiliary equipment, are physically separated from each other in the reactor building. A third loop, also consisting of a pump and associated piping, can pump suppression pool water directly into the reactor, if necessary.

During reactor isolation, the RHR system can be operated in the condensing mode to condense reactor steam; hence, the RHR system operates in conjunction with the reactor core isolation cooling system (RCICS). With the reactor isolated, reactor steam normally is directed to and condensed in the suppression pool via the relief valves and the RCIC turbine exhaust piping. However, the suppression pool temperature under these conditions is limited to 125°F in order that the water temperature rise due to a postulated subsequent design basis loss-of-coolant accident would not cause the pool temperature to exceed 170°F during the reactor blowdown. The condensing mode of RHR operation relieves the burden on the suppression pool by transferring a portion of the steam generated by decay heat to the RHR service water. The condensate is either dumped to the suppression pool or returned to the reactor vessel through the suction of the steam-turbine driven RCIC pump. Shortly after shutdown, both heat exchangers are used to handle essentially all of the decay heat. After about 1-1/2 hours, the capacity of one heat exchanger is adequate and the other may be transferred to the suppression pool cooling mode which utilizes the RHR heat exchangers to cool the suppression pool water by transferring heat to the RHR service water. This mode can be used in conjunction with the condensing mode or to provide long term suppression pool cooling following a loss-of-coolant accident blowdown.

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

The system is protected against overpressurization by relief valves and can be automatically isolated to protect the core from low water level in case of a break in the cleanup system. It is also automatically isolated when the Standby Liquid Control System is actuated.

The scope of our review of the RHR System for the GESSAR plant included piping and instrumentation diagrams, equipment layout drawings, failure mode and effects analysis, and performance specifications for essential components. Our review has included the applicant's proposed design criteria and design bases for the RHR and his analysis of the adequacy of those criteria and bases and how well the design conforms to these criteria and bases.

Based on our review, we concluded that the design of the RHR system is not single failure proof in the shutdown cooling mode, and, therefore, did not conform to General Design Criterion 34. This was identified by the ACRS as an area of concern requiring resolution acceptable to the NRC staff prior to the issuance of a PDA. An example of a single failure that could render the RHR system inoperable is a failure-to-open of the isolation valves in the RHR line leading from its associated recirculation loop.

GE proposed alternate methods to achieve this shutdown and satisfying the intent of GDC 34. These heat removal paths were from the main steam lines through the RHR heat exchanger to the suppression pool and through the safety relief valves to the suppression pool. The staff originally found these alternate paths unacceptable since one was prone to the same failure mode as the RHR and the other did not utilize safety grade equipment.

GE has stated that the design will be revised to 1) eliminate single failures which would affect both the RHR and the alternate paths and 2) upgrade equipment in the RHR and the alternate path to safety grade. We will review these changes when submitted. The staff concludes that the RHR system coupled with these alternate methods, if immune to single failures and incorporating safety grade equipment, is acceptable.

Our basis for acceptance is that the GESSAR reactor will conform to GDC's 19 and 34 in that the plant will have seismic Category I systems capable of bringing the plant to cold shutdown within approximately 24 hours, taking credit only for those actions that can be performed from the control room and assuming a single active failure in the systems.

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5.4.6 BWR Reactor Water Cleanup Systems

5.4.6.1 System Description

The reactor water cleanup system (RWCU) is used to maintain the chemical purity of the reactor coolant. The portions of the reactor water cleanup system up to and including the outermost isolation valve are part of the reactor coolant pressure boundary. The applicant's design objectives for the system are to: (1) prevent excessive loss of reactor coolant; (2) prevent the release of radioactive material from the reactor; (3) remove solid and dissolved impurities from the coolant; and (4) discharge excess water during power transients. In addition, the system is designed to minimize temperature gradients, to conserve reactor heat, and for maintainability during reactor operation.

The RWCU system flow rate is 154,000 lb/hr. The RWCU system will consist of two 50% capacity pumps, regenerative and non-regenerative heat exchangers, and two 50% capacity filter-dimineralizers. The demineralized water may be sent to the reactor through the shell side of the regenerative heat exchanger, to the main condenser hotwell, or to the liquid radwaste system.

The RWCU system is isolated from the reactor by two motor-driven isolation valves that close automatically in the event of RCPB leakage as discussed in Section 6.2.1.8 of this SER. The outermost drywell isolation valve will also close automatically in the event liquid poison is injected into the reactor by the Standby Liquid Control System or if the outlet temperature of the non-regenerative heat exchanger exceeds a predetermined level. Reverse flow is prevented by check valves in the return line to the feedwater system and downstream of the RCWU system pumps. Strainers in the outlet from the filter-demineralizers prevent resins from entering the reactor in the event of failure of a resin support. In the event of low flow or loss of flow in the system, flow is maintained in each filter-demineralizer by its own holding pump. Sample points are provided in the inlet to and outlet from each filter-demineralizer to determine demineralizer DF.

The reactor water cleanup system will be used to aid in maintaining the reactor water purity and to reduce the reactor water inventory as required by plant operations. The scope of our review of the reactor water cleanup system included the system's capability to meet the anticipated needs of the plant, the capability of the instrumentation and process controls to ensure operation within limits defined in Regulatory Guide 1.56 and the seismic design and quality group classification relative to Regulatory Guides 1.26 and 1.29. Our review has included single line diagrams and schematic diagrams along with descriptive information concerning the system design and operation and design requirements (industry codes) listed in Section 5.5.8.4 of GESSAR.

The basis for acceptance in our review has been conformance of the applicant's designs and design criteria to the Commission's Regulations and to applicable Regulatory Guides referenced above as well as industry standards.

Based on the foregoing evaluation, we conclude that the proposed reactor water cleanup system is acceptable and capable of performing its safety functions as listed in Section 5.5.8.1 of GESSAR.

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6.0 ENGINEERED SAFETY FEATURES

6.1 General

The purpose of the various engineered safety features (ESF) is to provide a complete and consistent means of assuring that the public will be protected from excessive exposure to radioactive materials, should a major accident occur in the plant. In this section of our report, we discuss the reactor containment system, the emergency cooling systems, and the provisions for maintaining the habitability of the control room after postulated accidents. Discussions of other engineered safety features are provided elsewhere in this report, as related to the particular systems they directly serve. As will be seen, certain of these ESF systems have functions for normal plant operations as well as their safety-related functions.

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

6.2 Containment Systems

The containment system for GESSAR includes a reactor containment structure, containment heat removal systems, a containment isolation system, a combustible gas control system, and a secondary containment system (gas control boundary) which includes the fuel building, auxiliary building, a shield building surrounding the primary containment and its recirculation system as well as the standby gas treatment system. The design of the containment system for GESSAR is similar to the design of the system for the previously reviewed Grand Gulf Nuclear Station which will be the first nuclear station to utilize the Mark III containment design.

The safety issues raised in the course of our review of the proposed containment systems, are basically the same as those issues raised during our review of Grand Gulf and other Mark III designs reviewed to date. During the review of Grand Gulf, the basic analytical approach and the design margins for the Mark III containment were established. The scope of the large and small scale Mark III test programs was also determined and results to date evaluated. The staff has available in the CONTEMPT computer code, the capability for calculating the pressure-temperature response of a Mark III containment. The results of our independent calculations were used to confirm the applicant's analysis. Our review is discussed below.

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6.2.1 Containment Funcational Design

The containment functional design refers to the performance capability of the reactor containment structure following postulated loss-of-coolant accidents. For GESSAR, a Mark III type containment maintains a fission product boundary in the event of a loss-of-coolant accident. Figure 6.2.1 shows the principal features of the Mark III contain ment concept. This design utilizes the effect of water pressure suppression and consists of separate drywell and containment volumes connected through a suppression pool by horizontal vents. This design is similar to that employed for other plants with Mark III containments such as the Grand Gulf and Clinton facilities. A comparison of the GESSAR design parameters with the Clinton facility is presented in Table 6.2-1 of this SER. Our review in this area included the temperature and pressure responses of the drywell and containment to a spectrum of loss-of-coolant accidents; suppression pool dynamic effects during a loss-of-coolant accident or following the actuation of one or more reactor coolant system pressure relief valves; the consequences of a loss-of-coolant accident occurring within the containment but outside the drywell; the capability of the containment to withstand the effects of steam bypass of the suppression pool; and the external pressure capability of the drywell and containment and the systems provided to limit external pressures. The review has considered General Electric's proposed design bases and design criteria for the containment and the analyses and test data in support of the adequacy of the criteria and bases.

The containment system is divided into two major subvolumes, a drywell enclosing the reactor system, and the primary containment surrounding the drywell and containing the suppression pool. The containment and the drywell volumes are connected, through the suppression pool by an array of horizontal vents in the drywell wall. The suppression pool serves as a heat sink in the unlikely event of a loss-of-coolant accident.

The primary containment is a free-standing steel structure consisting of a vertical cylinder, domed top, and a flat base. The net free volume of the primary containment is 1.168×10^6 ft^3 and the design pressure is 15 psig. To satisfy its design basis as a fission product leakage barrier, the primary containment is designed for a leakage rate of 0.3% of the volume per day at 15 psig. This is a recent design change from the originally proposed leakage rate of 1.0% per day.

An additional structure called the shield building, surrounds the primary containment. Its purpose, in conjunction with the fuel building and part of the auxiliary building, is to provide a secondary containment volume in which fission product leakage from the primary containment following a postulated loss-of-coolant accident can be diluted and held up prior to release to the environment. Our evaluation of the shield building design is included in Section 6.2.3 of this report.

Located within the primary containment is a substructure, called the drywell, which encloses the reactor and reactor coolant system. The drywell is an unlined concrete structure, enclosing a net free volume of about 274,500 ft^3 and designed for a differential pressure of 30 psid. The purpose of the drywell is to channel steam released during an unlikely loss-of-coolant accident through the vent matrix system to the

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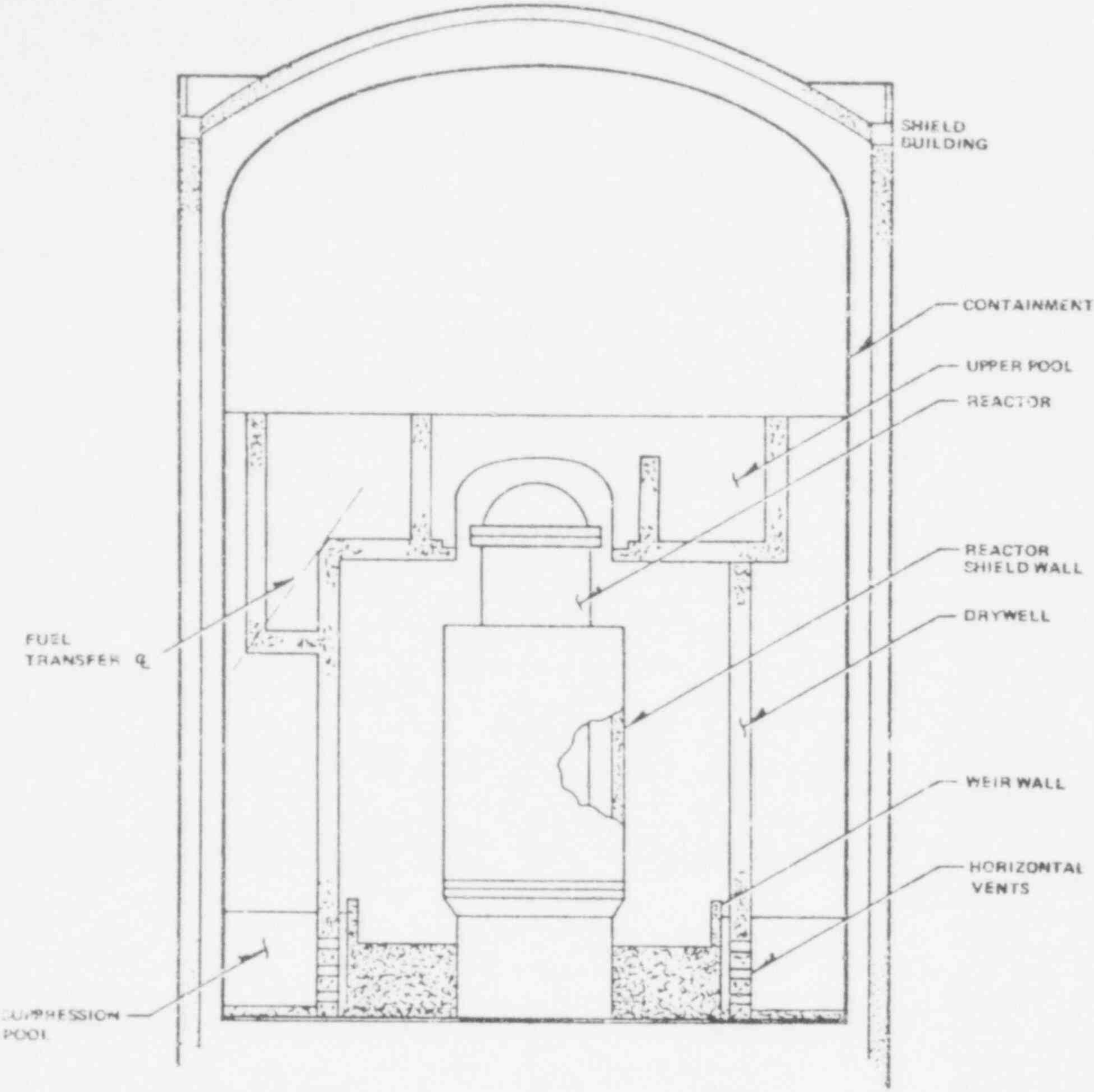

FIGURE 6.2-1 MARK III CONTAINMENT
(Containment and Shield Building)

POOR ORIGINAL

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TABLE 6.2-1

COMPARISON OF BWR CONTAINMENT DESIGNS

DRYWELL	MARK I (Brown's Ferry)	MARK II (Zimmer)	MARK III (GESSAR)	MARK III (Clinton)
type of construction	steel shell	steel-lined reinforced concrete	reinforced concrete	reinforced concrete
air volume	159,000 ft^3	184,000 ft^3	274,500 ft^3	241,500 ft^3
design pressure	56 psig	45 psig	30 psid	30 psid
leak rate	0.5%/day	0.5%/day	NA	NA
WETWELL				
type of construction	steel shell	steel-lined reinforced concrete	steel shell (containment)	steel-lined reinforced concrete
air volume	119,000 ft^3	103,000 ft^3	1.168×10^6 ft^3	1,457,500 ft^3
pool volume	85,000 ft^3	106,000 ft^3	163,700 ft^3 (long term)	180,550 ft^3 (long term)
design pressure	56 psig	48 psig	15 psig	15 psig
leak rate	0.5%/day	0.5%/day	0.3%/day	0.5%/day at 9 psig
thermal power	3293 MWt	2436 MWt	3758 MWt	3039 MWt
break area	4.8 ft^2	2.243	3.94 ft^2	3.23 ft^2
vent area	302 ft^2	274 ft^2	480 ft^2	410 ft^2
break area/vent area	.017	.008	.008	.008

suppression pool for condensation. While not a fission product barrier, the drywell must prevent gross bypass leakage to permit adequate performance of the pressure suppression feature.

For the Mark III design, the containment completely surrounds the drywell, and high energy lines penetrating the drywell must pass through the containment volume. Because the pressure suppression concept relies upon a controlled channeling of steam through the suppression system, the release of large amounts of high energy fluid in the containment must be prevented. Therefore, these lines are designed to low stress levels and high quality standards to preclude rupture inside the containment but outside the drywell. As an additional margin, the applicant has provided guard pipes on certain high energy lines between the drywell and containment to channel the flow from a break in the pipe back to the drywell where it will be condensed by the pool. The guard pipes will be designed to the operating pressure of the enclosed process pipe. Our evaluation of other potential bypass sources and containment bypass capability is discussed in Section 6.2.1.8 of this report.

The suppression pool is a 360-degree annular pool located in the bottom of the containment and retained between the containment wall and the drywell weir wall. The weir wall is a 360-degree, reinforced concrete wall located inside the drywell and 30 inches from the drywell wall. An additional volume of suppression pool water is stored in the upper containment pool, located on top of the drywell, during normal operation. This water is added to the suppression pool following a LOCA by the suppression pool makeup system discussed in Section 6.2.1.4. During normal operation, about 129,550 ft^3 of water is contained in the suppression pool and about 34,150 ft^3 of makeup water is stored in the upper pool. The suppression pool serves as a heat sink for postulated transients and accidents and as the source of cooling water for the emergency core cooling systems. In the case of transients that result in a loss of the main heat sink, energy would be transferred to the pool by the discharge piping from the reactor pressure safety/relief valves. In the event of a loss-of-coolant accident within the drywell, the horizontal vent system in the drywell wall would provide the energy transfer path.

Located in the vertical section of the drywell wall and below the suppression pool water level are 120 horizontal vent holes of 27.5" diameter and arranged in 40 circumferential columns of three vents. In the event of a loss-of-coolant accident, the pressure will rise in the drywell due to the release of reactor coolant, and force the level of water down in the weir annulus. When the water level has been depressed to the level of the first row of vents, the differential pressure will cause air, steam, and entrained water to flow from the drywell into the suppression pool. The steam will be condensed in the pool and the air driven from the drywell will be compressed in the primary containment. The net effect could result in approximately a 4 psi rise in average containment pressure. Peak drywell differential pressure is calculated by the applicant to be 21.8 psi.

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Figure 6.2-2 illustrates the drywell and containment pressure response as a function of time following a design basis loss-of-coolant accident. As also shown in Figure 6.2-2 the short-term containment response is shown in terms of two regions; one representing the volume between the suppression pool and the Hydraulic Control Units (HCU) floors, and the other representing the remainder of the containment volume. Although this response does not appear to significantly affect peak drywell differential pressure, it does result in pressure loads on some containment internal structures. These considerations are discussed more fully in Section 6.2.1.9.

Following the initial phase of the accident, containment and drywell pressure will continue to rise due to the input of core decay and sensible heat to the suppression pool. The long-term pressure rise will be limited to 9.8 psig by operation of the redundant containment heat removal system. Therefore, in the pressure response analysis of this type of containment two limiting conditions must be considered; the short term drywell differential pressure and the long-term containment shell pressure. Our evaluation of the applicant's analytical methods for each of these time periods (i.e., both long and short term) is discussed in Sections 6.2.1.2 and 6.2.1.3 of this report. The General Electric Company has also completed small-scale tests and is performing full-scale tests to support the Mark III short-term analytical model. Our review of these test programs is discussed below.

Both the drywell and containment are divided into a number of subcompartments by internal structures. Our evaluation of the subcompartment designs is discussed in Section 6.2.1.7 of this report.

6.2.1.1 Review of BWR Containment Technology

Two basic pressure suppression designs have preceded the Mark III containment, i.e., the Mark I, or "lightbulb-torus" and the Mark II, or "over-under". A comparison of design parameters for the three containment types is provided in Table 6.2-1. The wetwell and drywell of Mark I and II were connected by a vent system which entered the suppression pool vertically and was at a constant submergence. For both designs, the design basis loss-of-coolant accident for containment response was a recirculation line break. In both Mark I and II containments, the peak drywell pressure occurred at about 10 seconds following the accident, which was after vent clearing, and during the vent flow part of the transient. Wetwell peak pressures occurred in about 10-20 seconds due primarily to the compression of drywell air in the wetwell.

Mark II containments also experienced a short-term drywell deck differential pressure which could occur either at the time of vent clearing or later in the vent flow transient. Generally those plants with relatively large vent areas had vent clearing controlled peak deck differential pressures. In the long term both the drywell and wetwell reached a secondary peak pressure due to continued decay heat generation; however, this transient was less severe than the short term and therefore was not controlling for establishing containment design pressures.

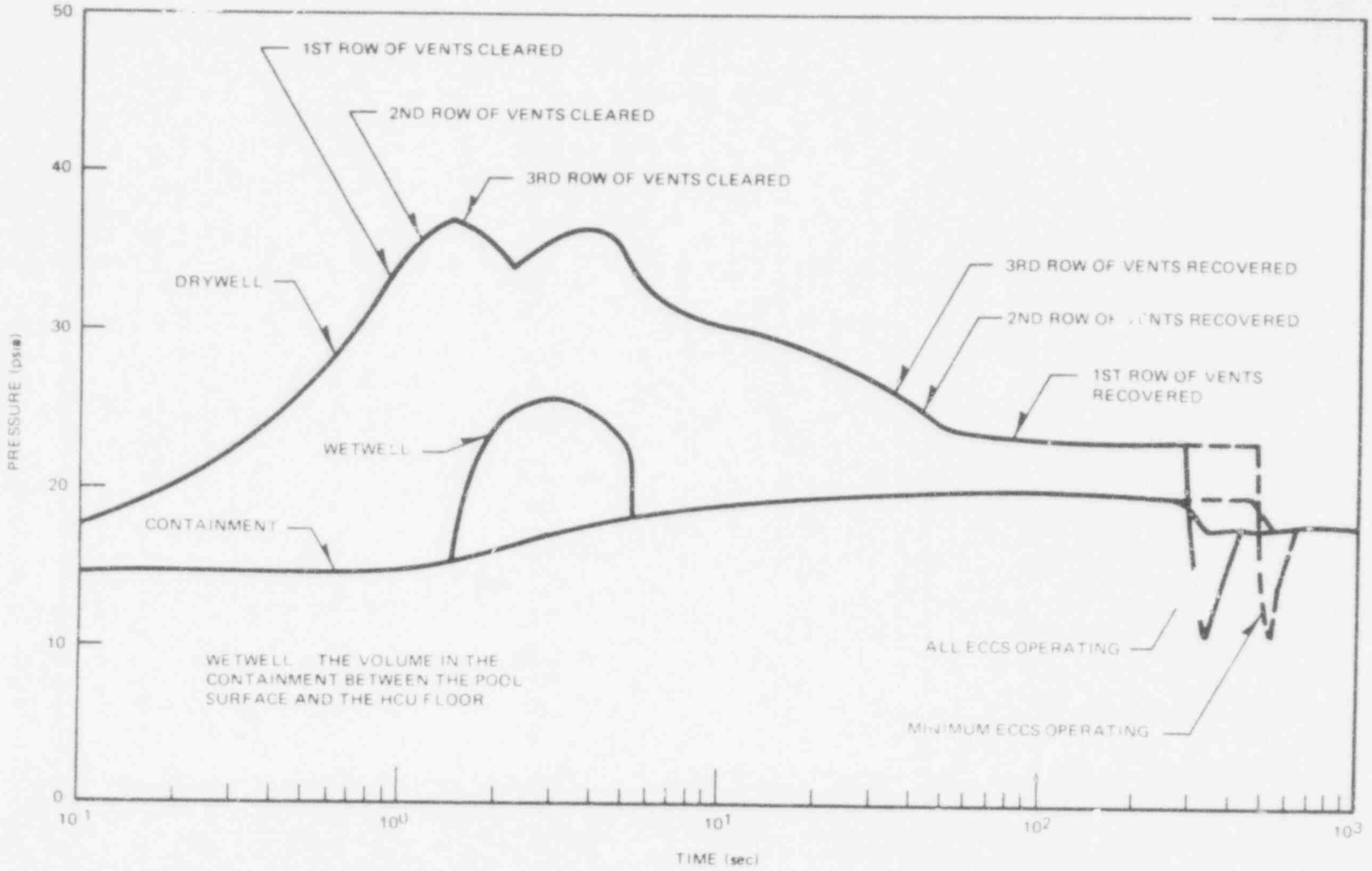

FIGURE 6.2-2 CONTAINMENT PRESSURE ANALYSIS - MAIN STEAM LINE BREAK

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For containment analysis, "The General Electric Pressure Suppression Containment Analytical Model" as described in NEDO-10320 and its supplement was used. This model consists of five separate submodels; blowdown, drywell, wetwell, vent clearing and vent flow. Based on a review of the analytical methods employed in the model, correlation with Humboldt and Bodega Bay test results, and comparison with CONTEMPT-PS results, the staff has previously concluded that the GE model was conservative and therefore acceptable for containment analysis.

The Mark III type containment proposed for GESSAR is different from the Mark I and II types of containments in three basic ways. First, the BWR-6 type reactor system proposed for GESSAR has relatively smaller recirculation lines than those on reactors of earlier design having nearly the same design power levels. The relative size of the main steam lines are approximately the same on both designs. On earlier designs using variable speed recirculation pumps, postulated loss-of-coolant accidents associated with the severance of the recirculation line resulted in higher peak drywell pressures than postulated LOCAs associated with a main steam line break. However, since the size of the recirculation line has now been reduced, LOCAs associated with both a recirculation line break and a steam line break result in nearly equivalent peak drywell pressures. Therefore, both of these postulated pipe breaks must be considered in determining the DBA-LOCA for Mark III containment pressure response.

Second, the vent system connecting the drywell and containment utilizes a circumferential arrangement of horizontal vents at three different elevations which leads to an additional functional dependence on vent clearing phenomena than the Mark I and II types. In addition, because of the relatively large vent areas provided, the peak drywell differential pressure is vent clearing controlled; i.e., the highest differential pressure across the drywell occurs during vent clearing. This places emphasis on the dynamics of vent clearing but reduces the impact of vent flow assumptions on drywell pressure.

Third, as the volume of the containment is about five times that of the drywell, the compression of drywell air into the containment during vent flow results in only a small (about four psi) rise in average containment pressure. This small effect leads to a long-term containment peak pressure which is not specifically related to the size of the reactor coolant break or the short-term pressure response.

As a result of the above, GE has proposed a new analytical model to evaluate the Mark III design. This model is entitled "The General Electric Mark III Pressure Suppression Containment System Analytical Model" and is described in GE topical report NEDO-20533, which the staff has reviewed and found acceptable.

6.2.1.2 Short-Term Pressure Response

As discussed above, both the main steam line break and recirculation line break result in nearly equal peak drywell pressures.

For the postulated double-ended rupture of a 26" main steam line the applicant has assumed a blowdown profile which is separated into an initial one-second period of steam only blowdown followed by two-phase, liquid water and steam blowdown due to liquid level swell in the reactor vessel. During steam blowdown, the mass and energy input rates to the containment were calculated assuming critical flow of an ideal gas. The two-phase blowdown rate was based on the frictionless Moody critical flow model and the average density of fluid inventory within the reactor vessel. The staff has previously reviewed and found acceptable these assumptions for determining blowdown rates.

The time at which the liquid level in the reactor vessel swells to the elevation of the steam line nozzles following the break determines the time at which the model changes from steam to two-phase blowdown assumptions. Peak drywell differential pressure can be sensitive to the level rise time since two-phase blowdown yields a greater rate of steam addition to the drywell than steam only blowdown and also introduces liquid water into the vent flow. Both of these effects increase drywell pressures.

In the GESSAR containment analysis GE has assumed a level rise time of one second and based on this assumption has calculated that the peak drywell differential pressure would be 21.8 psid (Figure 6.2.2). GE has also provided studies of level rise time as a function of operating conditions which indicate that the most rapid level rise would be about one second assuming a hot standby condition.

Following the postulated design basis loss-of-coolant accident, the drywell pressure will rise and accelerate the water in the vent annulus. At about 0.92 seconds, the first row of vents will be cleared of water and a mixture of air, steam, and water will flow into the suppression pool. The water in the vent annulus will continue to accelerate downward resulting in clearing of the second row of vents at about 1.17 seconds and the third row at about 1.52 seconds. The peak drywell differential pressure occurs at 1.40 seconds (main steam line break) and is a result of sufficient vent area being uncovered to reverse the pressure transient in the drywell. Due to this phenomenon the peak pressure is predominantly controlled by the dynamics of vent clearing and only partially influenced by vent flow assumptions.

In the analysis of the vent clearing transient, General Electric used the vent clearing model described in the "The General Electric Mark III Pressure Suppression Containment System Analytical Model," NEDO-20533. In this model the vent system is nodalized into six control volumes representing the vertical weir annulus and horizontal vents. Conservation of mass and momentum is applied to each control volume to determine fluid accelerations and vent clearing times. An effective vent length is used to simulate the effects of suppression pool inertia and turning loss coefficients are applied to account for changes in flow path direction and area. The loss coefficients currently used in the model are derived from generally accepted data and these coefficients are being confirmed during the large-scale Mark III testing program.

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The General Electric vent flow model has also been revised to consider the more complex Mark III horizontal vent geometry; however, the basic thermodynamic flow assumptions used for previous water-pressure suppression designs remain unchanged. For GESSAR the vent flow was computed on the basis of parallel path flow splits which are a function of the number of uncovered vents and geometric loss coefficients. These loss coefficients will also be confirmed experimentally on the large-scale facility.

Based on these analytical models, General Electric has determined that the postulated rupture of a main steam line would result in the highest drywell differential pressure and has calculated this pressure to be 21.8 psid. GE has stated in GESSAR that the drywell will be designed for a pressure of 30 psid which provides a margin of 37% above the peak calculated value. Both the Regulatory staff and our consultants, the Aerojet Nuclear Company, have reviewed the analytical model used for the pressure response calculation. We have also performed our own calculations of the drywell pressure response using the CONTEMPT-LT computer code. Our results confirm GE's analysis. Based on this confirmation, our review of the applicant's analytical model, and our consultants' recommendations, we conclude that the drywell design is adequate.

As shown in Figure 6.2.2 the short term containment response is calculated for two regions; the wetwell, which includes the volume between the suppression pool and the Hydraulic Control Unit (HCU) floor, and the remainder of the containment volume. As indicated on Figure 6.2.2, the two volume analysis does not show any significant effect on peak calculated drywell differential pressure. GE has recently made available the details of their analytical approach. The staff is presently reviewing this information and will report on this aspect of the analysis in a supplement to the SER post-PDA. In the interim, GE has committed to increasing the available flow area at the HCU floor, if required, to maintain an adequate drywell pressure margin. We consider this commitment satisfactory for the PDA.

GE has also provided analyses of the drywell pressure response for a postulated rupture of the recirculation suction line. GE has calculated that the peak drywell differential pressure for this break (19.9 psid) is less than that calculated for the steam line break.

The short-term blowdown rate is a sensitive parameter for a Mark III containment since the drywell pressure peaks very early in the transient. GE has used appropriate assumptions in the recirculation line blowdown model to accurately represent the short-term effect by including the mass inventory of the recirculation line. The model uses a break area equal to the cross section area of the suction line to account for the reactor vessel side of the break and an equivalent break area equal to one half the suction line area that simulates the external recirculation loop contribution. The effective break area for the loop side is derived from an analysis of the pressure-time history within the loop following the break and which calculates a mass flux approximately 50% of the mass flux predicted by the Mo , flow model using the initial fluid conditions within the pipe. Such modeling assumptions remain a valid consideration until the initial mass inventory within the external loop is depleted (∿ 2 seconds). Subsequent mass flow then becomes limited by the effects of critical flow through the jet pump nozzles. In summary, the total blowdown is calculated on the basis of a single effective volume representing the primary system, mass flux based on the Moody correlation and a break area profile which equals 1.5 times the suction line area up to 2.0 seconds and then becomes equal to the suction line plus jet pump nozzle area subsequent to 2.0 seconds.

We have reviewed GE's recirculation line blowdown model, and have also performed independent calculations of the mass flux from a recirculation suction line break using the RELAP-4 computer code. The results of these calculations show total mass release rates substantially lower than those calculated by GE. On this basis we believe that the blowdown rates presented in GESSAR are acceptable for use in the analysis of drywell pressure response to the postulated rupture of a recirculation suction line. Further, since the peak drywell differential pressure calculated by GE for this break is less than that for the main steam line we conclude that the latter is the most limiting pipe rupture for the drywell.

6.2.1.3 Long Term Pressure Response

Following the short-term blowdown phase of the accident, suppression pool temperature and containment pressure will increase due to the continued input of decay and sensible heat into the containment. Referring to Figure 6.2.2, at about 100 seconds after the accident, the drywell pressure has stabilized to approximately 3 psi above the containment pressure. This differential pressure corresponds to the submergence of the first row of vents. At some later time the drywell and containment pressures will equalize due to the return of air from the containment.

During this time period the ECCS pumps, taking suction from the suppression pool, have reflooded the reactor pressure vessel up to the level of the main steam line nozzles. Subsequently ECCS water will overflow out the break and fill the drywell up to the top of the weir wall, establishing a recirculation flow path for the ECCS coolant. Also during this time the Suppression Pool Makeup System has added water to increase the suppression pool inventory in the long term (see Section 6.2.1.4).

At about 30 minutes following the accident, the containment cooling mode of the Residual Heat Removal (RHR) System is activated and suppression pool water is circulated through the RHR heat exchangers, establishing an energy transfer path to the service water system and ultimate heat sink.

In the long-term analysis, GE has conservatively accounted for potential post-accident energy sources. These include decay heat, sensible heat, ECCS pump heat, and metal water reaction energy. GE has assumed that the only heat sink available in the containment is the suppression pool and the only mechanism for heat rejection is the RHR heat exchangers.

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The long-term model also assumed that the containment atmosphere is saturated and equal to the suppression pool temperature at any time. Therefore, the containment pressure is equal to the partial pressure of air and the saturation pressure of water corresponding to the pool temperature.

Based on the above assumptions GE has calculated the peak containment pressure to occur about 3 hours after a LOCA and to be 9.8 psig. Peak containment pressure occurs when the RHR system heat removal rate matches the containment heat generation rate. The design pressure of the containment is 15 psig which allows a 53% margin above the peak calculated value. On the basis of our review of the applicant's analysis and the pressure margin, we conclude that the containment design pressure for this plant is adequate.

6.2.1.4 Suppression Pool Makeup System (SPMS)

Included as part of the containment design is a SPMS. The Suppression Pool Makeup System provides water from the upper containment pool to the suppression pool following a loss-of-coolant accident. This increase in long-term suppression pool inventory provides additional pool heat capacitance, a minimum long-term drywell vent coverage of two feet, and accounts for any post-accident entrapment of water in the drywell and reactor vessel.

The origin of such a system for the Mark III containment is related to the pool dynamic forces imposed on containment structures following a loss-of-coolant accident. Test results have indicated that the extent and magnitude of such forces are proportional to the submergence of the horizontal vents. Therefore, decreasing the vent submergence, by a reduction in suppression pool water volume, results in less severe structural design requirements on containment internal structures.

This reduction in vent submergence has been confirmed not to affect the pressure suppression capability of the pool in the large scale testing of the Mark III by GE.

Two 24" lines connect the upper pool, located on top of the drywell, to the suppression pool. Each line contains two normally closed valves in series which open on a low-low suppression pool level signal in coincidence with a LOCA signal permissive. The low-low pool level signal will be set at eighteen inches below the low water level of the suppression pool during normal operation. Following a LOCA, dumping of the upper pool would start two to three minutes following the beginning of ECCS flow and would require about five minutes for completion. The makeup system dump lines are sized so that flow from one line exceeds the maximum ECCS pump flow.

The SPMS dump valves will also be signaled to open by a LOCA signal in series with a 30-minute timer where the timer is started by the LOCA signal. This initiation logic is independent of suppression pool level and is directed at ensuring that the combined upper and lower pool volumes are available for small primary system breaks which do not lower the suppression pool to the trip level.

The SPMS is designed to seismic Category I and Quality Group B in accordance with Regulatory Guide 1.29 and 1.26 respectively, and consists of redundant dump lines and valves. Instrumentation and control aspects of the SPMS are discussed in Section 7.3.7 of this SER.

6.2.1.5 External Pressure Design

The drywell structure is designed for an external pressure of 21 psid. A drywell vacuum breaker system is provided to control suppression pool water level in the weir annulus and prevent inadvertent flooding of the drywell. The system is not required to operate to maintain the structural integrity of the drywell. Two drywell vacuum relief assemblies are provided, each consisting of a 10-inch check valve in series with an 18-inch automatic valve.

The containment vessel is designed for an external pressure of 0.8 psid. A containment vacuum relief system, consisting of four, 36" lines, is provided to maintain external pressures within design limits. Each vacuum relief line connects the containment to shield building annulus and contains one check valve in series with a motor operated globe valve.

We have reviewed the drywell design external pressure and find that it is acceptable since it represents an upper limit on possible external pressures by assuming complete depressurization of the drywell to 0 psia. In addition, we find the drywell vacuum relief system to be acceptable since the valve arrangement for the penetrations reduces the potential for open lines and, due to the limited vacuum relief line size, an open line would still be within the bypass capability of the containment.

We have reviewed the applicant's sizing analysis for the containment vacuum breakers and we find that certain assumptions used may not be sufficiently conservative. We are currently pursuing additional analytical studies with GE in order to determine an appropriate basis for sizing of the containment vacuum breakers. This item is generic to all plants with Mark III containments utilizing the free standing steel shell type of construction. Acceptable resolution of this item can be postponed until the post-PDA period since the effect of the analyses with revised assumptions, if any, on the size of the vacuum breakers could easily be incorporated into the design. We will report resolution of these items in a supplement to the Safety Evaluation.

6.2.1.6 Test Program

The General Electric Company is presently conducting a large scale test program to verify the performance characteristics of the Mark III containment. Large scale testing was started in November 1973 following completion of a two-year small scale test program.

A total of 67 small scale tests have been performed by GE since June 1971. The test arrangement simulates a Mark III containment with a volumetric scale of approximately 1:2000. Small scale test data have been reported in "Mark III Confirmatory Test

Program Progress Report," NEDM-10848 and "Mark III Analytical Investigations of small Scale Tests Progress Report," NEDM-10976. The intent of these tests was basically proof of principle of a horizontal vent system and also a preliminary checkout of the vent clearing model. Correlations between test data and analytical predications for vent clearing times indicated reasonable agreement in this scale.

The large scale test program utilizes a facility which represents a segment of a Mark III containment. The nominal volumetric scale factor of the facility is 1/130 with the exception of the vent system and suppression pool. Vent system test sections in tions in the various stages of the test program. The original character of the programs was to be a confirmatory exercise to verify the short term analytical model described in Section 6.2.1.2. The scope of the program included testing beyond design basis conditions to investigate the margins available in pressure suppression systems. Additional "phenomena" tests are also planned (i.e., vent interaction) to confirm that their effect had been adequately treated in the analytical modeling.

A derivative of early tests, however, was the observation that containment structures could be subject to significant suppression pool hydrodynamic loads during blowdown (see Section 6.2.1.9). This has resulted in several additional test series whose objective was to generate design basis loads to be incorporated in the design of the affected containment structures.

Eleven large scale test series have been completed to date. Discussions of these and future test series are provided below. A list of completed test is provided in Table 6.2-2.

1. Series 5701 - 5703
   The primary objective of these tests was to verify short term analytical models for horizontal vents. Tests were run with one, two, and three vents open (unplugged) for three scaled break areas (50%, 100% and 200% of DBA) and centerline submergences of two to twelve feet. Based on the results of these tests we conclude that the GE vent clearing submodel calculates with sufficient conservatism the Mark III vent clearing response for the applicable range of vent submergences and drywell pressurization rates.

2. Series 5705 - 5706
   Eleven air blowdown tests were performed using the full scale (27-1/2 inches) test section with one of the three vents plugged. Impact targets were located above the test facility pool. Tests were run with submergences of 6 to 10 feet and pool surface to target clearances of 4 to 18-1/2 feet. The objectives of the tests were to obtain scoping data regarding pool dynamic response and impact loads on structures located above the suppression pool. Air blowdown tests were required to achieve air charging rates into the pool which were representative of an actual plant. We conclude that the test results provided an early indication of the range of pool swell and the magnitude of impact loads on small structures.

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TABLE 6.2-2

LARGE SCALE TESTS COMPLETED

Test Series	Blowdown	Vent Scale	No. of Tests	Primary Objective	Documentation
5701	Steam	Full	21	Vent Clearing	NEDM-13377
5702	Steam	Full	17	Vent Clearing	NEDO-20345
5703	Steam	Full	3	Vent Clearing	NEDO-20533
5705	Air	Full	4	Pool Swell	NEDO-20550
5706	Air	Full	7	Pool Swell	NEDE-20732-P
5801	Steam	1/3	19	Pool Swell	
5802	Steam	1/3	3	Pool Swell	NEDM-13407-P
5803	Water	1/3	2	Pool Swell	
5804	Steam	1/3	5	Pool Swell	
5805	Steam	1/3	51	Impact Loads	NEDE-13426P
5806	Air	1/3	12	Pool Swell	Not formally documented

3. Series 5801 - 5804

Twenty-nine tests were run using the one-third scale vent test section (vent area scaled) with a one-third scale suppression pool (pool area scaled). The flow restriction at the HCU floor was also modeled. The objectives of these tests series were to measure froth impingement loads on the HCU floor and two-phase pressure drop across the HCU floor, and to determine pool swell motion characteristics. Our review of these test data is currently in progress.

4. Series 5805

This test series utilized the same facility arrangement as Series 5801 - 5804 and included pipes, I-beams and grating situated above the pool. The objective of this series was to measure pool impact loads on representative containment structures. Our review of these test data is currently in progress.

5. Series 5806

Twelve air blowdowns were run in this test series utilizing the same facility arrangement as Series 5801-5804. The objectives of this series were to determine pool motion characteristics for large air mass fraction vent flows and to compare these one-third scale results to the previous full scale air tests. The results of these tests are scheduled to be formally documented in October 1975.

Integration of the pool dynamics test results into the GESSAR containment design is discussed in Section 6.2.1.9. Additional large scale tests are planned as discussed below:

1. A series of liquid blowdown tests will be conducted to indicate comparability to steam blowdowns.

2. A series of small break tests will be conducted to investigate pool stratification and vent chugging effects.

3. Tests will be performed with the suppression pool at an initial elevated temperature to determine steam condensation characteristics under such conditions.

4. A multi-vent series will be run employing a test section of three columns of three, nine-inch vent (one-ninth scale by area) to consider possible vent interactions.

As discussed in Section 6.2.1.1 and 6.2.1.2 we consider the basic design and performance of the Mark III containment system to be well established based on our review of the analytical models and the available margins incorporated in the design. Pool dynamic loads are a localized phenomenon which have received additional consideration as discussed in Section 6.2.1.9. We believe that those phenomena being covered in the future Mark III tests merit the additional evaluation but they do not represent design governing conditions at this time nor, in our judgment, will they escalate into design basis considerations as a result of these tests. In summary, we consider the remaining Mark III testing to be confirmatory in nature and will require that the tests and our evaluation of the test results be completed prior to issuance of the first operating license for a Mark III plant.

In several recent letters concerning plants with Mark III containments, including GESSAR, the Advisory Committee on Reactor Safeguards (ACRS) has commented on the progress of the confirmatory test program. In particular, the ACRS emphasized the importance of developing analytical models based on a first principles approach which can be used in conjunction with empirical test results. We are following this matter with GE on a generic basis and will report on their progress in model development in future supplements to this report.

6.2.1.7 Subcompartment Pressure Analyses

Within both the drywell and containment, internal structures form subcompartments or restricted volumes which are subject to differential pressures following postulated pipe ruptures. In the drywell there are two such volumes; the annulus formed by the reactor vessel and the biological shield, and the drywell head region which is a cavity surrounding the reactor pressure vessel head. In the containment various components such as the valves, heat exchangers, and filter/demineralizers of the Reactor Water Cleanup (RWCU) system are located in individual compartments.

GE has performed analyses of the above subcompartments including a nodalized analysis for the reactor vessel shield annulus to determine asymmetric pressure loadings. We have reviewed GE's modeling techniques and assumptions for each subcompartment and have performed confirmatory analyses. Based on the results of these comparisons and the 40% margin applied to the calculated results by GE, we conclude that the design differential pressures for the GESSAR subcompartments are acceptable.

6.2.1.8 Steam Bypass of the Suppression Pool

Possible bypass leakage paths from the drywell to the outer containment have been considered in our review of the Mark III containment. The control of such bypass paths is important to ensure that the design pressure of the containment is not exceeded for postulated design basis accidents.

There are two potential sources of steam bypass of the suppression pool associated with the Mark III containment used in GESSAR. First, since the drywell is of reinforced concrete construction, the potential exists for cracking of the drywell structure under accident loading conditions. This can allow direct leakage of blowdown steam to the containment volume. Second, parts of the Reactor Water Cleanup (RWCU) system are located within the primary containment but outside the drywell. This system has high energy pipe lines, connected to the reactor primary system, which do not have guard pipes. Therefore, postulated ruptures in these lines would result in blowdown of reactor coolant directly to the containment atmosphere without benefit of energy absorption in the suppression pool.

The design of the combustible gas control systems is such that the potential for inadvertent bypass of the suppression pool is minimized and there is no intentional bypassing (i.e., post blowdown) due to operation of the hydrogen control system (see Section 6.2.5 of this report).

In the case of postulated RWCU system pipe breaks the applicant has provided design features to terminate the blowdown prior to exceeding the design limits of the containment. Two isolation valves in series are provided on both the RWCU suction and return lines which will automatically isolate the RWCU system from the primary reactor system. Isolation signals will be generated by two leakage detection systems; one based on RWCU system flow comparisons and another based on compartment temperatures. In addition, a flow limiter is provided in the suction line to limit the rate of blowdown prior to isolation. Based on sensing leakage and isolation, GE has calculated that the containment pressure response assuming a RWCU pipe rupture would be less than 5 psig, which is below the containment design pressure of 15 psig.

In regard to bypass leakage associated with potential cracking of the drywell or other sources around penetrations, we conclude that the GESSAR containment should have an allowable bypass area of approximately one square foot ($A/\sqrt{K}$) for the spectrum of reactor coolant system breaks. The allowable bypass area is considered to be that leakage area between the drywell and containment which would result in containment pressurization to design pressure following a postulated loss-of-coolant accident. To mitigate the effects of bypass, a heat removal system is necessary. For GESSAR such a system is the containment spray system which is an operating mode of the Residual Heat Removal (RHR) system. GE has shown that starting the containment sprays following a 10-minute delay to satisfy the system's ECCS function provides a minimum bypass capability of about 0.93 ft^2 ($A/\sqrt{K}$) for small primary system breaks. We consider such capability adequate. GE has also made a commitment to automatically actuate the sprays when required. We find this to be acceptable pending our review of the electrical design details.

In addition, the drywell will be leak tested at approximately design pressure and at a low pressure prior to plant operation, and low pressure leak tests of the drywell will be performed periodically during plant lifetime (see Section 6.2.6 of this SER for a discussion of the leak test program.)

The acceptance criterion for the tests should be based on the measured leakage being less than 10% of the capability of the containment to accommodate bypass leakage at the test pressure. We have established the 10% acceptance criterion based on engineering judgement of the margin that should be allowed recognizing that (1) the structural response characteristics of the drywell under test conditions are less severe than would be experienced during a design basis accident and (2) allowance for potential deterioration of the leaktightness of the drywell during the plants lifetime (high pressure test is performed only one time, prior to operation) and during the time interval between periodic low pressure tests. We conclude that if the leakages

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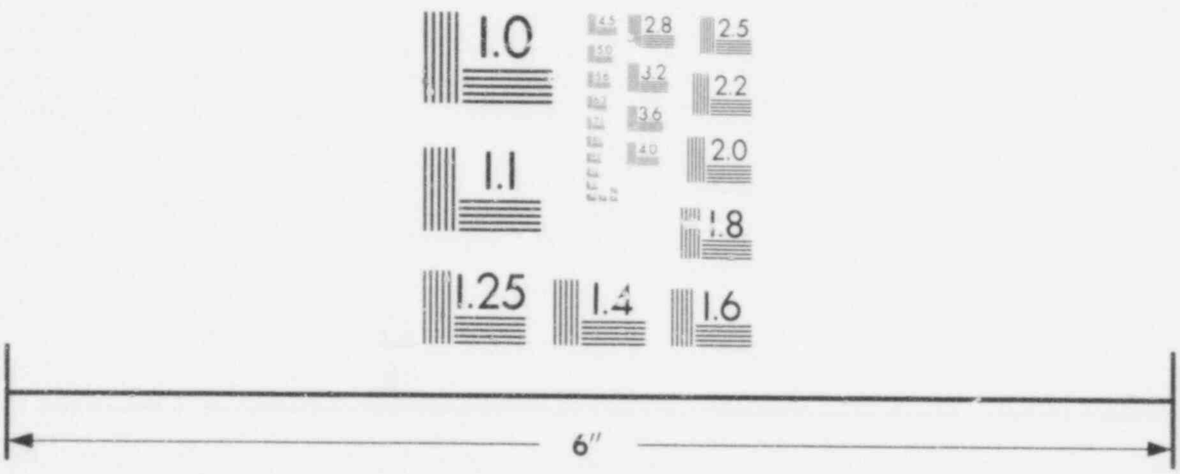
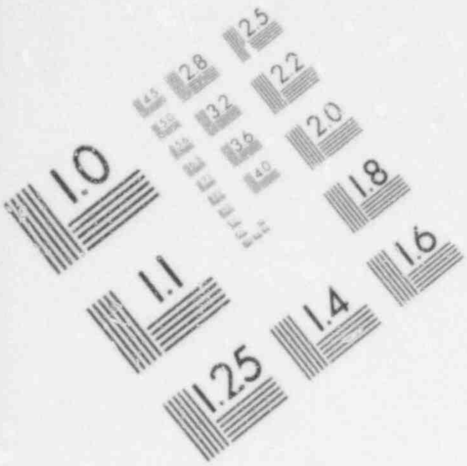
IMAGE EVALUATION
TEST TARGET (MT-3)





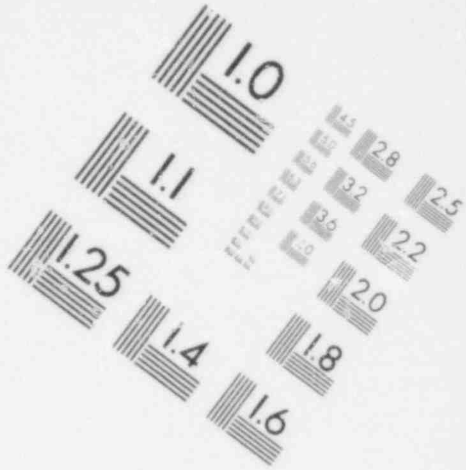
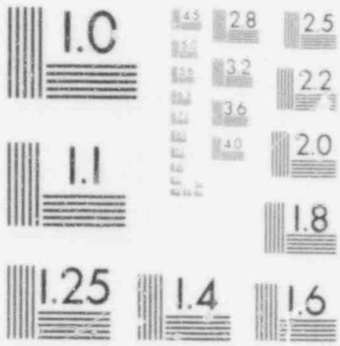
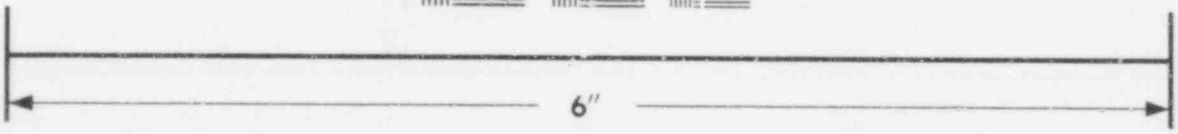
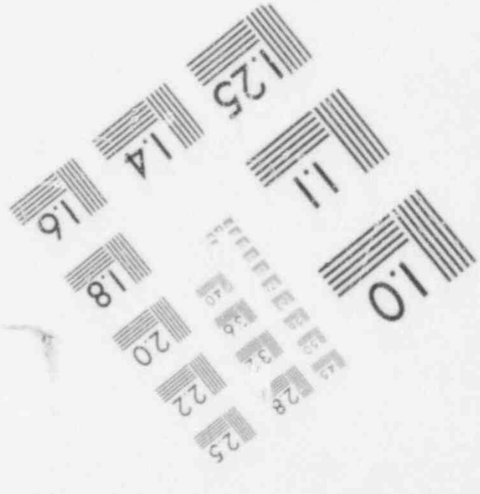


IMAGE EVALUATION TEST TARGET (MT-3)







are not in excess of the indicated limits, there would not appear to exist any potential bypass paths in excess of the design capability of the Mark III containment system.

6.2.1.9 Pool Dynamics

Several phenomena have been identified in our review of the Mark III containment that could result in dynamic loading of structures located in and above the suppression pool. They are related to (1) pool response to the loss-of-coolant accident (LOCA), and (2) pool response due to relief valve operation, generally associated with plant transient conditions. These phenomena are described in more detail below.

LOCA Pool Dynamics: Following a LOCA in the drywell, the drywell atmosphere will be compressed due to blowdown mass and energy addition to the volume. Following vent clearing an air/steam/water mixture will be forced from the drywell through the vent system and injected into the suppression pool, approximately 7-10 feet below the surface. The steam component of the flow mixture will condense in the pool, while the air will be released in the pool as high pressure bubbles. The continued addition and expansion of air causes the pool volume to swell resulting in an acceleration of the surface vertically upward. Due to the effect of buoyancy, air bubbles will rise faster than the pool water mass and will eventually break through the swollen surface and relieve the driving force behind the pool. Due to the dynamics of vent clearing and vent flow and the vertical motion of the pool water mass, structures forming the suppression pool boundary, structures located within the pool, and structures located above the pool could be subject to hydrodynamic loads.

Relief Valve Dynamics: Pressure waves are generated within the suppression pool when, on first opening, relief valves discharge high pressure air and steam into the pool water. This phenomenon is referred to as relief valve vent clearing loads which are imparted to pool retaining structures and structures located within the pool. These same structures can also be subject to loads which accompany extended relief valve discharge into the pool if the pool water is at an elevated temperature. This effect is known as steam quenching vibrations.

In the November SER we stated that GE, as part of the large-scale Mark III test program, had scheduled additional tests and was modifying the test facility to provide pool swell and impact loading data. We stated that we would require that the applicant modify the design of structures located above the pool if the forthcoming experimental data indicated the necessity for such changes. We indicated that we would review and evaluate test data from the test program regarding pool dynamic loading and establish appropriate design margins as part of our ongoing review effort for the Mark III containment. We concluded that the changes which could occur as a result of the tests were technically feasible without compromising safety and that we would require resolution prior to the FSAR.

Subsequent to issuance of the SER and the ACRS letter on the GESSAR, we have evaluated (as part of the post-CP effort) the design loads used in the Grand Gulf application

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for structures located within and above the suppression pool. These are contained in the report issued to the NRC staff by MP&L letter dated April 25, 1975. We concluded that in some instances the design loads were inadequately substantiated by test data or were based on what the NRC staff considered to be a non-conservative interpretation of the test data. We based this on our review of the quarterly progress reports issued through April 1975 by GE for the Mark III Confirmatory Test Program. As a result, we have reassessed our conclusions stated in the GESSAR SER with respect to the schedule for resolution of this area. Accordingly, in order to assure that the results of the ongoing GE test program in the area of pool dynamics is properly factored into the GESSAR design we required that this area be resolved prior to initiation of construction of affected components and structures.

We have sent a letter to GE dated October 24, 1975 which provides a set of design criteria developed by the NRC staff which we would find acceptable for issuance of a PDA. GE has subsequently agreed (by letter from I. Stuart of GE to V. Moore of NRC dated November 7, 1975) to adopt the staff's criteria, with two exceptions, as outlined in the following paragraphs. The first exception pertains to the dynamic loads generated during the clearing of the safety/relief valve discharge lines. GE has recently proposed a revised discharge design and is scheduled to submit additional information in support of their design in the near future. The second exception is related to impact loads on pipes at elevations between 17 and 19.5 feet above the suppression pool. GE is proposing a load of 30 psi in this region compared to the specification of 60 psi in the staff's criteria (see Figure 6.2-3). Both of these exceptions remain unresolved at this time and the PDA will be conditioned accordingly.

GE also plans to submit alternate criteria for NRC staff review on a post-PDA basis supported by additional evaluation of test data from the GE test programs in the area of pool dynamics. Since the suppression pool design features involved are principal architectural and engineering safety features of the plant design, no changes to these design criteria may be adopted in GESSAR for a referencing plant without NRC authorization.

GE as part of the principal architectural and engineering criteria for the design of GESSAR has committed to the course of action specified below for the resolution of the NRC staff's pool dynamic concerns:

1. <u>Small Structures Located at Elevations Less Than 19-1/2 Feet Above the Suppression Pool Surface</u>

   The applicant has made the following commitment:

   GE will either (a) locate these structures at elevations greater than 19-1/2 feet above the pool surface or (b) design these structures to load profiles and associated time histories specified by the NRC staff (attached figures). GE may also provide the NRC staff with additional test data (which is currently available) to justify the GE impact load versus time profiles for small

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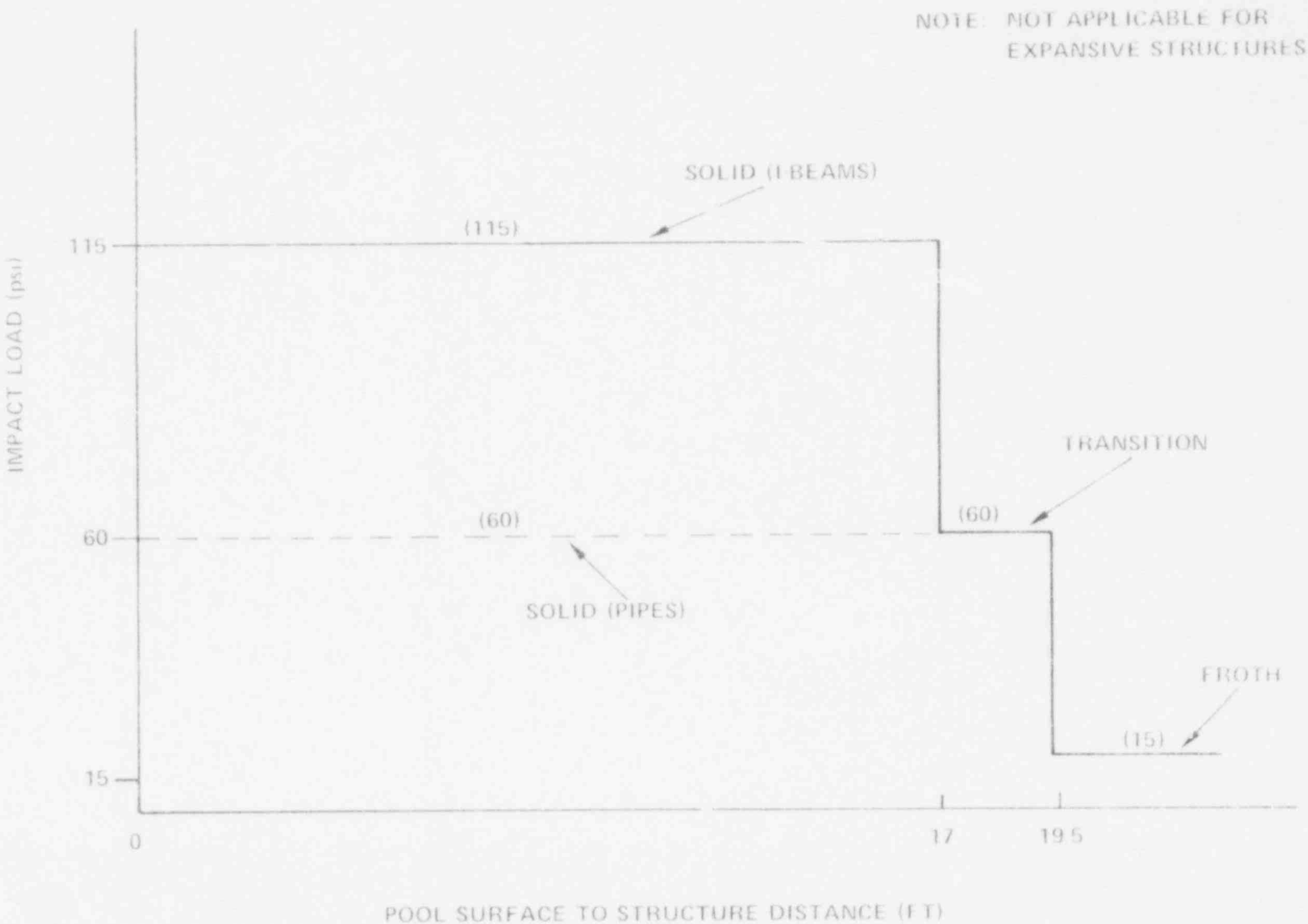

Figure 6.2-3 Mark III Pool Swell Impact Loads

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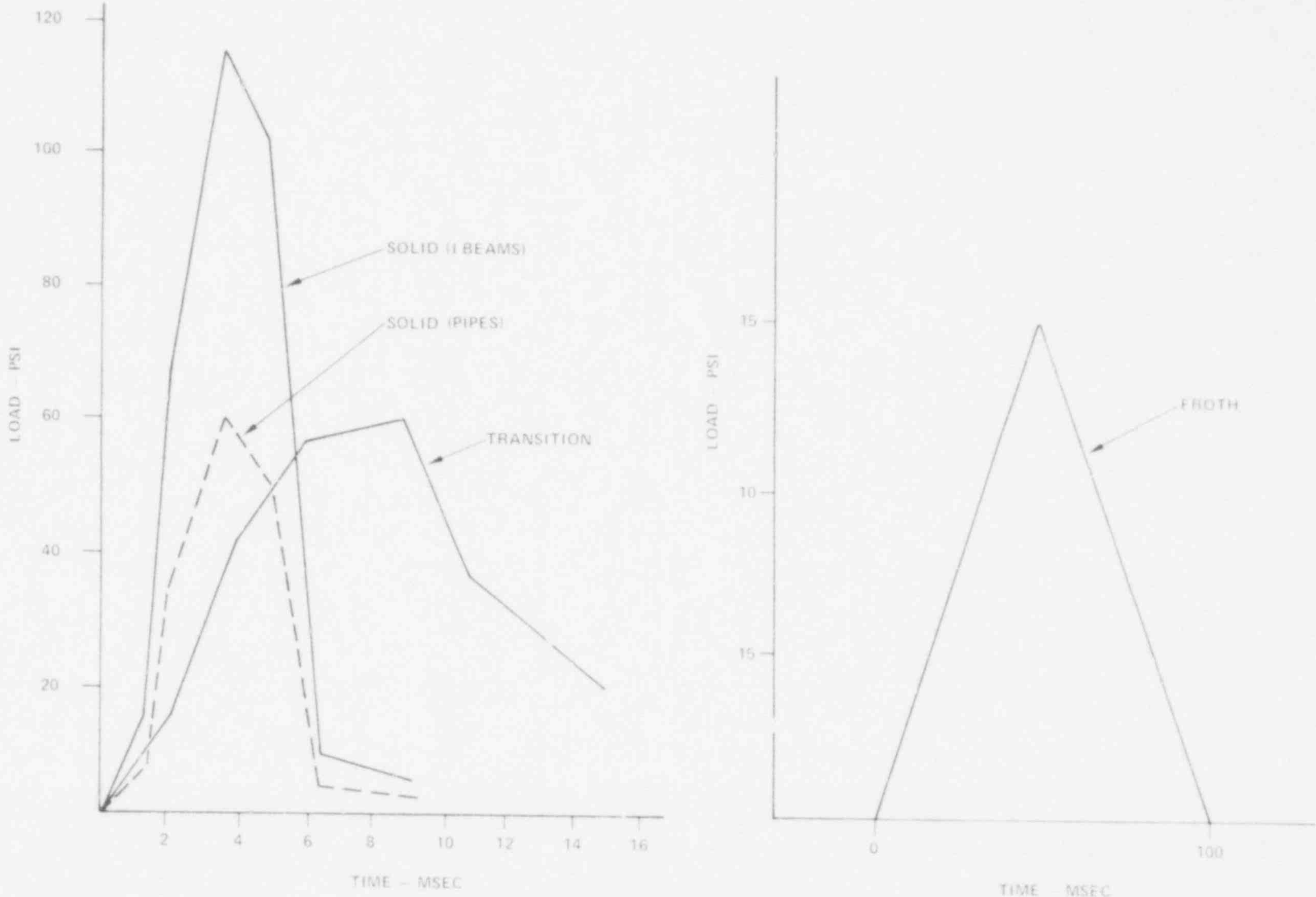

Figure 6.2-4 Pool Swell Impact Load Time Profiles

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structures (e.g., piping and beams). If the GE profiles cannot be substantiated to the satisfaction of the NRC staff, GE will design these structures in accordance with options (a) or (b) cited above.

The NRC staff finds this commitment acceptable based on the following:

(a) We believe that it is technically feasible to locate these structures at higher elevations where pool effects are negligible; and
(b) Sufficient test data have been made available in NEDO-11314-08 (preliminary), Information Report, Mark III Containment Dynamic Loading Conditions, to enable the NRC staff to conclude that the specified load profiles and associated time histories for small structures at their current location are acceptable.

2. Small Structures Located at Elevations Greater Than 19-1/2 Feet Above the Suppression Pool Surface

The applicant has made the following commitment:

Small structures located above 19-1/2 above the pool that could be exposed to froth impingement will be designed for a load of 15 psi and associated time history (see attached figures). The NRC staff finds this commitment acceptable since the specified design load and associated time history are adequately supported by the test data in NEDO-11314-08 (preliminary) and it is technically feasible to design such structures to the specified criteria.

3. Structural Protuberances From the Drywell and Containment Walls

The applicant has made the following commitment:

GE will extend these structures (e.g., the TIP Station, airlocks, and sumps) into the suppression pool. These structures will be designed for coincident loads due to air bubble (equal to peak drywell pressure for drywell protuberances and equal to 10 psi for containment protuberances) and pool drag (based on a pool swell surface velocity of 40 ft/sec).

The NRC staff finds this commitment acceptable since the design loads are adequately supported by the test data in NEDO-11314-08 (preliminary) and it is technically feasible to design such structures to the specified loads.

4. Expansive Structures

The applicant has made the following commitment:

Expansive structures (e.g., the main steam line pipe tunnel and the HCU floor) will be located at elevations greater than 19-1/2 feet above the suppression pool surface. Expansive structures located at elevations between 19-1/2 feet and 35 feet above the pool will be designed for a froth impingement load of 15 psi and associated time history (see attached figures) and a flow pressure differential of 11 psi. GE may also provide the NRC staff with additional information to justify the pool dynamic loads applied to these structures to support locating them at elevations lower than 19-1/2 feet above the suppression pool surface. However, should GE be unable to justify such designs to the NRC staff, GE will locate these structures at elevations greater than 19-1/2 feet above the pool surface and design these structures for the loads and associated time history cited above.

The NRC staff finds this commitment acceptable since the specified loads and associated time history for expansive structures at elevations greater than 19-1/2 feet above the pool are adequately supported by the test data in NEDO-11314-08 (preliminary) and it is technically feasible to design such structures to the specified loads without affecting any other aspect of the GESSAR design.

5. Asymmetric Loads

The applicant has made the following commitment:

GE will evaluate asymmetric loads based on (a) the relief valve load cases listed in Section A8 of NEDO-11314-08 (Preliminary) and (b) the unequal bubble load profile specified in Section 6.1.3 of NEDO-11314-08 (Preliminary).

The NRC staff finds this commitment acceptable since the specified load cases are adequately conservative and it is technically feasible to design such structures to the specified loads.

6. Other Structures

The applicant has made the following commitment:

GE will eliminate structures in the GESSAR containment design which are not included in the preceding categories unless the design of such structures can be justified to the NRC staff. In such cases GE will provide the NRC staff with additional justification to verify the bases for specification of the pool dynamic load versus time history applied to those structures. Should GE be unable to demonstrate to the NRC staff that such loads are adequately conservative, these structures will be eliminated such that the design conforms to the basic Mark III design as typified by the Grand Gulf design.

The NRC staff finds this commitment acceptable since the proposed alternatives are technically feasible as indicated by the Grand Gulf design.

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7. <u>Other Pool Dynamic Loads</u>

The applicant has made the following commitment:

For pool dynamic loads not specifically addressed in the above criteria GE will use the types, magnitudes, and combinations of loads identified in NEDO-11314-08 (Preliminary) as a basis for evaluating the structural design of affected containment structures.

The NRC staff finds this commitment acceptable since the design loads are adequately conservative and it is technically feasible to design such structures to the specified loads.

In summary, we have reviewed the applicant's program and have concluded that the principal architectural and engineering criteria for the design of affected components and structures have been adequately described.

Based on our review of the proposed GESSAR design and our review of these same areas on Grand Gulf (currently under construction), we maintain our conclusion that any changes which may be required as a result of our review are technically feasible without compromising safety. Such changes could include, as appropriate, relocation, local strengthening, or protection by incorporation of structures to preclude direct impingement of flow.

In addition, the staff has reviewed the ACRS comments on the Mark III containment design. The ACRS comments are contained in its letter on GESSAR. These same comments are also contained in the ACRS letters to the Commission on other BWR-6 applications. Specifically the ACRS stated that a more basic understanding of certain phenomena such as vent clearing, vent interaction, pool swell, pool stratification, and dynamic and asymmetric loads on the suppression pool and other containment structures is required. The ACRS further stated that the R&D program be expedited so that all design related issues are fully resolved prior to completion of construction of affected portions of the plants referencing GESSAR. In response to these comments, the NRC staff has expedited its review of these phenomena and we have actively pursued this matter with the applicant to ensure compliance with the ACRS recommendations.

To address these ACRS comments, the NRC staff prepared a series of questions which were transmitted to the applicant in letters dated March 18, 1975 and April 22, 1975.

In addition, the ACRS in its GESSAR letter, as well as in those for other BWR-6 applications recommended that the independent models developed by the NRC staff and its consultants "be used to evaluate the sensitivity of key design parameters, and to elucidate additional effects noted in the experimental programs such as oscillatory phenomena."

The NRC staff is continuing with its development of an independent model to analyze the Mark III containment and fully expects to satisfy this aspect of the ACRS recommendation with respect to the GESSAR Mark III containment.

In summary, we consider the remaining Mark III testing and analytical programs to be confirmatory in nature and will require that these programs be completed prior to issuance of an operating license for a Mark III plant. We believe that the concerns expressed by the ACRS are pertinent and merit additional evaluation, however, in our judgment, they will not affect the design bases for the GESSAR containment. We conclude that the information which has been developed to date relating to the ACRS concerns is sufficient to demonstrate the adequacy of the present design.

6.2.2 Containment Heat Removal

The containment heat removal system includes the piping, valves and mechanical components used to remove energy from the containment following a loss-of-coolant accident. For GESSAR this system is the Residual Heat Removal (RHR) System which, when operating in the suppression pool cooling or containment spray mode, removes energy from the containment to limit long-term containment post-accident temperatures and pressures. Our review in this area included process and instrumentation diagrams, descriptive information concerning system functioning and interaction with essential supporting systems, General Electric's proposed system design bases and criteria, and analyses in support of the adequacy of those bases and criteria.

The RHR System consists of two heat exhangers and three pumps. One heat exchanger and one pump form an independent loop and each loop is physically separated and protected to minimize the potential for single failures including loss of onsite or offsite power causing the loss of function of the entire system. The third pump is located in a separate room and can be connected to either loop. The RHR System is designed to Category I seismic criteria and will be routinely tested during plant operation to verify its availability.

Operating in the containment cooling mode, the RHR pumps take suction from the suppression pool, pass the flow through the RHR heat exchangers, and direct the cooled water to the suppression pool, the reactor vessel, or the containment spray headers. The locations of suction and return lines in the suppression pool facilitate mixing of the return water with the total pool inventory before the return water becomes available to the suction lines. Strainers are provided on the suction line inlets

Analyses of the minimum heat removal capability of the RHR system have been presented in GESSAR based on a service water temperature of 100°F and an overall heat exchanger duty of 610 Btu/° sec. These analyses indicate adequate heat removal capability to limit the suppression pool temperature to 173°F and the containment pressure to 9.8 psig following a loss-of-coolant accident. We find the values to be within acceptable limits and are appropriate interface values. The service water temperature,

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however, is a site dependent parameter which could be different for each plant referencing GESSAR. In turn, the detailed design characteristics of the heat exchanger design, which are based on service water temperature, are also site dependent. Therefore, each applicant for a standard plant must provide the above information and demonstrate that the available heat removal capacity (heat exchanger duty) for this plant is at least as great as that specified above.

General Electric has stated in GESSAR that adequate net positive suction head is available at the RHR pump inlets assuming the containment is at atmospheric pressure and the pool is at saturation temperature. These assumptions are consistent with the requirements of Regulatory Guide 1.1 and therefore acceptable. Provisions are made in the containment heat removal system to permit inservice inspection of system components and functional testing of active components.

We conclude that the containment heat removal system can be operated in such a manner as to provide adequate cooling to the containment following a loss-of-coolant accident and conforms to General Design Criteria 38, 39, and 40, and is acceptable.

GE has evaluated the potential for debris to clog the ECCS suction lines. Each ECCS pump draws water from the suppression pool through its own suction line and strainer assembly. The suction line ends in a tee arrangement in the pool, with each end of the tee capped with a 100% flow capacity strainer. GE has shown that the potential for ECCS or containment heat removal system degradation due to plugging of the screens is minimal. The reasons for this include: 1) GE has stated that all insulation in the drywell will be of such type that it minimizes the possibility of it breaking away from piping and being carried through the drywell vent system into the suppression pool, 2) Since the suction inlets are located about midway between the pool surface and pool bottom and since the screen surface area is large, resulting in low approach velocities, there is little potential for drawing debris, either from the pool bottom or surface, to the vicinity of the inlet lines, and 3) Due to the ramshead arrangement on each suction line, a 50% plugging of screen surface area can be tolerated without consequence to system performance.

Therefore, their design is acceptable.

6.2.3 Secondary Containment Functional Design

The secondary containment system includes the structures and system used to control and treat radioactive leakage from the primary containment in the event of a loss-of-coolant accident. For GESSAR the secondary containment system consists of the shield building, the fuel building, parts of the auxiliary building, the Shield Building Annulus Recirculation and Exhaust System (SBARES) and the Standby Gas Treatment System (SGTS). Our review in this area included schematic flow diagrams, descriptive information concerning system functioning and interaction with essential supporting systems, and General Electric's proposed system design bases and criteria.

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The shield building is a cylindrical, reinforced concrete structure completely enclosing the containment vessel. The Shield Building Annulus Recirculation and Exhaust System maintains the annulus formed by the shield building and containment at a negative pressure (approximately -5" w.g.) during normal operation and following a postulated loss-of-coolant accident. In the Chapter 15 accident analyses, the staff assumes that following an accident, a variable fraction of the SBARES exhaust is directed to the Standby Gas Treatment System (SGTS) for filtration prior to release to the atmosphere and the remainder is recirculated to the annulus where it is mixed in 50% of the annulus volume. The exhaust/recirculation split is a function of the flow rate in these paths as shown in Table 15.3-1 of this SER.

The Fuel Building and the ECCS and RWCU rooms in the Auxiliary Building are also part of the secondary containment system. These volumes are maintained at a negative pressure (-0.25" w.g.) during operation by normal plant ventilation systems. Following a postulated accident, the ECCS and RWCU rooms will be maintained at a negative pressure by the Standby Gas Treatment System (SGTS), which filters the exhaust flow prior to release to the atmosphere. The Fuel Building will be aligned to the SGTS in the event that high radiation in the exhaust flow is detected, or upon receipt of a loss-of-coolant accident signal.

The Shield Building Annulus Recirculation and Exhaust System provides active component redundancy, is designed to seismic Category I criteria, and is located within seismic Category I structures. Redundant components are separated and protected.

The Standby Gas Treatment System (SGTS) is designed to seismic Category I criteria and is located within seismic Category I structures. The SGTS consists of redundant exhaust fans and filtration trains each consisting of a demister, heat coil, prefilter, HEPA filters and charcoal filter. Redundant components are separated and protected.

Following a postulated loss-of-coolant accident the pressure in the secondary containment volumes could increase due to inleakage and the starting time required for the SGTS. Additionally, the annulus pressure and temperature will increase due to heat transfer through and expansion of the primary containment shell. GE has provided an analysis of the annulus pressure transient which considers the above phenomena and which indicates that the annulus will be maintained at a negative pressure of about -1" w.g. or less. We find this to be acceptable. GE has also provided similar analyses for the fuel building and ECCS and RWCU room volumes. The results indicate that these volumes will have reestablished a 0.25 inch w.g. negative pressure 60 seconds after the LOCA. We will require that the inleakage assumptions and drawdown times be verified during preoperational testing of each plant referencing GESSAR. In Amendment 27, GE demonstrated that the concurrent exhaust requirements from the fuel building (800 scfm), the ECCS pump rooms (1880scfm) and the annulus (1000 scfm) are within the capacity of one train of the SGTS (5000 scfm). We conclude that the capacity of the SGTS is adequate.

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Although the primary containment is completely enclosed by the secondary containment, there are systems which penetrate both the primary and secondary containment boundaries creating potential paths through which radioactivity in the primary containment could bypass the leakage collection and filtration systems of the secondary containment. A number of these lines contain physical barriers or design provisions which can effectively eliminate leakage such as water seals, closed seismic Category I piping systems, or vent return lines to a controlled region. The criteria by which potential bypass leakage paths are determined have been set forth in Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants."

GE has committed (in a letter from I. Stuart of GE to B. Rusche of NRC dated August 25, 1975) to providing design provisions to eliminate bypass leakage. GE will develop a design for positive leakage control systems, upgrade some piping systems to seismic Category I for the purpose of achieving credit for closed loops or water seals, and identify those containment water legs and loop seals which perform a sealing function and are presently contained in the Mark III design. Where positive leakage control systems are employed, such systems will be designed to meet Branch Technical Position CSB 6-3 and Regulatory Guide 1.96 as applicable.

We find the above commitments to eliminate bypass leakage acceptable for a PDA. GE will provide details of the design changes being employed and systems analyses by April 15, 1976, at which time we will perform a detailed review.

6.2.3.1 Containment Air Purification and Cleanup Systems

There are two engineered safety feature air cleanup systems proposed for GESSAR. They are the Standby Gas Treatment System and the Control Room Air Cleaning Unit.

The staff has analyzed the designs of engineered safety feature filtration systems designated by the applicant to operate in emergency situations with respect to the positions in Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmospheric Cleanup Systems Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." We find the applicant's design is in agreement with these positions and we have used an adsorption efficiency of 99% for iodine removal for these deep bed type charcoal systems in our accident consequence computations (see Section 15).

6.2.4 Containment Isolation System

The containment isolation system includes the containment isolation valves and associated piping and penetrations necessary to isolate the primary containment in the event of a loss-of-coolant accident. Our review of this system included the number and location of isolation valves, the valve actuation signals and valve control features, the positions of the valves under various plant conditions, the protection afforded isolation valves from missiles and pipe whip, and the environmental design conditions specified in the design of components.

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The design objective of the containment isolation system is to allow the normal or emergency passage of fluids through the contai ment boundary while preserving the integrity of the containment boundary to prevent or limit the escape of fission products from a postulated loss-of-coolant accident. GE has specified design bases and design criteria as well as the isolation valve arrangements used for isolation of primary containment penetrations.

No manual operation is required for immediate isolation of the containment. Automatic trip valves are provided in those lines which must be isolated immediately following an accident. Lines that must remain in service following an accident for safety reasons are provided with at least one remote manual valve. The containment isolation systems have been designed to the ASME Section III, Class 1 or 2, code and have been classified as Category I seismic design systems. GE has also described the instrument line penetrations of the primary containment, and justified that the design conforms to the positi ns of Regulatory Guide 1.11.

The environmental design criteria specified for isolation valves and other safety-related equipment located in the drywell are not entirely acceptable. We will require that GE qualify these components for a drywell negative pressure representative of limiting post-LOCA conditions. We will report on the resolution of this item in a supplement to this SER post-PDA.

GE has proposed that the containment purge system be operated continuously during normal operation to limit the buildup of activity and allow plant personnel unlimited access for surveillance and maintenance. The purge rate would be 4,300 cfm through 42-inch supply and exhaust lines. Fast-acting isolation valves are installed on the supply and exhaust lines to provide rapid containment isolation in the event of a loss-of-coolant accident.

The Regulatory staff considers that continuous purging of the primary containment through large penetrations is undesirable. This was identified by the ACRS during their review as an area of concern requiring resolution acceptable to the NRC staff prior to the issuance of a PDA. The staff could find continuous purging to be acceptable if GE adopts the design criteria set forth in Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations." This requirement has been made a condition of the PDA.

6.2.5 Combustible Gas Control

The combustible gas control systems incluse the piping, valves components, and instrumentation necessary to detect the presence of combustible gases within the primary containment and to control the concentrations of these gases. Our review of these systems included the potential sources of combustible gases and their yields, the accumulation of gases within each volume of the containment system, the capability

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to monitor the concentrations of the gases, and the capability to control and reduce the combustible gas concentrations by suitable means.

Following a loss-of-coolant accident, hydrogen may accumulate within the containment as a result of metal-water reaction between the fuel cladding and the reactor coolant, and as a result of radiolytic decomposition of the post-accident emergency cooling water. The applicant has analyzed the production and accumulation of hydrogen from the above sources using the guidelines of Containment Systems Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." GE has proposed a redundant hydrogen mixing system and redundant hydrogen recombiners to limit the hydrogen concentration within the containment to below 4 volume percent. A backup, controlled purge system is also provided in accordance with the above position.

The drywell mixing system is capable of purging any hydrogen that might be released from the reactor pressure vessel into the drywell through the horizontal vents in the drywell wall into the suppression pool and thence to the larger containment volume, thereby diluting the hydrogen concentration. This purging is accomplished by raising the pressure in the drywell so as to force the drywell atmosphere, including the hydrogen, through the suppression pool into the containment. The drywell is pressurized through two, six-inch lines, each of which is connected to a compressor. These lines contain a check valve and a butterfly valve in series. Operation of the mixing system would not be required for about 8 hours following a postulated LOCA at which time it would be manually actuated by the operator.

The proposed drywell mixing system provides several desirable features with respect to the potential bypassing of the suppression pool by the steam in the drywell. The first of these features is that the proposed purge flow path through the suppression pool eliminates any intentional bypass of the suppression pool in order to control the hydrogen concentration in the drywell. Additionally, the valving arrangement in the lines that penetrate the drywell minimize the potential for an inadvertently open line. Finally the proposed reduction in the size of these lines has reduced the bypass area of an inadvertently open line to a value within the bypass capability of the containment.

During operation of the mixing system the hydrogen concentration in the containment would continue to increase due to radiolysis. A thermal recombiner system is provided to maintain the long term hydrogen concentration to less than 4 v/o. Recombiner operation is manually initiated by the operator at about 15 days following a LOCA. The recombiner system has a design flow rate of 100 scfm and is located in the auxiliary building. It is designed to seismic Category I criteria and has redundancy in all active components.

We conclude that the design of the combustible gas control systems conform to the applicable regulations, guides and staff positions and is acceptable. We will require GE to submit results of their prototype test program for the thermal recombiner as they become available.

6.2.6 Containment Leakage Testing Program

The GESSAR containment design includes the provisions and features necessary to satisfy the testing requirements of Appendix J, 10 CFR Part 50. The design of the containment penetrations and isolation valves permit individual, periodic leakage rate testing at the pressure specified in Appendix J, 10 CFR Part 50. Included in the proposed program of leakage rate testing are those penetrations that have resilient seals, such as airlocks, equipment hatches, and fuel transfer tubes.

The proposed containment leakage testing program complies with the requirements of Appendix J, 10 CFR Part 50. Such compliance provides adequate assurance that con-containment leaktight integrity can be verified throughout the service lifetime and that the leakage rates will be periodically checked during service, on a timely basis, to maintain such leakages within the specified limits.

Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity release within the containment, the loss of the containment atmosphere through leakage paths will not be in excess of the acceptable limits specified for the site, i.e., the doses will be well within 10 CFR Part 100 guidelines. Compliance with the requirements of Appendix J constitutes an acceptable basis for satisfying the requirements of General Design Criteria 52, 53, and 54, of Appendix A of 10 CFR Part 50.

6.3 Emergency Core Cooling System (ECCS)

6.3.1 System Description

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

The ECCS consists of two high pressure systems and two low pressure systems. The former are the High Pressure Core Spray (HPCS) system and the Automatic Depressurization System (ADS). The latter are the Low Pressure Core Spray (LPCS) system and the Low Pressure Coolant Injection (LPCI) system, which is one mode of the Residual Heat Removal (RHR) system. The ECCS systems for the GESSAR reactor are functionally identical to General Electric 1969 product line facilities (LaSalle, Bailly, Zimmer).

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

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

The HPCS consists of a single motor-driven centrifugal pump and associated system piping, valves, controls and instrumentation. The system is designed to operate from offsite power or from a separate diesel generator. Suction is taken from the condensate tank or the suppression pool and piped to a spray sparger over the core (via two entry points at the shroud). Nozzles spaced around the sparger spray the water over the top of the core and into the fuel assemblies. The system is designed to function over the entire range of reactor coolant system pressures and break sizes. For small breaks the system will maintain reactor water level. For intermediate breaks that do not depressurize the reactor vessel rapidly, the system will depressurize the vessel. For large breaks, rapid depressurization occurs and the HPCS cools the core in the spray cooling mode until sufficient inventory is accumulated to terminate the transient.

The pump characteristics are selected to satisfy requirements for both high pressure, low flow rate deliveries for small breaks and low pressure, high flow rate deliveries for large breaks. When the cooling system is activated, the initial flow rate is established by primary system pressure. As reactor pressure decreases, the flow rate will increase until the required core spray flow rate is achieved when the differential pressure between the reactor vessel and primary containment reaches 200 psi. The pump is designed to deliver 6110 gpm at 200 psid, 1465 gpm at 1140 psid and has a shutoff head of 1370 psid.

The ADS reduces the reactor pressure so that flow from the LPCI and LPCS can enter the reactor to cool the core and limit the fuel cladding temperature. The ADS utilizes eight of the 19 safety-relief valves in the nuclear pressure relief system. Automatic opening of these valves requires coincident signals of reactor vessel low water and high drywell pressure along with a high discharge pressure indication of any LPCI or LPCS pump, but only after a timer delays operation of the relief valves for two minutes. If the operator determines that the initiation signal is false or depressurization is not required, the timer may be recycled. The ADS is redundant to the HPCS and is only required if the HPCS cannot maintain reactor water level following a LOCA. Similar to the HPCS, the ADS is not required for large breaks.

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The LPCS system consists of a motor-driven centrifugal pump (that can be powered by either normal offsite power or the standby a-c power system); a spray sparger in the reactor vessel; and piping, valves, instrumentation and controls to convey water from the suppression pool to the sparger.

The HPCS system operating in the low pressure mode serves as a redundant core spray loop to the one LPCS loop. The LPCS system protects the core in the event of a large break in the nuclear system and when the HPCS is unable to maintain reactor vessel water level. Such protection extends to the small break in which the ADS or HPCS has operated to lower the reactor vessel pressure to the operating range of the LPCS. The LPCS pump is designed to deliver 6110 gpm at 122 psid and has a shutoff head of 289 psid.

Since the number of fuel assemblies and the diameter of the core has changed compared to previous designs, spray distribution tests will be performed on a simulation of the GESSAR reactor to assure that an adequate amount of spray reaches every assembly. GE states that no significant differences are expected from other core geometries ppre-vously tested for spray distribution. GE will provide the results of these tests in a topical report to be submitted by the end of 1975. We will review that report when it becomes available. This commitment is acceptable for the PDA for GESSAR.

The LPCI system consists of three motor-driven centrifugal pumps (that can be powered by either normal offsite power or the standby, onsite a-c power system), associated piping, valves, controls and instrumentation. Each LPCI pump injects water from the suppression pool through a nozzle in the core shroud into the space between channel boxes over the active core. The suppression pool suction, vessel injection nozzle and connecting piping for each pump are separate and independent. Two of the pumps also function as RHR system pumps. These two pumps receive power from different a-c power buses. One of these buses also supplies power to the third LPCI pump, and the second bus supplies power to the LPCS pump.

The LPCI system provides cooling water following all LOCAs except those resulting from small breaks that can be controlled by the HPCS system. The LPCI is redundant to the LPCS system. Each LPCI pump delivers 7100 gpm at 26 psid and has a shutoff head of 225 psid.

As in previous BWR designs, the GESSAR reactor has the capability to use the LPCI pumps to spray water into the containment. Diversion of these pumps after a loss-of-coolant accident is automatic when required. In previous designs, an interlock prevented diversion of the LPCI pumps if the vessel water level was below 2/3 the active core height. In the proposed arrangement for this plant, this interlock would not be present but an interlock preventing LPCI pump diversion to containment spray until 10 minutes after a LOCA would be present. In support of this change, that proposed diversion of LPCI pumps at a specified time after a LOCA irrespective of vessel water level, GE has stated that diversion of LPCI pumps ten minutes after

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initiation of any loss-of-coolant accident will not significantly affect the performance of the ECC system. GE presented the results of their analyses that show that for various combinations of ECCS and liquid line breaks, the peak clad temperatures do not change for breaks greater than 0.02 ft^2 (2 in. diameter pipe) assuming a single ECCS failure. For breaks smaller than 0.02 ft^2, the HPCS failure is the only one that results in peak clad temperature differences. For these breaks, the peak clad temperature increases about 50-100°F (to about 700° or 800°F), as shown on Figure 6.3-39 of GESSAR.

We have reviewed the information submitted by GE and have determined that the performance of the ECCS is not significantly affected by the transferral of two LPCI pumps from core cooling to containment spray. We conclude that transferral after 10 minutes is acceptable.

The Regulatory staff addressed a concern (request for additional information 6.125) regarding the overall role of manual actions required to mitigate the consequences of a LOCA GE has agreed to provide this information for our review prior to the time that GE becomes committed to irrevocable final designs.

The following additional outstanding concerns should be addressed prior to submittal of the first BWR/6 FSAR: 1) results of spray distribution tests on the BWR-6 must be submitted to the Regulatory staff. The results in the topical report (NEDO-10846) are for BWR-4 and BWR-5 configurations, with results for BWR-6 promised later, 2) final copy of the proprietary version of the 8 x 8 Zr spray cooling test must be submitted for staff review. The resolution os these items will be discussed when our review of them is complete. Resolution of these items is not required prior to issuance of a PDA.

6.3.2 Performance Evaluation

In Section 6.3 of GESSAR, the applicant provided an analysis of the ECCS using the assumptions and calculational techniques described in 10 CFR 50.46 and Appendix K to 10 CFR 50.

On December 28, 1973 the Commission issued the Acceptance Criteria for Emergency Core Cooling System for Light-Water-Cooled Nuclear Power Reactors which is a new rule and replaces the Interim Policy Statement.

In Amendment 35, dated June 27, 1975, the General Electric Company submitted a LOCA analysis applicable to the GESSAR-238 reactor which complied with the requirements of paragraph 50.46 and Appendix K of 10 CFR Part 50. The analysis was performed using General Electric evaluation models as described in NEDE-20566 (Draft) submitted in August 1974, and the General Electric Refill/Reflood Calculation (Supplement to the SAFE Code Description) transmitted to the USAEC by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974. General Electric has submitted an additional report by letter from G. L. Gyorey to V. Stello, Jr., dated August 25, 1975, that discusses the way in which the GE Reflood model is used in the analysis of BWR's with in-shroud LPCI injection. The background of the staff review of the General Electric (GE) ECCS models is described in the staff Safety Evaluation Report (SER) issued in connection with the order dated December 27, 1974 for operating jet pump boiling water reactors. The bases for acceptance of the principal portions of the GE Evaluation Model are set forth in the staff's Status Report of October 1974 and the Supplement to the Status Report of November 1974 which are referenced in the December 27, 1974 SER. Together, the December 27, 1974 SER on operating plants, the Status Report and its Supplement described the basis for the staff's acceptance of the ECCS evaluation model. The General Electric evaluation model, in combination with the plant specific parameters, consitutes an acceptable ECCS evaluation in conformance with Appendix K to 10 CFR Part 50 and applicable to GESSAR-238 BWR/6 type reactors.

During the course of our review, we concluded that additional break sizes should be analyzed to substantiate the break spectrum curve. We also requested that other break locations (i.e., steam line, feedwater line, and core spray) be studied to substantiate that the limiting break location was the recirculation line. As part of the LOCA analysis additional BWR/6 single failure sensitivity analyses were performed to evaluate the effects of a single failure that could cause any manually-controlled electrically-operated valve to move to a position that adversely affects the ECCS. The analysis showed that these failures are less severe than those considered for the ECCS analysis.

We also investigated the effects of flooding caused by a LOCA. By letter from W. D. Gilbert to J. F. Stolz, dated August 11, 1975, GE submitted the results of a study on ECCS valves within the containment. The results show that all ECCS valve motors which must be operable during and after a LOCA are located outside the containment and will not become submerged due to the occurrence of a LOCA. Therefore, neither the short-term requirement nor the long-term cooling capability is affected by submergence effects.

The results of the ECCS Appendix K calculation (based on the limiting case of flat local peaking) show a peak cladding temperature of 2180°F; a peak local oxidation of 2%, and a maximum core average hydrogen generation of 0.25% for the worst large break assuming a failure of the LPCI diesel; a peak cladding temperature of 1680°F at a break area of 0.3 ft^2 assuming the failure of a LPCS diesel; and a peak cladding temperature of 1520°F at 0.1 ft^2 assuming a failure of the HPCS.

We have reviewed the evaluation of ECCS performance submitted by the General Electric Company for the GESSAR-238 and conclude that the evaluation was performed wholly in conformance with the requirements of 10 CFR 50.46(a). The GESSAR-238 ECCS performance assures conformance with: (1) the peak cladding temperature limit of 2200°F, (2) the maximum cladding oxidation limit of 17% of total cladding thickness before oxidation, (3) the maximum hydrogen generation core wide limit of 1% of the total metal in the cladding, (4) the core geometry remaining amenable to cooling,

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and (5) the long-term cooling requirement of maintaining acceptable core temperatures and decay heat removal.

An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, reactor operation under such conditions will not be authorized until the necessary analyses have been performed, evaluated, and determined acceptable.

During our Appendix K review of BWR/5 and BWR/6 type reactors, the NRC staff expressed a concern relating to recirculation valve closure during a LOCA. The results of a GE sensitivity study to evaluate the effect of a recirculation flow control valve fast closure coincident with a design basis LOCA and the worst postulated ECCS failure was submitted by a letter from A. J. Levine of GE to V. Stello, NRC, dated April 25, 1975. The results of this analysis show that the calculated peak cladding temperatures even with the worst failure of the ECCS and fast closure of the recirculation flow control valve remains below 2200°F. On this basis, we find the analysis acceptable.

In summary, we conclude that the ECCS for GESSAR meets all the criteria of paragraph 50.46 and the requirements of Appendix K to 10 CFR 50, and is acceptable.

6.4 Habitability Systems

The applicant proposes to meet General Design Criterion No. 19, (Control Room) of Appendix A to 10 CFR Part 50, by use of concrete shielding and by installing a dual fresh-air inlet system containing redundant six-inch-bed charcoal filters. This represents a system modification in response to a staff position that the initially proposed emergency ventilation system was inadequate.

The dual inlets are widely spaced to reduce the possibility of having both inlets exposed to contamination at the same time. Under most meteorological conditions one of the inlets will be free of contamination resulting from a postulated activity release. The free inlet will be used to supply sufficient make-up air to maintain the control room at a positive pressure. Redundant control room emergency air cleaning units each supply 2000 cfm of air.

The staff concludes that properly positioned inlets can provide the necessary degree of assurance against the possibility of both inlets being severely contaminated. The applicant has indicated the following locations:

Inlet 1: About 150 feet from the SGTS vent on the outermost wall of the control building (at elevation +35'-0").

Inlet 2: About 200 feet from the SGTS vent near the outer edge of the auxiliary building (separated from Inlet 1 by about 240 feet and at elevation 140'4").

These positions are considered adequate for a single unit facility. Based on average meteorology, the use of protective clothing will be necessary to protect against a beta skin dose estimated between 30 rem and 75 rem for the single unit facility.

For a multiple unit plant, we will require that the control room of one unit must meet GDC-19 when postulating an accident in the second unit. The dual inlets in one unit will not be as effective in this case if they line up in a straight line with the vent releasing th. activity. This results in a simultaneous contamination of both inlets. Thus, in the case of multiple unit plants, we must approve the inlet location on a case-by-case basis.

We conclude that, on a generic basis, the GESSAR design for control room habitability meets GDC-19 requirements with the possible exception of the dual inlet location on multiple unit plants. We will review the need for protective clothing o. e-by-case basis.

The control room design doe not specifically protect against a release of toxic gas such as chlorine. Thus, the hazardous material concerns as expressed in Regulatory Guide 1.78 will be reviewed on a case-by-case basis. The control room isolation system may, therefore, need to be modified for any site where gaseous chlorine is used for treatment of the circulating water or where substantial quantities of other hazardous gases are stored in the vicinity.

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7.0 INSTRUMENTATION AND CONTROLS

7.1 Introduction

GESSAR was submitted pursuant to the Reference System procedural option contained in the staff's study entitled, "Methods for Achieving Standardization of Nuclear Plants," issued March 5, 1973. In discussing the major advantages in the standardization of nuclear power plants, the staff stated in the study that "The most important advantage is the enhancement of reactor safety due to the concentration of staff effort on the in-depth review of standardized systems and on the resolution of generic safety-related issues that arise in the review, as well as in later construction and operation of the plant." To insure that this potential advantage is realized, the staff's review of the GESSAR instrumentation and control systems will not be substantially complete until after the preliminary designs are completed and evaluated by GE's and evaluation of the designs are submitted for review by the staff.

Major portions of the designs for the instrumentation and control systems proposed in GESSAR are different from those utilized in previously licensed BWR plants. These items include the use of ganged control rods, a revised control rod position indication and detection system, a method of increasing the reactivity insertion rate following a scram, a revised rod pattern control system, and a solid state 2-out-of-4 protection system. The design criteria and the conceptual designs for the instrumentation and control systems are discussed in GESSAR but the preliminary designs for these systems have not yet been submitted for staff review. Therefore, our review of the instrumentation and control systems is not complete and will continue after the PDA is issued.

This section of the staff's safety evaluation report discusses the status of our review and is based on the information provided in GESSAR and on discussions with GE representatives. Our review has concentrated on assessing the adequacy of the conceptual design and the proposed design criteria and on identifying those areas which will require additional effort after the preliminary designs are available and submitted to the staff. In assessing the adequacy of the proposed design criteria and the conceptual designs, we have based our conclusions on the requirements of the Commission's General Design Criteria and applicable Regulatory Guides for Power Reactors.

Additional guidelines by the staff regarding the implementation of the Reference System option was published in a report (WASH-1341) entitled, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants," dated August 1974. As discussed in this report, the staff anticipated that a fairly heavy review effort would be necessary during the post-PDA phase. In accordance with these procedures, the staff concludes that the specification of an acceptable set of design criteria is adequate for the PDA and that the review and approval of the preliminary designs for the instrumentation and control systems can be conducted as post-PDA item. The staff also concludes that the proposed design of the new instrumentation and control system is within the state-of-the art and that there is reasonable assurance that the system can

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be designed to meet the system design criteria. For the GESSAR review, the original schedule called for the preliminary designs and the applicant's evaluation of those designs to be submitted to the staff in the first quarter of 1975 and the staff's detailed review was expected to be completed in the summer of 1975. GE did not submit all the preliminary design information for Chapter 7 in early 1975. All of the information we need to complete our review has not been received as of July 1975. As a result, we do not expect to complete our review of the design until early 1976 possibly later depending on receipt of the required information. The results of the review will be reported in a supplement to the GESSAR safety evaluation report, and presented to the ACRS for their consideration.

7.2 Reactor Trip System

The design of the reactor trip system is in a conceptual design phase and GE has not yet developed a preliminary design for the equipment to be used in implementing the proposed conceptual design. However, it is known that the GESSAR design will utilize equipment and logic that are vastly different from all previous BWR designs.

The monitored system variables from which the reactor trip system input signals will be derived are the same as the variables monitored on previous BWR plants. However, the sensors will be analog devices with control room readout indication rather than the previously used digital, non-indicating sensors. The only exceptions to this are the input signals derived from valve position switches and the scram discharge volume water level switches.

GE has stated that all protection system sensors that do not have analog readout indicators will be testable during reactor power operation. In addition, GE has stated that for all sensors that must be removed from service during such tests, the design will include provisions for automatic indication of these bypasses in accordance with Regulatory Guide 1.47. GE and the staff have also reached agreement on the design criteria to be used in establishing the range of instrumentation and in selecting trip set points for safety-related functions. These criteria are:

1) The range selection for instrumentation shall be such as not to exceed the expected range of the process variable being monitored.
2) The accuracy of all the safety trip points will not be numerically larger than the accuracy that was assumed in the accident analysis.
3) The trip set points should be located in the portion of an instrument's range which is most accurate and must be located in a region with the required accuracy.
4) All safety trip points will be chosen to allow for the normal expected instrument system set point drift such that the technical specification limit will not be exceeded.
5) Verification of the above criteria shall be demonstrated as a part of the qualification test program required by IEEE Std 323-1974.

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Although the monitored system variables are the same for GESSAR as on previous plants, the significance of the scrams initiated from detection of turbine stop valve closure and turbine control valve fast closure has changed. Previously, we understood that these scrams were non-essential or back-up functions and that scrams initiated from higher reactor pressure and high neutron flux were adequate to prevent exceeding safety limits. We now understand that a reactor trip initiated directly from sensing a turbine trip is essential in order to assure that established safety limits for normal operation and expected transients are not exceeded.

As noted in Section 4.2.3 of this SER, GE has under development a fast scram system to mitigate the consequences of a turbine trip event near the end of the fuel cycle. This design replaces the prompt relief trip for which the staff did not complete its review for GESSAR.

As stated earlier, the proposed conceptual design for the reactor trip system will utilize logic that is different from all previous BWR plant designs as shown in Figure 7-1 of this SER. There will be four identical divisional logic channels and each of these four channels will receive input signals from four sensors per monitored system variable. Each of the four sensors associated with each monitored variable provides an input signal to each of the four divisional logic channels through isolation devices. The divisional logic channels utilize "2-out-of-4" logic for each set of four input signals to generate a trip signal, i.e , when 2-out-of-4 signals for a given input variable exceed the trip set point, a divisional logic trip output signal is produced. The divisional logic output signals are the input signals for the actuator logics which control the electric power for the scram pilot solenoid valves. The actuator logics utilize "1-out-of-2 taken twice" logic to initiate a reactor trip by deenergizing the scram pilot solenoid valves. The conceputal design arrangement described above is illustrated in Figures 7.2-3a through 7.2-3f of GESSAR. The manual scram logic and back-up scram valve logic will be "1-out-of-2 taken twice" as in present BWR designs. The functional arrangement of the solenoid-operated pilot scram valves, the solenoid-operated back-up scram valves, and the air-operated scram valves will also remain the same as in current BWR plant designs.

In Amendment No. 24 to GESSAR, the conceptual design of the reactor trip system was again changed but we have not completed our review of these recent design changes. The most significant changes appear to be that the reactor trip system busses and power supplies were reclassified as non-Class IE and a feature was added to trip open the power supply breakers on a LOCA signal. We do not understand the purpose of this feature, the source of the LOCA signal, or the method of implementing this feature since it appears that opening the breakers will also result in deenergizing portions of the neutron monitoring system, the nuclear boiler instrumentation system and the process radiation monitoring system. Amendment No. 24 did not include any discussion of the design change for our evaluation.

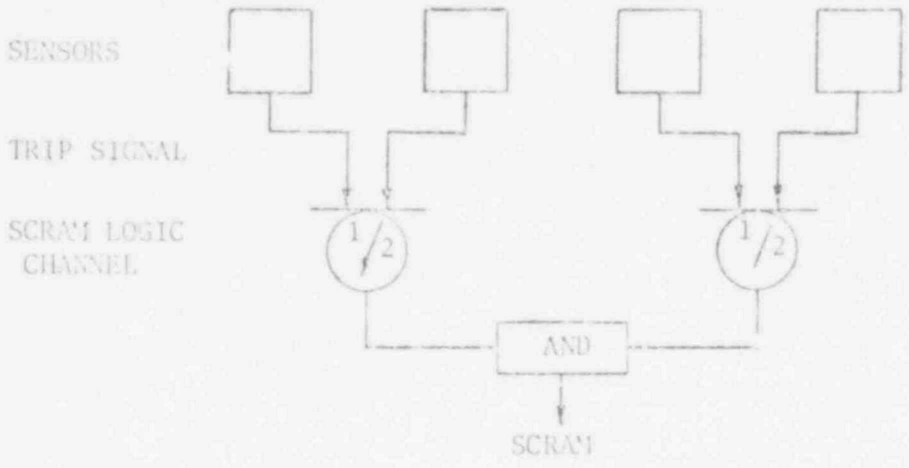

FIGURE 7-1a PREVIOUS GE DESIGN

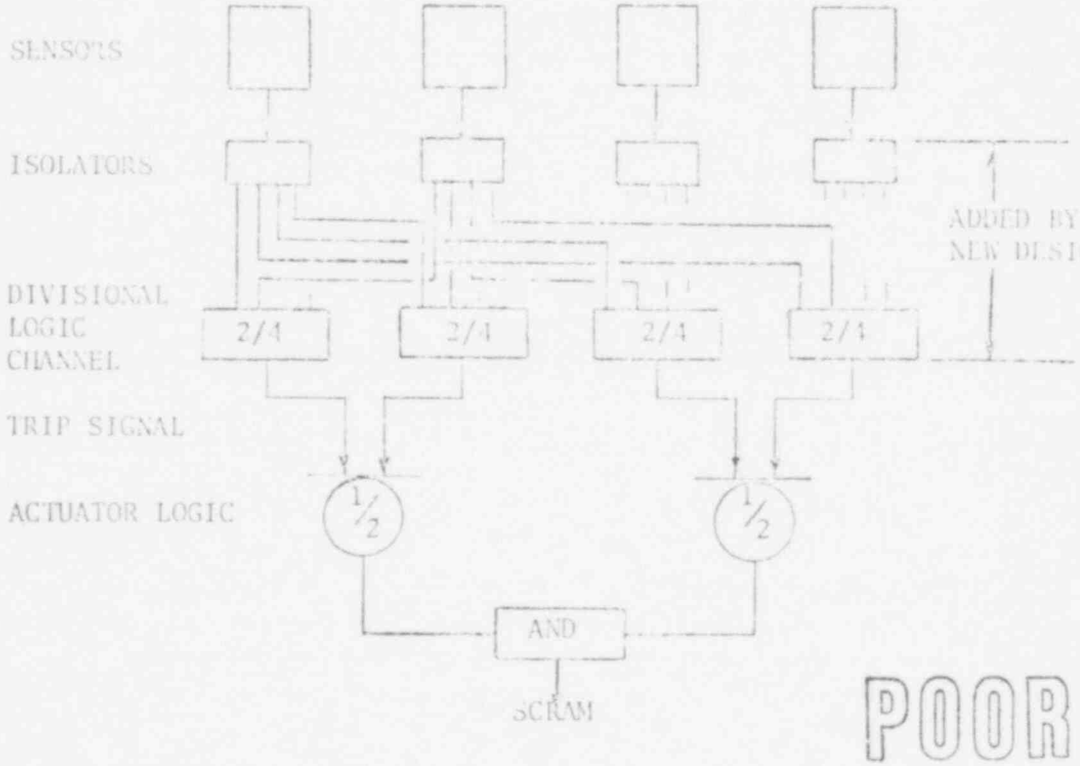

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FIGURE 7-1b BWR/6 DESIGN

FIGURE 7-1 REACTOR TRIP SYSTEM

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Another design change presented involves the conceptual design for bypassing the reactor trip from closure of the main steamline isolation valves. The design was changed such that this reactor trip signal can be bypassed by the reactor operator without regard to main steamline pressure. Previously the design proposed was such that the main steamline pressure signal also served to automatically remove the bypass when permissive conditions no longer existed. The design now proposed appears to be in violation of the requirements of IEEE Std 279-1971 which requires that the design include positive means to assure that operating bypasses are removed whenever permissive conditions are not met. We will report the results of our evaluation of these two recent design changes in a supplement to the GESSAR Safety Evaluation Report.

In response to our request, GE has described a pulse testing scheme that will be used for periodic testing of the reactor trip system as well as other portions of the new solid state protection system. Since a preliminary design has not yet been developed, it is not yet known which particular components will be included in the pulse testing. However, in response to a staff position, GE stated that the design will include provisions for manual testing to supplement the pulse testing provisions. We have concluded that the concept of a pulse testing scheme can form part of acceptable design provisions for periodic testing. When the preliminary design is developed and submitted for our review, we will perform an independent evaluation of the design to assure that the total periodic testing capability provides a means of duplicating, as closely as practical, the performance required to accomplish the safety function, e.g., a reactor trip. We will also assure that the requirements of IEEE Std 279-1971, IEEE Std 379-1972, General Design Criteria 21 and 22, and other applicable requirements are met. The results of that review will be reported in a supplement to the GESSAR Safety Evaluation Report.

Since a preliminary design of the reactor trip system has not been submitted, our review efforts have been directed principally toward understanding the proposed conceptual design and insuring that the proposed design criteria satisfy the Commission's regulations. With respect to the criteria listed or referenced in Figure 7.1-2 of GESSAR and identified by GE as applicable to the reactor trip system, we have concluded that the designs will be based on the application of current technology and the criteria form a generally acceptable basis for proceeding with development of a preliminary design and are acceptable for the PDA subject to the staff review and approval of a preliminary design. This aspect of the review will be conducted as a post-PDA item and the results will be reported in a supplement to the GESSAR Safety Evaluation Report.

7.3 Engineered Safety Feature Systems

7.3.1 Introduction

The design of the instrumentation and controls for the engineered safety feature systems, like the protection systems, is in a conceptual design phase. Although the functional performance required of the engineered safety feature systems is fundamentally the same as in previous BWR plant designs, the instrumentation and controls (i.e., the actuators, logic, and sensors) are not similar to any previous BWR designs.

Figure 7.1-1 of GESSAR identifies some of the engineered safety feature systems and Table 7.1-1 of GESSAR identifies some of the major areas of design changes.

Since a preliminary design of the instrumentation for the engineered safety feature systems is not yet available, our review has been directed at evaluating the proposed design criteria and the conceptual design described in GESSAR. The status of our review of the instrumentation and controls for the engineered safety feature systems is discussed in subsequent sections of this report.

The results of our evaluation presented in Section 7.2 of this report with regard to the use of analog sensors, the criteria for instrument range and trip point selection, and the adequacy of the periodic testing provisions for the solid state protection system are also applicable to the instrumentation for the engineered safety feature systems.

We have concluded that, except as discussed in Section 7.3.6 of this report, the criteria listed in Figure 7.1-2 of GESSAR and identified by GE as applicable to the engineered safety feature systems form a generally acceptable basis for proceeding with the development of preliminary designs subject to the detailed staff review and approval of the preliminary designs following the issuance of the PDA. In addition, in response to the staff's position, GE has committed to designing all essential auxiliary supporting systems (e.g., essential service water) in accordance with the same criteria used in designing the engineered safety feature systems they support. We have concluded that proposed design criteria for the engineered safety feature systems and their essential auxiliary supporting systems are acceptable for the PDA, subject to the detailed staff review and approval of the preliminary designs. The review of the preliminary designs for these system will be conducted during the post-PDA review phase. The results of that review will be reported in a supplement to the GESSAR Safety Evaluation Report.

7.3.2 Emergency Core Cooling Systems

7.3.2.1 High Pressure Core Spray System

The high pressure core spray (HPCS) system is automatically started by either low reactor vessel water level or high drywell pressure signals. To preclude the possibility of flooding the main steam lines through the inadvertent actuation of the HPCS, system logic has been added which automatically terminates HPCS flow on high water level but it will be effective only if no high drywell pressure signal exists. This change provides diverse initiation signals since failure of the reactor vessel water level signals will not prevent initiation of HPCS in the event of high drywell pressure. It also incorporates a feature, desired by GE for operational reasons, to terminate flow when HPCS may be inadvertently actuated because any inadvertent actuation is expected to be caused by low vessel water level rather than high drywell pressure.

The GE design also incorporates an interlock on the pump suction valve from the suppression pool and on the two valves in series in the test line to the condensate storage tank (valves MO F015, MO F010, and MO F011, respectively). GE stated that the purpose of this interlock is to maintain the quality of the water in the condensate

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storage tank. The interlock is effective only for manual control (at the component level) of the suppression pool suction valve. The interlock is not effective for either automatic initiation or manual initiation (at the system level) of the HPCS system. Therefore, failure of the interlocks cannot result in loss of the safety function. In this manner, the design incorporates a feature which precludes contamination of the condensate storage tank, a feature which is operationally desirable to GE, without degrading the reliability of the HPCS safety functions.

We have concluded that the proposed design criteria and the conceputal design for the HPCS system are acceptable for the PDA subject to the detailed staff review and approval of the preliminary design. We will report the results of our review of the preliminary design in a supplement to the GESSAR Safety Evaluation Report.

7.3.2.2 Automatic Depressurization System

The design of the automatic depressurization system (ADS) is identical in concept to previous BWR plant designs. We have concentrated our review of the conceptual design on attempts to resolve long-standing generic issues. At this time, the design incorporates no new features to improve testability of the ADS pilot solenoid valves. In addition, the proposed conceptual design does not meet the proposed design criterion that no single failure shall result in inadvertent opening of more than one relief valve. GE has stated that they are "...currently studying the effects of relief valve blowdown for various operating states." Based on discussions with GE, we understand that one goal of these studies is to demonstrate that inadvertent actuation of the ADS has acceptable safety consequences and therefore that the design criterion discussed above is unnecessary.

For the reasons discussed above and the fact that a preliminary design has not yet been submitted, we have not completed our review of the ADS. We discussed the subject of testability with GE and the alternatives that would be acceptable to the staff pending completion of the GE study of the effects of inadvertent actuation of the safety/relief valves. As a result of these discussions, it was agreed that if the study results show that the consequences of inadvertent actuation of the ADS are unacceptable, the design will be revised such that no single failure can result in opening of more than one safety/relief valve. We have concluded that this commitment is acceptable for the PDA and that if design changes are necessary they can be reviewed as a post-PDA item. We will report the resolution of this item in a supplement to the GESSAR Safety Evaluation Report.

With respect to periodic testing provisions in the ADS design, we have informed GE that it is our position that for the GESSAR design, improvements in the testability of the ADS are required. We have concluded that the design must include provisions for testing the pilot solenoid valves which control compressed air to the safety/relief valves but that the relief valves themselves need not be tested during reactor operation. In Amendment No. 24 to GESSAR, GE stated that a study is being made of methods to improve testability. We have concluded that identification of this development program is acceptable for the PDA subject to staff review and approval of the preliminary design for the ADS, including the design provisions for testing. This aspect of the review will be conducted as post-PDA item and the results will be reported in a supplement to the GESSAR Safety Evaluation Report.

7.3.2.3 Low Pressure Core Spray and Low Pressure Coolant Injection Systems

The conceptual design of the instrumentation and controls for the Low Pressure Core Spray (LPCS) system and the Low Pressure Coolant Injection (LPCI) mode of operation of the Residual Heat Removal (RHR) system are identical to previous BWR plant designs in their functional arrangement. There are many changes in the conceptual design of the instrumentation and controls used for other operating modes of the RHR system. These changes are discussed elsewhere in this report and are necessitated by the major changes in containment design.

The low pressure ECCS are divided as in previous plants with RHR Loop 'A' and LPCS comprising one division and RHR Loops 'B' and 'C' comprising the second division. The initiation logic is illustrated in Figure 7.3-3 and 7.3-8 of GESSAR. Two low reactor vessel water level signals or two high drywell pressure signals will initiate operation of the low pressure systems. Separate sensors will be used for each low pressure division and a third set of sensors will be used for HPCS which is the third ECCS division.

Since a preliminary design of the hardware to be used in implementing the concept is not yet available, we have concentrated on assuring that generic items will be resolved on GESSAR.

The conceptual designs of both the LPCS and LPCI systems include features that would prevent opening the injection valves until "the differential across the valves is reduced to a differential pressure equal to rated reactor vessel pressure minus discharge pressure of the ECCS loop with zero flow into the vessel," (GESSAR Page 7.6-83a). In response to our request, the applicant stated that "This feature reduces the size and power of the valve operation (sic) required. The reduction in valve stem and mechanical drive loads result in a more reliable valve drive train which is less susceptible to wear on repeated opening and closing during surveillance testing." In view of the proposed conceptual design, we do not fully understand this response. The differential pressure interlock is effective on both the manual signal used for surveillance testing as well as the automatic initiation signal used in the event of a LOCA. In addition, administrative procedures in the form of surveillance testing procedures could be used in lieu of an interlock to insure testing is not routinely conducted at high pressures. In response to a further request for additional information, the applicant stated that "General Electric is, however, considering deletion of these (differential pressure) switches and replacing them with switches which would not permit injection valves to open unless reactor pressure is below system design pressure as required by the current draft of ANSI N193." ANSI N193 is a draft standard currently in preparation by ANS Working Group No. 55.4. It has neither been approved by that committee nor accepted by the staff as a licensing basis. In any event, a recent draft, 1st Draft-Rev 1 June 1974, identified several proposed methods of providing

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isolation of low pressure systems connected to the reactor coolant pressure boundary. However, no particular method is required over another. In fact the draft indicates that its proposed criteria should be used in conjunction with other requirements that must be considered in implementing the system requirements.

During a meeting with GE in October 1974, it was agreed that GE would modify the LPCS and LPCI designs to provide diverse initiation signals that are not dependent on a non-diverse interlock. In conjunction with the changes that are necessary to accomplish this, GE will evaluate the new design to assure that adequate protection against high pressure is provided for the low pressure portion of these systems. These commitments were documented in Amendment No. 24 to GESSAR. We have concluded that these commitments are acceptable for the PDA. We will review the methods used to implement these commitments as post-PDA items and will report the results of our review in a supplement to the GESSAR Safety Evaluation Report.

7.3.3 Containment and Reactor Vessel Isolation Control System

The proposed conceptual design for the containment and reactor vessel isolation control system (CRVICS) is not similar to any previous BWR plant. The main steam line isolation control system will utilize different equipment and a different functional arrangement. Four instrument channels will be provided for each measured variable. The measured variables are the same as those used on previous plants. The instrument channels will be combined using "2-out-of-4" logic in each of four divisions. The four divisions will be arranged such that two divisions control the outboard isolation valves and the other two divisions control the inboard valves. Both divisions controlling a set of valves must trip to initiate closure of the main steam line isolation valves. The control arrangement for the main steam line isolation valves is illustrated in Figure 7.3-13 of GESSAR. The addition of a second air-operated valve in parallel with a similar valve provides two flow paths to vent compressed air from the operating piston of the main steam line isolation valve. However, the design includes no provisions to test the two valves independently.

We are currently reviewing the topical report APED-5750, "Design and Performance of General Electric Boiling Water Reactor Main Steam Isolation Valves," which was designated as an applicable reference in Amendment No. 24 to GESSAR. During a meeting with GE in October 1974, the staff and GE agreed that the review of the new control arrangement for the main steam isolation valves would be conducted on the topical report rather than as part of the GESSAR review. We have concluded that this is acceptable for the PDA. We will report the results of our review of the topical report APED-5750 and their applicability to GESSAR in a supplement to the GESSAR Safety Evaluation Report.

Other portions of the CRVICS will utilize logic similar to previous designs, i.e., 2-out-of-2 low vessel water level or high drywell pressure signals. We have concluded that these conceptual designs are acceptable for the PDA subject to the detailed staff review post-PDA and approval of the preliminary designs. This aspect of the review will be conducted as a post-PDA item and we will report the results of our review in a supplement to the GESSAR Safety Evaluation Report.

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7.3.4 Essential Service Water System

GE has not supplied all of the information requested in Regulatory Guide 1.70, "Standard Format of Safety Analysis Reports for Nuclear Power Plants." A preliminary diagram of the essential service water (ESW) system is provided in Figure 9.2-1 of GESSAR. The preliminary design of the instrumentation and controls is not available at this time. However, GE has stated that the ESW system will be initiated by the engineered safety feature actuation circuitry of the solid state protection system. The ESW instrumentation, control and power supplies will be separated into three divisions as are the engineered safety feature systems. As stated earlier in this report, GE has committed to designing all essential auxiliary supporting systems, such as ESW, in accordance with the same criteria applied in the design of the supported safety systems such as ECCS.

We have concluded that this commitment is acceptable for the PDA subject to staff review and approval of the preliminary design. This aspect of the review will be conducted as a post-PDA item and we will report the results of our review in a supplement to the GESSAR Safety Evaluation Report.

7.3.5 Flammability Control System

The design proposed for the control of combustible gas concentration in containment is based on mixing the drywell and containment atmosphere and the use of hydrogen recombination equipment when the atmosphere in either the drywell or containment approaches 4 percent hydrogen by volume as discussed in Section 6.2.5 of this SER.

Redundant mixing systems which are physically separated are provided. The applicant has stated that the instrumentation and controls of the mixing system are designed to meet IEEE Std 279-1971 requirements, but has not supplied an analysis to support this. No design details are provided for the instrumentation and control of the recombination equipment. However, the applicant has committed to provide equipment which will meet the requirements of IEEE Std 279-1971 and 308-1971.

Both mixing and recombination sub-systems are manually initiated as a result of high hydrogen levels indicated by the containment atmosphere monitoring system instrumentation. The applicant has provided the results of an analysis which shows that the mixing system will not be required until approximately 8 hours after a LOCA and that a recombiner would not be required until approximately 15 days after a LOCA. Manual actuation of the mixing and recombination subsystems is acceptable based on the time available before the operator needs to take action, and the redundant and alarmed instrumentation available to monitor the hydrogen concentrations.

The criteria for instrumentation and control circuit design, fabrication, and testing satisfy the staff's requirements for the PDA stage of review and are acceptable.

7.3.6 Standby Gas Treatment System

The standby gas treatment system (SGTS) will be automatically initiated by any one of the following signals:

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a) LOCA signal

b) Auxiliary building ECCS pump room high radiation

c) Shield building high radiation

d) Fuel building high radiation

e) Containment pressure control exhaust high radiation

f) Drywell bleed-off pressure line open

We have reviewed the description of the proposed design of the instrumentation and controls for the SGTS, the simplified functional control diagram provided in Figure 7.3-20 and the proposed design criteria identified in Figure 7.1-2. We have concluded that the proposed design criteria are unacceptable because IEEE Std 308-1971 (as modified by Regulatory Guide 1.32), Regulatory Guide 1.6 and General Design Criteria 17 and 18 are not included. These criteria have been identified by GE as applicable to other engineered safety feature systems and we have concluded that they are equally applicable to the SGTS. We will report the resolution of this item and the results of our detailed post-PDA review of the preliminary design in a supplement to the GESSAR Safety Evaluation Report.

7.3.7 Suppression Pool Makeup System

The suppression pool makeup system (SPMS) transfers water from the upper containment pool to the suppression pool following a loss-of-coolant accident. The conceptual design for the instrumentation is illustrated in Figure 7.3-25 of GESSAR. The dumping of the upper pool is initiated automatically by either a LOCA signal in coincidence with 1-out-of-2 low suppression pool water level signals or by a 30-minute timer that is started by a LOCA signal. These initiation signals are interlocked with the reactor mode switch such that these signals are blocked when the mode switch is in the refueling position. The purpose of this interlock is to prevent dumping the upper pool when fuel assemblies are being moved in the upper pool during refueling. The dumping of the upper pool may also be initiated manually.

There will be two independent instrumentation systems designed as described above. One system controls the two normally closed valves in series in one dump line and the other system controls the valves in the second dump line.

We have reviewed the proposed conceptual design, the proposed design criteria and the applicant's evaluation of the effects of inadvertent dump of the upper pool. We have concluded that the proposed design is not susceptible to single failures that could cause inadvertent dumping and that the proposed design criteria and conceptual design are acceptable for the PDA. We will report the results of the review of the preliminary design of the suppression pool makeup system in a supplement to the GESSAR Safety Evaluation Report after issuance of the PDA.

7.3.8 Containment Spray System

The criteria to be used in designing the instrumentation and controls for automatic initiation of containment spray will be identical to criteria used for other engineered safety feature systems as discussed in Section 6.3.1 of this SER.

We have concluded that the proposed design criteria committed to by GE are generally acceptable for proceeding with the development of a design and are acceptable for the PDA. We will report the results of our detailed post-PDA review of the preliminary design in a supplement to the GESSAR Safety Evaluation Report.

7.3.9 Indication of Bypasses

The applicant has stated that provisions will be made for indication of bypassed or inoperable status conditions of the safety systems of the station. The applicant has not provided a design for these provisions, but has stated his commitment that they will conform to the provisions of Regulatory Guide 1.47. We conclude that this commitment is acceptable and that there is reasonable assurance that the applicant can develop an indication system that conforms to the provisions of Regulatory Guide 1.47.

7.4 Safe Shutdown Systems

7.4.1 Reactor Core Isolation Cooling System

The reactor core isolation cooling (RCIC) system provides coolant inventory makeup during reactor shutdown and is activated in time to preclude conditions that lead to a need for the ECCS. In addition to being classified as a safe shutdown system, the RCIC is also classified as an engineered safety feature because, together with the HPCS, it provides the protection necessary in the event of a rod drop accident.

The RCIC system actuators, logic and sensors function differently than previous plants. The system is initiated by low reactor vessel water level signals utilizing a "1-out-of-2 taken twice" logic. GE has stated that the RCIC system will be designed in accordance with all the criteria and design requirements applicable to an engineered safety feature as shown in Figure 7.1-2 of GESSAR.

We have concluded that the proposed design criteria for RCIC are acceptable for the PDA subject to detailed staff review and approval of the preliminary design. This aspect of the review will be conducted as a post-PDA item and we will report the results of our review in a supplement to the GESSAR Safety Evaluation Report.

7.4.2 Standby Liquid Control System

GE has stated that the standby liquid control system (SLCS) is identical to the design used in the Zimmer Nuclear Plant.

Since a preliminary design is not yet available, we have not completed the review necessary to assure resolution of the generic problems associated with the interlock which prevents operation of both pumps. The concern is that improper implementation of this interlock could result in a design such that a single failure could disable both pumps. We will report further on this aspect of the design in a supplement to the GESSAR Safety Evaluation Report.

The design of the SLCS and the reactor water cleanup (RWCU) system includes an interlock designed to isolate the RWCU system when the SLCS is initiated. We have concluded that this interlock can be disabled by a single failure. In Amendment No. 24 to GESSAR, GE stated that a study has been conducted to evaluate the effects of simultaneous operation of the SLCS and the RWCU system. GE stated that the analysis indicates that under such conditions, the SLCS will continue to accomplish its intended safety function with substantial margin. On the basis of these statements, we agree with the conclusion reached by GE that interlock design changes are not needed.

We have reviewed the proposed design criteria for the SLCS listed in Figure 7.1-2 of GESSAR. We have concluded that these criteria are acceptable for the PDA subject to staff review and approval of the preminary design. This aspect of the review will be conducted as a post-PDA item and we will report the results of our review in a supplement to the GESSAR Safety Evaluation Report.

7.5 Safety-Related Display Instrumentation

This section of GESSAR presently does not completely satisfy the information specified in Regulatory Guide 1.70. In response to our request that GE be required to eliminate obsolete information, GE deleted certain figures in GESSAR and stated that the main control board arrangements for BWR-6 would be submitted in the first quarter of 1975. During a meeting with GE in October 1974, it was agreed that GE will provide specific information such as lists of the indications and controls to be provided and the physical arrangements of control board panels and that we will review this information during the post-PDA phase.

In the course of our review we have also reached agreement with GE on the design criteria for some of the safety-related display instrumentation. The two major areas on which agreement was reached are:

(1) Post-accident monitoring and safe shutdown display instrumentation will be qualified for the accident environment, will utilize redundant channels with at least one channel recorded, will be capable of being energized from onsite emergency power supplies, and will be designed in accordance with IEEE Std 279-1971. GE also agreed that the indicators and recorders will be designed to function satisfactorily following (but not necessarily during) the safe shutdown earthquake without any maintenance or repair following the earthquake.

(2) Safety-related display instrumentation which is used to indicate the need for manual action by the reactor operator will be designed in accordance with protection system criteria if the accident analysis takes credit for correct performance of that manual action.

We have concluded that these agreements and the information pertaining to safety-related display instrumentation presently contained in GESSAR are an acceptable basis for the PDA. We will report the results of our review of the specific information which will be submitted in 1975 in a supplement to the GESSAR Safety Evaluation Report.

7.6 All Other Instrumentation Systems Required for Safety

7.6.1 General

The systems discussed in this section of this report are:

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a) Refueling interlocks
b) Reactor vessel instrumentation and controls (excluding those used for safety systems, engineered safety features and control systems which are discussed in other sections of this report)
c) Process radiation monitoring system
d) Area radiation monitoring system
e) Reactor water cleanup system
f) Leak detection system
g) Process computer system
h) Containment atmosphere monitoring system
i) Neutron monitoring system
j) Fuel pool cooling and cleanup system

We have reviewed the information provided in Table 7.1-1 which describes the changes in the designs compared to previous plants. We have concluded that the identified differences are primarily in the electronic equipment used or are attributable to the larger plant size described in GESSAR. On this basis we have concluded that these systems are acceptable for the PDA. It is intended that the safety evaluation report for a standard design application be a self-sufficient document in order to provide a firm and clear baseline for utility application reviews in the future. Therefore, we will continue our review of these systems during the post-PDA review phase without reliance on the comparison with previously approved designs. We will report the results of our evaluation of the preliminary designs in a supplement to the GESSAR Safety Evaluation Report.

7.6.2 Reactor Pressure Relief Instrumentation

In Amendment No. 23 to GESSAR, GE provided additional information pertaining to the effects of the reduction in the number of relief valves to be provided for the GESSAR standard design. Initially the proposed design had 22 relief valves for the reactor coolant system. The design now proposed has 19 valves. Although the relief capacity per valve has been increased, the reduction in the number of valves resulted in less total relief capacity. The net effect of this design change is that the safety analysis now takes credit for 50% of the safety/relief valves opening at their lower relief set points. The previous design relied on all valves (except one which was assumed failed) opening at their higher safety valve spring set point pressure. In previous designs the instrumentation used to open the valves at the relief set point was classified as non-safety. Now this instrumentation must be classified as a protection system since it is relied on to initiate opening of the valves at their relief set point.

We have not completed our review of this instrumentation. However, in Amendment No. 23 to GESSAR, GE stated that this instrumentation will be upgraded to provide redundancy and independence equal to that required for protection systems, e.g., ADS. We have concluded that this commitment is acceptable for the PDA. We will report the results of our review of the specific design criteria and the preliminary design for this instrumentation in a supplement to the GESSAR Safety Evaluation Report.

7.7 Control Systems

7.7.1 Reactor Manual Control System (RMCS)

GE prov ed the following description of the function of the Rod Pattern Control System and its provisions for ganged rod withdrawal. This information was provided in a letter from John A. Hinds, Manager, Safety & Licensing, General Electric Company to John F. Stolz, Directorate of Licensing, USAEC, dated December 28, 1973.

"The purpose of the rod pattern control system is to limit the worth of any control rod such that no undesirable effects will result from a rod drop accident. The rod pattern control system will enforce operational procedural controls by applying rod blocks before any rod motion can produce high worth rod patterns.

A. System Description - Definition of Terms

1. The rod pattern control system (RPCS) is a dual channel system using like components in each channel. The control logic for the RPCS is contained in a logic device such as the processor portion of a mini computer. This logic device, has, in permanent storage, the identification of all rod groups and logic control information required to prevent high worth rods. The logic device is hardwired and is not to be site programmable, except through controlled engineering design change.

2. There is a logic device for each channel. The logic for these two devices is the same and both channels receive the same data inputs but from different sources.

   a) There is a dual rod position probe for each drive. Two sets of reed switches are provided for rod position information and provides, through different connectors, inputs to different rod position multiplexer cabinets.

   b) Two different rod position multiplexer cabinets are provided, one for each channel. These cabinets transmit rod position data to the rod position information cabinets. A rod position information cabinet is provided for each channel. These cabinets decode the multiplexed data and provide rod position data to the RPCS logic devices for all rods. Rod position is the primary data input for RPCS.

   c) Other inputs to the RPCS logic devices include reactor power level mode of operation, identification of selected rod, drive mode requested by the operation (sic) and special modes of operation such as shutdown margin tests.

   d) A means of comparing the outputs of the RPCS logic devices provide a way of monitoring the performance of the two channels. Both channels must be operable and with identical outputs before rod motion is permitted. Failed comparisons and logic device failures are indicated in the control room.

   e) RPCS outputs are transmitted to the two activity control sections of the Reactor Manual Control System in the form of a rod select and drive permissive interlock. The two RPCS channels provide inputs separately to the two separate activity controls. These two inputs are then treated as other rod block interlocks and are further compared to the analyzer portion of reactor manual control.

   f) An addition to the existing full core status display, a continuous full core rod position display is provided from one of the rod position system cabinets. A new display is provided which will show the LPRM values for all the rods in any RPCS rod group.

B. System Operation

a) From 0% power and with all rods full in to 50% rod density, only rods in the same sequence can be moved. Once a sequence is selected and rods moved, startup must proceed in the chosen sequence, either A or B. Up to 50% rod density a rod may be continuously withdrawn or notched out.

b) From 50% rod density to 25% reactor power level, only rods in a defined group may be moved. The rods in a given group are symmetrically distributed in the core and permanently identified in the RPCS. The position of the group rods must remain within 1 notch or a rod block will occur. All rods in the core will belong to a group which typically numbers from 2 to 8 rods. Rods in a group must therefore be notched out in rotation until all group rods are again in the same position.

c) Above 25% power level the RPCS is no longer needed because no high worth rod pattern can be established. This is due to the core void fraction established. The RPCS is therefore switched off. From this point up to 100% power, the Rod Block monitor is in service to limit flux peaking.

d) Coming down in power the above sequence of operation occurs but in reverse order. At 35% reactor power level decreasing, an alarm will be initiated to alert the operator of pending RPCS constraints. At this point, the operator must get into proper sequence for shutdown.

C. Allowable System Bypasses

a) Upon failure of one channel of the RPCS the failed channel may be taken out of service. By procedural control the output of the remaining good channel would then be input to both activity controls of reactor manual control.

b) At time of startup or shutdown both channels of RPCS must be operable.

c) A means for bypassing failed reed switches will be provided for unique failures.

d) A means for handling drives which are valved out of service will be provided.

The purpose of ganged rod withdrawal is to facilitate startup or shutdown between 50% rod density and 25% power.

A. Description of System and Operation

1. At present each rod has its own binary code identification determined by identification cards mounted on the transponder card. To incorporate ganged rod withdrawal, each rod has two identifications--its unique identification and its group identification.

   a) The above group identification contains the same identical rods as used in the RPCS group for that rod.

   b) Reactor manual control system is modified to give group rod identification commands as well as single rod commands. An operator action is required to move rods in the ganged rod withdrawal mode.

   c) Using the whole core rod position display it is possible to monitor the motion of all rods in any rod group. The LPRM strings associated with the rods in the rod group are displayed on a new separate control room display.

2. Operation in the ganged rod mode allows continuous withdrawal of an RPCS rod group above 50% rod density and below 25% power.

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a) The requirement for all rods in a group to be within ±1 notch position is still in effect. A rod block will occur whenever any rod in a rod group is more than ±1 notch out of step. The out of step rod may be operated in the single notch mode to correct the rod pattern fault. Upon correction of the 1 notch block, ganged rod withdrawal may be then initiated."

Although GE classifies the reactor manual control system as a power generation system and non-essential to safety, GE has stated that the system shall meet the following safety design bases:

(1) The circuitry provided for the manipulation of control rods shall be designed so that no single failure can negate the effectiveness of a reactor scram.
(2) Repair, replacement or adjustment of any failed or malfunctioning component shall not require that any element needed for reactor scram be bypassed unless a bypass is normally allowed.
(3) The reactor manual control system instrumentation and controls are designed in accordance with the specific regulation requirements shown in Figure 7.1-2 of GESSAR.
(4) Inhibit control rod motion whenever instrumentation is incapable of monitoring core response to rod movement.
(5) Inhibit control rod withdrawal in time to prevent local fuel damage as a result of erroneous control rod manipulation.

GE has also documented its commitment to comply with our position that these portions of the RMCS which provide safety functions must be designed in accordance with all requirements applicable to a protection system. We have concluded that the safety design bases above and this commitment are acceptable for the PDA. After GE submits a preliminary design for the equipment for the RMCS, we will review that design to assure that the safety design bases above have been implemented in accordance with protection system design criteria. We will report the results of our evaluation in a supplement to the GESSAR Safety Evaluation Report.

7.7.2 Other Control Systems

GE has stated that the following control systems do not have a safety design basis: the Feedwater Control System and the Pressure Regulator and Turbine-Generator Controls. GE has also stated that the effects of failures in these control systems have been analyzed and that the potential consequences of such failures are acceptable.

When the preliminary design of the safety systems is submitted for review, we will review the interaction between these control systems and the safety systems to assure that the requirements of IEEE Std 279-1971, Section 4.7 are met.

Other control systems do have a safety design basis identified by GE. These systems and their safety design bases are:

(1) Recirculation Flow Control System - The recirculation flow control system shall function so that no abnormal operational transient resulting from a malfunction in

the recirculation flow control system can result in damaging the fuel or exceeding nuclear system pressure.

(2) Gaseous Radwaste Control System - The safety objective of the gaseous radwaste system is to process and control the release of gaseous radioactive waste to the site environs so that the total radiation exposure to persons outside the controlled area is as low as practicable and does not exceed applicable regulations.

(3) Liquid Radwaste Control System - The safety objective of the liquid radwaste system is to control the release of liquid and solid radioactive waste material to the environs and to package these wastes in suitable containers for offsite shipment and burial.

We have concluded that this information is acceptable for the PDA. After receipt of a preliminary design for this instrumentation, we will review the design to determine that the safety design bases have been implemented in conformance with appropriate safety system design criteria, including IEEE Std 279-1971, Section 4.7, Control and Protection System Interaction. We will report the results of this post-PDA review in a supplement to the SER. (Refer to Section 11 of the SER for additonal information on our evaluation of the radwaste systems.)

7.8 Instrumentation Interfaces with Balance of Plant Systems

As discussed in Section 1.8.2 of this SER related to interfaces, all interface conditions with the remainder of the plant must be clearly identified and specified. With respect to the instrumentation, control and electric power systems, we have notified GE that we will need a list of systems or components which are not within the "nuclear island" but which are necessary to support the conclusion that the "nuclear island" systems are acceptable. For each such system, the GESSAR application must specify the criteria and design bases which must be met by the balance of plant systems to insure that the nuclear island systems perform acceptably.

The goal of our review is to insure that these interface requirements are sufficiently specific in order to preclude the need to re-evaluate the GESSAR design on specific plant applications utilizing the GESSAR design.

Since the preliminary designs for several instrumentation systems have not yet been developed, we have been unable to proceed to a review of the interface requirements for the balance of plant system. We will report the results of this effort in a supplement to the GESSAR Safety Evaluation Report post-PDA.

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8.0 ELECTRIC POWER SYSTEMS

The design of the electric power systems is primarily the responsibility of the applicant submitting a utility application for a construction permit or submitting a balance-of-plant standard design application for a PDA. We originally concluded that much of the information contained in Chapter 8 of GESSAR was typical of the information to be supplied by future applicants, and was not within the scope of the GESSAR nuclear island review. GE recently informed us that it was their intent that Section 8.3 of GESSAR be included in the nuclear island review. We will pursue our detailed review of this section with GE during our post-PDA evaluation of the details of the hapter 7 designs. Additional specific information in the interface area will be provided by GE for our review during the Final Design Approval (FDA) review phase for GESSAR. Our evaluation of GESSAR onsite electrical power for the PDA is discussed below.

We have reviewed the proposed design criteria for the standby power instrumentation and control systems. We have concluded that the criteria listed in Figure 7.1-2 of GESSAR form a generally acceptable basis for developing a design for the electric power systems on any plant referencing GESSAR in its construction permit application. The conceptual design for the electric power systems provides for a three division arrangement for both the a-c and d-c power systems. We have concluded that this arrangement is compatible with the functional requirements of the engineered safety feature systems which also have three divisions and that this is in accordance with Regulatory Guide 1.6 and is, therefore, acceptable.

The HPCS onsite power supply is also within the scope of GESSAR. GE has referenced the topical report NEDO-10905, "High Pressure Core Spray System Power Supply Unit," May 1973. We are reviewing this topical report separately from the GESSAR application. The instrumentation and controls for the HPCS power supply system are being reviewed on the GESSAR docket and are discussed in GESSAR Section 7.3. As stated in Section 7.3.2.1 of this report, the preliminary design for the HPCS instrumentation will be reviewed during the post-PDA phase and we will report the results of our review in a supplement to the GESSAR safety evaluation report.

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9.0 AUXILIARY SYSTEMS

The scope and description of the auxiliary systems are presented in Section 9.0 of GESSAR. In the course of our review, we have primarily focused our attention on the design of the safety-related systems, their associated interfaces with the Balance of Plant (BOP), and the manner in which these objectives have been achieved.

The auxiliary systems within the scope of the nuclear island that are necessary to assure safe plant shutdown include the Nuclear Island Essential Service Water System, Control Building Chilled Water System, Control Room Air Conditioning System and the onsite standby diesel generator subsystems (i.e., diesel generator compartment ventilation systems, fuel oil storage and transfer system, cooling water system, starting air systems, and lubrication systems)

The nuclear island systems necessary to assure safe handling of fuel and adequate cooling of the spent fuel include new and spent fuel storage systems, the fuel pool cooling and cleanup system, the fuel handling system and a portion of the fuel building ventilation system.

We have reviewed those nuclear island auxiliary systems whose failure would not directly prevent safe shutdown but could, directly or indirectly, be a potential source of radiological release to the environment. These systems include: compressed air system, demineralized water makeup system, closed cooling water system, auxiliary building ventilation system, process sampling system, portions of the equipment and floor drainage system, and main steam line isolation valve leakage control system.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

The description and evaluation of the new dry fuel storage vault, Section 9.1.1 of GESSAR, in conjunction with the general arrangement drawings Figures 1.2-2 through 1.2-8 were reviewed as the basis for our evaluation.

The general arrangement drawings indicate the new fuel storage facility will be housed in the Fuel Building. The Fuel Building and the Auxiliary Building surround the Reactor Building and all form a portion of the Reactor Island Facility. The storage racks will be designed such that the spacing between fuel elements will assure K_{eff} will not exceed 0.90 when fuel is stored dry nor result in K_{eff} exceeding 0.95 should the vault be flooded. To prevent the accumulation of water in the vault, a drain will be provided to direct any entering water to the equipment drain subsystem. The storage capacity and number of storage racks will vary depending upon the design cycle time of the fuel. If the applicant selects an 18 month fuel cycle, the new fuel storage facility will be capable of containing 42 percent of

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full core fuel or 308 fuel assemblies. If a 12 month fuel cycle is selected, storage capacity equal to 30 percent of a full core or 220 fuel assemblies will be provided.

The fuel elements are loaded into the racks by lowering them through holes in the top of the rack using a general purpose grapple and the five ton general purpose building crane. The racks are designed to meet seismic Category I requirements. Handling of any other loads above the stored new fuel other than a new fuel assembly shall be prohibited by administrative controls. Procedural controls will limit the handling to one fuel assembly at a time and at a height not to exceed 2 feet above the top of the racks.

The height of the storage racks is such that the fuel bundles extend above the top of the racks thereby assuring that the grapple cannot engage the fuel rack and accidentally impose uplift forces on the rack castings during the handling of fuel. Hold down bolts are used to restrain the racks and maintain their spacing during the SSE.

No considerations have been given to the concerns relating to sharing of the storage facility since only one unit is considered in this Safety Evaluation Report.

Based on the arrangement concept presented in Figure 1.2-2 through 1.2-8 and our evaluation of the design, we conclude that the criteria and bases for the new fuel storage facility design are acceptable.

9.1.2 Spent Fuel Storage

The description and evaluation of the spent fuel storage facility, Section 9.1.2 of GESSAR and the general arrangement drawings, Figures 1.2-2, 1.2-3, 1.2-4, 1.2-7, 1.2-8 and Figure 9.1-2 were reviewed as the basis for our evaluation.

The general arrangement drawings indicate the spent fuel storage facility is housed within the fuel building.

The top entry spent fuel storage racks will be designed to maintain space geometry that precludes criticality under normal and abnormal conditions. A full fuel array of spent fuel in the storage racks will be maintained to less than a K_{eff} of 0.90 with a normal water level of 25 feet of water above the stored fuel. Under abnormal conditions, such as an accidental dropping of equipment or other events causing horizontal movement of the fuel, K_{eff} will not exceed 0.95.

The racks are bolted to the rack support structure in a fashion to faci[illegible] rack rearrangement or replacement without draining the pool. The holddown bolts maintain the minimum spacing of the racks for geometric reactivity control. Each rack is capable of storing 10 fuel assemblies. The design precludes the accidental insertion of fuel bundles between racks and will meet seismic Category I requirements.

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The spent fuel storage pool will have sufficient storage rack capacity for 117 percent of one full core fuel load, if the applicant selects an 18 month fuel cycle and 105% of a full core if a 12 month fuel cycle is selected. In either case, temporary storage space for 25 percent of one full core fuel load will be provided in a storage pool in the containment. The number of fuel assemblies in one full fuel load depends upon the choice of either an 18 month or 12 month fuel cycle (308 elements for an 18 month cycle and 220 elements for a 12 month cycle). The total storage space for fuel assemblies in the pool is 857 (for an 18 month cycle) and 769 (for a 12 month cycle). The core consists of 732 fuel assemblies. It will not be possible to remove and store the entire core in the spent fuel storage pool while the previous refueling load is stored without using the storage racks in the containment pool.

The discussion of spent fuel handling equipment and procedures is discussed in Section 9.1.4 of GESSAR. Handling of any loads other than one fuel bundle or control rod assembly over the spent fuel storage array is prohibited by administrative control for the small, general purpose crane and by structural barriers for the cask crane. The height of lift of the general purpose crane will be limited to 2 feet above the top of the spent fuel storage racks.

The fuel building and fuel storage facilities will be designed to seismic Category I requirements. The concrete sides and roof of the fuel building will be designed to prevent tornado borne missiles from damaging the stored fuel and permit maintaining a slight negative pressure in the building by the heating and ventilation system.

Stainless steel liner plates seal the interior pool surfaces. Plate joints are fitted with leak chases that direct leakage to a sump as well as allowing testing and monitoring the leak tightness of the plate joints.

Based on the information provided, we conclude that the criteria and bases for the spent fuel handling equipment and spent fuel storage facility meet the requirements of Regulatory Guide 1.13 and form the basis for an acceptable design.

9.1.3 Spent Fuel Cooling and Cleanup System

The description and evaluation of the spent fuel pool cooling and cleanup system, Section 9.1.3 of GESSAR and Figures 9.1.3a through 9.1.3c were reviewed as the basis for our evaluation.

The spent fuel pool cooling and cleanup system will be designed to remove the decay heat from the spent fuel assemblies and maintain pool water level in the containment pool and fuel building pools. Limiting the radioactivity concentration, the corrosion product buildup in the water and maintaining its clarity will also be accomplished by the cleanup system.

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The cooling system will be designed to be capable of maintaining the pool water temperature below 125°F under normal operating conditions and maximum normal heat load. The maximum normal heat load has been defined as the sum of the decay heat released by the average spent fuel batch discharged from the equilibrium fuel cycle at the earliest refueling time plus the decay heat being released by the batch discharged at the previous refueling. In addition, the system is capable of being lined up to and removing the heat released to the containment pool through the drywell head.

The applicant proposed to supplement the above cooling system when larger heat loads would exist if more than the above anticipated number of fuel elements are in storage by utilizing the RHR system. If the pool water temperature exceeds the stated 125°F, the RHR systems will be used to supplement the available spent fuel cooling system.

The RHR system will only be used to supplement the fuel pool cooling when the reactor is shutdown. The reactor will not be started up should either the fuel pool water exceed 150°F or whenever portions of the RHR systems are needed to cool the fuel pool.

The heat from the spent fuel pool cooling system and the RHR system will be removed by the station service water system.

Table 3.2.1 of GESSAR indicates that, with the exception of the makeup water supply and the filter/demineralizer vessels, the spent fuel pool cooling and cleanup system will be designed to the positions set forth in Regulatory Guides 1.26, "Quality Group Classifications and Standards," and 1.29, "Seismic Design Classification."

Section 9.1.3.3 of GESSAR states that redundant loops of the Essential Service Water System (both seismic Category I) can be used as a source of makeup in the event of failure of the normal makeup system. GE has not, as yet, provided us with the detailed design of this system. However, we consider a commitment to provide a seismic Category I makeup water system independent of the RHR system as a satisfactory design basis for the PDA stage and will review and report on the design of this system post-PDA.

Based on the information provided for heat removal requirements and the indication that operation of the spent fuel pool cooling system is based on an interface inlet station cooling water temperature of 100°F, we conclude that based on an 18 month refueling cycle, the design criteria and bases are acceptable.

9.2 Water Systems

The auxiliary water systems in GESSAR consist of portions of the service water system, the closed cooling water system, the demineralized water system, the potable and sanitary water system, the plant chilled water system and the condensate storage facilities. Many of these systems are designed entirely by the utility applicant and perform no safety-related function and are therefore not appropriate to GESSAR.

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Others like the essential service water system do perform a safety-related function, but are not entirely within the scope of GESSAR.

9.2.1 Essential Service Water System (ESWS)

Information provided in GESSAR indicates there will be three independent Essential Service Water Systems. The design basis for the ESWS are listed in Sect on 9.2.1.1.1.1 of GESSAR.

The essential service water system will be designed to seismic Category I requirements. Two redundant trains of the system will be separated and protected to ensure that sufficient equipment would remain in operation to permit safe shutdown following an accident such as flooding or steam release from equipment failure, pipe whip and jet forces resulting from pipe rupture, missiles caused by equipment failure, and fire.

These two redundant ESWS will be used to cool both essential and nonessential components for all plant operational modes; however, all nonessential components will be isolated automatically during an accident condition. Since at least one service water pump will be operating in each train during all plant operating modes, the service water system required to shutdown the plant safely will be readily available. If a loss of offsite power occurrence coincides with a LOCA, the pumps will momentarily stop until transfer to standby diesel generator power is completed and the pumps are restarted automatically according to the diesel generator loading sequence.

A standby pump will be provided for each train of the ESWS. If a pump in either train fails to start, the standby pump in that train will be started manually from the control room.

The ESWS will be operated at a lower pressure than the residual heat removal system, the fuel pool cooling and the cleanup system, and the closed cooling water system. These three systems are potential sources of radioactive contamination to the ESWS. Liquid radiation monitors will be provided in each of the two main discharge headers to alert the operator so that the affected train can be isolated remote-manually from the control room.

The applicant indicates that seismic Category I sumps and pumps will be located through the plant to detect system leakage. Leakage will be indicated by abnormal increase in the frequency of operation of the sump pumps as indicated in the control room.

The ESWS flow requirement for the reactor island (see GESSAR Table 9.2.2) is establishe on the basis of a LOCA coincident with a loss of offsite power, and a maximum inlet water temperature of 100°F at the system interface, for the flows noted, with the utility-applicant balance of plant (BOP).

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The third independent essential service water system within the GESSAR-NI scope of supply, which also interfaces with the BOP, will be designed to the same requirements of the two other essential service water systems. This third system will transfer the rejected heat from the High Pressure Core Spray (HPCS) diesel engine and its associated accessories.

The Essential Service Water System is an interface between the GE nuclear island and the balance of plant. The part of the system within the nuclear island consists of three distinct, mechanically and electrically separated piping systems. The system supplies cooling water to many safety related components as listed in Table 9.2.2 of GESSAR.

The requirements on the utility-applicant's portion of the system include: (1) integration of the BOP piping with that of the nuclear island piping, (2) the mechanical and electrical separation of the three piping systems external to the nuclear island including separate pumps, and (3) meeting the design requirements listed in Table 9.2.8 of GESSAR.

We conclude from our review that the design criteria and basis for the three Essential Service Water Systems are acceptable.

9.2.2 Closed Cooling Water Syst

The closed cooling water system (CCWS) is a closed-loop, non-safety-related system and will not be required for safe shutdown of the reactor. It provides cooling water to non-essential equipment in the drywell, containment, auxiliary building and radwaste building.

Piping for the CCWS is routed so that a pipe break will not flood or damage any safety-related equipment.

The CCWS is designed to provide a closed cooling water loop between nonessential systems, which are potentially radioactive, and the service water used for cooling and thus will serve as a barrier between potentially contaminated systems and the service water that discharges to the environment. A radiation monitor will be provided to indicate inleakage to this system from a contaminated system. Leakage will be detected by pressure instrumentation in the system supply header, monitoring of local sump flow and system surge tank level.

Any cooling water leakage from the system will be collected in the floor drains and sumps, and piped to the radwaste system for processing.

We conclude the system design criteria and bases are acceptable.

9.2.3 Demineralized Water Makeup Systems

The Demineralized Water Makeup System described in Section 9.2.3 of GESSAR is to be described in the utility-applicant's SAR (including the raw water treatment and demineralization). GE states that this system is not safety-related. The only portions of the system having a safety function will be the containment isolation valves and the piping between the valves of those portions of the system penetrating containment. Its power generation bases are described in GESSAR and include that routing of pipes will not flood or damage safety-related equipment when the break is outside the cubicle containing the safety-related equipment and further breaks inside the cubicle containing safety-related equipment will not affect the redundant system.

We conclude that the requirements contained in Section 9.2.3 provide an adequate basis for the utility-applicant to provide a Demineralized Water Makeup System and is, therefore, acceptable.

9.2.4 Plant Chilled Water System

The plant chilled water systems for the Nuclear Island is comprised of three systems: the drywell nonessential chilled water system, the reactor island nonessential chilled water system and the control building chilled water system.

The nonessential drywell chilled water system consists of two 100 percent systems comprised of chillers and water distribution piping systems that are connected to the six drywell coolers and the two reactor water cleanup (RWCU) pump room coolers. Remotely operated solenoid valves permit isolation of any drywell coil in the event it develops a detectable leak.

The nuclear island nonessential chilled water system consists of two 50% capacity chillers, coils and piping. Chilled water circulation to the fan coil units will be accomplished by three 50% capacity (one standby) pumps placed in parallel. Fail-safe positions on the system controls and equipment safety controls will be provided. Local display devices will be provided to indicate all vital parameters. Flow switches will prohibit the chiller from operating unless there is water flow through the evaporator and the condenser.

The SAR information and Figures for these nonessential systems indicates they can be isolated by a LOCA signal or remote manual operation from the control room for isolation due to a system piping failure.

Based on our review of the above two systems, we conclude the design criteria and bases for the systems are acceptable.

The Control Building Chilled Water System delivers 45°F chilled water via two 100 percent chillers to the control building air conditioning units cooling coils and the Auxiliary Building emergency electric switchgear room coolers during normal operating modes, shutdown of the reactor and abnormal conditions including a LOCA. Redundancy plus electric and mechanical separation are provided to ensure operation under all conditions. It will be designed to seismic Category I requirements, powered by the engineered safety feature buses, housed in a seismic Category I structure and protected from flooding and tornado missiles. Temperature controllers and flow switches will automatically control the systems.

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Based on our review, we will require GE to demonstrate, during the post preliminary design approval stage, that the coolant inventory in the control building chilled water system has been sized to accommodate conservative leakage rates, over a 30 day period, in each system following an accident (the non-seismic demineralized water system normally provides the makeup). We conclude that the design criteria and bases for the system are acceptable, and that the makeup tank volume can be resolved after the issuance of a PDA.

9.3 Process Auxiliaries

The process auxiliaries in GESSAR include the compressed air system, the process sampling system, the equipment and floor drainage system, the liquid poison system, the failed fuel detection system, and the main steam isolation valve leakage control system.

9.3.1 Main Steam Isolation Valve Leakage Control System (MSIVLCS)

GE is providing an MSIVLCS to control and minimize the release of fission products after a LOCA that could bypass the SGTS. The MSIVLCS consists of 2 independent systems to accomplish the leakage control function. The inboard system is powered from one electrical division and the outboard from the other division of the emergency power supply. The outboard system is connected to all steam lines between the outboard MSIV and the downstream block valve. All steam lines are connected in parallel to the outboard bleed header. Suction is drawn on the space between the valves and exhausted to a building volume served by the standby gas treatment system. The inboard system is connected between the main steam isolation valves. An individual bleed line is provided for each steam line. The exhaust is vented to a building volume served by the SGTS.

The instrumentation and controls are to be designed to applicable nuclear safety-related instrumentation and control system standards.

All piping systems and components will be in accordance with current applicable codes, addenda, code cases as well as being designed to meet seismic Category I requirements.

Following our review, GE has informed us of the following modifications:

(a) The exhaust from both inboard and outboard blowers will be discharged to the recirculation system ring header located in the lower portion of the shield building annulus to provide for treatment of exhausts prior to release.

(b) The blowers will draw air from the shield building annulus area to mix with the gas and vapors drawn from the entrapped volumes to provide cooler air for the blower fans.

(c) The leakage control system piping has been changed from 1 to 1-1/2 inch schedule 80 to 1-1/2 to 2 inch schedule 160 pipe to accommodate increased flow rates.

(d) The single blower capacity for the inboard system (formerly 50 cfm at -20 inches of water) has been changed to 100 cfm at -60 inches of water.

(e) The single blower and capacity for the outboard system (was formerly 50 cfm at -20 inches of water) has been changed to two blowers each having a capacity of approximately 100 cfm at -60 inches of water.

(f) The open drains from the inboard and outboard systems have been provided with check valves to prevent the ingress of air.

The operation of the system is limited by a series of pressure sensors and timers which serve interlocks designed to preclude system actuation prior to the pressure in the main steam lines decaying to the pressure of which the Leakage Control System (LCS) is designed to operate. The interlocks also preclude continued operation of any portion of the LCS which fails to achieve a subatmospheric condition in its respective steam line after a preset time. The MSIVLCS is manually initiated and controlled and shall be designed to permit actuation in a time period of about 20 minutes following a postulated design basis LOCA. The required actuation time period shall be consistent with loading requirements on the emergency electrical buses, with reasonable times for operator information, decision, and action, and shall be consistent with the time required for main steam line pressure decay.

The applicant has provided additional information which incorporated all of the proposed MSIVLCS modifications except for drawing air from the shield building annulus. The applicant informed us they prefer to draw air from the auxiliary building ECCS room because it is cooler than shield building annulus air and is more effective in cooling MSIV steam leakage before entering the MSIVLCS blowers. They also proposed (1) that MSIV technical specification leakage be relaxed from between 11.5 SCFH to 100 SCFH per MSIV depending on the site, and (2) that MSIV leak rate verification testing frequency of once a year be allowed. However, the staff has used the 11.5 SCFH leakage value in their accident analysis calculation.

We have reviewed the additional information and discussed remaining concerns with the applicant and reached the following resolutions:

(1) We were concerned there could be momentary escape of steam from the MSIVLCS to building volumes other than the shield annulus building via the dilution air flow damper or the drain line check valve during the initial 2-1/2 minute depressurization period if the blower failed to operate thus resulting in a positive pressure being established in the low pressure manifold. The applicant intends to prevent positive pressure in the manifold by use of a delay timer which will block opening of any of the MSIVLCS isolation valves for approximately 15 seconds after there is indication the blower is operating. This should ensure the manifold is maintained subatmospheric.

(2) The applicant will provide for MSIV steam packing leakage processing by piping it to a sump funnel location in the auxiliary building served by the Standby Gas Treatment System. It is expected that at 35 psia, steam packing leakage should be negligible; however, the piping will provide a means of controlling such leakage if present.

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(3) Based on the present design containment leak rate of 0.3%/day which includes the leakage from the MSIV's, the staff is assuming an MSIV leakage limit of 11.5 SCFH for each value to meet Regulatory Guide 1.3 limits. The setpoint for the flow element timer in the inboard LCS must be set at 11.5 SCFH and has been made a condition of the PDA.

The staff conducted an analysis of the activity released to the site during the depressurization phase of the inboard MSIV-LCS operation assuming that an inboard MSIV is open or only partially closed following a LOCA. The resulting offsite doses for the average site described in Section 2.3.4 exceeded Part 100 guidelines and were therefore unacceptable.

Based on this analysis, the staff will require that GE provide interlocks on each inboard LCS to prevent the operation of any individual inboard LCS if the associated inboard MSIV in the steam line connected to that portion of the LCS is not fully closed following a LOCA. GE has not agreed to provide these interlocks; therefore, they will be made a condition of our PDA.

We conclude, subject to the above condition, that the MSIV-LCS design criteria and bases are acceptable for the PDA.

9.3.2 <u>Compressed Air Systems</u>

The Nuclear Island - BOP concept safety interface matrix table in Section 1.10 of GESSAR indicates that no safety-related interfaces exist with the Compressed Air System. Further, Section 9.3.1 indicates that the compressed air system will be designed, described and supplied by the utility-applicant to provide the Nuclear Island requirements.

The SAR, Section 9.3.1, indicates the two instrument and service air systems are non-safety related systems because the essential equipment utilizing air such as the main steam isolation valves and drywell ventilation isolation valves will be provided with accumulators plus connected piping which will be designed, built, installed and tested to ASME Section III Code, and the staff's equivalence to Quality Group C and seismic Category I requirements.

In acknowledgement of the potential for the gas containers or accumulators being a source of missiles, it is stated that they will be provided with safety valves set in accordance with ASME Section VIII, they shall be located so that no credible failure shall generate missiles or otherwise impair the safety functioning of safety-related equipment, and the supporting structures will be designed to absorb the thrust loads developed assuming a failure of the largest pipe connected to the accumulator.

9.3.3 <u>Equipment and Floor Drainage System</u>

The Equipment and Floor Drainage System, comprised of sumps, level switches, and pumps serve areas for the control rod drive seals, the drywell equipment, the RCIC and ECCS and other systems that support or protect the nuclear steam system. These systems will be designed to comply with seismic Category I and IEEE 279 and 308 requirements.

The following specific ECCS pumps and heat exchangers are located in individual below grade watertight compartments: HPCS, RHR A and B pumps and heat exchangers, RHR C pump, and LPCS. The sump pump in each compartment will be capable of delivering 50 gpm which is adequate for minor leakage. A pipe break or major leak will only flood that one compartment. The equipment and floor drainage will be returned to suitable vessels for processing and/or disposal. Two pumps will be supplied for sumps located in inaccessible areas such as the drywell equipment and floor sumps, with each pump having the above rated capacity. Sumps located in accessible areas where maintenance during operation can be accomplished, such as the reactor building equipment and floor drains, will be provided with a single full capacity sump pump.

In reference to the Equipment and Floor Drainage System for the Nuclear Island, Section 3.4 states that the plant finished grade will be one foot above maximum flood level including coincident waves and resultant runup. The discussion in SAR Section 9.3.3, Equipment and Floor Drainage System failed to address the precautionary measures taken to prevent inflow by way of any and/or all Nuclear Island Drainage Systems (including the remaining buildings within the scope of the Nuclear Island such as the Diesel Generator Building, Control Building, and Fuel Building and the associated fuel storage facilities during the above described flood conditions).

We will require, at the FDA phase, supplementary information to satisfactorily demonstrate that design measures (such as stop check valves) have been properly taken to prevent inflow via the Equipment and Floor Drainage System. We conclude the design criteria and bases for the system are acceptable.

9.4 Air Conditioning, Heating - Cooling and Ventilating Systems

The description of the control room HVAC, Section 6.4 of GESSAR, and Figure 6.4-1 were reviewed as the basis for our evaluation.

The control room HVAC consists of two subsystems which will be provided to assure normal outside air HVAC and minimum make-up air cleaning and control room pressurization. The functional operation of the normal HVAC provides filtered, heated or cooled outside air to the control room and the control equipment room and maintains these rooms at a positive pressure with respect to the ambient barometric pressure. The functional operation of the minimum make-up air cleaning system is to discharge or recirculate normal ventilation air through the two rooms. During conditions of high radioactivity at the outside air intakes, the system will recirculate the room air through a filter unit containing particulate and charcoal filters. A small amount of outside air is taken from either of two air intakes and processed through the filter assemblies and air conditioner unit to maintain a positive pressure in the rooms. The system can be completely isolated from the outside if required for a sufficient period of time to protect personnel during extreme adverse outside air contamination.

Another function of the system is to detect smoke in the outside air intakes and in the system ducting to provide for isolation of the outside air intakes, open recirculation dampers or switch to a once through purge operation of a given area of the rooms or the equipment racks by automatic operation of the necessary dampers.

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The normal HVAC system takes outside air from a missile protected intake located on the far side of the control building located near grade elevation (+ 1'0") where it is processed through one of two 100% capacity unit air conditioning systems. This processed air ventilates the control room, control equipment room, and the HVAC equipment room. Two 100% capacity room return air fans are provided to discharge the air to the building exhaust or recirculate a portion back to the air inlet processing system. A positive pressure is always m..ntained by mixing dampers in the recirculation ducting. Two 100% capacity equipment rack return air fans are provided to perform the same function.

The minimum make-up air cleaning system consists of two 100% capacity air filtration units, a missile protected outside air intake located on the auxiliary building side at elevation (35' - 0"), the two unit air conditioning units of the normal system and two sets of return air fans of the normal system. On a high radiation signal this system is placed in operation and the normal system outside air intakes and exhaust air isolated.

The applicant has performed a single failure analysis assuming a failure of any individual component in this complete system. We have reviewed the results of this analysis and find the system can withstand any single active failure and perform its safety function.

Based on the above, we conclude the design criteria and bases meets the requirements of General Design Criterion 19 and the applicable portions of the position set forth in Regulatory Guide 1.52 and are, therefore, acceptable.

9.4.1 Balance of Control Building (BCBHVAC)

The description and evaluation of the BCBHVAC, Section 9.4.1 the GESSAR and Figure 9.4.1 were reviewed as the basis for our evaluation.

The BCBHVAC consists of the main control building outside air intake, three inlet filter fan assemblies with two being provided with heating coils and the other a cooling coil, three exhaust fans and associated ducting. The system serves all areas of the control building except the control room and the control equipment room.

The applicant states that the system is not required to perform a safety function. Based on our review we agree with GE that the BCBHVAC system is not required for a safety function except for the seismic Category I control dampers provided in the control room return and exhaust ducts which isolate the balance of the system to maintain the required control room negative pressure. Based on the above, we conclude the design criteria and bases for the BCBHVAC are acceptable. This conclusion is based on the fact that the failure of the BCBHVAC system during an accident will not result in the control room operators receiving an unacceptable dose of radiation.

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9.4.2 Auxiliary Building (ABHVAC)

The description and evaluation of the ABHVAC, Section 9.4.2 of GESSAR and Figure 9.4-2 were reviewed as the basis for our evaluation.

The ABHVAC consists of a normal HVAC system and an emergency operating condition system. The functional operation of the normal HVAC system is to provide ventilation air to the reactor water cleanup pump rooms, the ECCS pump rooms (RHR pumps A, B & C, LPCS pump, HPCS pump and the RCIC pump). Single zone HVAC systems will be provided to provide ventilation air to the electrical area and corridors and battery room. The steam tunnel is provided with a cooling system to maintain the tunnel space at 125°F during normal plant operation. This temperature is 10°F below that at which the 3000 cycle test of the MSLIV control valves was conducted (APED-5750, Supplement 1).

The normal HVAC system consists of two outside air filter fan units supplied by a single outside air intake, series isolation dampers in the ducting of the two main areas it serves, two variable vane exhaust fans and series isolation dampers in the main exhaust duct. The system normally operates with one outside air filter fan unit and one exhaust fan unit. The other fans are provided for standby service. All of the above areas are maintained at a negative pressure by the variable vane exhaust fan.

Each of the two single zone closed HVAC systems consist of one cooling coil, ducting and an exhaust fan located downstream of the battery rooms. Each system is capable of ventilating its area. The battery room exhaust fans draw air from the electrical areas and thereby create a slight negative pressure in the electrical areas, and the battery room exhaust blower suction lines have been provided with back draft dampers. As discussed in Section 8.3.2.1.2.4 of GESSAR, the ventilation system is designed to keep the gases generated during battery charging below explosive levels at all times.

Our review of the normal ABHVAC system, including an independent failure mode and effects analysis, indicates that the system can perform its normal function for the normal mode of operation.

During the loss-of-coolant accident the normal system is isolated by series dampers and the Standby Gas Treatment System (SGTS) will maintain the essential pump rooms at a negative pressure and filter any radioactivity from the air prior to discharge to the environment. The SGTS evaluation is presented in Sections 6.2.3 and 11.3 of this SER. Each pump room has its own seismic Category I fan cooling unit which is powered from the same essential power source as the pump it cools. These units will limit the temperature in the pump room to 148°F.

Our review of the emergency mode of operation using the SGTS included an independent failure mode and effects analysis up to the connection to the SGTS. The results of this analysis indicate that the system can withstand a single active failure and still perform its function.

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Based on the above, we conclude the design criteria and bases are acceptable.

9.4.3 Radwaste Building (RWBHVAC)

The description and evaluation of the RWBHVAC, Section 9.4.3 and Figure 9.4-3 of GESSAR were reviewed as the basis for our evaluation.

The RWBHVAC system will be provided to ventilate the radwaste control station, the radwaste storage cells and non-contaminated areas. The system will maintain the storage cells and the non-contaminated area at a negative pressure and the control station at a positive pressure.

The system will be a once-through type consisting of a roughing filter, heating and cooling coils, two 100% supply fans and associated ductwork. The system will be exhausted to the plant vent by two 100% capacity exhaust fans. Provisions are included to remove airborne particulates and monitor the air for high radiation levels before it is released. The control station will be exhausted separately to permit maintaining the required positive pressure.

Our review of the system shows that the dual fan system will be capable of ventilating the radwaste area and single active failures will not preclude its function. The storage cell area will be isolated by series dampers if activity is detected.

Based on the above, we conclude the system design criteria and bases are acceptable.

9.4.4 Fuel Building (FBHVAC)

The description and evaluation of the FBHVAC, Section 9.4.6 of GESSAR and Figure 9.4-6a were reviewed as the basis for our evaluation.

The FBHVAC system consists of four subsystems to control the fuel building and standby gas treatment system equipment rooms at a negative pressure and to process any radioactivity that could be released due to a fuel handling accident. The subsystems are (1) the dual outside air pressure supply fan units, (2) two 50% capacity fuel building recirculation heating and cooling fan units; (3) the dual variable vane exhaust fans which normally exhaust the above mentioned areas and control them at a negative pressure, and (4) the standby gas treatment system which will provide for filtration of radioactivity from the above areas and maintain them at a negative pressure during accident conditions.

Our review of the integrated system, including an independent failure mode and effects analysis, indicates that the system can withstand a single active failure and still perform its safety function.

Radiation monitors will be provided in the fuel building which will automatically initiate isolation of the building and start the SGTS for the emergency mode of operation.

The system required for emergency operation will be designed to seismic Category I requirements including the appropriate isolation dampers.

The SGTS evaluation is given in Sections 6.2.3 and 11.3 of this SER.

Based on the above, we conclude the design criteria and bases for the FBHVAC meet the positions set forth in Regulatory Guides 1.13 and 1.52, and are, therefore, acceptable.

9.4.5 Diesel Generator Building Heating and Ventilation System

All three standby power diesel generator compartments (Division I & II diesel generators are utility-applicant supplied and Division III HPCS diesel generator is supplied as a part of GESSAR-NI) will each be provided with a seismic Category I ventilation system powered from the engineered safety feature bus. They will be sized to remove the full rated load radiant heat generated in the compartment concurrent with a maximum engine room temperature of 120°F. Supplemental electric heaters will be provided to maintain a minimum room temperature of 50°F with a -10°F winter ambient.

The system does not include the separate diesel engine combustion air intake or exhaust system provided each compartment. Based on our review we conclude that the design criteria and bases for the system are acceptable.

9.4.6 Diesel Generator Fuel Oil Storage and Transfer System

The Diesel Generator Fuel Oil Storage and Transfer System is within the GESSAR-NI scope of supply and responsibility for all three standby diesel generators except for the Diesel Generator Fuel Oil Storage Tanks which are a utility-applicant responsibility.

Each engine will have its separate storage and transfer system consisting of two tanks (utility-applicant supplied), two transfer pumps and one day tank. Portions of the system within the GESSAR-NI scope will be designed in accordance with ASME Section III Class 3, seismic Category I requirements. The bulk storage capacity for each engine will be the longer of the two following requirements (1) seven day supply when providing post-LOCA maximum load demands, or (2) the time required to replenish the onsite supply from outside sources.

The design will protect the systems from adverse environment conditions such as tornadoes, hurricanes and earthquakes either by design or locating it underground or within protective structures. System components will be protected against corrosion including the underground pipes which will have protective coatings and wrappings. The storage and day tanks will be located a sufficient distance from the plant control room to preclude danger to control room personnel or equipment from a fire in the fuel storage system. The day tanks are vented to the atmosphere and equipped with flame arrestors. In addition the day tanks for Division I and II diesel engines are of such size (800 gallons) that they will be placed in a separate, diked, fire rated enclosure within the diesel generator room.

Provisions will be made to replace degraded fuel by tank truck. Manifold piping will be provided to prevent the withdrawal of fuel from more than those storage tanks associated with one diesel engine.

Based on our review, we conclude that the design criteria and bases for the Diesel Generator Fuel Oil Storage and Transfer System are acceptable.

9.4.7 Diesel Generator Cooling Water System

The Diesel Generator Cooling Water Systems are designed to remove the heat released by the engine air intercoolers, lubrication oil coolers and engine jacket water closed loop heat exchangers for all three diesel engines by means of three separate essential service water systems designed to ASME Section III Safety Class 3. Therefore, the failure of one system will not impair the operation of the other engine.

The engine jacket closed loop system (treated water) passes through a three-way temperature control valve to maintain the required water temperature. The heat exchanger will be in accordance with ASME Section III Class 3 requirements. Components of the cooling water system will be designed to seismic Category I requirements.

The engine jacket closed loop cooling water system will be provided with a keep-warm feature to enhance quick "first try" starting of the engine.

Based on our review, we conclude that the design criteria and bases for the Diesel Generator Cooling Water systems are acceptable.

9.4.8 Diesel Generator Starting Air System

Each of the diesel generator units will be provided its own separate compressed air starting system. Each system consists of two 100 percent capacity, redundant sections both having an air compressor and receiver. The piping and valves between the receivers and engine will be designed to ASME Section III Class 3 and seismic Category I requirements. In addition redundant starting air-admisssion valves to the engine starting air-admission manifold will be provided each engine. The failure of any one starting system in no way affects the ability of the other to perform its required safety function.

To provide further protection against starting air valve malfunction, filters and a strainer will be installed just upstream of the starting air valve for protection against airborne contamination.

The compressors are normally automatically controlled by pressure switches on their respective receiver tanks. They may also be manually operated. Automatic water traps and drain valves will be provided for the removal of water in the starting systems. Each independent starting system will have sufficient capacity for 30 seconds of cranking (equivalent to 5 automatic or manual starts) without recharging the air receivers. It will require 30 minutes for the compressor to fully recharge the air receiver from the minimum pressure set point on the receiver.

Based on our review, we conclude that the design criteria and bases for the Diesel Generator Air Starting Systems are acceptable.

9.4.9 Diesel Generator Lubrication System

The self-contained closed Diesel Generator Lubrication System is designed to supply sufficient lube oil at a controlled pressure, temperature and clean condition to ensure adequate lubrication of wearing parts and cooling of the diesel engine as required. It consists of an oil sump in the engine frame, an engine driven positive displacement pump, an oil cooler, an oil strainer and filter. The above pump directs oil from the sump through a filter to a three-way thermostatically controlled valve which directs the oil through or around the oil cooler, through a cartridge type filter and then back to the engine bearings and sumps. A pressure regulating valve bypasses excess oil flow to the sump. The cooler, a shell and tube type heat exchanger will be designed in accordance with the latest edition of ASME III Class 3 requirements. The essential service water system transfers the rejected heat to the ultimate heat sink.

Standby preheating of the lube oil is accomplished by an AC motor driven pump which directs oil from the sump through an electric heater and then back to the sump in order to enhance the engines "first try" starting reliability.

Based on our review, we conclude that the design criteria and bases of the Diesel Generator Lubrication System are acceptable.

9.5 Fire Protection

9.5.1 Fire Protection System

Section 9.5 of GESSAR describes details of the Fire Protection System which will be provided in the GESSAR-Nuclear Island.

All safety-related wiring in the control room, cable tunnels, and other areas of the Nuclear Island is flame resistant. Due to divisionalized wiring of motors, and associated controls plus fire rated walls or physical separation, the loss of one of the redundant systems would not jeopardize plant safety. Cable trays are distance separated or utilize fire rated barriers to avoid the loss of redundant channels of safety-related cabling. Electrically shorted equipment or overheated wires should not propagate as ordinary combustibles and can be controlled and extinguished by hand applied extinguishing media. Piping and cabling penetrations will be fire stopped to preserve the fire rating of the walls. The seismic Category I safety-related structures are constructed of reinforced concrete walls (except for the free standing steel containment) having many walls with a fire rating of four hours due to the structural requirements. Other concrete walls are of at least two hour rating. Non-concrete walls are at least one hour rating. The fire rating of the fire resistive doors will be consistent with its wall rating. Materials selected for construction are noncombustible or are rated as having a low flame spread rating. The reactor island does not use natural gas nor are other combustible gases piped into the buildings; therefore, no special systems for combustible or explosive gas control are provided.

In addition the utility-applicant, in meeting the positions set forth in Regulatory Guide 1.39, should not introduce large quantities of highly inflammable substances into any areas of the facility.

The fire protection and detection systems are non-seismic Category I. However, the design is such that the failure of the system or its parts will not adversely affect seismic Category I items. There will be three supply systems serving (a) Auxiliary, Fuel and Reactor Buildings, (b) Control Building, and (c) the Radwaste Building. The wet standby piping systems have been routed outside those areas containing safety-related equipment. In the event of a failure of the standpipe, flooding can be controlled by closing the appropriate post indicator valve located outside the building and thereby prevent water damage to safety-related systems. Equipment and building drains minimize the accumulation of combustible and non-combustible liquids which are piped to the Radwaste Building for concentration and disposal.

The fire protection and detection systems are designed, specified, and installed in accordance with the current criteria published by the National Fire Protection Association and other applicable Codes and Standards as listed in Section 9.5.1.1.5 of the SAR.

To preclude the loss of habitability of the control room, it has been provided with a smoke and toxic contaminate removal system. In addition the reactor can be safely shutdown from either the control room or the standby shutdown panel located in the Auxiliary Building.

Fires could make a room or area uninhabitable without self-contained breathing apparatus due to production of significant toxic and asphyxiant atmospheres. These self-contained breathing apparatus will be provided by the utility-applicant (BOP) to permit personnel to remain in the control room to extinguish fires and to permit personnel to enter other areas of the plant to extinguish fires.

Section 1.1.3 states for the purposes of GESSAR, only a single standard plant unit will be considered. Therefore the hazards of multiple units were not discussed.

The extinguishing systems provided consists of (a) water systems, (b) hand extinguishers type Class ABC (20A-80BC) (This designation indicates it is a multipurpose extinguisher for extinguishing Class A, B and C fires per Underwriters Laboratories testing requirements), (c) 15 lb. CO_2 hand extinguishers, (d) 2-1/2 gallon pressurized portable water extinguishers, and (e) a manual or automatically activated (with a 30 second delay) CO_2 total flooding extended discharge system for each individual diesel engine compartment.

The applicant has identified the following potentially [illegible]le fires that could affect Category I safety related structures within the Nuclea[illegible]land:

a. diesel generator compartments - the fuel oil day tank within each compartment, 800 gallons for Division I and II diesel engines and 550 gallons for Division III diesel engine.
b. reactor containment building - lubrication oil or hydraulic oil reservoirs (maximum in one fire is about 400 gallons) for the recirculation pump valve control system.
c. all buildings have a potential for electrical equipment and cable insulation fires.

Figures 9.5-3 through 9.5-11f of the SAR, plus Table 9.1 below, presents information on the particular fire detection, protection and suppression systems provided the facility.

As indicated in Table 9.1 the control building has not been provided with an automatic fire suppression system. Rather manually applied fire extinguishants (water, hose and/or 2-1/2 gallon pressurized water portable extinguishers or 15 lb. CO_2 hand extinguishers) will be employed. Figure 9.5-11b indicates that within the control room the fire extinguishants will be hand extinguishers type ABC (20A-80BC) or 15 lb. CO_2 extinguishers. In addition the following areas are provided with an early warning fire detection system which will be appropriately placed within (a) the reactor control console and termination cabinets as well as the subfloor sections, (b) cable tunnels, (c) HVAC Equipment rooms, (c) main stairwells, all levels, (d) Elevator Equipment Room, (e) Battery Room, (f) RCIC Pump Room, ECCS Pump Rooms, (g) Diesel Generator Rooms, (h) Remote Shutdown Panel, (i) Electrical Switchgear Rooms, (j) CRD Maintenance Rooms, and (k) SGTS Rooms. The above systems are zoned by floor level and will be provided with audible alarms. A digital readout display system will be provided on the control room fire control panel.

Each diesel generator compartment will be provided with a Supervised Product of Combustion Detector System which will activate the automatic CO_2 fire extinguishant. Since each diesel engine compartment ventilation system is independent and has no cross connections to the other buildings of the Nuclear Island, activation of the CO_2 sy tem will (a) stop the ventilation system (b) deactivate the air compressors, (c) close the air supply ventilation louvers, and (d) hold open a smoke overpressure ventilation damper located at the compartment ceiling level. The engine combustion air and exhaust systems are independently ducted into and out of the compartments, and therefore the diesel generator will remain running at the start of the fire. Further, remote manual shutoff valves have been provided in the fuel oil line to the day tank and the engine. These valves enable the day tank oil to be isolated from the engine, at the operator's discretion, by closing the valves. To prevent flash over after the fire has been suppressed, wet standby pipe and hose reels outside the main compartment entrance may be utilized in the cooldown period.

Section 9.5.1 of the SAR also sets forth the interface and other requirements that must be met by the utility-applicant in order to meet the GESSAR-NI fire protection system overall design objectives.

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TABLE 9.1

	Control Bldg.	Auxiliary Bldg.	Fuel Building	Reactor Containment Bldg.	Diesel Generator Bldg.	Radwaste Bldg.
A Manual and 30 Second Time delay Automatic Fire Protection CO_2 total flooding extended discharge system activated by products of combustion.	No	No	No	No	Yes	No
Manual Fire Suppression System Provided	Yes	Yes	Yes	Yes	Yes	Yes
a. Modified Class III Standpipes and Hose Reels provided, and located such that the distance from any fire to a hose reel will not exceed 100 feet.	Yes	Yes	Yes	No	Yes (located outside main compartment entrance)	Yes
b. Portable Class ABC (20-80BC) provided, located such that the distance from any fire to an extinguisher will not exceed 50 feet.	Yes (including control room)	Yes	Yes	Yes (Elev. 11'-0") (Elev. 37'-1") (Elev. 84"-0")		Yes
c. 15 lb. CO_2 hand extinguishers provided, located such that the distance from any fire to an extinguisher will not exceed 50 feet.	Yes (including control room)		Yes	Yes (Elev. 11'-0")		Yes
d. 2-1/2 Gallon Pressurized Water Portable Extinguishers provided, located such that the distance from any fire to an extinguisher will not exceed 50 feet.	Yes, such as offices, change rooms chart rooms					

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TABLE 9.1 (Continued)

	Control Bldg.	Auxiliary Bldg.	Fuel Building	Reactor Containment Bldg.	Diesel Generator Bldg.	Radwaste Bldg.
e. Products of Combustion Detection System	Control Room, Cable Chase, Termination Cabinets Elect Equipment Rooms	RHR, RCIC, LPCS & HPCS Rooms, Battery Rooms Electric Switchgear Room		Stairs & Elevator Tower	Diesel Engine Compart-ments	Oil Separation Room

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Currently, the staff is in the process of developing a Branch Technical Position regarding fire protection systems.

Based on our review, we conclude that the design criteria and bases plus the essential interface data meet the requirements of General Design Criteria No. 3 and, therefore, forms an adequate basis for acceptance at the PDA stage. As a result of investigations presently being conducted in the development of a Branch Technical Position on the Fire Protection System, further requirements may be imposed so that unacceptable damage will not result from a fire.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

The radioactive waste management system for the General Electric Standard Plant (GESSAR) will be designed to provide for the controlled handling and treatment of liquid, gaseous and solid radioactive wastes, and for monitoring of all major release points of radioactive materials. The liquid waste system will process liquid wastes from such sources as equipment drains, system leakage, condensate demineralizer regenerant solutions, laboratory and decontamination liquids, and detergent wastes. The liquid waste will be processed and recycled for reuse if the plant water balance requires makeup and if the water quality is adequate.

Gaseous wastes will consist of offgases from the main condenser air ejector, vents from equipment containing radioactive materials, and leakage from systems and components containing radioactive material that is released via the building ventilation systems. The offgases from the main condenser air ejector will be treated by catalytic recombination to reduce the volume of offgases and by charcoal adsorption to selectively delay fission product noble gases before release to the environment. Certain equipment vents and building ventilation exhausts will be treated by high efficiency particulate air (HEPA) filters and charcoal adsorbers to remove radioactive particulates and radioiodine prior to release to the atmosphere.

Solid wastes generated during plant operation will consist of spent demineralizer resins, evaporator bottoms, discarded radioactive components and tools, and miscellaneous dry solid wastes. Wet solid wastes will be solidified and drummed. Dry, compressible materials will be compacted and drummed. Drummed wastes will be shipped to a licensed burial site.

The GE design objective for liquid and gaseous effluents are described in Sections 11.2.1 and 11.3.1 of GESSAR and indicate their intent to meet "proposed" Appendix I to 10 CFR Part 50. Specifically, they state that the objective of the liquid treatment system is to limit releases to less than 5 Ci/yr and for the gaseous system, it is to keep the dose to individuals offsite to as low as practicable.

On May 5, 1975, the Commission published in the FEDERAL REGISTER (40FR19439) its decision in the rulemaking hearing concerning numerical guides to meet the "as low as practicable" criterion for effluents. The rule sets forth criteria which, if met, provide one acceptable method of establishing compliance with the "as low as practicable" requirement of 10 CFR 50.34a and 50.36a. These criteria are based on doses to individuals and the incremental cost of processing equipment versus the dose reduction to the population within 50 miles of the facility. As a result of these features, conformance to the guidelines of 10 CFR Part 50, Appendix I will be on a site-by-site basis. That is, individual utility license applications will be reviewed to assure their conformance with 10 CFR 50.34a and 50.36a.

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The following sections present our evaluation of the liquid, gaseous, and solid radioactive waste treatment systems. The liquid, gaseous, and solid waste systems will be designed to accommodate the waste produced during operation of a single unit at a maximum thermal power level of 3758 MWt. The radwaste treatment system, as discussed in GESSAR, includes provisions to process wastes produced by plants employing either regenerable deep bed condensate demineralizer units (with optional ultrasonic cleaning of resins), or by plants employing powdered (Powdex) type filter-demineralizers for condensate polishing.

Our evaluation and calculation of annual releases of radioactive materials are based on the parameters and calculational models given in Appendices B and C to WASH-1258. The models in WASH-1258 are based on data obtained from operating reactors, and represent a realistic assessment of the expected releases over the life of the plant during normal operation, including anticipated operational occurrences.

11.2 Liquid Waste Treatment Systems

11.2.1 System Description

Treatment of the liquid waste is dependent on the source, activity and composition of the particular liquid waste and on the intended disposal procedure. The liquid waste system is divided into three subsystems: the waste collector subsystem (low conductivity), the floor drain neutralizer subsystem (high conductivity) and the detergent waste subsystem.

The liquid radwaste treatment systems are designed to permit complete recycle of processed liquids during normal operation. Processed liquids will be handled on a batch basis to permit optimum control and release of radioactive materials. In the event of the release of treated liquid wastes, samples will be analyzed to determine the type and amount of radioactivity in each batch. Based on the analytical results, these wastes will either be released through the circulating water discharge or processed through the detergent evaporator and released as vapor. In our evaluation, we considered that processed high purity and low purity liquid wastes would be released as liquids, and that processed detergent wastes would be released as vapor.

The waste collector (low conductivity) subsystem receives liquids from the drywell, containment building, auxiliary building, fuel building, radwaste building and turbine building equipment drains and the decantate from the cleanup phase separator. Depending on whether the design of the condensate polishing system uses either the deep bed or Powdex demineralizers, the waste collector subsystem will also receive the decantate from the condensate phase separators (Powdex systems), condensate demineralizer wash water, and ultra-sonic resin cleaning rinse water (deep bed systems). These wastes will be collected in one of two 57,000 gallon low conductivity tanks, processed in a batch through a traveling bed filter, and collected in a 3,000 gallon filtrate tank.

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The filtered wastes will then be processed through two 350 gpm mixed bed demineralizers in series and will normally be routed to the condensate storage tank for reuse in the reactor. If storage capacity is not available, the processed liquids will be manually diverted to one of two 50,000 gallon excess water tanks. From the excess water storage tanks, the liquids may be sent to condensate storage, released at a controlled rate to the discharge canal, or routed to the detergent evaporator and then released as vapor. General Electric expects that all waste treated through this system will be recycled for use within the plant, but has evaluated the releases based on 10% discharge. In our evaluation, we considered that 34,000 gpd of low conductivity liquids will be processed through the system and that 10% of the treated liquids will be discharged to the environment.

The floor drain neutralizer (high conductivity) subsystem collects liquids from the drywell, containment, auxiliary building, fuel building, radwaste building and turbine building floor sumps, demineralizer regenerant solutions, laboratory drains and non-detergent decontamination solutions. The wastes will be collected in one of three 18,000 gallon high conductivity tanks, neutralized, and processed through a 40 gpm waste evaporator. The evaporator condensate will be processed through a 350 gpm distillate demineralizer, a 350 gpm backup demineralizer, and recycled to condensate storage. If the condensate storage tanks are full, the processed liquids will be routed to the excess water tanks. From the excess water tanks, water may be either recycled to condensate storage, released to the discharge canal, or routed to the detergent evaporator and released as vapor. The applicant estimates that 9300 gpd of high conductivity liquids will be processed through this subsystem, and that 10% of the processed liquids will be released to the discharge canal. Our analysis is based on an input rate of 8200 gpd and discharge of 10% of the treated liquids to the environment. The applicant's estimates of liquids are based on his projections of the design capability. Our estimates are based on the parameters given in WASH-1258.

Laboratory wash water and wastes containing detergents (laundry, personnel and equipment decontamination wastes) will be collected in a 1500 gallon detergent waste tank. These wastes will be filtered and evaporated in a 5 gpm detergent evaporator, which is vented to the atmosphere. The applicant considers that 1100 gpd of detergent waste will be processed through this system, with 100% of the evaporator distillate released to the atmosphere. Our analysis is based on 450 gpd of detergent wastes being processed, with 100% release of the processed liquids to the atmosphere.

Bottoms from the waste evaporator and the detergent evaporator will be collected in a 25,000 gallon concentrated waste tank and transferred to the solid waste system for solidification, packaging and shipment offsite. Spent demineralizer resins will be collected in a 10,000 gallon spent resin tank and transferred to the solid waste system for packaging and shipment offsite. Resin sludges from the phase separator tanks will be dewatered and transferred with the filter sludges to the solid waste system for loading into shipping containers.

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11.2.2 Liquid Waste System Evaluation

Using the methods and parameters of WASH-1258, we have calculated the liquid waste releases. GE has also calculated what the liquid releases will be and their values are presented in parentheses following ours. In general, the liquid releases will be less than 1 Ci/yr/reactor (GE - 4.6 Ci/yr), excluding tritium and dissolved gases. We have normalized these releases to 2 Ci/yr to compensate for equipment downtime and anticipated operational occurrences. We also calculate that 6×10^{-4} Ci/yr (GE 3.6×10^{-6} Ci/yr) of particulate material, will be released to the atmosphere due to evaporation of detergent wastes. Based on reported releases at operating boiling water reactors, we calculate that 20 Ci/yr/reactor (GE 12 to 20 Ci/yr) of tritium will be released in the liquid effluents. The principal reason for the applicant's higher estimate of the quantity of material to be released in liquid effluents is the applicant's use of lower decontamination factors for the demineralizers in the low conductivity system. The bases for our decontamination factors are found in WASH-1258.

All major processing components are redundant. There are spare collector tanks in all three subsystems of the liquid radwaste treatment system. There is a spare travelling belt filter, a backup demineralizer in the waste collector subsystem, and a spare waste evaporator in the floor drain-neutralizer subsystem. In the event of equipment failure in the low conductivity system, it will be possible to divert the wastes to the high conductivity tank. The only equipment items that are not redundant in this system are the detergent waste filter and the detergent waste evaporator. If, in the event one of these equipment items is unavailable, detergent wastes will be released without treatment, the liquid effluents would be increased by 0.06 Ci/yr, which is a small fraction of our calculated total release from the plant.

Overflows from the low conductivity tank, the filtrate tanks, the spent resin tank and the cleanup phase separator tanks are piped to the radwaste building equipment drain sump. Overflows from the detergent drain tank and the high conductivity tank are piped to the radwaste building floor drain sump. These provisions are acceptable.

Processed liquids will be released to the discharge canal through a line that is equipped with a radiation monitor that will alarm and automatically terminate the release if the concentration of radioactive material in the effluent exceeds a predetermined level. We consider that this design provides adequate storage and control of releases of radioactive materials, in accordance with General Design Criterion 60 of Appendix A to 10 CFR Part 50, and find it acceptable.

The liquid radwaste system equipment will be designed to Seismic and Quality Standards listed in Section 3.2 of this SER.

The liquid radioactive waste system includes the equipment and instrumentation to control the release of radioactive materials in liquid effluents. Our review of the liquid radwaste system included normal operation, anticipated operational occurrences, design provisions incorporated to preclude uncontrolled releases of radioactive materials in liquids due to leakage or overflows, and the quality group classification and seismic design criteria. We have reviewed the applicant's system descriptions, process flow diagrams, piping and instrumentation diagrams, and design criteria for the components of the liquid radwaste treatment system. We have performed an independent calculation of the releases of radioactive materials in liquid effluents based on the calculational methods of WASH-1258.

11.2.3 Conclusions

Based on our evaluation, we conclude that the design supplied in GESSAR utilizes state-of-the-art technology. Also, there is reasonable assurance that the concentration of radioactive materials in liquid effluents will be a fraction of the limits in 10 CFR 20, Appendix B, Table II, Column 2, for the expected and design releases, for plants whose discharge canal flow equals or exceeds the minimum discharge canal flow of 1500 gpm specified by the applicant. Adequate control of releases of radioactive materials in liquid effluents is provided in accordance with General Design Criterion 60 of Appendix A of 10 CFR Part 50.

Compliance with the design objective doses to individuals at or beyond the site boundary is site dependent, and will be reviewed for individual license applications.

11.3 Gaseous Waste Treatment Systems

11.3.1 System Description

The gaseous radioactive waste treatment systems for the GESSAR plant will consist of a charcoal delay system for treating the offgas from the main condenser air ejector, and iodine and particulate control systems for certain building ventilation systems. The release of radioactive materials in the offgas from the turbine gland sealing system will be negligible, since non-radioactive steam from an auxiliary boiler will be used to seal the turbine glands. During startups, a mechanical vacuum pump will be used to evacuate the main condenser. The discharge from the vacuum pump will contain radioactive gases that will be released without treatment through a roof vent on the turbine building to the atmosphere, along with the offgases from the charcoal delay system. Building ventilation system exhausts are normally released via roof vent. The turbine building ventilation system is outside the scope of GESSAR, and will be reveiwed for individual license applications.

Offgases from the main condenser air ejector will be treated through a low temperature charcoal delay system to provide for decay of the radioactive noble gases before release to the atmosphere. The offgases from the main condenser will contain principally, hydrogen and oxygen from decomposition of water, air from condenser inleakage, fission and activation gases, and water vapor. The offgases from the last stage of the condenser air ejector will be diluted with steam to maintain hydrogen concentrations below 4%. The pressure boundary of the system is designed to maintain its integrity in the event of a hydrogen explosion. The gas mixture will be heated, passed through a catalytic recombiner to react hydrogen and oxygen and

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passed through a condenser and moisture separator. The condensate will be returned by gravity to the main condenser hotwell, and gases will be passed through a ten minute delay line to provide for radioactive decay of activation and short lived fission product gases. The gases will be cooled to 45°F in a glycol cooler, filtered through a high efficiency particulate air (HEPA) filter, and dried to a dewpoint of approximately -90°F by a desiccant drier. The gases will be further cooled to approximately 0°F and passed through a train of eight three-ton charcoal beds in series. The gases exiting the charcoal delay train will be filtered to remove charcoal fines and particulate matter, and released to the atmosphere via a roof vent.

In passing through the charcoal delay beds, the higher molecular weight xenon and krypton will be preferentially adsorbed on the charcoal surface and delayed with respect to the flow of the carrier gas (air). The delay time will be affected by a number of variables, such as moisture content of the carrier gas, charcoal type, carrier gas flow rate, poisoning of the bed by impurities, and by temperature gradients in the bed due to fission product decay heating. Based on an air in-leakage rate of 30 scfm into the condenser, with the air dried to a dewpoint of -90°F, and beds filled wtih 8 x 16 mesh coconut base charcoal at 0°F, GE has estimated delay times of 46 hours for krypton and 42 days for xenon. The applicant has based his evaluation on the results of small scale experiments that he has performed. The applicant is currently gathering verifying data on a large scale system at the KRB reactor. We have accepted the applicant's adsorption data (dynamic adsorption coefficients) pending confirmation by the large scale experiments, and have based our evaluation on the above delay times. Topical Report NEDE-10751-1P, which provides the results of large scale tests, has recently been submitted for our review. We will report the results of our review when it is completed after issuance of a PDA.

Leakage from components and systems containing radioactive materials will be released to the atmosphere via the building ventilation systems. Volatile radioactive materials will be released to the containment building atmosphere as a result of relief valve actuations and exhausted without treatment through the containment purge system. Building ventilation systems within the scope of GESSAR that are sources of radioactive gaseous effluents are the containment purge, the drywell purge, the shield building annulus, the auxiliary building, the fuel building and the radwaste buildings. The turbine building ventilation system is also a source of radioactive gaseous effluents; however, it is not included in the scope of the application and will be reviewed for individual license applications that reference GESSAR.

The containment building air is cooled and recirculated at a rate of 42,500 scfm. In addition, the air in the containment dome is recirculated by two 6900 scfm fans. The applicant's design provides for the containment to be at a slight negative pressure by means of a 4300 scfm low volume purge. Fresh makeup air is supplied by a

4300 scfm supply fan. In the event it is required to purge the containment rapidly, the applicant's design also provides 25,000 scfm purge and supply fans.

During normal operation, the drywell is a closed system with air being recirculated and cooled at a rate of 92,000 scfm. When necessary, the drywell may be purged via a 4300 scfm drywell purge vent fan. The shield building annulus will be maintained at a negative pressure by redundant 4200 scfm capacity exhaust and recirculating fans. Areas in the auxiliary building containing ECCS pumps and heat exchangers will be maintained at a negative pressure via redundant 4200 scfm capacity supply and exhaust fans. The fuel handling areas of the fuel building will be maintained at a slight negative pressure by means of redundant 2000 scfm supply and exhaust fans. The ventilation exhaust air from each of the above areas will be monitored and released to the atmosphere without treatment. In the event radioactivity levels in any area are above a predetermined level the exhaust from that area will be diverted to the SGTS.

The radwaste building will be maintained at a slight negative pressure by means of redundant 17,600 scfm capacity supply and exhaust fans. The exhaust air will be filtered through HEPA filters and monitored for radioactivity content before release to the atmosphere. In the event of a high radioactivity level in the exhaust stream, an alarm will sound in the control room and the cells containing radwaste equipment will be isolated. The exhaust can be terminated by a remote manual switch. General Electric does not provide radioiodine control systems for the radwaste building ventilation system, but proposes to employ charcoal adsorbers on vents from selected tanks and equipment.

11.3.2 Gaseous Waste Treatment System Evaluation

We calculate that the noble gas releases from the GESSAR plant will be 5700 Ci/yr, and that 0.28 Ci/yr of I-131 will be released through the building ventilation systems. The applicant estimates 5000 Ci/yr of noble gases and 0.16 Ci/yr of I-131 will be released. The applicant's lower estimate of I-131 releases is based on a smaller rate of steam leakage to the turbine building. Our value for this parameter is 1340 pounds per hour. The basis for our parameter is given in Appendix B to WASH-1258.

All major equipment in the off-gas system is redundant. The glycol cooler equipment is not redundant, but will be located in a non-radioactive area so that it will be accessible for maintenance in the event of equipment malfunction. We find that the off-gas system design has sufficient redundancy to provide reasonable assurance that the system will have the capability to perform its intended function.

We find that continuous purging of the containment directly to the environment without treatment is unacceptable for the reasons listed in Section 6.2.4 of this SER. Either an internal recirculation iodine cleanup system for containment atmosphere cleanup before purging or a charcoal filtration system for continuous operation during purging is required. GE has not agreed to provide either system; therefore, we will make this item a condition of the PDA.

The system seismic and quality classifications are discussed in Section 3.2 of this SER. The staff concern regarding the capability of the SGTS to handle gaseous effluents from the plant ventilation systems is discussed in Section 6.2.3 of this SER.

Offgases from the charcoal delay system and the mechanical vacuum pump will be released to the atmosphere via a roof vent on the turbine building. The design of this vent is outside the scope of GESSAR and will be reviewed on a case-by-case basis when applications referencing GESSAR are received. The major inputs to this roof vent will be monitored and controlled individually and the release of radioactive materials from the vent will be monitored. The offgases from the charcoal delay beds will be monitored and the release automatically terminated if the radioactivity exceeds a predetermined level. The radioactivity in the SGTS exhaust will be monitored and will annunciate radioactivity in excess of predetermined levels in the discharge from the SGTS.

We find that the applicant's design provides adequate control of releases of radioactive materials in gaseous effluents from the condenser offgas system, in accordance with General Design Criterion 60 of Appendix A to 10 CFR Part 50. The provisions for monitoring and control of releases of radioactive materials in gaseous effluents from the turbine building is outside the scope of GESSAR, and will be evaluated on a case-by-case basis when applications referencing GESSAR are received.

11.3.3 Conclusions

The Gaseous Waste Treatment Systems include process equipment and associated instrumentation to collect, store, handle, process and control releases of radioactive materials in gaseous effluents produced as the result of operation of the GESSAR standard plant. The gaseous waste systems include all plant systems that have a potential to release radioactive materials in gaseous effluent to the environment, including building ventilation systems. Our scope of review of the gaseous waste treatment systems has included the applicant's system descriptions, schematic flow diagrams and piping and instrumentation diagrams (P&ID's), the applicant's design objectives for releases of radioactive materials during normal operations, including anticipated operational occurrences, and the capability of the applicant's proposed system to meet the concentration limits of 10 CFR Part 20, Appendix B, Table II, Column 1, for the design conditions and during periods of equipment downtime. Based on this review we conclude that the design utilizes state-of-the-art technology and meets the requirements of 10 CFR 20. We have reviewed the applicant's analysis of the expected releases of radioactive materials in gaseous effluents and have performed an independent calculation of these releases based on the methods and parameters given in Appendix B and C of WASH-1258. We have reviewed the Quality Group and Seismic Design Classification of the proposed treatment systems and the provisions to prevent and withstand hydrogen explosions. We have reviewed the capability of the system to prevent uncontrolled releases of radioactive materials to the environment in gaseous effluents.

As stated in Section 11.1 of this SER, the ability of the design to meet the requirements of 10 CFR 50.34a will be determined on a site-by-site basis.

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Our acceptance criteria for Quality Group and Seismic Design classifications are set forth in Section 3.2. Our acceptance criteria for instrumentation provided to control releases of radioactive materials is based on the requirements of General Design Criterion 60.

We find that the applicant has provided adequate control of releases of radioactive materials in effluents from the condenser offgas system in conformance with the requirements of General Design Criterion 60.

We find the following item does not conform to our acceptance criteria and will be made a condition of the PDA.

Continuous purging of the containment directly to the environment without treatment is unacceptable, and a closed containment with an internal iodine removal system or a continuous purge system with a charcoal filtration system should be provided. Additional information is needed in the following areas to verify the applicant's design. This information is confirmatory and may be submitted during the FDA review. We will report on this in a supplement to our SER.

(1) The applicant's values for the dynamic adsorption coefficients (K_D) for xenon and krypton in charcoal delay systems at low temperatures (0°C) will be confirmed by large scale tests.
(2) Following an in-plant measurement program, the applicant will identify for our review, tanks and components that require charcoal adsorbers on vent lines.

The following items are not included in the scope of GESSAR or are site dependent and will be reviewed for individual license applications:

(1) The capability of the system to meet the dose design objectives of Appendix I to 10 CFR Part 50 for noble gases and iodine;
(2) The capability of the system to limit concentrations and radioactive materials in gaseous effluents to those given in Table II, Column 1 of Appendix B to 10 CFR Part 20 at points at our beyond the site boundary;
(3) The treatment systems, monitoring and control systems, and release points for the turbine building ventilation system.

11.4 Solid Waste Management Systems

The solid waste management system consists of subsystems for handling, storing, solidifying, drumming and shipping wet solid wastes, and for compacting and packaging dry, compressible wastes generated as a byproduct of reactor operation. Wet solid wastes consist of spent filter-demineralizer sludges, spent demineralizer resin beads, diatomaceous earth filter media, and evaporator bottoms. Filter-demineralizer sludges and spent demineralizer resin beads will be collected, dewatered and transferred to a 170 ft^3 shipping container.

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Bottoms from the waste evaporator and the detergent evaporator will be collected and pumped to the shipping containers where a measured amount of cement will be added and mixed by a disposable mixing blade.

The container will be closed and placed in a storage area. Storage will be provided for twelve 170-ft^3 containers, which is sufficient capacity to store one month's production of containers during normal operating conditions or 10 days production of containers during periods of design basis condenser tube leakage. In Amendment 25 to GESSAR, General Electric stated that they "will work with Industry Committees to assess the consequences of free water present in solid radioactive wastes. Should it be established by the Industry Committees and the AEC that free water in solid wastes is a safety problem which requires resolution, GE will establish a program for developing equipment which will detect the presence of free water in solid radioactive wastes. This development program, if required, will be presented to the AEC for their evaluation."

Dry, compressible wastes consisting of spent air filters, rags, clothing, paper, small contaminated tools and solid laboratory waste will be compacted into 55-gallon drums, capped, and stored prior to shipment offsite.

GE estimates that for a plant employing deep bed condensate demineralizer resins, approximately 140 containers will be generated annually containing about 1700 curies.

Based on experience at operating BWR's, we estimate that 110 containers per year containing approximately 2400 curies will be generated at plants using either Powdex or deep-bed condensate demineralization systems, with essentially the same isotopic content as calculated by the applicant. We estimate that 450 drums of compressed dry wastes will be shipped from the site annually, containing a total of less than 5 curies of radioactive materials.

We find the seismic and quality classification of the components and the structure acceptable. We conclude that the design supplied in GESSAR utilizes state-of-the-art technology and meets the requirements of 10 CFR 20. We also conclude that the system has sufficient capacity and redundancy to perform its intended function during periods of normal operation, including anticipated operational occurrences.

11.5 Process and Effluent Radiological Monitoring Systems

11.5.1 System Description

The process and effluent radiological monitoring systems will be designed to provide information to operations personnel on radiation levels in plant process streams, to initiate operation of emergency systems, to provide inputs to the reactor protection system, and to record the rate of release of radioactive materials in plant effluents.

The applicant has provided radiation monitors to monitor and control the releases of radioactive materials in gaseous effluents from the offgas system vent, the containment, drywell and shield building annulus purge exhausts, and the ventilation exhausts from the radwaste, auxiliary and fuel buildings.

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The applicant provides the capability to obtain liquid samples from the effluents from the excess water storage tank and the waste demineralizer. Provisions have also been made to obtain gaseous samples upstream and downstream of the offgas system, upstream of the steam jet air ejector and from points within the offgas system. The applicant has provided an area monitor in the charcoal bed vault to detect leakage from the charcoal delay beds.

11.5.2 System Evaluation

The provisions for process and effluent radiological monitoring include the instrumentation and controls for monitoring and controlling the releases of radioactive materials in plant effluents and monitoring the level of radioactivity in process streams. The scope of our review included the provisions for monitoring and controlling the release of radioactive materials in plant effluents in accordance with General Design Criteria 60 and 64 and Regulatory Guide 1.21, and for monitoring radioactivity levels within the plant in process streams in accordance with General Design Criterion 13.

The basis for acceptance in our review has been conformance of the applicant's design, design criteria, and design bases for the process and effluent monitoring systems to the Commission's Regulations as set forth in the General Design Criteria and to applicable Regulatory Guides, as referenced above, as well as staff technical positions and industry standards.

The radiation monitors for the turbine building ventilation exhaust are outside the scope of GESSAR, and will be reviewed for individual license applications.

The type of instrument, range, set point, sensitivity, calibration frequency, and provisions for maintenance and testing are outside the scope of GESSAR and will be reviewed for individual license applications at the FSAR stage.

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12.0 RADIATION PROTECTION

The staff has evaluated the proposed radiation protection program presented in GESSAR. The review was conducted to determine that the program satisfies the following objectives: (1) to assure that radiation exposures to operating personnel and to the general public will meet the requirements of 10 CFR Parts 20 and 50, and (2) to assure that occupational radiation exposures (ORE) to operating and construction personnel during normal operation and anticipated operational occurrences (including refueling, purging, fuel handling and storage, radioactive material handling, processing, use, storage and disposal, maintenance, routine operational surveillance, and inservice inspection and calibration) will be as low as practicable (ALAP).

12.1 Shielding

The shielding for GESSAR is designed primarily to protect operating personnel and the general public from radiation emanating from the reactor, power conversion, process and auxiliary systems while maintaining suitable access for operation and maintenance. To meet the design objectives, the facility design classifies all areas of the plant into radiation zones based on the access requirements of the area. Zone dose rate values are based on operating experience from the large BWR plants that have been in operation for many years. Both operating conditions and shutdown conditions have been considered in designing the radiation shielding to meet the zone dose rate criteria in specific areas. On the basis of our evaluation of the radiation dose rates in the various areas of the plant where shielding separates the sources from normally occupied areas, we conclude that shielding is appropriately utilized and will be conservatively designed.

Early in the review, it was not apparent that layout and other design features for radioactive fluid processing, transporting and storage equipment indicated that adequate consideration had been given to assuring that ORE will be ALAP. Equipment design and layout were compared with the practices in Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposures As Low As Practicable (Nuclear Reactors)," which was formulated primarily with regard to areas within shielding where experience has shown that the major ORE occur. In response to our concerns relative to Regulatory Guide 8.8, GE supplied additional information and assure us that equipment design and layout measures will be consistent with Regulatory Guide 8.8 although the guide is not specifically referenced. The design has been under continuous review by a competent health physicist, as recommended by Regulatory Guide 8.8, and the extensive experience with BWR plant operation acquired by the applicant through interaction with utility BWR plants is utilized.

In a number of instances the use of proper personnel practices and procedures to assure that ORE will be ALAP is emphasized. The utility purchasing the standard plant will ultimately be responsible for assuring that ORE will be ALAP and these aspects will be reviewed on specific applications.

Examples of specific plant design features for minimizing personnel radiation exposure during maintenance have been provided. Even though detailed plant arrangements and equipment designs are not available at this stage of the design, GE has considered further design activities for lowering dose during maintenance. These include selecting low maintenance, reliable equipment, designing for fast access and egress to maintenance areas, choosing quick disconnect and replacement parts and equipment, provisions for temporary shielding, adequate working space and provisions for remote viewing, and other such measures. The utility applicant proposing to use the BWR-6 system and plant will have to specify in more detail the restrictions and controls that will be implemented to assure that ORE will be ALAP.

The area radiation monitoring system has the objective of indicating and recording abnormal gamma radiation levels in areas where radioactivity is present or may be inadvertently introduced, and to monitor the radiation levels in areas where personnel may need to be for whatever reason. Table 12.1.3 of GESSAR indicates area monitoring. Area radiation monitors are able to receive power from an Engineered Safety Feature power supply. We will review the specific area radiation monitoring system details for specific utility applications.

Operating procedures will be the responsibility of the utility applicant.

The analysis that determines whether the plant design assures that ORE will be ALAP is the estimate of exposure, and is covered in GESSAR Section 12.1.6 and in answers to several of our questions. GE has estimated ORE on the basis of experience gained in operating BWR plants. While this experience should place an upper limit on expected ORE, the many improved design features incorporated since the earlier designs should result in ORE that are lower than those experienced in the older operating plants. Using data that the applicant has provided in GESSAR, the utility applicant that references the BWR-6 Nuclear Island will be able to provide a dose assessment in their application, based on the way they expect to operate the plant. It is our position that the plant design provided in GESSAR is acceptable and provides assurance that occupational radiation exposures can be ALAP, provided the operation by the utility applicant applies additional principles of a good radiation protection program, as presented in Chapter 12 of the Regulatory Standard Review Plans and Regulatory Guide 8.8.

12.2 Ventilation

The proposed ventilation system has the objective of providing effective protection for operating personnel against possible airborne radioactive contamination. Appropriate design features have been incorporated to make certain that the objectives are met and that airborne radioactivity levels for normal operation, including anticipated operational occurrences, are within the limits of 10 CFR Part 20, Appendix B, Table I for areas within plant structures and on the plant site where construction workers and visitors are permitted.

The staff has evaluated the doses to personnel potentially occupying the BWR-6 Mark III containment during a reactor pressure relief valve actuation at full power.

Estimates of noble gas and ^{16}N concentrations in the containment annulus following steam releases to the suppression pool have been made. Calculations based on these source estimates, the estimated twice yearly frequency of occurrence, the containment location of personnel and assumed egress time factors, dose factors for specific radionuclides, and the fraction of gamma infinite dose as a function of gamma energy for conditions of finite geometry, show that personnel occupying containment could receive an annual dose equivalent of 0.025 Rem to the whole body and 1.53 Rem to the skin.

Dose calculations were also made for noble gases and ^{16}N concentrations following an isolation scram to the suppression pool that is estimated to occur once per year in the Mark III containment. Again, based on our estimates of source terms and occupancy factors, our calculations show that a dose equivalent of 0.044 Rem to the whole body and 2.3 Rem to the skin could be received due to this event.

Doses received during the occurrence of each type event will be less than the quarterly limit of 10 CFR 20.101. Although we consider the parameters used to calculate the source terms and the doses to be reasonable, the probability of an operator being in the location considered when the venting occurs is not large and therefore the actual doses should normally be less than those calculated. Under these circumstances we feel that neither restrictive measures nor administrative controls will be required.

The airborne radioactivity monitoring system consists of monitors in various building exhaust control systems. It is stated that this system will provide a clear indication to operations personnel when abnormal amounts of radioactivity exist in the exhaust from the buildings involved. Sensitivity of these monitoring systems, and other system details are left to final design stage. It is the staff's position that the systems should be sufficiently sensitive and extensive to be able to detect a level of airborne radioactivity in any room in the pertinent building at MPC level. The utility applicant will have to review the need for airborne radioactivity monitoring instrumentation, since it has been our experience that the use of such systems is dependent on the kind of health physics program that is developed by the utility radiation manager. Fixed airborne radioactivity monitoring, to meet the requirements of 10 CFR Part 20, 20.103 and 20.201, will be reviewed for specific applications

12.3 Health Physics

This section will not be addressed, since the utility applicant supplies the entire health physics program.

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15.0 ACCIDENT ANALYSIS

15.1 General

Two basic groups of events pertinent to safety are separately evaluated in this section: abnormal operational transients, and accidents. In order for the analysis of events in either group to be acceptable, it is required that an accurate model of the reactor core be used, and that all appropriate systems whose operation (or postulated misoperation) would effect the event be included. Transients are analyzed to assure that they will not cause damage to either the fuel or to the reactor coolant pressure boundary. Accidents, which are far less likely to occur than transients, may result in some fuel damage; they are analyzed to determine the extent of fuel damage expected and to assure that reactor coolant pressure boundary damage, beyond that assumed initially by the accident, will not occur.

The acceptability criteria of analysis results for transients are that no fuel barrier (clad) damage occurs (a sufficient, but not necessary, condition to meet this requirement is that the MCPR remain above 1.07) and that peak nuclear vessel pressure not exceed 110% of the design pressure (ASME Codes, Section III, Class I are met if nuclear system pressure remains below 1375 psig, which is 110% of the 1250 psig design pressure). These two requirements demonstrate, respectively, that the first radioactive material barrier (the clad) and the second barrier (the pressure vessel) are protected for abnormal operational transients.

For design basis accident analyses, which evaluate situations that require functioning of the engineered safety features (including containment), it is necessary to assure that no catastrophic fuel failures and no damage beyond that already assumed to the RCPB occur. This is done by insuring that peak fuel enthalpy remains below 280 cal/gram, the limit used in Regulatory Guide 1.77 for the PWR rod ejection accident analysis and accepted by the staff for use as a fuel safety limit for BWR's. The 280 cal/gm energy density value provides a conservative maximum limit to ensure that core damage from postulated events will be minimal and that both short-term and long-term core cooling capability will not be impaired. Also, the peak clad temperature must remain below 2200°F (10 CFR 50.46). GE will meet these limits.

For postulated accidents for which fuel damage is calculated, the extent of damage is determined by correlating fuel energy content, cladding temperature, fuel rod internal pressure, and cladding mechanical characteristics. These correlations are substantiated by fuel rod failure tests and are presented in Section 3.5 and Section 6 of GESSAR.

15.2 Abnormal Operational Transients

Abnormal operational transients are the result of single equipment failures or single operator errors that can reasonably be expected during any mode of operation. The applicant has provided analyses of various abnormal operational transients in GESSAR.

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These analyses include such events as process system control malfunctions, inadvertent control rod withdrawal, turbine trip, loss of electrical load, and variations in operating parameters.

Eight nuclear system parameter variations are listed as potential initiating causes or threats to the fuel and reactor coolant pressure boundary. These parameter variations in the analyzed transients are as follows:

(a) Nuclear System Pressure Increase. Transients analyzed in this group included loss of load events such as a generator trip, turbine trip, loss of condenser vacuum, closure of one or all of the main steam line isolation valves, and malfunction of the reactor primary system pressure regulator.
(b) Reactor Water Temperature Decrease. These transients included events that might cause a power surge by reduction of the reactor primary coolant water temperature. They included malfunction of the feedwater control in a direction to increase feedwater flow, loss of a feedwater heater, shutdown cooling malfunction, and inadvertent activation of an auxiliary cold water system.
(c) Reactivity Insertions. These transients include rod withdrawal transients from zero reactor power, hot critical condition, and from full power; fuel assembly insertion, and control rod removal errors during refueling.
(d) Reactor Water Inventory Decrease. These transients included events leading to decrease in the inventory of reactor primary coolant such as loss of auxiliary power, loss of feedwater, pressure regulator failure in a direction to cause decreasing reactor system pressure, inadvertent opening of a safety relief valve and opening of condenser bypass valves.
(e) Primary Coolant Flow Decrease. These transients included failure of one or more recirculation pumps or malfunction of the recirculation flow control in a direction to cause decreasing flow.
(f) Reactor Coolant Flow Increase. These transients included events that might increase the recirculation flow and thus induce a positive reactivity insertion. They included a malfunction of the recirculation flow controller in a manner to cause increasing primary coolant flow and the startup of a recirculation pump that had been on standby.
(g) Core Coolant Temperature Increase. The transient analyzed in this category was loss of shutdown cooling.
(h) Excess Coolant Inventory. The transient analyzed in this group was feedwater controller failure to maximum demand.

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The need for increasing the negative reactivity insertion rate is related to the operating objective of GE and the utility applicants to continue full rated power operation into the end-of-cycle-life period. From current experience with BWR operation it may be expected that GESSAR and similar plants with control rod drive systems of current design may find it necessary to reduce power somewhat during the last 10-20% of each cycle. This situation will occur, if it occurs at all, as the reactor core approaches its equilibrium fuel cycle.

In the event of a sudden loss of normal heat removal capability which can occur as a result of loss-of-load transients such as turbine trip or generator trip, sudden reactor coolant system pressure increases will occur. The increased pressure causes collapse of steam voids that were present in the core, which in turn causes a power increase due to the positive reactivity effect of void collapse. This power increase then tends to further increase pressure. The above cycle of events is terminated by reactor scram (rod insertion), but toward end-of-fuel-cycle in BWR's the time required to achieve effective power reduction from rod insertion is somewhat increased. This is because the rods have further to travel from their end-of-cycle position (mostly or completely withdrawn from the core and ready for insertion from below the core) and because the rods must reach the more reactive region (which is nearer the top of the core at end-of-cycle) to achieve full effectiveness.

This operating condition has been studied by GE, the NRC staff and our consultant (Brookhaven National Laboratory). Analyses of core dynamics performed for certain events (such as turbine trip without bypass) for an equilibrium core operating at full power near its end-of-cycle, and assuming a number of conservative assumptions, show that without further analyses, BWR's employing control rod drive systems of current design might require a limited decrease in acceptable power level during the last 10-20% of each near equilibrium operating cycle. The staff will complete its review of this matter on the GESSAR 238 Nuclear Island application (Docket No. STN 50-447) and on other applications in which the General Electric Company is the common supplier for the nuclear steam supply system. If it is determined that additional negative reactivity insertion capability is needed or desired, design changes will be required if full rated power operation is desired during the end of each operating cycle as the core approaches its equilibrium reload pattern.

To cope with this potential end of cycle problem, GE originally proposed a prompt relief trip (PRT) system. However, during the course of the GESSAR review, GE modified the GESSAR design to include a fast scram system to replace the earlier PRT design (both PRT and "fast scram" are discussed in more detail in Section 4.2.3).

The staff has not had an opportunity to review the design of the fast scram system. The fast scram system will also result in changes to the transient analyses presently in GESSAR Section 15.1.

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In Amendment 26 to GESSAR, GE stated that all plant transients which employ the rapid insertion (fast scram) as an analysis parameter will meet the performance requirements shown in GESSAR required to comply with established safety criteria. The fast scram system can meet these performance requirements by inserting the present control rods at a faster rate.

The fast scram is a reasonable extension of the existing control rod scram system which has been approved and operated for many years on previous BWR plants. The staff therefore believes that there is reasonable assurance that the scram system can meet the performance requirements established in GESSAR and limit the consequences of transient events to acceptable levels. However, GE is required to submit information on the design and performance of this system and we will review this information when the details of the fast scram system are available.

15.3 Design Basis Accidents

In order to demonstrate the effectiveness of the engineered safety features, we computed typical offsite doses resulting from the Loss-of-Coolant, Fuel Handling, and Control Rod Drop Accidents. Our acceptance criteria are that the doses from these postulated accidents (as evaluated by the Regulatory staff) be within the exposure guidelines of 10 CFR Part 100. As indicated in Regulatory Guide 1.3, the doses considered appropriate at the CP or PDA stage are not more than 150 rem thyroid and 20 rem whole body.

The charcoal filters of the Standby Gas Treatment System have been given credit for 99% efficiency in removal of all species of iodine during the LOCA, because they comply with the intent of Regulatory Guide 1.52. We have evaluated the consequences of a LOCA with credit given for mixing within the annulus.

On the basis of our experience with the evaluations of the steam line accident for BWR plants of similar design, we have concluded that the consequences of this accident can be controlled by limiting the permissible radioactivity concentrations in the reactor coolant so that potential offsite doses are small. We will include limits in the Technical Specifications on the coolant activity concentrations such that the potential two-hour doses at the exclusion radius, as calculated by the Regulatory staff for these accidents, will be appropriately small fractions of the guideline values of 10 CFR Part 100.

15.3.1 Loss-of-Coolant Accident (Radiological Considerations)

A design basis loss-of-coolant accident has been postulated for the GESSAR design. Estimates of dose consequences from containment leakage and operation of the main steam line isolation valve leakage collection system (MSIV-LCS) have been made. No estimates have been made of doses due to the proposed continuous operation of the containment purge system or due to the operation of any other leakage control systems. These are potentially significant release paths to the environment. GE intends that the design of the containment purge system

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will preclude the release of any activity in the event of a LOCA and further intends that there will be no path for fission product release from the operation of leakage control systems other than the MSIV-LCS (i.e., all other systems will have positive means of preventing leakage).

The calculations performed in the original SER issued November 1974 were based on a containment leak rate of 1%/day and various bypass leakages. These calculations indicated that the design would not be adequate for a site typical of those previously reviewed by the staff in recent licensing cases. The design has subsequently been improved by letter from I. Stuart dated 8/22/75; the applicant specifies a total design basis primary containment leakage rate of 0.3%/day and details the leak paths and the fraction of the primary containment leakage through each path. Leakage through the main steam line isolation valves (MSIV) is included in the total design basis leak rate of 0.3%/day. Because the MSIV leak path is treated separately, it was not included in the containment leakage calculation. This assumption is considered sufficiently conservative because testing will be required in the plant technical specifications to demonstrate that the containment leakage, excluding the main steam line isolation valve leakage, does not exceed this reduced leak rate.

The data for the model used for the containment leakage calculation are presented in Table 1, and the leakage paths are illustrated in Figure 1. This model assumes that the primary containment leakage to the annulus goes directly to the intake of the Shield Building Annulus Recirculation and Exhaust System (SBARES) with a fraction (pathway T_2, Figure 1) being exhausted to the Standby Gas Treatment System (SGTS) and the balance (pathway T_1, Figure 1) being recirculated to the annulus where it is assumed mixed in 50 percent of the annulus free volume. The split between exhaust fraction and recirculation fraction is a function of the flow rates in these respective paths.

The data used to estimate the consequences due to operation of the MSIV-LCS are presented in Table 2. The model assumes the source is uniformly distributed in the drywell free volume and that all valves are leaking at the Technical Specification limit of 11.5 cfh per valve (see Section 9.3.1 of this report). The discharge from the MSIV-LCS is assumed to be physically directed to the return header of the SBARES and mixed in 50% of the annulus free volume prior to release to the atmosphere. This calculation assumes the following requirements will be met by the applicant:

a) To support the assumption of no actuation of the inboard LCS if the inboard MSIV fails to close (actuation would result in the direct release of containment atmosphere at 448.9 cfm for 1 minute), a positive interlock will be provided on each inboard MSIV which precludes actuation of the inboard LCS unless the inboard MSIV is fully closed.

b) To eliminate the potential for substantial additional exposures during the 0-2 hour time period, the setpoint for the flow element

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TABLE 1

ASSUMPTIONS USED TO ESTIMATE LOCA DOSES FROM CONTAINMENT LEAKAGE FOR THE GESSAR DESIGN

Power Level	3758 MWt
Operating Time	3 years
Total Leak Rate (Containment Plus MSIV)	0.3%/day
Core Fraction Available for Leakage from Containment:	
Iodines	25%
Noble Gases	100%
SGTS Filter Efficiency for all Iodines	99%
Primary Containment Volume	1.168×10^6 ft^3
Shield Building Annulus Volume	5.04×10^5 ft^3
Mixing Fraction in Annulus Volume	50%

Shield Building Recirculation System Flow Rate, cfm:

Time Period	Exhaust (T_4)	Recirculation
0 - 27 seconds	0	0
27 seconds - 30 minutes	1000	4000
30 minutes - 2 hours	700	4300

Primary Containment Leak Paths:

Time Periods	Percent of Total Leakage TO Annulus (T_1)	To SGTS (T_2)	To Environs (T_3)	Through MSIV*
0 - 27 seconds	60.5	0	8	31.5
27 seconds - 1 minute	48.4	12.1	8	31.5
1 minute - 30 minutes	48.4	20.1	0	31.5
30 minutes - 2 hours	52.	16.5	0	31.5
2 hours - 30 days	52.	16.5	0	31.5

X/Q Value Assumed for Exclusion Area Boundary (0 - 2 hours) 1.0×10^{-3} sec/m^3

*Not included in Containment Leakage Calculation. See Tables for MSIV Leakage Assumptions.

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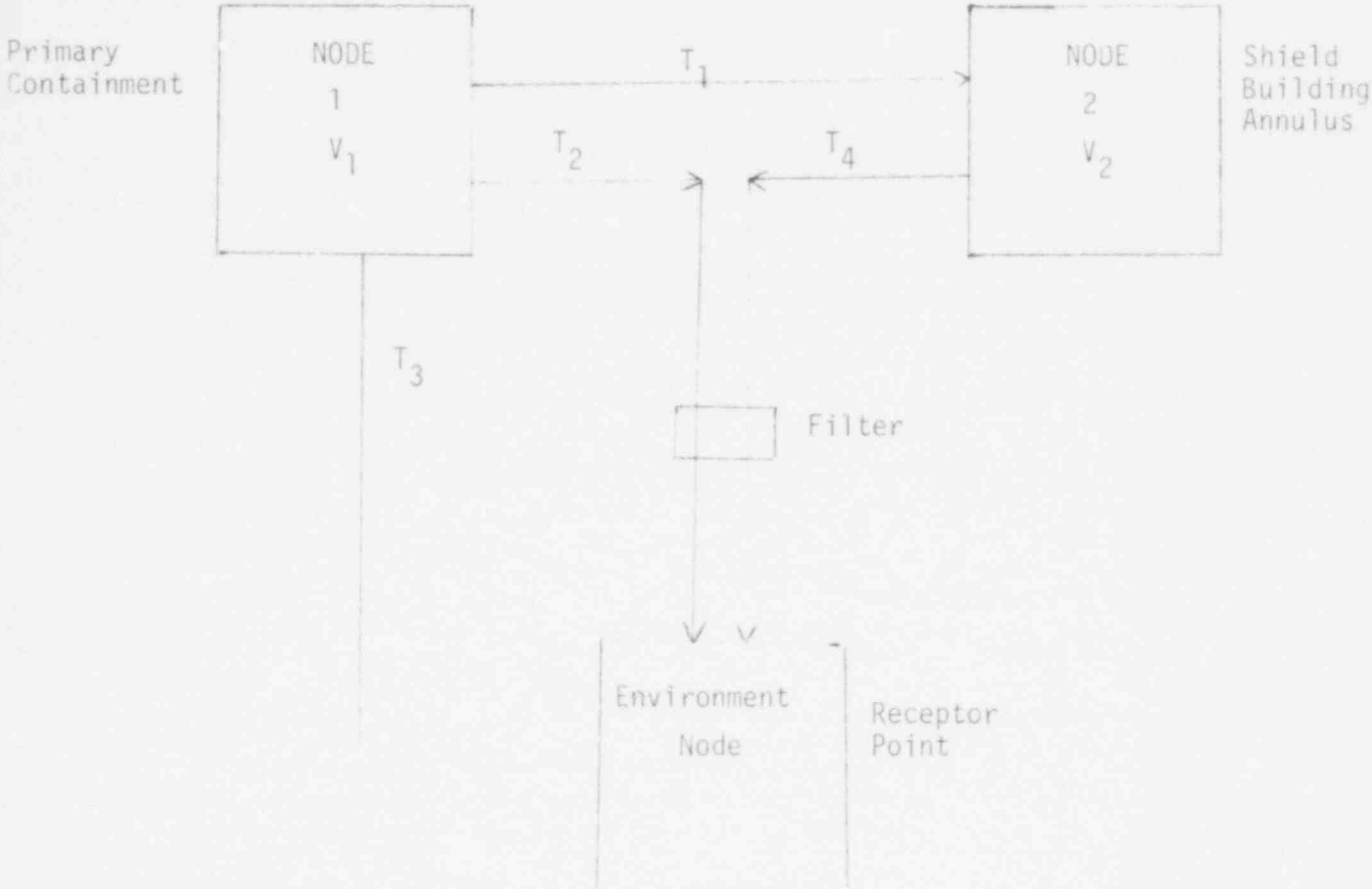
FIGURE 1

GESSAR CONTAINMENT LEAKAGE DOSE MODEL



T_1 = transfer from Node 1 to 2

T_2 = transfer from Node 1 to the environment thru a filter

T_3 = transfer from Node 1 to the environment

T_4 = transfer from Node 2 to the environment thru a filter

V_1 = volume of primary containment

V_2 = available volume of annulus (V_2 = 50% of actual free volume)

TABLE 2

ASSUMPTIONS MADE TO ESTIMATE CONSEQUENCES OF
MSIV-LCS OPERATION FOLLOWING A
POSTULATED LOCA AT A GESSAR PLANT

Power Level, Mwt	3758
Operating Time, years	3
Core Fraction Released to Drywell:	
Noble Gases	100%
Iodine	25%
Iodine Filter Efficiency (SGTS)	99%
Drywell Free Volume, ft^3	274,000
Shield Building Annulus Free Volume, ft^3	504,000
Shield Building Mixing Fraction	50%
Exhaust Flow Rate, cfm	
0 - 30 mins.	1000
30 mins. - 30 days	700
Recirculated Flow Rate, cfm	
0 - 30 mins.	4000
30 mins. - 30 days	4300
MSIV Leak Rate, cfh/valve (4 lines)	11.5
Delay Time to Release	2.9 hours

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timers in the inboard LCS will be 11.5 cfh. This low setpoint will assure that the transport time to the LCS will be greater than two hours and no contribution from this leak path will need to be added to the two-hour dose.

c) To assure mixing of the LCS releases in the shield building annulus volume, the discharge from all points of the LCS will be routed to the return header of the SBARES as shown in Figure 9.4-8a of GESSAR.

We will appropriately condition the PDA on points a) and b).

Because of the low leak rate past the valves, no activity is assumed to reach the MSIV-LCS for 2.9 hours, thereby precluding any 0 - 2 hour exclusion area boundary dose. This assumption is based on a delay of 10 minutes to actuate the MSIV-LCS, a one minute depressurization, and 163 minutes at a flow rate of 11.5 cfh to force the balance of steam out of the pipe so that contaminated atmosphere reaches the MSIV-LCS. This delay was calculated assuming laminar flow. Table 3 presents the assumption used to estimate the appropriate delay time.

To estimate doses for a standard plant like GESSAR, some assumptions must be made concerning certain site-related parameters such as meteorology. The staff selected the value of 10^{-3} sec/m^3 for the short term (0 - 2 hours) exclusion area boundary X/Q value which is considered to envelope about 70% of the sites in the U.S. as discussed in Section 2.3.4. The results obtained for GESSAR may then be extrapolated to any particular site simply by multiplying the consequences presented in Table 4 by the ratio of the site X/Q value to the value used here.

However, with regard to the estimate of the dose consequences for the low population zone boundary, the use of four time periods to calculate the dose complicates the estimate for the standard plant. Therefore, the staff assumed a spectrum of atmospheric diffusion conditions which are representative of locations in the U.S. assuming a two-mile LPZ distance. These X/Q values are presented in Table 5 and the containment leakage dose, the MSIV-LCS operation dose, and the total LOCA does estimate for these two release paths for each example are presented in Table 6.

Assuming a two-mile LPZ boundary, we conclude that the GESSAR design will meet the appropriate exposure guidelines at a majority of sites in the U.S. Larger LPZ boundary distances can compensate for sites with less favorable atmospheric diffusion conditions than those considered in Table 5.

TABLE 3

ASSUMPTIONS USED TO ESTIMATE DELAY TIME IN MAIN STEAM LINE

Pipe volume between valves	150 ft^3
Initial temperature	500°F
Initial pressure	35 psia
Mole volume	18.45 liters
Moles of steam in pipe volume prior to depressurization	230
Moles of steam in pipe volume after depressurization (isothermal expansion to 12 psia)	78.9
Valve leak rate	11.5 scfh or 14.5 moles/hour
Leakage into pipe volume during depressurization	0.242 moles
Leak rate into pipe volume after depressurization	0.184 pipe volume/hour
Time to release one pipe volume assuming plug flow	326 minutes
Assuming laminar flow is twice plug flow: Time to release one pipe volume	163 minutes

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TABLE 4

ESTIMATED CONSEQUENCES FROM CONTAINMENT LEAKAGE FOLLOWING A POSTULATED LOCA AT A GESSAR PLANT

	0 - 2 Hour Doses**, Rem	
	Thyroid	Whole Body
Hypothetical Exclusion Area Boundary	17	10

*Dose calculation model used is that presented in Regulatory Guide 1.3.

** Example only. The acceptability of a particular application referencing GESSAR will be determined based on the meteorological dispersion values derived from an analysis of onsite data. The doses would be a direct ratio of the X/Q determined at the site to the X/Q used in this analysis (10^{-3} sec/m^3).

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TABLE 5

COMPARISON OF X/Q VALUES AT 2 MILES
FOR VARIOUS EXAMPLE ATMOSPHERIC DIFFUSION CONDITIONS
AND REGULATORY GUIDE 1.3

Source	X/Q Values, sec/m³ — Time Periods, hours: 0 - 8	8 - 24	24 - 96	96 - 720
Regulatory Guide 1.3	1.13-4	2.3-5	8.14-6	1.76-6
Example A	1.1 -4	7.0-5	2.7 -5	7.0 -6
Example B	3.5 -4	6.6-5	2.6 -5	6.9 -6
Example C	3.2 -5	2.1-5	9.1 -6	2.6 -6
Example D	5.1 -5	4.3-5	2.5 -5	9.3 -6
Example E	8.8 -5	5.7-5	2.2 -5	5.6 -6

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TABLE 6

ESTIMATED 0 - 30 DAY DOSE CONSEQUENCES*
FOLLOWING A POSTULATED LOCA AT A GESSAR PLANT
FOR EXAMPLE ATMOSPHERIC DIFFUSION CONDITIONS

Meteorology at 2 miles (X/Q's) from:	From Containment Leakage		From Operation of the MSIV-LCS		From Both Leak Paths	
	Doses, rem		Doses, rem		Doses, rem	
	Thyroid	Whole Body	Thyroid	Whole Body	Thyroid	Whole Body
Regulatory Guide 1.3	18	4	21	4	39	8
Example A	37	7	58	9	95	16
Example B	56	14	69	12	125	26
Example C	12	2	19	3	31	5
Example D	31	5	53	7	84	12
Example E	30	6	48	7	78	13

* Examples only. The acceptability of a particular application referencing GESSAR will be determined based on the meteorological dispersion values derived from an analysis of on-site data.

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15.3.1.1 Hydrogen Purge Dose Analysis

The applicant will provide redundant hydrogen recombiners for the purpose of controlling the concentration of hydrogen within containment following a design basis LOCA. For use in the event of failure of both recombine , the applicant will provide a backup purging mode. Purging of the containment will be conducted through the standby gas treatment system to minimize the radiological consequences. We have evaluated the additional dose an individual might receive due to purging the containment following the design basis LOCA for various example atmospheric diffusion conditions. The assumptions and input parameters we used in evaluating the consequences of this mode of operation and the calculated doses are listed in Table 7. The purge dose, when added to the computed LOCA doses would have to be within 10 CFR Part 100 guidelines to be acceptable. This will be reviewed on a case-by-case basis, but this does not appear to be limiting for four of the five examples of atmospheric diffusion conditions.

15.3.2 Fuel Handling Accident

In this accident, it is assumed (See GESSAR Section 15.1.41.5.1.2 for discussion) that a fuel assembly is dropped during refueling operations, and that as a result of the fall, 98 fuel rods are damaged. We have evaluated the GE analysis and conclude that the assumption of 98 fuel rods damaged is acceptable. The assumptions for this accident are also consistent with the conservative assumptions of Regulatory Guide 1.25 and are given in Table 15.3-8. Activity released to the environs is assumed to be released through the SGTS deep-bed, charcoal filters within a 2-hour period.

Using an assumed value for X/Q of 1.0×10^{-3} for calculational purposes, the resulting dose would be about 2 rem thyroid and about 5 rem whole body. Thus, the consequences of the loss-of-coolant accident are more limiting.

The staff will review the consequences of the fuel handling accident at particular sites for each CP application referencing GESSAR.

15.3.3 Control Rod Drop

The postulated control rod drop accident assumes that a bottom entry control rod has been fully inserted and becomes stuck in this position, unknown to the reactor operator. The drive is then assumed to become uncoupled and fully withdrawn. The rod subsequently falls from the core, inserting an amount of reactivity corresponding to its reactivity worth.

In evaluating the radiological consequences of this accident, we made assumptions given in Table 8 that are based upon the applicant's analytical model as presented in the PSAR.

In the analysis, the rod is assumed to drop out of the core during startup, which occurs 30 minutes after shutdown from full power operation. This is assumed to cause 770 fuel rods to have a total energy in excess of 170 cal/gm which perforate, releasing 100% of their contained noble gases and 50% of their contained halogens,

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TABLE 7

ASSUMPTIONS AND ESTIMATED CONSEQUENCES*
FOR A HYDROGEN PURGE OF A GESSAR PLANT
FOLLOWING A POSTULATED LOCA

Power Level	3758 Mwt
Containment Volume	1.442×10^6 ft^3
Purge Duration	30 days
Holdup Time	15 days
Purge Rate	30 cfm
Filter Efficiency for Iodine	99%

Meteorology at 2 Miles from:	4-30 day X/Q, sec/m^3	Doses, rem	
		Thyroid	Whole Body
Regulatory Guide 1.3	1.76×10^{-6}	10.	0.6
Example A	7.0×10^{-6}	39.	2.
Example B	6.9×10^{-6}	38.	2.
Example C	2.6×10^{-6}	14.	0.8
Example D	9.3×10^{-6}	52.	3.
Example E	5.6×10^{-6}	31.	2.

* Examples only. The acceptability of a particular application referencing GESSAR will be determined based on the meteorological dispersion values derived from an analysis of on-site data.

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TABLE 8

REFUELING ACCIDENT

Shutdown Time	24 hours
Total Number of Fuel Rods in the Core	46,116
Number of Fuel Rods Involved in the Refueling Accident	98
Power Peaking Factor	1.5
Iodine Fractions Released from Pool	
Elemental	75%
Organic	25%
Filter Efficiencies	
Elemental	99%
Organic	99%
X/Q Values, Sec/m^3	
0 - 2 hours	1×10^{-3}

CONTROL ROD DROP ACCIDENT

Number of Fuel Rods Involved	770
Fraction of Fission Product Inventory Released to Coolant	
Noble Gases	100%
Iodines	50%
Iodine Fraction Released to Condenser	10%
Iodine Fraction Plated Out in Condenser	50%
Condenser Leak Rate	0.5%/day
X/Q Values, Sec/m^3	
0 - 2 hours	1×10^{-3}
0 - 8 hours	1×10^{-4}
0 - 24 hours	1×10^{-4}

to the reactor coolant system. The perforated rods are assumed to have operated at a level 50% greater than the average rod (1.5 peaking factor). Of the halogens released, 90% are assumed to be retained in the primary system and half of the remainder is removed by plate-out. Thus, all of the noble gases and 2.5% of the halogens in the affected rods are assumed to be available for release. Detection of a high radiation signal in the main steam lines automatically closes the main steam line isolation valves, shuts down the mechanical vacuum pump and closes the isolation valve downstream of the pump. The activity entrained in the condenser is assumed to be released at ground level from the turbine building by leakage from the condenser at the rate of 0.5% of the condenser volume per day for a duration of one day.

The calculated two-hour doses, assuming a X/Q of 1.0×10^{-3} for calculational purposes, are 22 rem thyroid and about 1 rem whole body. The doses for the course of the accident are calculated to be about 13 rem to the thyroid and < 1.0 rem whole body, assuming a X/Q of 1.0×10^{-4} for calculational purposes.

The staff will review the consequences of the control rod drop accident at the particular site for each CP application referencing GESSAR.

15.4 Anticipated Transients Without Scram (ATWS)

Anticipated Transients Without Scram (ATWS) was identified as a generic area of concern by the ACRS during their review requiring resolution acceptable to the NRC staff. The Regulatory staff's requirements with respect to ATWS are provided in the staff's technical report on "Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, dated September 1973. As applied to GESSAR, these requirements are that plant changes will have to be made to make ATWS consequences acceptable. In a letter to the staff dated February 19, 1974, GE made a commitment to provide measures to accomplish this. The program for implementation of these measures in GESSAR is described in Appendix A, paragraph II.B of WASH-1270.

By letter dated November 6, 1974, GE submitted Amendment 23 to GESSAR which stated that based on topical report NEDO-20626, "Studies of BWR Designs for Mitigation of Anticipated Transients Without Scram," they had concluded that the GESSAR design already satisfies the requirements of WASH-1270.

Topical report NEDO-20626 is being reviewed by the staff on a generic basis. Based on our review of NEDO-20626 to date, we do not agree with GE that GESSAR already satisfies the requirements of WASH-1270. We expect to complete our review and publish our results by the end of 1975. We will require that this area of concern be resolved prior to the FDA review. We believe that appropriate measures to make the consequences of ATWS acceptable are technically feasible and conclude that any changes required as a result of our review should be incorporated in the first application referencing GESSAR.

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publish our results by the end of 1975. We will review this matter to assure that it is appropriately resolved on a schedule consistent with the overall staff generic position for resolution of ATWS.

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17.0 QUALITY ASSURANCE

17.1 General

The description of the Quality Assurance (QA) Program for the General Electric (GE) BWR/6 Mark III is contained in Section 17 of the GE Standard Safety Analysis Report (GESSAR). Our evaluation of the QA Program is based on a review of this information and discussions and meetings with GE to determine how their QA Program complies with the requirements of Appendix B to 10 CFR Part 50 and the applicable Regulatory guides.

17.2 Organization

GE's Nuclear Energy Products Division (NED) is responsible for the design, manufacturing, inspection and testing of Boiling Water Reactors (BWR), and these activities are accomplished by the Boiling Water Reactor Operations (BWRO) Division. The BWRO Division has four major departments (Note Figure 17.1-1 of GESSAR): Product & Quality Assurance Operations (P&QAO), Nuclear Fuel Department (NFD), BWR System Department (BWRSD), and BWR Projects Department (BWRPD). Each of these departments is headed by a General Manager who reports directly to the BWRO Deputy Division General Manager. The major functions of these four departments are summarized and described in Figure 17.1-8 of GESSAR.

The Deputy Division General Manager has established a BWRO Quality Council chaired by the Manager of P&QAO and consisting of Managers from the major organizations in the Division. The Quality Council is responsible for assuring total quality uniformity and consistency throughout the design and manufacturing of BWR and for keeping the Deputy Division General Manager abreast of quality-related matters.

The QA organizations within BWRO and their reporting levels are as follows (Note Figure 17.1-1):

(1) The Managers of Quality Assurance and Product Assurance report to the General Manager of P&QAO.
(2) An NFD QA Manager reports to the General Manager of the NFD.
(3) The Managers of Control and Instrumentation Manufacturing (C&IM), Reactor Equipment Manufacturing (REM), and Design Engineering, each of whom has a QA Manager reporting to him, report to the General Manager of the BWRSD.
(4) The Manager of Engineering Equipment Procurement Installation (EEPI) who has a QA Manager reporting to him reports to the General Manager of BWRPD.

The responsibilities of these QA Managers are as follows:

P&QAO Manager - establishes quality-related BWRO and Division Policies and Instructions; audits the NED functional organizations and their procedures and practices to assure conformance with BWRO and Division Policies and Instructions; verifies

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compliance of the overall QA Program with applicable codes, standards, regulations, and respective contractual commitments; conducts selected engineering design reviews independent of the design verification and review groups.

QA Manager of NFD - provides quality assurance planning and QA Program implementation for equipment (fuel elements, channels, and fuel element components) manufactured by NFD in Wilmington, North Carolina, as well as purchased material and equipment used in the manufacture of NFD products.

QA Manager of C&IM - provides quality assurance planning and QA Program implementation for equipment (sensors, instrument panels and racks, peripheral electrical sensing and control equipment, and servicing tools and equipment) manufactured by C&IM in San Jose, California, as well as purchased material and equipment used in the manufacture of C&IM products.

QA Manager of REM - provides quality assurance planning and QA Program implementation for equipment (control rods, control rod drives, steam separators and CRD hydraulic control modules) manufactured by REM in Wilmington, North Carolina, as well as purchased material and equipment used in the manufacture of REM products.

QA Manager of Design Engineerign - defines the design process; provides training and educational programs; conducts internal audits, and coordinates necessary corrective action.

QA Manager of EEPI - defines QA requirements to vendors of engineered equipment and assures vendor compliance with BWRO requirements through surveillance audits, and quality documentation review and approval; directs the QC Site Representative who performs site audits of quality-related activities during field installation.

The above QA Managers report to the same organizational level as managers who have product scheduling, expediting, and fabricating responsibilities. The QA organizational structure and functional responsibility assignments are such that verification of conformance to establish quality requirements is accomplished by those who do not have direct responsibility for specifying, producing or expediting products. The personnel in key quality assurance functions have direct access to top-level NED management. The QA Managers have the authority, independence, and organizational freedom to identify quality problems; verify implementation of the solutions, and prevent further processing, delivery, installation, or utilization of nonconforming items until proper dispositioning has occurred.

Based on our review and evaluation of the reporting level, duties and responsibilities of persons responsible for QA functions, we find that they have sufficient delegated authority to preclude undue cost and schedule influence on their activities, and have sufficient responsibility to properly establish and implement an effective QA Program. Therefore, it is the staff's conclusion that the QA organizational structure presented in GESSAR is acceptable and meets the requirements of Appendix B to 10 CFR Part 50.

17.3 Quality Assurance Program

All aspects of the QA Program were reviewed and considered to determine how each of the enighteen elements described in Section 17 of GESSAR interrelates with each other and forms the total body of the QA Program. The significant portions of the QA Program in GESSAR are described in the following paragraphs which formed the basis of our evaluation and which led to our conclusion that it meets the requirements of Appendix B to 10 CFR 50.

The QA Program of GE applies to all BWR safety-related structures, systems, and components, including fuel, throughout all phases of design, manufacturing, inspection and testing. GE commits to complying with the requirements of Appendix B to 10 CFR Part 50 and with those ANSI Standards and rough draft standards addressed in the NRC Gray Book "Guidance on Quality Assurance During Design and Procurement Phase of Nuclear Power Plants" dated June 7, 1973. GE will give NRC timely notification of any proposed significant changes in the QA Program.

The BWR quality system documentation, such as QA Manuals and Procedures, is structured from the GE Corporate Product Quality Policy. Responsibility for the final review and issuance of the overall BWR QA Program rests with the BWRO Deputy Division General Manager. Communication by the Deputy Division General Manager's office, through the Division Product Quality Policy and Division Instructions, states that Quality Policies, Manuals, and Procedures are mandatory requirements which must be implemented and enforced. Controlled copies of the QA Manuals and Procedures are distributed to a predetermined list of key personnel. Revisions to these documents are distributed to these personnel with instructions for replacement and disposition of obsolete documents.

Provisions are established requiring a formal training and indoctrination program for all GE personnel performing quality related activities to assure that they are instructed in the proper interpretation and implementation of the QA Policies, Manuals, Procedures, and their requirements. The training program provides for training of inspectors, testers, shop personnel and engineers.

The QA Program provides for an acceptable design control system for structures, systems and components which is documented and controlled by procedures and instructions. These procedures and instructions describe the responsibilities and interfaces of each organizational unit which has an assigned design responsibility. They also include measures to assure that:

(1) The design requirements are defined and that design activities will be carried out in a planned, controlled, and orderly manner.
(2) Appropriate quality requirements and standards are specified in design documents.
(3) Suitable materials, parts, components, and processes are applied.
(4) Design verification methods are properly selected.
(5) The designs are verified for adequacy by individuals or groups not having

responsibility for the original design.

(6) Design changes are controlled to the same level as was applied to the original design including review and approval by the same organization that performed the original review and approval unless another responsible design organization is designated by BWRO management.

Design documents are reviewed for interface compatibility in order to assure that there is no conflict in the design objectives and that the product resulting from the interfacing designs will function as planned. Design documents are furnished to the applicant and/or his agents to provide for interface compatibility review and coordination. Controlled distribution lists of design documents are maintained to assure proper and timely distribution to responsible individuals and organizations. Measures are established and documented for the preparation, review, approval and control of procurement documents to provide assurance that Regulatory requirements, design bases, and quality requirements are included or referenced in the procurement documents. Quality requirements are prepared, reviewed, approved, and issued by GE QA organizations for inclusion in the procurement document. Purchase specifications are reviewed and signed off by QA on an Engineering Review Memorandum prior to issuance. Reviews of procurement documents by qualified engineering and QA personnel provide assurance that quality requirements are complete and correctly stated, and that they can be controlled by the supplier and verified by BWRO QA personnel. Quality-related changes in procurement documents are subject to the same level of control as was exercised in the preparation of the original procurement documents.

The QA Program provides for the control of purchased material, equipment and services. These provisions require, where appropriate, for source evaluation and selection; review of procurement requirements; QA review and concurrence of QA Programs; objective evidence of quality furnished by the subcontractor; inspection or audit at the source; and examination or review of items or services upon delivery or completion.

GE requires that identification and control measures by established for relating an item (batch, lot, components, part) at any stage of production, from material receipt through fabrication and shipment, to the applicable quality documentation.

Manufacturing and quality assurance personnel, procedures, and equipment associated with special processes are qualified to the requirements of applicable codes and standards. Documented evidence of the validity of such qualifications is maintained for all such personnel, procedures and equipment.

GE requires inspection of materials, equipment, processes and services be in accordance with established QA plans, procedures, or instructions to provide assurance that the items conform to applicable drawings, specifications, codes, standards, and regulations. The inspection documents define or reference appropriate quantitative criteria (dimensions, tolerances, and operating limits) or qualitative criteria (comparative workmanship samples or visual standards) for determining that important activities have been satisfactorily accomplished. The inspection record (traveler) provides space for recording or stamping the identification of the inspector and enables one to determine the acceptance status of the item(s) being inspected. Inspections are required to be performed by quality control personnel who are organizationally independent from the manufacturing personnel who performed the work being inspected. These personnel report to the various BWRO QA Managers and include inspectors, test technicians, and QA representatives who have been trained to meet established performance standards. They are periodically evaluated to assure their competence in applicable quality control technologies. Inspection results are documented and evaluated by quality assurance personnel prior to release of a product for shipment to provide assurance that inspection requirements have been satisfied and proper records have been prepared.

Each of the BWRO QA organizations have provisions for the disposition and control of nonconforming items and for determining its cause and corrective action. Disposition decision to "use as is" or to "repair" a nonconforming item requires a review and approval by a Material Review Board consisting of technical persons from Engineering and QA. Originally there were not adequate provisions in GESSAR for reporting nonconformances of safety-related structures, systems, and components dispositioned "accept as is" or "repair" to the utility. The staff discussed this concern with GE, and they have amended GESSAR to describe and report nonconformances of safety-related equipment to the utility which could affect end use.

Provisions are established for conducting a comprehensive system of planned and documented audits to verify product quality and compliance with the QA Program. The audits are designed to assure compliance with all aspects of 10 CFR 50, Appendix B, including the quality-related aspects of design, procurement, manufacturing, storage, shipment, and reactor site activities. The P&QAO, by delegation from the NED General Manager through the Division Product Quality Policy, has the responsibility for the conduct of QA audits of each of the departments in BWRO. In addition each BWRO department is required, by Division Policy, to conduct internal QA audits of its products and all elements of the BWR QA Program. BWRO suppliers are subject to audit by cognizant BWRO QA organizations. The construction site is audited by a resident Site QC representative from QAEE&I who performs surveillance of applicant and AE conformance with BWRO supplied installation and test documents.

These audits are performed to determine the adequacy of QA-related practices, procedures and instructions; compliance with procedures, instructions, and policy directives of the QA Program; the effectiveness of the implementation of the QA Program, procedures, instructions, and policy directives; the adequacy of work areas, activities, processes, documents and records; product compliance with applicable engineering drawings and specifications; and implementation of corrective action in accordance with applicable procedures.

17.4 Conclusion

Based on our review and evaluation of Section 17 of GESSAR we find that the QA Program description provides for a comprehensive system of planned and systematic controls which adequately demonstrates GE's ability to comply with each of the eighteen criteria of Appendix B to 10 CFR Part 50. GE QA personnel are required to be actively involved in all quality related activities throughout the design, procurement, fabrication, inspection, testing, shipping, preoperational testing and auditing phases of the nuclear island. We find that the QA Division has sufficient delegated independence and authority to effectively establish and execute their QA Program without undue influences from those organization elements directly responsible for costs and schedule.

We conclude that the Quality Assurance Program as described in GESSAR, as amended, complies with the requirements of Appendix B to 10 CFR Part 50 and is acceptable.

Certain areas of GESSAR are such that GE has responsibility for design but not for procurement and construction such as containment. In such areas we have reviewed GE's QA program and determined that it meets the requirements of 10 CFR 50 Appendix B for their area of responsibility. When we review the application of a utility applicant referencing GESSAR, we will conduct a separate QA review of all such areas to assure that the total QA program for the project complies with 10 CFR Part 50 Appendix B for all portions of the plant.

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

The application of the General Electric Company for a PDA for GESSAR was reviewed at the 179th meeting of the ACRS on March 7, 1975. The application was also reviewed at Subcommittee meetings in Washington, D. C. on July 1, 1974, September 11, 1974 and January 18, 1975, and in Bloomington, Minnesota on November 9, 1974. A copy of the ACRS report to the Commission on the GESSAR application is included as Appendix F to this SER.

Our responses to the ACRS comments are discussed in the SER sections referenced for each ACRS comment listed below.

18.1 ACRS Comments Requiring Resolution Prior to Issuance of the PDA

18.1.1 Seismic Capability of the Offgas System (3.2.1)

GE has committed to designing the offgas system in accordance to the criteria listed in Appendix B (ETS Branch Technical Position 11-1), of this SER. As discussed in Section 3.2.1, the staff considers this item resolved.

18.1.2 Provision to Satisfy the Single-Failure Criteria for the RHR System (5.4.5)

GE has proposed alternate methods of achieving a cold shutdown condition within 24 hours which takes credit only for those actions which can be performed from the control room. These alternate methods employ safety grade equipment and are not subject to the same single failure which make the RHR inoperable. The staff considers this item resolved.

18.1.3 Requirement to be Met for Continuous Purging of Containment (6.2.4)

We indicated in Sections 6.2.4 and 11.3.3 that we could find continuous purging acceptable if GE would agree to filtration of the purge effluent and to reduction of the size of the purge lines. GE has not agreed to either of these changes, therefore, we will make them conditions of the PDA.

18.1.4 Evaluation of the ECCS to Meet 10 CFR 50.46, Appendix K (6.3.2)

GE has submitted and the staff has reviewed and accepted the ECCS evaluation for GESSAR. The staff considers this item resolved.

18.2 ACRS Comments on Matters that the Staff Plans to Review after the PDA is Issued

In their Report to the Commission, the ACRS stated that they wished to be kept informed of the status of certain items as the development of further information continues. These were:

(1) I&C design (Section 7)
(2) Fast Scram (4.2.3)
(3) BWR/6 fuel modification (1.8.1)

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(4) Ongoing R&D to verify the Mark III design (6.2.1.6)

(5) Interfaces between GESSAR and the BOP (1.10.2)

(6) ATWS (15.4)

(7) Recirculation Pump Overspeed during LOCA (5.4.1)

(8) Improved Testability of ADS (7.3.2.2)

We will regularly advise the ACRS of the progress of our review of these items.

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19.0 CONCLUSIONS

Based on our analysis of the proposed GESSAR-238 Nuclear Island preliminary design, we have determined that, subject to the conditions discussed herein, and to the satisfactory completion of the post-PDA items discussed above, we conclude that for the portion of the nuclear reactor design covered by GESSAR-238:

1. The General Electric Company has described, analyzed and evaluated the proposed design including, but not limited to, the principal architectural and engineering criteria for the design; the interface information necessary to assure compatibility between the submitted design and the balance of nuclear power plant; the envelope of site parameters postulated for the design; the quality assurance program to be applied to the design, procurement and fabrication of safety related features of the nuclear island design; the design features that affect plans for coping with emergencies in the operation of the reactor or major portion thereof; and has identified the major features and components incorporated therein for the protection of the health and safety of the public.
2. Such further technical or design information as may be required to complete the safety analysis will be supplied prior to or in the final design application;
3. Safety features or components which require research and development have been identified by the General Electric Company and it has described, and will conduct, research and development programs reasonably designed to resolve safety questions associated with such features or components;
4. On the basis of the foregoing, there is reasonable assurance that: (i) such safety questions will be satisfactorily resolved at or before the issuance of the operating license for the first nuclear power plant referencing the GESSAR-238 nuclear island design; and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, a facility can be constructed and operated without undue risk to the health and safety of the public, provided the site characteristics conform to the site parameters specified in GESSAR-238 as discussed above, and otherwise conform to the Part 100 requirements, and provided further that the balance of plant is properly designed and constructed in conformity with the interface requirements specified in GESSAR-238, as discussed above.
5. The General Electric Company is technically qualified to design the nuclear island facility described in the GESSAR-238 document.

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APPENDIX A

CHRONOLOGY - REGULATORY REVIEW

OF GENERAL ELECTRIC COMPANY'S STANDARD

SAFETY ANALYSIS REPORT

(GESSAR)

Docket No. STN 50-447

April 30, 1973	General Electric Company (GE) letter submitting report entitled "General Electric Standard Safety Analysis Report" (GESSAR), Volumes 1-7 for an acceptance review, pursuant to AEC Policy Statement issued in the form of a press release on 3/5/73. GESSAR describes a standard 3579 MWt boiling water reactor and consists of safety information for a complete BWR-6/Mark III containment system. (Project No. 484 assigned).
July 20, 1973	AEC letter advising that GESSAR is sufficiently complete for us to initiate our detailed review, upon receipt of the appropriate number of copies. The letter also encloses a list of deficiencies, the response for which we should receive as an amendment within 30 days.
July 26, 1973	GE letter transmitting additional copies of GESSAR for docketing.
July 30, 1973	GESSAR docketed
July 31, 1973	Meeting held in San Jose, California, between AEC and GE representatives.
July 31, 1973	AEC letter advising that GESSAR has been docketed and transmitting a related Federal Register Notice.
July 31, 1973	GE letter transmitting proprietary information (Figures 4.3-22 and 11.3-2 and Table 11.3.2) in support of the non-proprietary descriptions contained in GESSAR.
August 6, 1973	GE letter transmitting a notary page insert for GESSAR.
August 8, 1973	Notice of Receipt of GESSAR published in the Federal Register (38 F.R. 21444).
August 10, 1973 (notarized 8/8/73)	GE letter transmitting Amendment No. 1, which submits part of the information requested by AEC letter dated 7/20/73.
August 17, 1973	GE letter submitting Amendment No. 2, which furnishes answers to questions contained in AEC's 7/20/73 letter, and other additional information.
August 24, 1973	GE letter transmitting Amendment No. 3, which submits further information in connection with AEC's 7/20/73 letter.
August 31, 1973	GE letter transmitting Amendment No. 4, which contains additional information requested by letter dated 7/20/73.
August 31, 1973	Meeting between AEC and GE representatives to discuss GESSAR.
August 31, 1973	AEC letter transmitting the staff's review schedule for GESSAR.

August 31, 1973	GE letter submitting proprietary information to be included in GESSAR -- Tables 4.2.1 and 4.2.1a, which consist of fuel data and fuel cladding properties, respectively.
September 28, 1973	GE letter transmitting Amendment No. 5, which provides information related to the core power distribution study (Appendix 4A), requested in questions 4.1.1, 4.1.4, and 4.1.6 in AEC's letter dated 7/20/73.
September 28, 1973	GE letter transmitting the proprietary portion of Amendment No. 5 (Table 2 of Appendix 4A and several figures).
October 5, 1973	GE letter transmitting Amendment No. 6, which consists of updated figures of building design and equipment arrangements, corrections of typographical errors and clarification of portions of the text where obvious discrepancies exist.
October 12, 1973	GE letter transmitting Amendment No. 7, which consists of revised and new pages, tables and figures.
October 24, 1973	GE letter providing a tabulation that itemizes questions posed in other BWR/6 projects (i.e., Grand Gulf, Perry, River Bend, Douglas Point, Allens Creek, Clinton) for which answers will be provided in future GESSAR amendments.
November 1, 1973	AEC letter transmitting (Q-1's) a request for additional information.
November 6, 1973	AEC letter requesting information in connection with the staff's review of anticipated transients without scram (ATWS) in water-cooled reactor power plants.
November 8, 1973	AEC letter granting the withholding of proprietary information submitted by letters dated April 30, 1973 (Figure 4.3-22); August 31, 1973 (Tables 4.2.1 and 4.2.1a); and September 28, 1973 (Table 2 of Appendix 4A and several figures), pursuant to Section 2.790(b) of 10 CFR Part 2.
November 21, 1973	AEC letter requesting additional information concerning safety-related and control systems; integrated leak rate; site characteristics; General Design Criterion 4; engineered safety features; electric power; auxiliary systems, et al.
November 27, 1973	Meeting held between GE and AEC representatives to discuss GESSAR schedule, technical matters to be resolved, and the future use and expected benefits of GESSAR.
December 7, 1973	GE letter reaffirming its position that proprietary Figure 11.3-2 and Table 11.3.2 should be treated in accordance with 10 CFR 2.790.
December 11, 1973	Meeting between AEC and GE representatives to discuss the seismic design of GESSAR.
December 12, 1973	AEC letter requesting additional information.
December 13, 1973	Summary of meeting held on 11/27/73.
December 14, 1973	GE letter transmitting Amendment No. 8, which responds to questions forwarded by AEC's letter dated 11/1/73, including 16 pages of proprietary information (submitted by letter dated 12/17/73). Amendment No. 8 identifies all interfaces between the nuclear island and the balance-of-plant.
December 28, 1973	GE letter submitting information relating to interface and electrical areas requested in the 11/27/73 meeting.

January 11, 1974	GE letter transmitting Amendment No. 9, which responds partially to AEC letter dated 11/21/73 (part 2 of Q-1 list).
January 18, 1974	GE letter transmitting Amendment No. 10, which is in reference to AEC letter dated 12/12/73.
January 22, 1974	Meeting held between AEC and GE representatives to discuss LOCA flooding of the containment drywell ΔP test, suppression pool swell, and liner corrosion allowance.
February 8, 1974	GE letter transmitting Amendment No. 11, which addresses all questions raised by AEC in the first round of questions and identifies those questions for which residual information will be provided. Amendment 11 includes proprietary information.
February 12, 1974	AEC letter expressing concerns about delays that have occurred in the review to date of GESSAR and transmitting a revised review schedule.
February 19, 1974	GE letter consisting of a rebuttal to AEC's letter dated 2/12/74.
February 19, 1974	GE letter responding to AEC's letter dated 11/6/73 regarding ATWS.
February 27, 1974	Meeting held between GE and AEC representatives to discuss the GE scram system including the control rod drive position detection and indicating system; the rod pattern control system: and the use of ganged control rods.
February 28, 1974	Meeting held between AEC and GE representatives to discuss the adequacy of GE's responses to the interface questions.
March 15, 1974	Meeting held between AEC and GE representatives. Areas of discussion included design details beyond the scope of the standard format, new regulatory positions, outstanding items and resolved items.
March 20, 1974	AEC letter requesting additional information (Part 1 of Q-2).
March 26, 1974	AEC letter transmitting request for additional information (Part 2 of Q-2).
April 3, 1974	Meeting held between AEC and GE representatives to discuss site parameters.
April 11, 1974	AEC letter requesting additional information (QA program included) and discussing concerns related to the availability of information needed on preliminary instrumentation design, as discussed in meeting of 2/27/74.
April 11, 1974	AEC letter granting the withholding of proprietary (1) Figure 11.3-2 and Table 11.3.2 on the basis of reasons contained in letters dated 4/30/73 and 12/7/73; (2) pages (14) and responses to AEC questions 4.71 and 4.72 submitted by letter dated 12/17/73 (amendment No. 8); and (3) page 6.2-116 responding to question 6.86 submitted by letter dated 2/8/74 (Amendment No. 11).
April 17, 1974	Meeting between AEC and GE representatives to discuss site parameters.
April 19, 1974	GE letter transmitting Amendment No. 12, which clarifies inconsistencies in Chapter 7.
April 19, 1974	GE letter commenting on use of GESSAR questions on projects.
April 25, 1974	GE letter stating its position in connection with the information requested by AEC on 4/11/74.

May 1, 1974	Issued summary of meeting held on 4/3/74.
May 2, 1974	Issued summary of meetings held on 2/27 - 2/28/74; and 3/15/74 (summary relating to 2/27 - 2/28/74 meeting, dated 5/1/74; summary for 3/15/74 meeting dated 4/30/74).
May 7, 1974	AEC letter requesting additional information relating to Sections 7.22 and 5.39.
May 10, 1974	GE letter transmitting Amendment No. 13, which answers Q-2's.
May 14, 1974	Issued summary of meeting held on 4/17/74.
May 17, 1974	GE letter submitting Amendment No. 14, which consists of proprietary GESSAR pages 4.2-101d, 4.3-42c, and a 14 page report entitled "Fuel Rod Heat Transfer Model." Amendment 14 is in reference to AEC letters dated 3/20, 3/26 and 4/11/74.
May 30. 1974	AEC letter advising that AEC review of the new Section 1.11 for Chapter 1, which was included in Amendment 13, will not commence until the fall of this year.
June 17, 1974	GE letter transmitting Amendment No. 15, which responds to AEC letter dated 5/7/74.
June 11, 1974	Meeting between GE and AEC representatives for the purpose of discussions relating to drywell structural proof test and high pressure leak test. AEC staff reaffirmed its position with respect to both tests.
June 27, 1974	Issued summary of meeting held on 6/11/74.
June 28, 1974	GE letters (2) submitting the non-proprietary and proprietary (Question 4.15) portions of Amendment No. 16, which is in reference to AEC letter dated 11/1/73.
July 1, 1974	ACRS subcommittee meeting held.
July 2, 1974	AEC and GE representatives met to discuss Reg. Guides 1.31 (testing for weld delta ferrite) and 1.44 (testing for non-sensitization of welds).
July 12, 1974	GE letter transmitting Amendment No. 17, which clarifies and updates portions of the text; and provides information relating to main steam line leakage control system.
July 29, 1974	AEC letter making proprietary findings on May 17 & June 28 submittals.
July 30, 1974	AEC letter requesting additional information on Chapter 7 of GESSAR.
August 2, 1974	Amendment 18 filed.
August 16, 1974	Meeting to discuss schedule and procedural considerations.
August 23, 1974	Amendment 19 filed.
August 30, 1974	Amendment 20 filed.
September 3, 1974	Issued summary of management meeting held on August 16, 1974.
September 6, 1974	Meeting held with GE representatives, which included topics relating to interfaces, the main steam lines isolation valve leakage control system, and additional Chapter 9 information needed by the staff.

September 12, 1974	Letter to GE notifying them of a one month delay in publication of our SER due to the number of outstanding items.
September 20, 1974	Issued summary of meeting held on September 6, 1974.
September 27, 1974	Amendment 21 filed. This amendment addresses the outstanding items listed in the September 12, 1974 staff letter to GE.
October 21, 1974	Amendment 22 filed. This amendment addresses the outstanding items listed in the September 12, 1974 staff letter to GE.
October 29, 1974 through November 1, 1974	Meeting between the staff and GE to discuss the Instrumentation and Control designs.
November 6, 1974	Amendment 23 filed. This amendment addresses the outstanding items listed in the September 12, 1974 staff letter to GE.
November 8, 1974	Amendment 24 filed. This amendment addresses the agreements reached at the I & C meetings between GE and the staff on October 29, 1974 through November 1, 1974.
November 9, 1974	ACRS Subcommittee meeting held in Minneapolis, Minnesota.
November 13, 1974	Safety Evaluation Report issued.
November 21, 1974	GE letter concerning the identification of interfaces between the NSSS and the BOP portions of the standard plant.
December 13, 1974	Amendment 25 filed. This amendment addresses outstanding issues listed in the September 12, 1974 letter to GE.
December 13, 1974	Issued Supplement 1 to the Safety Evaluation Report.
December 20, 1974	GE letter requesting a letter from the AEC relating to the approval of the GESSAR-238 design.
December 23, 1974	Amendment 26 filed. This amendment addresses outstanding issues listed in the September 12, 1974 letter to GE.
January 13, 1975	AEC letter transmitting a copy of letter to TVA informing them that we do not intend to review on the Hartsville docket sections of GESSAR that are acceptable, in reference to GE letter dated December 20, 1974.
January 16, 1975	Meeting with GE to discuss RHR single failures and alternate shutdown methods.
January 18, 1975	ACRS Subcommittee meeting held.
January 29, 1975	Issued summary of meeting held on January 16, 1975.
February 3, 1975	Amendment 27 filed. This amendment addresses outstanding issues listed in the September 12, 1974 letter to GE.
February 18, 1975	Amendment 28 filed. This amendment addresses outstanding issues listed in the September 12, 1974 letter to GE.
February 22, 1975	Issued Supplement No. 2 to the Safety Evaluation Report.
March 3, 1975	Amendment 29 filed. (Same as Amendment 28.)
March 4, 1975	Issued Supplement No. 3 to the Safety Evaluation Report.
March 6, and March 8, 1975	ACRS Full Committee meeting held.

March 14, 1975	Meeting held with GE for the purpose of discussing the drywell leakage test acceptance criteria.
March 14, 1975	ACRS Report issued.
March 18, 1975	NRC letter requesting information related to our review of the containment design.
March 27, 1975	NRC letter requested information related to Amendment 27.
March 28, 1975	GE letter responding to our letter dated March 18, 1975.
March 31, 1975	Amendment 30 filed. This amendment addresses the post-LOCA Hydrogen control system as well as providing revised pages.
April 8, 1975	GE letter in reference to NRC letter dated March 27, 1975.
April 17, 1975	Amendment 31 filed. This amendment provides Appendix K ECCS Analysis as well as outstanding issues listed in the September 12, 1974 letter to GE.
April 17, 1975	Meeting with GE to discuss the drywell instrumentation requirements for the structural proof test.
April 22, 1975	NRC letter requesting information related to ongoing review of BWR plants with pressure suppression-type containment.
May 1, 1975	Amendment 32 filed. This amendment consists of revised pages for Sections 7.5 and 7.6 of GESSAR.
May 8, 1975	GE letter submitting report relating to case studies 18 and 19 for soil structure interaction analysis.
May 12, 1975	Amendment 33 filed. This amendment discusses alternate methods of shutdown of the reactor should the RHR system not be available.
May 12, 1975	NRC letter consisting of clarifications relating to leak and structural proof tests.
May 23, 1975	NRC letter requesting additional information related to the instrumentation and control systems.
June 2, 1975	NRC letter advising that information concerning tornado missile velocites for GESSAR plant submitted by Amendment 31 is unacceptable.
June 2, 1975	Amendment 34 filed. This amendment provides additional information regarding safety/relief valve setpoints and drywell proof test instrumentation.
June 6, 1975	GE letter requesting staff conclusions relating to accident off-site dose evaluation of GESSAR.
July 1, 1975	Amendment 35 filed. This amendment provides additional ECCS analyses.
July 9, 1975	NRC letter relating to interface.
July 10, 1975	Meeting with GE to discuss the offsite doses.
July 14, 1975	NRC letter reflecting the staff's positions on the design of the RHR system.
July 16, 1975	Amendment 36 filed. This amendment provides additional instrumentation and control information.
July 22, 1975	NRC letter relating to staff conclusion regarding the accident offsite dose evaluation of GESSAR.

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July 28, 1975	GE letter reflecting its position on the RHR systems, which is in reference to NRC letter dated July 14, 1975.
July 31, 1975	NRC letter requesting additional information related to instrumentation and control.
July 31, 1975	ACRS Subcommittee meeting on containment and drywell testing held in Chicago, Illinois.
August 4, 1975	Issued summary of meeting held on July 10, 1975.
August 20, 1975	GE letter containing additional information relating to bypass leakage requirements for Mark III containment.
August 22, 1975	Meeting held to discuss instrumentation and control review of GESSAR-251 NSSS, which included discussions applicable to GESSAR-238 nuclear island design.
August 22, 1975	GE letter supplementing the information submitted by GE letter dated August 29, 1975.
September 2, 1975	Amendment 37 filed. This amendment submits information regarding hydrodynamic loads associated with postulated LOCA and safety relief valve discharge events.
September 2, 1975	Issued summary of meeting held on August 22, 1975.
September 15, 1975	Meeting held in Bethesda, Maryland between NRC and GE representatives for the purpose of discussing outstanding items on containment systems.
September 26, 1975	NRC letter requesting further justification for withholding 10 CFR 50, Appendix K information submitted as a part of Amendment 35 from public disclosure.
September 26, 1975	GE letter confirming its position on containment and bypass leakage as discussed in the September 15, 1975 meeting.
October 14, 1975	Amendment 38 filed. The amendment deletes Section 1.11 from GESSAR-238 NI, which contains an identification of the 238 NSSS interfaces and the BOP.
October 21, 1975	NRC letter relating to interlock requirements for the GESSAR-238 NI main steam isolation valve leak control system.
October 24, 1975	NRC letter transmitting acceptable set of design criteria for the Mark III containment.
October 31, 1975	GE letter requesting the staff to reevaluate its position regarding the MSIV-LCS interlock.
November 7, 1975	GE letter transmitting set of design criteria for the Mark III containment to NRC.
November 7, 1975	GE letter submits Amendment 39.
November 21, 1975	GE letter incorporating additional design information in GESSAR-238.
December 16, 1975	GE letter affirming commitments relating to design and design criteria.

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APPENDIX B

BRANCH TECHNICAL POSITION - ETSB NO. 11-1 (Rev. 1)

DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER REACTOR PLANTS

A. Background

An aspect of nuclear power plant operation is the control and management of liquid, gaseous and solid radioactive waste[1/] generated as a byproduct of nuclear power. We have established acceptable design guidance, seismic and quality group classifications, and quality assurance provisions for radioactive waste management systems including steam generator blowdown systems. For the purpose of this position paper, the radioactive waste management systems are considered to begin at the interface valve(s) in each line from other systems provided for collecting wastes that may contain radioactive materials and to terminate at the point of controlled discharge to the environment, at the point of recycle back to storage for reuse in the reactor, or at the point of storage of packaged solid wastes prior to shipment offsite to a licensed burial ground. The steam generator blowdown system begins at, but does not include, the outermost containment isolation valve on the blowdown line and terminates at the point of controlled discharge to the environment, at the point of interface with other liquid waste systems, or at the point of recycle back to the secondary system.

Except as noted below the positions set forth in this paper do not apply to the reactor coolant cleanup system, the condensate cleanup system, the chemical and volume control system, sumps and floor drains provided for collecting liquid wastes, the boron recovery system, building ventilation systems (heating, ventilating and air conditioning) and chemical fume hood exhaust systems. Positions set forth in this paper regarding provisions to control releases of radioactive materials in liquids due to tank overflows apply to all plant systems, outside reactor containment, having the potential to incur such releases.

The design and construction of radioactive waste management and steam generator blowdown systems should provide assurance that radiation exposures to operating personnel and to the general public are maintained at low and acceptable levels, by assuring that these systems are designed to quality standards conducive to increasing system reliability, operability, and availability. In development of this design guidance, the NRC staff has reviewed a number of designs and concepts submitted in license applications and operating system histories. The NRC staff has been guided by current industry practices and the cost of design features, taking in account the potential impact on the health and safety of operating personnel and the general public.

[1/] Radioactive waste used in this guide means liquid, gaseous, or solids containing radioactive material resulting from operation of a LWR which by design or operating practice may be or will be processed prior to final disposition.

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The design guidance given in this position paper provides reasonable assurance that equipment and components used in the radioactive waste management and blowdown systems are designed, constructed, installed and tested on a level commensurate with the health and safety of the public and plant operating personnel. Instrumentation and controls associated with the waste management and blowdown systems should be designed to a quality commensurate with their intended function.

This position paper sets forth minimum branch requirements and is not intended to prohibit the implementation of more rigorous design codes, standards, or quality assurance measures than those indicated herein.

In addition to the design guidance given for radwaste systems, recommendations are given for provisions to preclude the inadvertent release of radioactive materials in liquids due to spills or overflows from both radwaste and non-radwaste system tanks located inside or outside of plant structures.

B. Branch Technical Position

I. Systems Handling Radioactive Materials In Liquids

a. The liquid radwaste treatment system, including the steam generator blowdown system downstream of the second containment isolation valve, should meet the following criteria:

(1) The systems should be designed and tested in accordance with the codes and standards listed in Table 1, to include the provisions in (2) below and in Section IV of this position paper.

(2) Materials for pressure retaining components should conform to the requirements of one of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Manufacturer's material certificates of conformance with material specifications may be provided in lieu of certified materials test reports.

(3) Foundations and adjacent walls of structures that house the liquid radwaste system should be designed to the seismic criteria described in Section V to a height sufficient to contain the liquid inventory in the building.

(4) Equipment and components used to collect, process, and store liquid radioactive waste need not be designed to the seismic criteria given in Section V.

b. All tanks located outside reactor containment and containing radioactive materials in liquids should be designed to prevent uncontrolled releases of radioactive materials due to spillage in buildings or from outdoor storage tanks. The following design features should be included for tanks that may contain radioactive materials:

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(1) All tanks, both inside and outside the plant including the condensate storage tank(s) should have provisions to monitor liquid levels and to alarm potential overflow conditions.

(2) All tanks should have overflows, drains, and sample lines should be routed to the liquid radwaste treatment system.[1/]

(3) Indoor tanks should have curbs or elevated thresholds with floor drains routed to the liquid radwaste treatment system.[1/]

(4) Outdoor tanks should have a dike or retention pond capable of preventing runoff in the event of a tank overflow and have provisions for sampling collected liquids and routing them to the liquid radwaste treatment system.

II. Gaseous Radioactive Waste (Radwaste) System

a. The gaseous radwaste treatment system, including systems provided for treatment of normal offgas releases from the main condenser vacuum system for a BWR and for the treatment of gases stripped from the primary coolant for a PWR should meet the following criteria:

(1) The systems should be designed and tested in accordance with the codes and standards listed in Table 1, to include the provisions in (2) below and in Section IV of this position paper.

(2) Materials for pressure retaining components should conform to the requirements of one of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Manufacturer's material certificates of conformance with material specifications may be provided in lieu of certified materials test reports.

(3) Those portions of the gaseous radwaste treatment system which by design are intended to store or delay the release of gaseous radioactive waste, including portions of structures housing these systems, should be designed to the seismic design criteria given in Section V of this position paper. For systems that normally operate at pressures above 1.5 atmospheres (absolute), this should include isolation valves, equipment, interconnecting piping, and components located between the upstream and downstream values used to isolate these components from the rest of the system (e.g., waste gas storage tanks in a PWR). For systems that operate near ambient pressure and retain gases on charcoal adsorbers, only the tank elements and the building housing the tanks are included (e.g., charcoal delay tanks in a BWR).

[1/] Retention by an intermediate sump or drain tank, designed for handling radioactive materials and having provisions for routing to the liquid radwaste system is acceptable.

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III. Solid Radioactive Waste (Radwaste) System

a. The solid radwaste system consists of slurry waste collection and settling tanks, spent resin storage tanks, phase separators, and tanks, equipment, and components used to solidify wastes prior to offsite shipment. The solid radwaste handling and treatment system should meet the following criteria:

(1) The system should be designed and tested in accordance with the codes and standards listed in Table 1 to include the provisions in (2) below and in Section IV of this paper.

(2) Materials for pressure retaining components should conform to the requirements of one of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Manufacturer's material certificates of conformance with material specifications may be provided in lieu of certified materials test reports.

(3) Foundations and adjacent walls of structures that house the solid radwaste system should be designed to the seismic criteria given in Section V of this position paper to a height sufficient to contain the liquid inventory in the building.

(4) Equipment and components used to collect, process or store solid radioactive waste need not be designed to seismic criteria referenced above.

IV. Additional Design, Construction, and Testing Criteria

In addition to the requirements inherent in the codes and standards listed in Table 1, the following criteria, as minimum, should be implemented for components and systems considered in this guide.

a. The Quality Assurance provisions described in VI of this guide should be applied.

b. Pressure retaining components of process systems should utilize welded construction to the maximum practicable extent. Process piping systems include the first root valve on sample and instrument lines. Flanged joints or suitable rapid disconnect fittings should be used only where maintenance or operational requirements clearly indicate that such construction is preferable. Screwed connections in which threads provide the only seal should not be used except for instrumentation connections where welded connections are not suitable. Process lines should not be less than 3/4-inch. Screwed connections backed up by seal welding, socket welding or mechanical joints may be used on lines 3/4-inch or greater, but less than 2 1/2-inch, nominal size. For lines of 2 1/2-inch nominal size and above, pipe welds should be of the butt-joint type. Backing rings should not be used in lines carrying resins or other particulate material. All welding constituting

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the pressure boundary of pressure retaining components should be performed in accordance with ASME Pressure and Vessel Code Section IX.

c. Completed process systems should be pressure tested to the maximum practicable extent. Piping systems should be hydrostatically tested in their entirety except at atmospheric tank connections where no isolation valves exist. Testing of piping systems should be performing in accordance with applicable ASME or ANSI codes, but in no case less than 75 psig. The test pressure should be held for a minimum of 30 minutes with no leakage indicated. Testing provisions should be incorporated to enable periodic evaluation of the operability and required functional performance of active components of the system.

V. Seismic Design Requirements for Radioactive Waste Management Systems and Structures Housing Radioactive Waste Management Systems

a. Seismic Design Requirements for Gaseous Radioactive Waste Management Systems[1/]

(1) For the evaluation of support elements in the gasesous waste system, a simplified seismic analysis procedure to determine seismic loads may be used. The simplified procedure consists of consideration of the system as a single degree of freedom system and picking up a seismic response value from applicable floor response spectra, once the fundamental frequency of the system is determined. The floor response spectra should be obtained analytically (Section V.b) from the application of Regulatory Guide 1.60 design response spectra normalized to OBE level maximum ground acceleration at the foundation of the building housing the gaseous radwaste system.

(2) The allowable stresses to be used for the system support elements should be those given in the AISC Manual of Steel Construction, 7th edition 1970, including the one-third allowable stress increase provision for load combinations involving earthquake loads. For design of concrete foundations of the system, where applicable, use of the ACI 318-71 code with one-third increase in allowable stress for seismic loads is acceptable.

(3) The construction and inspection requirements for the support elements should comply with those stipulated in AISC or ACI Codes as appropriate.

b. Seismic Design Requirements for Buildings Housing Radwaste Systems

(1) Define input motion at the foundation of the building housing the radwaste systems. The motion should be defined by normalizing the Regulatory Guide 1.60 spectra to the OBE maximum ground acceleration selected for the plant.

[1/] For which seismic capabilities are required in Section II(3).

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A simplified analysis should be performed to determine appropriate seismic loads and floor response spectra pertinent to the location of the systems; i.e., an analysis of the building by a "several degrees of freedom" mathematical model and the use of an approximate method to generate the floor response spectra for radwaste systems and the seismic loads for the buildings. No time history or dynamic analysis is required.

(2) The simplified method for determination of seismic loads for the building consists of (a) calculation of first several modal frequencies and participation factors for the building, (b) determination of modal seismic loads by item (1) input spectra, and (c) combination of modal seismic loads by the square root of the sum of squares (SRSS) rule.

(3) With regard to generation of floor response spectra for radwaste systems, methods such as the Biggs or other equivalent procedures which give approximate floor response spectra without need for performing a time history analysis may be used.

(4) The load factors and load combinations to be used for the building should be those given in the ACI-318-71 Code. The allowable stresses for steel components should be those given in the AISC Manual of Steel Construction, 7th edition, 1970.

(5) The construction and inspection requirements for the building elements should comply with those stipulated in the AISC or ACI Code as appropriate.

(6) The foundation media of structures housing the radwaste systems should not liquify during the Operating Basis Earthquake.

c. In lieu of the requirements and procedures defined above, optional shield structures constructed around and supporting the radwaste systems may be erected to protect the radwaste systems from effects of housing structural failure. If this option is adopted, the procedures described in Section V.b only need to be applied to the shield structures while treating the rest of the housing structures as non-seismic Category I.

VI. Quality Assurance for Radioactive Waste Management Systems

A quality assurance program should be established that is sufficient to assure that the design, construction, and testing requirements are met. The quality assurance program should include the following:

a. Design and Procurement Document Control - Measures should be established to insure that the requirements of this position paper are specified and included in design and procurement documents and that deviations therefrom are controlled.

b. Control of Purchased Material, Equipment and Services - Meausres should be established to assure that purchased material, equipment and construction services conform to the procurement documents.

c. Inspection - A program for inspection of activities affecting quality should be established and executed by, or for, the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity.

d. Handling, Storage, and Shipping - Measures should be established to control the handling, storage, shipping, cleaning and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration.

e. Inspection, Test and Operating Status - Measures should be established to provide for the identification of items which have satisfactorily passed required inspections and tests.

f. Corrective Action - Measures should be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment and nonconformances are promptly identified and corrected.

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TABLE 1

EQUIPMENT CODES

EQUIPMENT	CODES			
	Design and Fabrication	Materials[2]	Welder Qualification And Procedure	Inspection And Testing
Pressure Vessels	ASME Code Section VIII, Div. 1	ASME Code Section II	ASME Code Section IX	ASME Code Section VIII, Div. 1
Atmospheric or 0-15 psig tanks	ASME Code[3] Section III, Class 3, or API 620 & 650, AWWA D-100	ASME Code[4] Section II	ASME Code Section IX	ASME Code[3] Section III, Class 3 or API 620; 650 AWWA D-100
Heat Exchanger	ASME Code Section VIII, Div. 1 and TEMA	ASME Code Section III	ASME Code Section IX	ASME Code Section VIII, Div. 1
Piping and Valves	ANSI 31.1	ASTM or ASME Code Section II	ASME Code Section IX	ANSI B 31.1
Pumps	Manufacturers[1] Standards	ASME Code Section II or Manufacturers Standard	ASME Code Section IX (as required)	ASME[3] Section III Class 3; or Hydraulic Institute

NOTES:

(1) Manufacturer's standard for the intended service. Hydrotesting should be 1.5 times the design pressure.
(2) Material Manufacturer's certified test reports should be obtained whenever possible.
(3) ASME Code stamp and material traceability not required.
(4) Fiberglass reinforced plastic tanks may be used in accordance with Part M, Section 10, ASME Boiler and Pressure Vessel Code, for applications at ambient temperature.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Appendix C
APR 1 9 1974

Mr. John A. Hinds, Manager
Safety and Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95114

Dear Mr. Hinds:

At various times the AEC staff has discussed with the General Electric Company the subject of appropriate classification requirements in boiling water reactor (BWR) plants for main steam system components. These discussions have included consideration of (a) structures, components and systems that are not classified as safety-related items but are located downstream of the isolation valves; (b) those not specifically designed to seismic Category I standards; and (c) those not housed in seismic Category I structures.

To date, BWR plant reviews have resulted in various approaches for different individual applications. While these different approaches have resulted in acceptable levels of safety in each case, they have required time-consuming customized reviews. The GESSAR BWR/6 application, under review as part of our standardization program, includes this portion of the BWR plant.

In the course of the GESSAR review, we have identified a systematic basis for classification of such components and structures that will result in an acceptable and uniform design basis for the main steam-lines (MSL) and main feedwater lines (MFL) in the standardized plant. Although it is recognized that a significant portion of the equipment involved in this classification scheme may include equipment outside the normal scope of supply of the General Electric Company, specifically the shutoff valves in the main steam and feedwater lines and the equipment beyond those valves, the implementation of these requirements defines acceptable standardized requirements with respect to quality and seismic design for the BWR/6 nuclear steam supply system and the power conversion system.

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This approach involves specification of appropriate safety requirements for those portions of the MSL and MFL that are housed in Seismic Category I structures (e.g., the auxiliary building), and includes suitable restraints near a shutoff valve outside the containment isolation valves. The portions of the MSL and MFL on the turbine side of the shutoff valves would be designed in accordance with quality group and certification procedures as outlined in Attachment A of this letter.

To implement this approach, Attachment 1 of our letter of November 19, 1973, would be amended to provide acceptable seismic and quality requirements for BWR/6 steam system components as shown in the attachments to this letter. We believe this provides an appropriate standardized approach to MSL and MFL classification and is acceptable as an alternate to the guidelines currently specified in Regulatory Guides 1.26 (March 23, 1972) and 1.29 (August 1973).

As we have discussed with you, a suitable interface restraint should be provided at the point of departure from the Class I structure where the interface exists between the safety and nonsafety-related portions of the MSL and MFL.

Sincerely,

J M Hendrie

Joseph M. Hendrie, Deputy Director
for Technical Review
Directorate of Licensing

Enclosures:
Attachment A (Classification Requirements for Main Steam System Components Other than the Reactor Coolant Pressure Boundary)
Attachment B (Sketch - AEC Quality Group and Seismic Category Classifications Applicable to Main Steam System Components in BWR/6 Plants)

cc: L. S. Gifford

ATTACHMENT A

Classification Requirements for BWR/6 Main Steam and Feedwater System Components Other than the Reactor Coolant Pressure Boundary.

ITEM	SYSTEM OR COMPONENT	QUALITY GROUP CLASSIFICATIO
1.	Main Steam Line (MSL) from second isolation valve to and including shutoff valve.	B
2.	Branch lines of MSL between the second isolation valve and the MSL shutoff valve, from branch point at MSL to and including the first valve in the branch line.	B
3.	Main feedwater line (MFL) from second isolation valve and and including shutoff valve.	B
4.	Branch lines of MFL between the second isolation valve and the MFL shutoff valve, from the branch point at MFL to and including the first valve in the branch line.	B
5.	Main steam line piping between the MSL shutoff valve and the turbine main stop valve.	D (1)
6.	Turbine bypass piping	D
7.	Branch lines of the MSL between the MSL shutoff valve and the turbine main stop valve.	D
8.	Turbine valve, turbine control valve, turbine bypass valves and the main steam leads from the turbine control valve to the turbine casing.	D (1,2) or Certification
9.	Feedwater system components beyond the MFL shutoff valve.	D

(1) All inspection records shall be maintained for the life of the plant. These records shall include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

(2) All cast pressure-retaining parts of a size and configuration for which volumetric methods are effective shall be examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards may be used as an alternate to radiographic methods. Examination procedures and acceptance standards shall be at least equivalent to those defined in Paragraph 136.4, Examination Methods of Welds - Non-Boiler External Piping, ANSI B31.1-1973.

POOR ORIGINAL

ATTACHMENT A

(3) The following qualification shall be met with respect to the certification requirements:

1. The manufacturer of the turbine stop valves, turbine control valves, turbine bypass valves and main steam leads from turbine control valve to turbine casing shall utilize quality control procedures equivalent to those defined in General Electric Publication GEZ-4982A, "General Electric Large Steam Turbine-Generator Quality Control Program".

2. A certification shall be obtained from the manufacturer of these valves and steam loads that the quality control program so defines has been accomplished.

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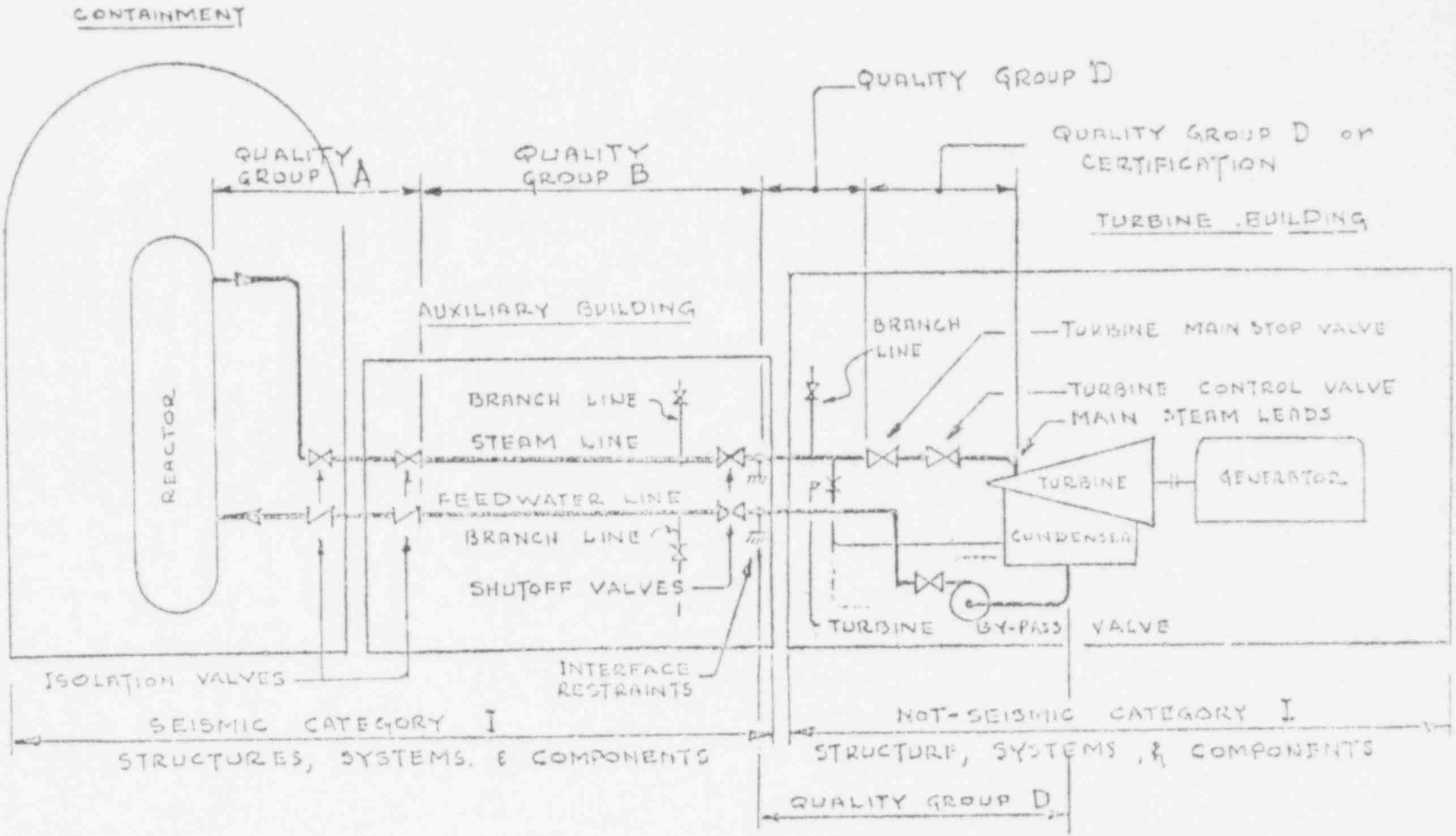
ATTACHMENT - B

APR. 17, 1974



AEC QUALITY GROUP AND SEISMIC CATEGORY CLASSIFICATIONS

APPLICABLE TO POWER CONVERSION SYSTEM COMPONENTS

IN BWR/6 PLANTS

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Appendix D

Technical Report On

the

General Electric Company

8 x 8 Fuel Assembly

5 February 1974

Regulatory Staff

U. S. Atomic Energy Commission

ERRATA
Technical Report on the General Electric Company
8 x 8 Fuel Assembly
Dated February 5, 1974

Pg. 1 - last line change "evaluation" to "evaluations"

Pg. 2 - line 8 add "and" after word "diameter;"

Pg. 4 - line 3 change "0.088" to "0.060."

Pg. 5 - line 6 change "two" to "one"

Pg. 6 - line 2 delete "(,)"
line 4 delete "(7)"

Pg. 7 - Line 16 change "one-half" to "up to three quarters"

Pg. 9 - line 6 change "effect" to "affect"

Pg. 14 - line 1 change "2.3" to "2.10"

Pg. 20 - line 14 "less severe" to "similar"; line 15 change "than" to "and" to read "... the consequences of these events are similar for 8 x 8 assemblies and for 7 x 7 assemblies."

Pg. 25 - last line change "not" to "no"

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Table of Contents

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1.0 Introductior

The current fuel in General Electric Company boiling water reactors is sintered, slightly enriched uranium dioxide pellets sealed in Zircaloy tubes. Bundles of these fuel rods are contained within a square open-ended Zircaloy channel box to form fuel assemblies. The General Electric Co. has recently modified the design of these fuel assemblies and licensees propose to reload assemblies of this new design as replacements for depleted assemblies of the old type.[1, 2, 3, 4, 5] This report presents the results of the Regulatory staff's generic review of 8 x 8 fuel assemblies as used both in partial and full core reloads. As part of the staff's review of the General Electric Company BWR-6 class of reactors, which are currently under consideration for construction permits, the staff is continuing its review of the 8 x 8 fuel assemblies as used both in partial and full core reloads. As part of the staff's review of the General Electric Company BWR-6 class of reactors, which are currently under consideration for construction permits, the staff is continuing its review of the 8 x 8 fuel assemblies used in these new reactor designs. The staff's review of reload assemblies considered the effects that the changes in the fuel design have no normal operation, abnormal operational transients and accidents. However, the staff's review considered only generic aspects of the fuel design such as the adequacy of design methods, the comparative performance of the old and new fuel designs, and the applicability of accident analysis methods. The plant specific aspects of the review, such as compliance with the Interim Acceptance Criteria, including the effects of fuel pellet densification, any necessary revisions to Technical Specification requirements, and the radiological consequences of postulated accidents will be addressed in separate evaluation for the individual plants.

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2.0 MECHANICAL DESIGN

The reload fuel assemblies consist of 63 fuel rods and one unfueled, capture-spacer rod in a square 8 x 8 array within a square channel box. The rods are spaced and supported at the top and bottom by stainless steel tie plates. The rods are also held in alignment by spacer-guides located along the assembly. As shown in Table I the 8 x 8 fuel assembly is similar to the current 7 x 7 design. The major mechanical changes are the larger number of rods; the reduction in the rod diameter; the introduction of the asymetrically located unfueled spacer-capture rod; and the use of fully annealed, rather than cold worked, Zircaloy cladding. Other changes, which have also been incorporated in the most recent 7 x 7 designs include shorter, chamfered and undished pellets and a hydrogen getter. However, the designs of both assemblies have the same objective, that is maintenance of clad integrity during normal operation and abnormal transients. The designs of both are also based on the same stress criteria, that is, the ASME Boiler and Pressure Vessel Code, Section III. In evaluating the performance of the fuel, the design analyses considered stresses due to external coolant pressure, internal gas pressure, thermal effects, spacer contact, and flow induced vibration. Other effects which were considered included pellet-cladding mechanical interaction, stress corrosion cracking, fretting, and densification. Verification of the adequacy of the design of the 8 x 8 assemblies is based on analysis, mechanical tests, operating experience of previous designs, in-pile tests of a prototypical fuel rod and similar fuel rods, and an out-of-pile test of an assembly of similar design.

Much of the previous experience with fuel rods and assemblies is applicable to the 8 x 8 fuel assemblies. [6]These rods ranged in diameter from 0.3 [illegible] to 0.593 inches, in clad thickness from 0.022 to 0.088 inches, and in pellet-clad diametral gap from 0.002 to 0.016 inches. Rods have been irradiated for up to 6 years and had peak exposure of 30,000 MWD/T. Although rods identical to the 8 x 8 design have not been tested by GE, the background of experience is sufficient to enable GE to design rods of new design with confidence in their durability.

Confidence that the vibration and fretting characteristics of the 8 x 8 assemblies are known, is based on rod vibration experiments (7) and the operating experience with other types of fuel assemblies in general and the 7 x 7 design in particular. The 7 x 7 and 8 x 8 assemblies are very similar in this regard. The fuel rods in both are of similar design, are made of the same material and have nearly the same natural frequency. The fuel rod spacer grids in both types of assembly also are of similar design, are made of the same materials and exert the same spring force. Both operate at the same pressure and temperature with nearly identical fluid velocities and quality.

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TABLE I

MECHANICAL DESIGN COMPARISON

ASSEMBLY		
Rod Array	7 x 7	8 x 8
Number of Fueled Rods	49	63
Rod Pitch, In.	0.738	0.640
FUEL ROD		
Active Fuel Length, In.	144	144
Gas Plenum Length, In.	11.25	11.25
Fill Gas	He	He
FUEL		
Material	UO_2	UO_2
Pellet Diameter, In.	0.477	0.416
Pellet Immersion Density, % TD	95.0	95.0
CLADDING		
Material	Zr-2	Zr-2
Thickness, In.	0.037	0.034
Outside Diameter, In.	0.563	0.493
CHANNEL		
Material	Zr-4	Zr-4
Thickness, In.	0.080	0.080
Outside Dimension, In.	5.438	5.438
Length, In.	162 1/8	162 1/8
SPACERS		
Number	7	7
Material		
Grid	Zr-4	Zr-4
Springs	Inconel.	Inconel.

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Further verification of the adequacy of the design has been provided by the testing of an assembly of similar design for 7000 hours in high pressure, two-phase flow loop.[8] This test was performed by ASEA-Atom, a Swedish BWR manufacturer and a General Electric Company licensee, as part of a fuel development program.

A comparison of the significant parameters of this test assembly[8] and the GE 8 x 8 assemblies indicate that the wear and fretting characteristics would be similar. The most significant differences are that the test assembly had no unfueled spacer-capture rod, and had four latern springs supporting a fuel rod, where the GE assemblies have only two. However, the vibration and fretting in this test would be expected to be at least as severe as in a GE 8 x 8 assembly since the axial pitch of the spacers was larger and the rods thinner walled and smaller in diameter. Inspection at 1.5 month intervals and the conclusion of the test revealed no significant fretting wear.

Although the design of the unfueled spacer-capture rod is new, it is based on experience with similar designs. Five 6 x 6 fuel assemblies with eccentrically located fueled spacer-capture rods which have a locking tab design identical to the 8 x 8 have operated in the Humboldt Bay reactor. Visual examination of these assemblies has revealed no deficiencies. Assemblies with eccentrically located fuel spacer rods with a different locking tab design have operated in the Dresden-1, KRB, Tarapur and Garigliano reactors. Twenty four assemblies with unfueled rods have operated in the Big Rock reactor.

A number of mechanical tests have been performed on 8 x 8 fuel assemblies and components in order to demonstrate their integrity. Dead weight loading of the 7 x 7 type assembly spacer grids has demonstrated that they are adequate to withstand all expected loads. Although, as GE has stated, the 8 x 8 assembly spacer-grids are stronger than 7 x 7 spacer-grids, this will be verified by dead weight crushing tests of the new spacer design.[7] Testing of the spacer locking tab has shown that it can satisfactorily resist a shear load, and further verification tests which more closely simulates the actual loading are to be performed.[7] Other tests have been made to determine the bending stiffness of the assembly and the force exerted by the fuel rod expansion springs. In addition, a channel box removal and replacement test and an instrumented shipping test have been performed.

The methods used by the GE to calculate the effects of fuel pellet densification and cladding creep have been previously submitted and reviewed by the Staff.[11] These methods and the Staff conclusions apply equally to both the 8 x 8 and 7 x 7 fuel designs.

Performance of the fuel during operation will be indirectly monitored during operation by measurement of the activity of the primary coolant and condenser steam air-ejector off-gas. During refueling, fuel assemblies will be tested for radioactivity leakage selected assemblies will be examined visually for bowing and some rods will be given standard non-destructive and destructive examinations as part of the normal fuel surveillance program.[8] Additional surveillance of ten rods in each of two 8 x 8 fuel assemblies will also be performed. This program will consist of dimensionally characterizing the rods prior to irradiation and then the visual dimensional, ultrasonic and eddy current examination of them during each refueling.[8]

Accident induced loads and stresses have been calculated for both the 7 x 7 and 8 x 8 assemblies using the same methods. The limiting accident loads result from a steam line break. The pressure differences following a steam line break are less than 10% greater than normal operating pressure differences. As in normal operation, the pressure differences in an 8 x 8 assembly following a steam line break are 5 to 10% greater than in a 7 x 7 assembly. The loads following a steam line break are well below the allowable loads.

The behavior of the two fuel designs under seismic loading is nearly identical. This is so because the stiffness of the fuel channel and the weight of the fuel assembly are the same for both designs. Only these two parameters need to be considered since the stiffness of the bundle of fuel rods is small compared to the channel, and the clearance between the channel and the rod bundle is small compared to the limiting deflection of the channels. The predicted loads from the postulated safe shutdown earthquake are one-half the allowable loads.

We conclude that based on operating experience with similar fuel, the results of an out-of-pile test of an assembly of similar design, the increased thermal margins which the 8 x 8 fuel has, the Technical Specification requirements to monitor and limit off-gas and coolant activity, and the existence of a continuing fuel rod surveillance program which includes destructive and non-destructive post irradiation examinations, the cladding integrity of the 8 x 8 fuel will be maintained during normal operation and abnormal operational transients and significant amounts of radioactivity will not be released. Furthermore, we conclude that accidents or earthquake induced loads will not result in an inability to cool the fuel and safely shutdown the reactor.

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3.0 NUCLEAR DESIGN

The nuclear design of the 8 x 8 reload assemblies is similar to that of the equivalent 7 x 7 reload assemblies as shown in Table II. The U-235 enrichments for the individual fuel rods, the number and distribution of fuel rods containing gadolinia, and the water-to-fuel ratio are similar in the two designs. However, two features which might effect the nuclear characteristics differ in the proposed 8 x 8 reload assemblies and the equivalent 7 x 7 reload assemblies. First, there are 64 rods in the 8 x 8 assembly, compared to 49 in the 7 x 7 assembly. Second, the 8 x 8 assembly has a water filled rod near the center of the assembly and the 7 x 7 does not.

The major items of interest from the standpoint of nuclear design of the 8 x 8 reload fuel assembly are the uncontrolled and controlled (all control rods in) reactivity, the change in reactivity of the assembly with burnup, the local peaking in the assembly, the Doppler reactivity coefficient, the delayed neutron fraction, and the void reactivity coefficient. Values of these parameters as a function of burnup for an infinite lattice of 8 x 8 reload assemblies were presented[1,2,3,4,5] and compared with values for an infinite lattice of 7 x 7 assemblies of similar enrichment. In general, the values for the 8 x 8 lattice differed by less than 10% from those of the 7 x 7 lattice.

The same calculational techniques were used in calculating the lattice parameters for the 8 x 8 reload assemblies and those equivalent 7 x 7 assemblies. The particulars of the design of the assembly do not directly enter reactor calculations since homogenized parameters for the assembly (e.g., few group cross-sections, diffusion coefficients) are used as input. The 8 x 8 reload assemblies are neutronically similar to the 7 x 7 assemblies (i.e., similar enrichment, water-to-fuel ratio and gadolinia content), and we believe the calculational techniques are of equivalent accuracy for an 8 x 8 assembly as for a 7 x 7 assembly. The local peaking factor for the 8 x 8 reload assemblies is reported to decrease monotonically with exposure, while that of the equivalent 7 x 7 assemblies is reported to decrease with an exposure of about 10 GWD/t, then increase slowly. This behavior was explained, in response to a staff question, in terms of differences in the shift in the position of the peak local power rod within the bundle as a function of exposure. The effect of the water is to increase moderation in the interior of the bundle and reduce the rod to rod power peaking. Voiding of the water rod would decrease the reactivity of the bundle and would depress the flux in the center of the bundle. (Voiding of the water rod is equivalent to increasing the void fraction in the assembly of about 1%).

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TABLE II

Nuclear Design Comparison

	8 x 8	7 x 7
Pellet Outside Diameter, in.	0.416	0.487
Rod Outside Diameter, in.	0.493	0.563
Rod-to-Rod Pitch, in.	0.640	0.738
Water-to-Fuel Ratio	2.60	2.43
U Bundle Weight, lbs.	404.6	427.8
Cladding Thickness	34	32
K_∞, cold uncontrolled	1.166	1.163
K_∞, cold-controlled	0.981	0.988
Max. Local Peaking Factor	1.22	1.24
Average U-235 content, %	2.62	2.63
Number Gadolinia containing pins	4	4
Relative gadolinia content of gadolinia containing pins	2	1
Number of water rods	1	0

2.59 w/o U-235 8 x 8 Assembly; 2.50 w/o U-235 7 x 7 Assembly

	8 x 8	7 x 7
Pellet Outside Diameter, in.	0.416	0.477
Rod Outside Diameter, in.	0.493	0.563
Rod-to-Rod Pitch, in.	0.640	0.738
Water-to-Fuel Ratio	2.60	2.53
U Bundle Weight, lbs.	404.6	412.8
Cladding Thickness, mils.	34	37
K_∞, cold uncontrolled	1.148	1.129
K_∞, cold-controlled	0.966	0.960
Max. Local Peaking Factor	1.22	1.30
Average U-235 content, %	2.50	2.50
Number gadolinia containing pins	4	4
Relative gadolinia content of gadolinia containing pins	1	1
Number of water rods	1	0

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We have reviewed the nuclear design of the 8 x 8 reload fuel assemblies by comparing their properties with equivalent 7 x 7 assemblies and conclude that the nuclear design of the 8 x 8 reload assemblies is acceptable.

4.0 THERMAL-HYDRAULIC DESIGN

During normal operation and abnormal operational transients, the design objective for both types of assembly is to maintain clad integrity and prevent the release of significant amounts of radioactivity. The fuel damage limits and thermal-hydraulic criteria used to evaluate the performance of the fuel is the same for both designs. During normal steady state operation the Minimum Critical Heat Flux Ratio (MCHFR) is heald above 1.9. For abnormal operational transients, the clad strain is limited to less than 1% and the MCHFR is maintained greater than 1.0. These design bases are the same as the design bases for fuel previously reviewed and accepted for boiling water reactors.

In general, the 8 x 8 fuel has greater thermal margins to these design limits than 7 x 7 fuel. The design value of linear heat generation rate for normal operation is 13.4 kw/ft for an 8 x 8 fuel and 17.5 to 18.5 kw/ft for 7 x 7 fuel. Based on previous experience, this lower thermal duty combined with the other design changes is expected to result in fewer clad perforations. During normal operation, the hot channel MCHFR in the 8 x 8 assemblies is expected to be greater than 2.3 which is 11% greater than the hot channel MCHFR expected for 7 x 7 assemblies. The LHGR which is calculated to produce 1% strain in the cladding is 1.8 times the design value for 8 x 8 fuel and only 1.5 times the design value for 7 x 7 fuel. Similarly, the LHGR which produces fuel pellet center-line melting is 1.4 times the design value for 8 x 8 fuel as compared to 1.2 times the design value for 7 x 7 fuel.

Since the 8 x 8 assemblies are different than the 7 x 7 assemblies, we reviewed the thermal-hydraulic design methods to determine their applicability to the new fuel design. The differences are the modified flow geometry and the introduction of an unfueled rod. The portions of the thermal-hydraulic design methods which might be affected by these differences and which we reviewed are the techniques used to calculate flow rate and critical heat flux in the 8 x 8 assemblies.

The methods used to calculate flow and pressure drop in the 8 x 8 assemblies are the same as that used for the 7 x 7 assemblies. However, empirical constants are varied to adjust the results to the specific fuel design. Tests have been made to determine these empirical constants for an 8 x 8 geometry and to confirm the method of calculating friction, acceleration and elevation pressure drop. Furthermore, the fuel assembly support casting orifice is the major flow resistance and, therefore, the flow distribution between fuel assemblies is insensitive to differences in the hydraulic characteristics of the fuel assemblies. The methods of hydraulic analyses are the same as those previously reviewed and accepted for boiling water reactors and are equally applicable for 8 x 8 fuel assemblies.

The correlation used to calculate the critical heat flux in the 8 x 8 assemblies is the same Hench-Levy correlation used in evaluation of 7 x 7 assemblies. Introduced in 1966, the Hench-Levy correlation has been the accepted basis for determining thermal margin for a variety of General Electric boiling water reactors. The 8 x 8 fuel assembly is, except for the inclusion of an unheated rod and the change in hydraulic diameter, very similar in geometrical and thermal-hydraulic characteristics to the 7 x 7 fuel assembly.

We have previously reviewed (12) the effect of an unheated rod and the applicability of a CHF correlation such as the Hench-Levy correlation which is based on average fluid conditions and concluded that the effect of the unheated rod is not significant. We have also reviewed the effect that the changes in subchannel hydraulic diameters might have on thermal performance and conclude that the subchannel flow in the 8 x 8 assembly is more balanced than in the 7 x 7 design and should result in improved thermal performance. Therefore, we conclude that the Hench-Levy correlation is equally applicable to both the 8 x 8 and the 7 x 7 assemblies.

Because the Hench-Levy correlation does not specifically account for non-uniform axial heat flux distributions and rod-to-rod variations in power, as exist in fuel assemblies, a lower limit line to the then existing critical heat flux data was chosen as the form of the correlation. In addition, for added conservatism, the steady state design CHF was to be such that it did not exceed the Hench-Levy CHF divided by 1.9.

In order to overcome these shortcomings of the Hench-Levy correlation and to provide a data base that is more representative of actual fuel assembly performance, General Electric constructed the ATLAS facility which has the capability to test full size, full power 8 x 8 rod bundles. Except for the method of heating the rods (electrical resistance heating) and differences in grid spacer design, the 8 x 8 rod bundles tested in the ATLAS loop are similar to fuel assemblies. The large body of critical heat flux data obtained from the ATLAS facility for both 7 x 7 and 8 x 8 of rods in 16, 49, and 64 rod bundles has provided the foundation for developing a new correlation called GEXL (General Electric Critical Quality X_C - Boiling Length) which GE proposed as a replacement for the Hench-Levy correlation. A new thermal design method (GETAB, General Electric Thermal Analysis Basis), which uses GEXL and appropriate design parameters to determine the maximum power capability of a fuel assembly during normal operation and abnormal operational transients and accidents conditions, is also proposed.

The Regulatory staff is now reviewing GEXL, GETAB, the Hench-Levy correlation, and the ATLAS rod bundle data. General Electric has informed the Regulatory staff that all operating BWR plants have been provided with GETAB with the instructions that, in the interim, operating thermal limits be determined by either the Hench-Levy correlation or GETAB, choosing the method that provides the more conservative result.

At this time the staff agrees that the operating plant thermal margins should be predicted on the basis of the method (i.e., either Hench-Levy or GETAB) which yields the more conservative result, on this basis, use of the Hench-Levy correlation for the 8 x 8 fuel design would be acceptable.

5.0 ABNORMAL OPERATIONAL TRANSIENTS

To assure the safety of the plant, the results of the analyses of abnormal operational transients are required to indicate that the fuel and the reactor coolant pressure boundary (RCPB) are not damaged. The fuel damage criteria are a minimum critical heat flux ratio (MCHFR) of unity and a cladding strain of one percent. The RCPB damage criteria is the system design pressure (as specified in the ASME Boiler and Pressure Vessel Code, Section III). These damage limits for 8 x 8 fuel are the same as previously reviewed and accepted for 7 x 7 fuel in boiling water reactors.

Abnormal operational transients are the result of single equipment failures or single operator errors that can reasonably be expected to occur during anticipated modes of station operation. The types of failures and errors considered are the same for both types of fuel. The transients resulting from these failures and errors can cause variations in both system parameters such as core flow, core power, pressure and coolant level, and in local parameters such as flow and power in a single assembly. System parameters are primarily a function of the core average nuclear, thermal and hydraulic characteristics.

Since the characteristics of the 8 x 8 assemblies are similar to those of the 7 x 7 assemblies, the 8 x 8 fuel has no significant effect on these transients. However, for the determination of local parameters, the characteristics of the 8 x 8 fuel may be significant. It has been reported(1) that the thermal margin of the hot assembly has been analyzed using the conservative fuel type and the results demonstrate that the fuel damage limits are not exceeded. The results of three limiting events, i.e., a seizure of one recirculation pump, the continuous withdrawal of a control rod, and the misorientation of an assembly indicate that the consequences of these events are less severe for 8 x 8 assemblies than for 7 x 7 assemblies. Analyses of all transients have been made(3) considering both the 7 x 7 and 8 x 8 assemblies and the results indicate that the fuel damage limits are not exceeded.

6.0 ACCIDENTS

Analyses of the design basis accidents are made to evaluate the capability of the engineered safety features to mitigate the consequences of postulated accidents and control the possible escape of fission products. The four postulated design basis accidents are the a) loss-of-coolant, b) steam line break, c) fuel handling, and d) control rod drop accidents.

6.1 Rod Drop Accident

The rod drop accident analysis is not significantly affected by a change from a 7 x 7 to an 8 x 8 assembly. The kinetics model uses homogenized cross sections and is not directly involved with the details of the lattices. The local peaking factors of interest are also similar for both types of assemblies. Analyses of the rod drop accident demonstrate that the dropping of a maximum worth sequenced control rod will not result in a peak fuel pellet enthalpy which exceeds the damage limit of 280 cal/gm.

6.2 Refueling Accident

The method of determining the number of rods which might fail following the dropping of an assembly is equally applicable to both designs. Since the types of assembly are similar, the total amount of fission products released from the 8 x 8 assemblies in a refueling accident would not be significantly greater than from the 7 x 7 assemblies.

6.3 Steam Line Break

The radiological consequences of a postulated steam line break outside of the primary containment are dependent on the amount of primary coolant lost during the accident and the concentration of the radioactivity in the coolant. The amount of coolant lost is primarily a function of system parameters which would not be significantly changed by introduction of 8 x 8 fuel assemblies. The concentration of radioactivity in the coolant is limited by Technical Specifications and is also unchanged. Therefore, the radiological consequences of a postulated steam line break accident are unchanged by the use of 8 x 8 fuel assemblies.

6.4 Loss of Coolant

The analysis of the performance of the ECCS and the response of the 8 x 8 fuel assemblies following postulated loss-of-coolant accidents has been made using the assumptions and calculational techniques described in "Part 2 - General Electric

Evaluation Model, Appendix A, Acceptable Evaluation Models including their Conservative Assumptions and Procedures" which is contained in the Commission's Interim Policy Statement, entitled "Criteria for Emergency Core Cooling Systems for Light-Water-Power Reactors" and publisehed in the Federal Register on June 29, 1971. The Commission Rule, "Acceptable Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors," dated December 28, 1973, is intended to replace the Interim Policy Statement. Conformance with this new rule, which includes revised criteria and revised features of the evaluation model, will require re-analysis of the ECCS performance. When the requisite evaluations are submitted to the Director of Regulation, as required by the implementation schedule contained in the rule, the staff will make its review and conclusions. Our current review is only concerned with compliance with the Interim Policy Statement. Since the 8 x 8 fuel assemblies are a different design than the 7 x 7 assemblies considered in the General Electric Evaluation Model described in NEDO-10329, and referenced in Part 2 of Appendix A to the Interim Policy Statement, the staff has reviewed the evaluation model to determine its applicability to the new fuel design. The features of the new fuel design which are different from the old design and significant in determining applicability of the evaluation model are: a) smaller diameter fuel rods; b) larger number of fuel rods in each assembly, and c) an unfueled central rod. The features of the evaluation model which might be affected by these changes in the design of the fuel assembly and which we reviewed include applicability of the transient critical heat flux correlation, the thermal radiation and the spray cooling convective heat transfer in an 8 x 8 array, and the effect of the unfueled rod on heat transfer.

As discussed in a preceding section of this report, we have reviewed the differences in the thermal and hydraulic characteristics between an 8 x 8 fuel assembly and the 7 x 7 assembly, and concluded that the steady state critical heat flux correlation is equally applicable to both designs. In addition, GE has nearly completed an extensive series of steady-state critical heat flux tests on full-scale, 8 x 8 heater bundles with varying inlet conditions, ar power distributions which are representative of expected conditions in a BWR. These tests will provide a large additional set of critical heat flux data applicable to the 8 x 8 fuel design. General Electric and the staff are now in the process of evaluating this data and its applicability to the conditions following a loss-of-coolant accident. Upon the completion of this evaluation and during the review of the re-analysis required by the new rule, the staff will re-examine the acceptability of the current critical heat flux model.

We have also reviewed the differences in thermal radiation and spray cooling characteristics between the 8 x 8 and the 7 x 7 fuel assemblies and conclude that the procedures used to calculate the heatup of an 8 x 8 fuel assembly following a loss-of-coolant accident are consistent with the approved General Electric

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Evaluation Model. Our conclusion is based on independent calculations usi g a computer program developed for the staff(13) and the results of full-scal , stainless steel 8 x 8 rod array, heater bundle spray cooling and flooding test(14, 15).

The adequacy of the thermal radiation model for an 8 x 8 fuel bundle has been verified by comparison of the predictions of clad temperature using both the GE(14) and staff's(16) computer programs to the results of steady-state heater bundle tests which had no spray cooling. The staff's computer program under-predicts the temperature of rods in the bundle by not more than 25°F, but overpredicted the temperature of some rods by as much as 150°F. The GE program predicted temperatures which were from 50 to 75°F lower than the staff's calculations. The temperature overprediction of the corner and unfueled rods may be due to local differences in emissivity. Although comparison of the gray body view factors for individual rods used in the two programs revealed no reason for the difference between the GE and staff results, the simpler nodalization of the heater rods in the GE program could account for the difference.

The adequacy of both the GE and staff heatup models, including both convective cooling to the spray and rod-to-rod radiation, was demonstrated by comparing predictions to the results from transient tests of the 8 x 8 stainless steel heater bundle. The predictions were based in part on the conservative values of spray cooling convective heat transfer coefficient specified in the IAC evaluation model. The other parameters, such as heat-generation, emissivity and thermal properties, were best estimate values. The staff's calculations are as much as 40°F lower, and as much as 80°F higher than the measured temperatures. The predictions reported by GE have approximately the same inaccuracy. These differences are within the uncertainties of the test results.

he General Electric Company has also completed a test witnessed by the staff on an 8 x 8 Zircaloy heater bundle, but has not yet reported the results. Previous tests have shown that a heatup model which is based on the results of tests with stainless steel rods can predict the thermal response of Zircaloy rods within the uncertainty of the experimental measurements. For most reactors which have jet pumps, the heatup transients are short, that is, approximately two minutes long, result in moderate temperatures, that is below 2000°F, and the degree of uncertainty is acceptably small. However, for transients which are longer and result in higher temperatures, such as occur in reactors without jet pumps, additional experimental verification of the applicability of analytical methods derived from stainless steel heater bundle tests to Zircaloy clad rods are required. Therefore, the results of this Zircaloy bundle test will be submitted and reviewed prior to use of fuel in reactors without jet pumps.

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We reviewed the effect that the unfueled rod might have on heat transfer. Inspection of the test results indicate the convective cooling of the rods prior to wetting of the unfueled rod is insensitive to the location of a rod relative to the unfueled rod. That is, rods immediately adjacent to the unfueled rod heat up at the same rate as rods which are separated from the unfueled rod by one row of rods. However, the unfueled rod is beneficial since after the rod wets, it acts as a thermal radiation sink. Wetting of the unfueled rod is not included in either the GE or staff computer program models.

We conclude that the General Electric Evaluation Model as described in NEDO-10329 and including the requirements specified in Part 2 of Appendix A of the Interim Policy Statement when modified as described in NEDE-10801 to account for differences between the design of the 8 x 8 and 7 x 7 assemblies, is applicable to the evaluation of the ECC performance of 8 x 8 assemblies in a General Electric boiling water reactor which has jet pumps.

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CHRONOLOGY

REGULATORY PEVIEW OF GENERAL ELECTRIC COMPANY 8X8 FUEL ASSEMBLY

September, 1973	General Electric Company submits report "General Design Information for General Electric Boiling Water Reactor Reload Fuel Commencing in Spring, '74," NEDO-20103.
September, 1973	General Electric Company, Nucelar Fuel Department, submits report "Dresden 3 Nuclear Power Station, Second Reload License Submittal."
September 14, 1973	Memo to A. Giambusso, AEC from P. D. Raymond, "Nine Mile Point Unit 1 - Second Refue' ng."
October 15, 1973	Niagara Mohawk Power Corporation submits report "Nine Mile Point Unit 1 Safety Analysis for Type 5 and Type 6 Reload Fuel."
October 17, 1973	Memo to V. Stello, AEC, from D. Ross and T. Novak, "Review of GE 8X8 Reload Fuel Assemblies."
October 24, 1973	Memo to V. Stello, AEC, from W. Minners, AEC, "Review of GE 8X8 Reload Fuel Assemblies."
November, 1973	Northern States Power Company submits report "Monticello Nuclear Generating Plant - Second Reload Submittal."
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December 6, 1973	General Electric Company. Nucelar Fuel Department submits Supplement B, "Dresden 3 Nuclear Power Station, Second Reload License Submittal."
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December 18, 1973	ACRS meeting on GETAB and applications to LOCA analyses for 8X8 assemblies.
January 8, 1974	ACRS Subcommittee on Fuels Meeting, Washington, D.C.
January 10, 1974	ACRS Meeting, Washington, D. C.
January 24, 1974	ACRS Subcommittee on Fuels Meeting, Denver, Colorado.
January 30, 1974	AEC - General Electric Meeting.
February 5, 1974	Letter from J. A. Hinds to V. Moore.

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APPENDIX E

REVIEW AND EVALUATION OF GETAB

(General Electric Thermal Analysis Basis)

FOR BWRs

By

Technical Review

Directorate of Licensing

United States Atomic Energy Commission

September 1974

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APPENDIX E

REVIEW AND EVALUATION OF GETAB FOR BWRS

1. INTRODUCTION

During anticipated abnormal operating transients in a boiling water reactor, a criterion of no fuel rod damage is applied. Historically, the thermal-hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical heat flux at which boiling transition is calculated to occur has been adopted as a convenient limit.

Since 1966 this limit for BWR fuel assemblies has been based on the Hench-Levy correlation(1) which was formulated as a lower limit line to the existing rod bundle critical heat flux data. To allow sufficient margin for uncertainties, the steady state operating conditions of General Electric Company reactors were limited such that during anticipated abnormal transients the calculated heat flux was always less than the lower limit critical heat flux line. That is, during transients, the critical heat flux ratio was always greater than unity.

Based on recent extensive critical heat flux data obtained with full size, full power rod bundles in the ATLAS test loop, the General Electric Company has developed a new method of critical heat flux correlation. With this General Electric Critical Quality (X_c)-Boiling Length Correlation, (GEXL)(2,3) critical power, the fuel assembly power at which boiling transition is expected to occur, is based on the correlation of the critical quality and boiling length. The basic form of GEXL is identical to the well verified CISE (Italy) correlation. In contrast to the Hench-Levy correlation, which is a lower limit line to the data, the GEXL correlation is a best fit to the ATLAS data. GE proposes to determine thermal limits using a new thermal design method, the General Electric Thermal Analysis Basis (GETAB)(2,3) which incorporates the GEXL correlation. The uncertainties associated with the GEXL correlation and the reactor steady state operating parameters are combined statistically. The steady state operating conditions are to be limited such that during anticipated abnormal transients, more than 99.9% of the fuel rods in the core are expected not to experience boiling transition. That is, during transients, the minimum critical power ratio is to be greater than a value determined by the magnitude of these uncertainties. A typical value is 1.05.

This report presents the results of our review of the GEXL correlation and the GETAB method. We have reviewed the GEXL correlation and its basis including the experimental data and analytical methods used to determine the correlation. We have also reviewed the experimental methods used to obtain the data, including the design and operation of the ATLAS test loop. Finally, we reviewed the application of the correlation to the design and operation of boiling water reactors. As discussed at the end of this report, based on this review we conclude that the GEXL correlation and a statistical application of the correlation, similar to the proposed GETAB method, are acceptable.

2. ANALYTICAL

A. Critical Heat FLux Correlations

a. Methods

Several methods, as described by Keeys, et al[4] have been proposed for the prediction of heat input for, and the position of, critical heat flux in non-uniformly heated tubes. These prediction methods have the following features:

(a) In the "Average Heat Flux" concept the critical heat flux is assumed to be correlated thus:

$$\phi_c = f\,(G,\ D,\ P,\ \Delta H_{in},\ L), \qquad (1)$$

where L is the total heated length, ΔH_{in} is the inlet subcooling; G is mass velocity, D is the equivalent diameter, P is the pressure, and ϕ_c is the average heat flux. Inspection of these parameters show that the critical power is assumed to be independent of the form of the heat flux spatial distribution. Although this method does not permit the prediction of the critical heat flux location, it is simple and has been shown by Lee[5], to give predictions within 10% of experimental data for tubes with large L/D ratios and moderate peak-to-average heat flux form factors.

(b) In the "Local Conditions" concept the expression for critical heat flux is:

$$\phi_c = F\,(G,\ D,\ P,\ X_c) \qquad (2)$$

This method assumes that the critical heat flux location will occur at a local heat flux, ϕ_c, and a local steam quality, X_c, irrespective of the axial heat flux distribution. Examples of this type of correlation are those of Thompson and Macbeth[6], Tong (W-3)[7], and Gellerstedt, et al, (B&W-2)[8]. However, the need for a correction to the predicted uniform critical heat flux, for the case of non-uniform axial heat flux, is described in Reference 9 for the W-3 correlation [8], and in Reference 10 for the B&W-2 correlation[9]. The Hench-Levy correlation[1], for use in BWR rod bundles, is similar in form to equation 2.

$$\phi_c = f\,(G,\ P,\ X_{CB}) \qquad (3)$$

where ϕ_c is the local critical heat flux, and X_{CB} is the bundle average critical steam quality. The equivalent diameter does not appear as the correlation is applicable only to GE BWR rod bundles. Since 1966, the Hench-Levy correlation, in the form of a lower limit line to the then existing rod bundle data, has served as the basis for predicting the thermal margin in BWRs.

(c) In the "Boling Length" concept the critical quality is correlated in the following form:

$$x_c = f\ (G, P, D, L_B), \qquad (4)$$

where x_c is the steam quality at dryout conditions and L_B is the length over which boiling takes place. Examples of this type of correlation are those of Bertoletti, et al [11], and Hewitt.[12] The "boiling length" type of correlation has the demonstrated advantage of being able to correlate critical heat flux data for both uniform axial heat flux as well as non-uniform axial heat flux. Since the axial (and radial) heat flux distribution in a BWR fuel bundle is not uniform, the correlation of Bertoletti et al [11] was chosen as the basis for the new GE correlation called GEXL (General Electric Critical Quality X_c - Boiling Length). As used for GE rod bundles, GEXL relates the bundle average critical quality, X_{CB}, to boiling length.

A recent comparison of the correlations described above (excluding GEXL), and adaptations of some of the methods to use with rod bundles, to BWR rod bundle critical heat flux data is described by Guarino, et al.[13] The compared data comprised 785 points and included uniform heat flux, radially non-uniform heat flux, and axially non-uniform heat flux.

b. GEXL

The GEXL correlation is a variation of the critical quality vs boiling length correlation of Bertoletti, et al[11] which was based on single tube data, but was shown to apply, with good success, to a large amount of rod bundle critical heat flux data.[11] Subsequently, the similar (X_c vs L_B) correlation of Hewitt, et al,[14] for single uniformly heated tubes, was applied to rod bundles by Marinelli and Pastori[15] on the basis of ascribing the flow rate attending each rod to that which exists within a zero shear interface between rods (the CISE criterion).

The GEXL correlation is of the form:

$$X_{CB} = f\ (G, D, P, L_B, L, R)$$

where the terms are as previously defined, L is the heated length, and R is a weighting factor which characterizes the local rod-to-rod peaking pattern with respect to the most limiting rod. In addition, R is dependent on lattice dimensions (7x7 or 8x8) and grid spacer configuration. Since R, in effect, accounts for the flow and enthalpy distribution within the bundle, it can be interpreted as being the bundle average analog of subchannel analysis.

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The form of the CISE correlation[11] is:

$$X_c = \frac{a\,L_B}{b+L_B} \qquad \text{where } a = f(P,G), \quad b = f(P,G,D)$$

where L_B is defined as the heated length over which the steam quality is greater than zero. The form of the GEXL correlation is similar to that of CISE; however, in GEXL, L_B is defined as the distance from the initiation of bulk boiling to the boiling transition point.*

An example of the ability of the critical quality - boiling length CISE correlation,[11] which is the basis for GEXL, to bring critical heat data for various axial heat flux distributions into a single curve is shown in Figure 1. These data, for 1000 psia steam, from Keeys, et al,[4] include the following heat flux distributions: uniform, exponential decrease, and symmetrical chopped cosine. Figure 2, from Reference 2, the report under review, shows the identical results for Freon-114 wherein critical heat flux data for uniform, cosine, half cosine, inlet peak, and outlet peak heat flux distributions as well represented by a single critical quality-boiling length curve.

c. Data Basis for GEXL

The GEXL correlations (7x7, 8x8) are based on experimental data which cover the following ranges:

Pressure: 800 to 1400 psia
Mass Flux: 0.1×10^6 to 1.25×10^6 lb/hr sq. ft.
Inlet Subcooling: 0 to 100 Btu/lb
Local Peaking: 1.61 corner to 1.47 interior
Axial Profile: Uniform
Cosine (1.39 max. to avg. at 72 in. from inlet)
Inlet Peak (1.60 max. to avg. at 51 in. from inlet)
Outlet Peak (1.60 max. to avg. at 93 in. from inlet)
Double Hump (1.46 and 1.38 max. to avg. at 51 and 108 in. from inlet)
Lattice: 7x7 and 8x8
Rod Bundles: 16, 49, 64 rods
Heat Lengths: 6 ft, 12 ft, 12-1/3 ft

*Keeys, et al[4] use L_B as defined in GEXL, whereas Hewitt, et al[14] use L_B as defined by Bertoletti, et al[11]. However, this difference is not important as long as the particular definition is consistently applied.

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The GEXL correlations are based on data of which the overwhelming portion were obtained in the ATLAS loop and the remainder in the Columbia University test loop. The data used for GEXL are:

Lattice	No. of Rods	Axial Profile	Heated Length	No. of Points
7 x 7	16	Uniform	6 ft*	84*
	16	Cosine	12 ft	223
	49	Cosine	12 ft	127
8 x 8	16	Cosine	12-1/3 ft	211
	64	Cosine	12-1/3 ft	1058

*Columbia University data.

From the above, it can be seen that the GEXL correlations (7x7 and 8x8) are based on 1803 data points of which 1285 were obtained with full size (49 and 64 rods), full length (12 ft and 12-1/3 ft) rod bundles. Except for 84 points which were obtained with a uniform axial heat flux, the data were obtained with a cosine heat flux distribution.

After development of the correlations they were compared to additional data which represent the whole range of parameters. The data used for comparison are:

Lattice	No. of Rods	Axial Profile	No. of Points*
7 x 7	16	Uniform	456
	16	Cosine	121
	49	Cosine	470
	16	Inlet Peak	484
	16	Outlet Peak	477
	49	Outlet Peak	32
	16	Double Hump	434
8 x 8	16	Cosine	131

*Includes 220 points from the Freon loop

Inspection of these comparisons shows that the 7x7 GEXL correlation, which was based on uniform and cosine axial profiles, accurately predicts the whole range of data.

While the formulation of the GEXL correlations (7x7 and 8x8) relied very heavily (1269 out of 1803 points) on data taken with the 8x8 lattice, only a small fraction (131 out of 2605 points) of the data used to check the GEXL correlations were for the 8x8 lattice. In addition, about half of these confirming data for the 8x8 lattice, were for Freon and all were for a cosine heat flux distribution. GE is performing additional tests with 8x8 lattices and non-uniform axial profiles. These additional tests will include profiles with a peak toward the outlet. Inlet peaked profiles may also be included. Although these tests can provide additional confirmation of the 8x8 GEXL correlation predictive capability, they are not required for two reasons. First, the 7x7 GEXL correlation, which was based solely on data from uniform and cosine axial heat flux profiles tests, accurately predicts boiling transition for the other tested profiles. There is no reason to believe that the 8x8 GEXL correlation would not perform similarly. Second, in the application of GEXL, the standard deviation of the uncertainty in the 8x8 GEXL correlation will be increased to account for the less complete data base. The standard deviation of 2700 experimental critical power ratios (ECPR) about the 7x7 GEXL correlation is 3.6%. The standard deviation of 1299 ECPR about the 8x8 GEXL is 2.8%. In applying the 8x8 GEXL to the determination of BWR thermal limits, the standard deviation will be increased to at least 3.4%, which is the square root of the sum of the variance of the 8x8 experimental results and the variance of the means of the 7x7 data for each flux shape.

B. Subchannel Analysis Method

The subchannel analysis methods used to develop and complement the GEXL correlation have been reviewed and evaluated by our PNL consultant.[(6)] This evaluation indicates that the primary General Electric subchannel analysis model has a reasonable basis. The basic formulations and computations of the model are typical of subchannel analyses available in the open literature; however, formulations of the exchange mechanisms between subchannels contain some unique features. The inclusion of the particular formulations of the turbulent mixing and void drift exchange models is one of the most significant aspects in the GE subchannel analysis formulation and is the primary reason that the GE model does a good job of predicting subchannel flow and enthalpy data for simulations of BWR rod bundles.

As part of its subchannel analysis package, GE has included a subchannel critical power correlation which correlates GE rod bundle data with subchannel parameters of mass velocity, quality and boiling length determined by the subchannel analysis. It is important to note that tube and annulus correlations were not directly used in the subchannel correlation, but instead were merely used to establish the important parameters for correlation of the boiling transition data for rod bundles. The complete subchannel analysis method correlates the experimental data within $\pm 6\%$. Because of its subchannel nature, the subchannel analysis method is a valuable tool for use in parametric design studies of BWR fuel bundles. Thus, GE has two methods which each correlate the GE data well, namely, the subchannel analysis method and the bundle average GEXL method.

However, it does not logically follow that since the two empirical methods provide comparable critical power results that any one of the individual factors in either of the correlations can be justified on that basis alone. Each function must stand on its own merits in conjunction with the empirical or semi-empirical correlating scheme of which it is a part. Consequently, the arguments associated with the justification of the bundle average R factor via the subchannel critical power results are not only circular in nature, since both critical power correlations are based on the same rod bundle data, but are unnecessary since the GEXL correlation stands on its own merits as a method of predicting critical bundle power.

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C. Data Comparison

1. Comparison of ATLAS with Hench-Levy Correlation

As the Hench-Levy correlation[1] presently forms the basis for predicting thermal limits for BWRs, this correlation was independently compared, by our ANC consultant,[17] to the rod bundle boiling transition data obtained in the ATLAS test loop. The comparison, consisting of 5868 data points, showed that, except for the case of uniform axial and radial heat flux distribution, the Hench-Levy correlation, which is a lower limit line, is not conservative.* That is, the experimental critical heat fluxes are generally less than those predicted by the correlation.

Figure 3, from Reference 17, shows critical bundle power as a function of inlet subcooling with mass velocity as a parameter. The consistency shown by these data is typical of that obtained in the ATLAS test loop.

A comparison between measured and predicted critical heat flux (Hench-Levy) for uniform axial and radial heat flux distribution, is shown in Figure 4.[17] It can be seen that almost all of the measured heat fluxes are greater than the predictions thus showing that, for these conditions, the Hench-Levy correlation is conservative. This result is not too surprising as the experimental basis for establishing the Hench-Levy correlation was primarily comprised of uniform axial and radial heat flux rod bundle data. However, as shown in the following two figures, the greater the departure from uniform axial and radial heat flux distribution, the greater becomes the disparity between the Hench-Levy correlation and measured critical heat fluxes, with Hench-Levy being the higher.

Figure 5,[17] which compares critical heat flux for uniform axial heat flux and non-uniform radial peaking to predictions, shows that about one-half of the data points are less than the Hench-Levy prediction. When corner peaking is combined with non-uniform axial heat flux distributions (cosine, inlet peak, outlet peak double hump), the comparison in Figure 6[17] results. Here it is seen that, for the vast majority of data points, the Hench-Levy predictions substantially exceed the measured critical heat fluxes. Thus, it can be concluded that, for rod bundle heat flux distributions and heat lengths which correspond to those found in a GE BWR (non-uniform axial heat flux with corner peaking, 12 ft. heated length) the Hench-Levy correlation does not provide a lower limit line.

Two possible reasons why the Hench-Levy correlation does not provide a lower limit line to the ATLAS data which were obtained with reactor fuel assembly-like rod bundles are:

*This argument as to "conservative" does not embrace the use of Hench-Levy i.e., the present requirement that MCHFR using Hench-Levy be greater than 1.9 during state operation.

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1) The approximately 700 critical heat flux data points for 4 and 9 rod bundles, which form the basis for the Hench-Levy correlation, were obtained with uniform axial heat flux with some data obtained with corner peaking or interior rod peaking. As shown in Figure 7, from Reference 4, the critical heat flux, for a single tube, is substantially greater with uniform axial heat flux than with a cosine heat flux distribution.

2) For a 19 rod bundle, using 1000 psia steam, Matzner et al[18] demonstrated that, over the heated length range of 1-1/2 ft. to 9 ft., the bundle average critical heat flux increases as heated length decreases. At a given steam quality, a pressure of 1000 psia, and a mass velocity of 1×10^6 lb/hr sq. ft., a decrease in the heated length from 9 ft. to 5 ft. increases the critical heat flux by about 40%.

Since the heated lengths of the 4 and 9 rod bundle critical heat flux data which formed the basis for the Hench-Levy correlation varied from 3 to 5 feet, it follows that these data provided greater critical heat flux than would have been obtained had the heated length been more representative of actual fuel rods; i.e., 12 ft.

From the above considerations, it can be concluded that the critical heat flux data used to develop the Hench-Levy correlation were high with the result that the correlation is not a lower limit line when applied to data obtained in long (12 ft.) rod bundles with non-uniform axial and radial heat flux profiles.

b. Comparison of Hench-Levy and GEXL for Rod Bundle Power

It was previously shown that, for non-uniform axial and radial heat flux distributions, the critical heat flux predicted by the Hench-Levy correlation generally was substantially greater than that measured. Since GEXL is a close representation of the experimental boiling transition data, it might be anticipated that, on a bundle power basis, Hench-Levy will provide higher values than GEXL. That is the case shown below.

Figures 8 and 9 show critical power, for a 7 x 7 rod bundle, as a function of inlet subcooling, for mass velocities of $.5 \times 10^6$ and 1×10^6 lb/hr sq. ft., respectively. The two upper curves in Figure 8 represent the Hench-Levy correlation and GEXL, respectively, along with experimental data. It is seen that the Hench-Levy correlation predicts about 5 percent greater power than GEXL or the data. The two lower curves represent the operating curves for each of the correlations; Hench-Levy is based on critical heat flux divided by 1.9 whereas GEXL is, in a typical case, based on critical bundle power divided by 1.2.* Note that, with respect to the operating curves, GEXL permits higher power than Hench-Levy at low inlet subcoolings while the converse holds true at high inlet subcoolings. At a mass velocity of 1×10^6 lb/hr sq. ft., as shown in Figure 9, both the Hench-Levy correlation and its operating curve are always higher than the respective GEXL curves.

*As discussed in section 4, use of the arithmetic mean, rather than the geometric mean, in the statistical analysis, will increase the required CPR from 1.20 to 1.24. This would correspondingly lower the GEXL operating curves on both Figures 8 and 9.

c. Comparison of ATLAS Data with GEXL

Our ANC consultant,(19) independently compared all of the ALTAS data, consisting of 5868 data points, to the GEXL correlation (February 1974 version). The ANC comparison has basically substantiated the claim by GE that the GEXL correlation fits the ATLAS data with a standard deviation of 3.5%, the value quoted prior to the second data submittal. ANC has determined the standard deviation to be 3.7% based on the data from all 69 assemblies and the February 1974 version of the GEXL correlation.

No significant error trends in the correlation are observed with respect to the input variables (pressure, mass flux, axial power shape, radial power shape, inlet subcooling, quality). While small systematic differences between assemblies are shown, these differences are not associated with any particular phenomenon.

The variation of the ratio of the difference between the measured and calculated power to calculated power with axial flux shape is shown in Figure 10. When compared to the uniform and cosine power shapes which formed the basis for the 7 x 7 and 8 x 8 GEXL correlations, the inlet peak shape is seen to be slightly conservative while the outlet peak and double hump shapes are slightly nonconservative. The same relative positions among the various axial power shapes is shown, in terms of predicted to measured power, on p. 5-5 of Reference 3. However, it should be noted that these comparisons are applicable only to the 7 x 7 lattice as data with inlet peak, outlet peak, and double hump axial shapes have not, as yet, been obtained with the 8 x 8 lattice.

General Electric has limited the application of the GEXL correlation to conditions where the inlet subcooling is 100 Btu/lb or less. However, as no trend with respect to inlet subcooling is observed over the entire range of subcooling, this restriction does not appear to be necessary.

The sensitivity of the 7 x 7 and 8 x 8 GEXL correlations to pressure, mass flux, and R factor was evaluated. This parameter study has shown that at a particular combination of conditions it is possible to predict negative critical quality at positive values of boiling length. As shown in Figure 11, this inconsistency occurs at the higher pressures ($P > 1200$ psia), at higher mass fluxes ($G > 1 \times 10^6$ lb/hr sq. ft.), and at high values of R factor ($R > 1.2$). The consequences of such a condition are that it is impossible to obtain a convergence of the GEXL critical quality-boiling length curve and the energy balance quality-boiling length curve. This problem occurs only at short boiling lengths where boiling transition does not occur for BWR conditions. If required, it may be possible to achieve a solution by ignoring the first 15-20 inches of boiling length. Only four ATLAS data points had conditions where a negative quality was predicted by GEXL. For two of these points, a reasonable solution was obtained by ignoring the first 20 inches of boiling length. For the other two points, convergence was obtained, but the results were unsatisfactory. Another anomaly in the behavior of GEXL was observed at a high R factor ($R = 1.25$) where at 800 and 1400 psia, the curve for $G = 1 \times 10^6$ lb/hr sq. ft. crosses the curves for $G = .75 \times 10^6$ and $G = 1.25 \times 10^6$ lb/hr sq. ft. However, the differences between the two highes mass flux curves is very small over the entire boiling length range.

Unmodified, the GEXL correlation fails to predict accurately the location of boiling transition. To correct this, GE has formulated a location predictor correction to modify the GEXL correlation for use in the prediction of the boiling transition location. An error trend with respect to boiling length is observed in Figure 12 where the location residual is plotted vs. boiling length. This figure shows that at shorter boiling lengths, the correlation, using the location predictor correction, generally predicts the location downstream of the measured position while, at the longer boiling length, the correlation generally predicts the location upstream of the measured position. The shorter boiling lengths are characteristic of the inlet peaked axial profile. Figure 13 shows the location residual vs. axial power shape. Generally, the scatter for the prediction of location is greater than for the prediction of power.

While this review shows that the GEXL correlation does a good job of predicting the critical power for the ATLAS data, it must be borne in mind that the correlation is completely empirical and no attempt has been made by General Electric to make phenomenological explanations or justifications for any of the terms in the correlation. Consequently, the correlation should be used only for heat transfer predictions which are within the range of the thermal-hydraulic conditions from which it was derived.

D. Evaluation

In view of the fact that the Hench-Levy correlation does not, as originally thought, provide a lower limit line to the recent (ATLAS) critical heat flux data which were obtained in rod bundles which closely simulate reactor fuel assemblies, it is reasonable to ask: "What assurance is there that the new correlation, GEXL, will not be found to be inadequate, with regard to the prediction of fuel assembly thermal margin, at some future date?" This question can be answered by the following:

While fairly extensive, the data used to develop the Hench-Levy correlation were obtained from rod bundles which did not duplicate the number of rods, the axial and radial heat flux profile, or the heated length of reactor fuel assemblies. However, test operating conditions (flow rate, pressure, inlet temperature) did duplicate reactor conditions.

In contrast, GEXL is based on more than four thousand boiling transition data points, many of which were obtained from full size, full length, rod bundles with a wide range of axial and radial heat flux profiles. Furthermore, the spacer grids used in critical heat flux tests were very similar to those used in a fuel assembly, and in addition, had the same axial spacing. A wide range of operating conditions (flow rate, pressure, inlet temperature) duplicating those of reactor conditions were used in performing the tests. Since the test sections (49 and 64 rods) are, except for the method of heating, virtually duplicates of fuel assemblies when the axial heat flux distribution of the test and fuel assembly coincide, the ATLAS test data can be considered to be in the nature of calibration.

Based on the above evidence, there is high assurance that the ATLAS test assemblies and tests duplicate the thermal performance of fuel assemblies. From this, it follows that the GEXL correlations (7 x 7 and 8 x 8), which are based on ATLAS data, can be expected to

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faithfully mirror the thermal performance of BWR fuel assemblies for conditions which fall within the prescribed limits of the correlations. Based on the very detailed, independent evaluation of GEXL, the Hench-Levy correlation, and the ATLAS data, by our ANC consultant[20] the revelation of some anomalies in the GEXL correlation under certain extreme conditions, does not seriously flaw GEXL or its utility as a prediction method. In total, the ATLAS data - GEXL correlation combination provides a distinct improvement over the presently used Hench-Levy method.

3. EXPERIMENTAL

A. The ATLAS Heat Transfer Facility

The GEXL boiling-transition correlation is based on data measured in the ATLAS heat transfer facility. ATLAS was constructed by the General Electric Company for the purpose of doing steady state and transient thermal-hydraulic tests of full scale electrically heated rod bundles which simulate reactor fuel.

In preparation for this review, a team of four AEC Regulatory staff members and consultants visited San Jose, the site of ATLAS, to witness boiling transition tests and discuss the operation of the loop with members of the ATLAS operations staff. The following discussion is based on information obtained at that time, together with a written description of ATLAS provided by GE and is directed at areas relating to accuracy and reliability of the ATLAS test results. More details on ATLAS can be found in a letter, dated July 31, 1973, from J. A. Hinds to Dr. J. M. Hendrie.[20]

a. The Loop

ATLAS is an all stainless-steel loop designed to operate with water at wide ranges of conditions up to the following maxima:

2250 psig system pressure
655°F system temperature
1000 gpm test section flow
17.2 MW test section power

It can therefore be used for the full range of steady state testing appropriate to boiling (and pressurized) water reactors. Furthermore, power, flow, and pressure controls are available to simulate a wide variety of transient and accident conditions.

The power supply consists of four silicon controlled rectifier units each comprised of 96 SCR cells balanced in impedance to equally share the load. Voltage to the test section is controlled manually by operator adjustment of a 0-10v demand signal to a feedback control system. This control system alters the firing phase at the gates of all SCR cells so as to reduce the error between the demand and output voltage to within $\pm$ 1/4%. For transient tests there is provision for automatically following a programmed power history with a time constant of less than 10 ms.

The AC ripple component of the rectified voltage is 5 to 6v compared to a full power voltage of 180v. The contribution of this ripple to the test section power varies from 0.5% at full power to 1% at 30% full power and is accounted for by a compensated, Hall-effect wattmeter. The output of this wattmeter is displayed as a digital reading in kilowatts and is available to the data acquisition and control systems. Calibrated DC shunts are also used to calculate the test section power. They measure the current from each SCR unit and the resultant calculated power agrees with the wattmeter measurement within ± 1.%.

Redundant measurements are also made of other parameters which affect boiling transition:

- test section inlet temperature is measured to ± 1°F by an RTD and checked by three Chromel-constantan thermocouples

- test section pressure is measured ± 5 psi by a Heise gauge and the pressure drop by the differential pressure transducer

- test section flow is measured to ± 1% by both a turbine flowmeter and an orifice/servomanometer

b. Test Sections

The test sections consist of a number of heater rods arranged in an array identical to that of the nuclear fuel being simulated and held by grid spacers of the appropriate design and location. The heater array is housed within a flow channel which accurately simulates the fuel channel wall.

In the tests to date, heat has been generated ohmically in the heater wall. The axial distribution of the heat flux is, therefore, dependent on the local wall thickness which is determined by drawing the tube over a variable mandrel. Boiling transition is detected by electrically insulated, ungrounded Chromel-constantan thermocouples with Inconel sheaths silver soldered to the inner surface of the heater wall. For axially uniform power distributions boiling transition is known to occur at the downstream end of the test section, in general, so there is no problem in locating the thermocouples at the correct axial location. For nonuniform cases however, boiling transition occurs over a range of axial locations, usually between 0.7 and 0.9 times the channel length. Therefore, in these cases, GE installs a larger number of thermocouples, selecting a variety of axial locations based on their experience with non-uniform test sections. Once tests begin on a particular test section, it soon becomes apparent where boiling transition tends to occur and thermocouples in this region are monitored preferentially. The error associated with this detection procedure is minimal for two reasons:

- considerable experience has shown that boiling transition is initiated just upstream of a spacer grid on one of the higher powered rods, and thermocouples were attached accordingly.

- if, in spite of the GE experience, boiling transition occurs between two planes of thermocouples, only a minimal power increase (1 to 2% according to experience) will cause the boiling transition zone to advance to the thermocouple plane.

Test sections geometries in ATLAS have included 4 x 4 and 7 x 7 heater bundles arranged in the 7 x 7 reactor fuel assembly array and 8 x 8 bundles arranged in the 8 x 8 fuel assembly array, in each case using the appropriate heater diameter and length.

The grid spacers of the heater bundles were spaced at the same axial intervals as fuel bundle spacers. Except for overall dimensions, in the case of the 16 rod bundles, and the design of the lantern spring the bundle spacer materials and dimensions are the same as the fuel assembly spacers. Stiffer springs were provided for the test bundles in order to resist the magnetic forces present in the tests. Based on comparative tests with 8 x 8 bundles in the Freon loop, GE stated that the critical power in the bundles with unmodified spacers was equal to or greater than in the bundles with stiffer springs.

The axial flux shapes tested are those shown in Figure 14. They were chosen to represent the widest range of shapes anticipated during the core life. Considering the wide range of peaking factors included in the ATLAS program the tests appear to simulate as closely as possible the geometry and power distributions expected to occur in BWR fuel.

c. Test Procedures

The following procedure is used to measure boiling transition at steady state conditions. The inlet temperature, flow and pressure are selected and held constant by the loop operators. Errors between the selected and measured values signal alarms which the operators may cancel by correcting these parameters. The test section power is slowly increased by operator manual adjustments while the operators continually monitor the operating conditions and the strip-chart records of the thermocouple signals intended to indicate boiling transition. The onset of boiling transition is identified by an increase in heater temperature of about 25°F. At this point, all thermocouple signals are checked on a Metrascope to assure that no thermocouple which is not connected to the strip-chart recorder is indicating boiling transition, and the operators signal the on-line computer to record all the pertinent data. The engineering data required to assess the results and to proceed to the next run are pointed out, the subcooling is changed to a new value and the next run done in the same manner, until the desired range of subcoolings is covered. The flow rate is then changed and the procedure repeated.

For transient tests, the flow rate is varied by timer circuits which actuate an air-operated flow control valve and the power is varied by a programmned function generator. The raw data, including heater thermocouple signals, can be sampled as often as 50 times per second and recorded on magnetic tape for subsequent processing.

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B. Comparison of ATLAS Data with Columbia University Data

As evidence of the accuracy of the ATLAS loop results, GE repeated a test series run earlier in the Columbia University Heat Transfer Facility. A comparison of the two sets of results is shown in Figures 15 through 19. Estimates of the percentage difference between curves drawn through each set of data, shown in the lower right hand corner as Δ lie between -5% (the ATLAS data are lower) and +3% (the ATLAS data are higher). In view of the scatter which typifies boiling transition test data, the Columbia University and ATLAS loops agree remarkably well.

C. Evaluation

In general, the ATLAS Heat Transfer Facility compares favorably with any facility in the world constructed for steady-state and transient boiling transition tests. Furthermore, it incorporates special features:

- automatic alarm system to ensure required test conditions are closely met

- specially designed controls and data acquisition system to facilitate transient tests

- highest test section power to any loop

which make ATLAS superior to other facilities and ensure a valuable source of data useful in the safe design of BWRs.

4. GETAB APPLICATION

General Electric proposes to establish design and operational thermal limits based on the GEXL correlation. These limits were previously based on the Hench-Levy correlation. The GEXL correlation is based on a larger amount of more representative data than the Hench-Levy correlation. The GEXL correlation is a best-fit of the data while the Hench-Levy correlation is a lower limit of the data.

GE proposes to state the thermal limit in terms of the critical power ratio (CPR) which is not only a consequence of the form of the new GEXL correlation, but is also more representative of the available thermal margin. Previously, the thermal limit was stated in terms of the critical heat flux ratio (CHFR), which is not directly related to the thermal margin. The use of CPR rather than CHFR as a thermal limit more clearly defines the thermal margin available.

GE also proposes to combine the effect of the uncertainties in the GEXL correlation with the uncertainties in the reactor operating variables in determining the thermal limits. Previously, only nominal values of the operating variables were used in determining heat flux relative to the CHF limit line. Although statistical analyses have been applied to the previous CHF thermal limit in order to evaluate the effect of uncertainties in the operating variables, the direct incorporation of uncertainties in the proposed CPR thermal limit assures that uncertainties are considered during design and operation of the reactor.

GE proposes that transients caused by single operator error or equipment malfunction shall be limited such that considering uncertainties in defining the core operating state, more than 99.9% of the fuel rods would be expected to avoid boiling transition. The application of this design basis to the determination of steady state operating limits is in two steps.

First, a statistical model is used to calculate the minimum critical power ratio (MCPR) for which less than 0.1% of the rods are expected to experience boiling transition. Second, a transient model is used to calculate the change in CPR resulting from transients. The steady-state operating limit is determined as the sum of the largest change in CPR due to any of the transients considered and the MCPR at which less than 0.1% of the rods are expected to experience boiling transition. The transient model (NEDO-10802)[21] is the same model previously used in calculating the Hench-Levy CHFR limit, and is not a subject of this review. The staff is reviewing this subject separately. The statistical procedure uses a computer program which calculates the CPR of the bundles in the core assuming a given power distribution and values of the operating variables. Using the calculated values of CPR, the probability of boiling transition occurring is summed for all rods in the core. Successive trials using random variations in the operating variables are performed until the mean and standard deviation of the probability of boiling transition occurring in the core is found.

The probability of boiling transition occurring is calculated based on the standard deviation of the ATLAS data relative to the GEXL correlation assuming a normal distribution. Because only tests with a symmetrical cosine axial profile are included in the 8 x 8 data, the magnitude of the uncertainty in the 8 x 8 GEXL correlation is increased to be comparable to the larger variability of the 7 x 7 data which included four other axial profiles. GE originally used the anti-log of the mean of the logarithm of the number of rods expected to experience boiling transition, in the determination of the CPR limit. The use of this geometric mean reduces the uncertainty interval. The procedure has been modified, by GE,[22] to use the arithmetic mean. This resulted in an increase in the MCPR for which less than 0.1% of the rods are expected to experience boiling transition.

The random variations in operating variables are based on estimates of the uncertainties in each variable.[23,24] A review and evaluation of these variables has shown that the variables which contribute significantly to the overall uncertainty have been considered. The estimated value of these uncertainties and the basis for the value depend on the specific design and equipment of each reactor and will be evaluated for each reactor at the time Technical Specifications are issued.

The proposed design basis appears at first to be a departure from the intent of the previous basis. The intent of the previous basis was that boiling transition would not reasonably be expected to occur on any rod in the core when at the thermal limit (i.e., the worst fuel assembly had a calculated MCHFR of unity). Under the proposed

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basis for core-wide transients, 0.1% of the rods in the core would be expected to experience boiling transition at the thermal limit (e.g., a MCPR of 1.05 on the worst assembly in a typical reactor). However, the proposed limit includes uncertainties in the reactor operating variables and the previous limit does not. If uncertainties in operating variables are not considered, there is a 95% confidence that with a CPR of 1.05 there is a 95% probability that boiling transition does not occur in the worst bundle. Therefore, a MCPR of 1.05 is roughly equivalent to a MCHFR of unity and both design bases provide similar assurance that boiling transition would not occur following core wide transients.

However, for local transients, the proposed design basis is a departure from the previous basis. Previously, the calculated MCHFR in any assembly was limited to unity and no rods were expected to experience boiling transition. Under the proposed basis, all of the rods in a fuel assembly could be expected to experience a boiling transition without violating the proposed basis, since all of the rods in one bundle comprise only approximately 0.1% of the rods in a core. For example, if a MCPR of 0.95 were calculated for the worst fuel assembly, that is, boiling transition would be predicted to occur, the proposed basis would not be violated. Therefore, the proposed design basis provides less thermal margin following a localized transient than the previous basis.

We conclude that the proposed design basis (i.e., more than 99.9% of the fuel rods in the core would be expected to avoid a boiling transition caused by single operator errors or equipment malfunctions) is acceptable when applied to core-wide transients such as a turbine-trip or pump-coastdown transient. We also conclude that the method used to calculate the MCPR thermal limit is an acceptable method by which power distribution and uncertainties in the GEXL correlation and the reactor operating parameters can be included in the determination of whether the design basis is met. However, we conclude that applying the proposed design basis to local transients such as control rod withdrawal, is inappropriate. Therefore, we require that the MCPR thermal limit determined for core wide transients also be used as the MCPR thermal limit for local transients.

5. STAFF FINDINGS AND CONCLUSIONS

The staff has reviewed the General Electric Thermal Analysis Basis and its application to reactor design and operation. Included in the review were the GEXL correlation, which is the basis for GETAB; the analytical methods used to develop this correlation; the experimental results from which the correlation was synthesized; and the experimental methods used to obtain the data.

Based on our review of the design and operation of the ATLAS test facility, the staff concludes that the steady state and transient tests had accurately controlled and measured test conditions. Comparison among the results of tests conducted on the ATLAS and Columbia loops verified reproducibility and lack of bias of the experimental results.

The experimental results were mainly obtained from full size, full length rod bundles which duplicated fuel assemblies in all respects that could significantly affect boiling transition. The tests were performed with a range of test conditions (flow, pressure,

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temperature and power) and heat flux distributions, both axial and radial, which equaled or exceed those expected to occur in a fuel assembly. Therefore, we conclude that the experimental results represent the thermal performance of GE 8 x 8 and 7 x 7 fuel assemblies.

Based on an independent comparison of the ATLAS data to the GEXL correlation, we conclude that the data can be conservatively treated as normally distributed about the correlations with a standard deviation of 3.6% and 3.4% for the 7 x 7 and 8 x 8 GEXL respectively. While small systematic differences between assemblies with different power distributions are shown, the correlation is slightly conservative with respect to the most probable distributions, (i.e., inlet peak and symmetrical cosine). Although the correlation has some anomalies at extreme conditions, GEXL can predict within a defined uncertainty the thermal performance of GE 8 x 8 and 7 x 7 fuel assemblies for the expected range of reactor normal steady state operation and abnormal operating transients.

General Design Criteria 10 requires that "acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences." We conclude that the proposed design bases (i.e., transients caused by single operator error or equipment malfunction shall be limited such that considering uncertainties in monitoring the core operating state, more than 99.9% of the fuel rods would be expected to avoid boiling transition) meets the criterion when applied to core - wide transients. However, we require that the MCPR limit derived for core-wide transients also be used as the Safety Limit applicable to local transients such as a control rod withdrawal. We also conclude that the statistical model used to derive the MCPR limit is acceptable.

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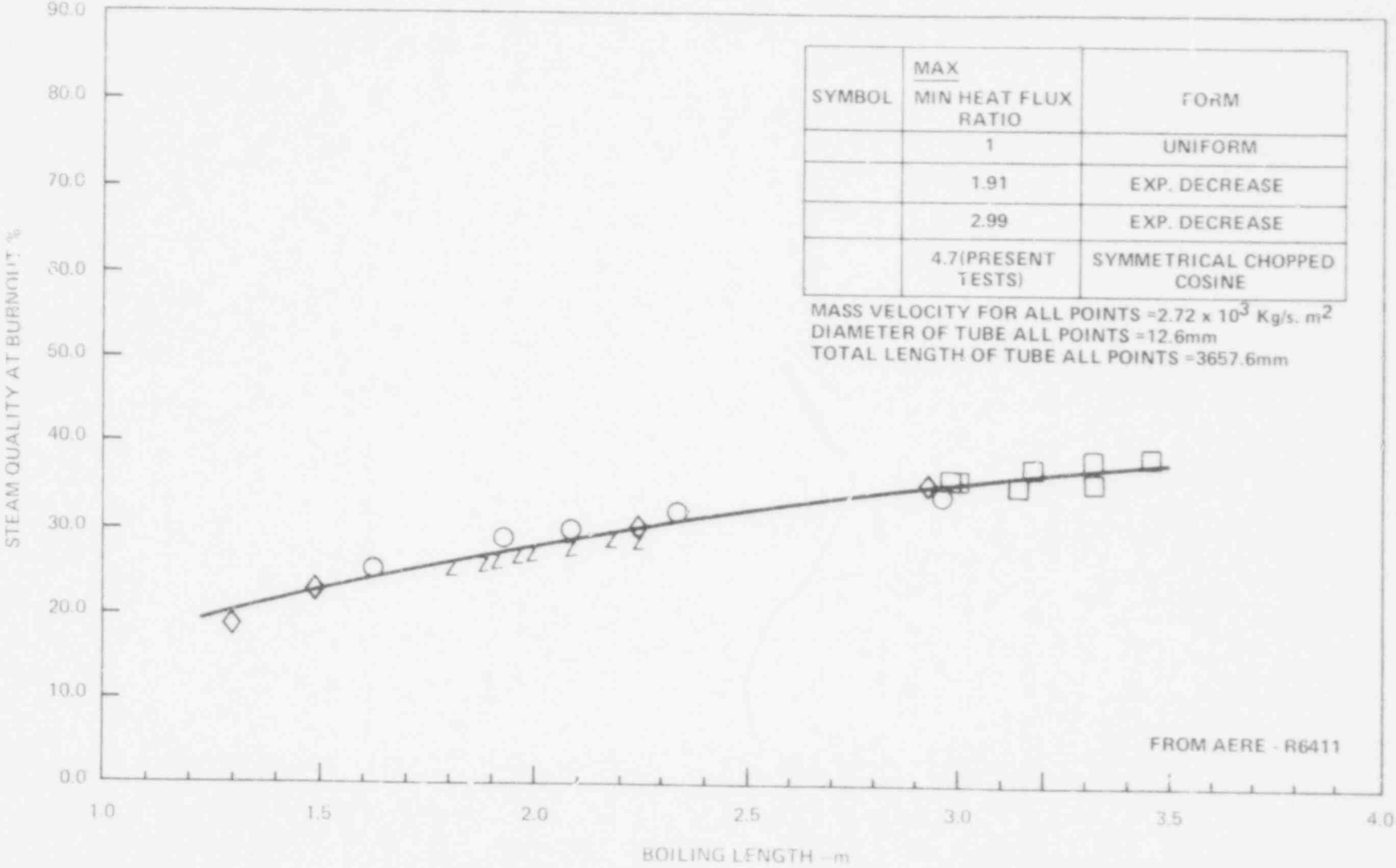

Figure 1 COMPARISON OF BURNOUT CONDITIONS ON A BOILING LENGTH BASIS FOR VARIOUS HEAT FLUX DISTRIBUTIONS

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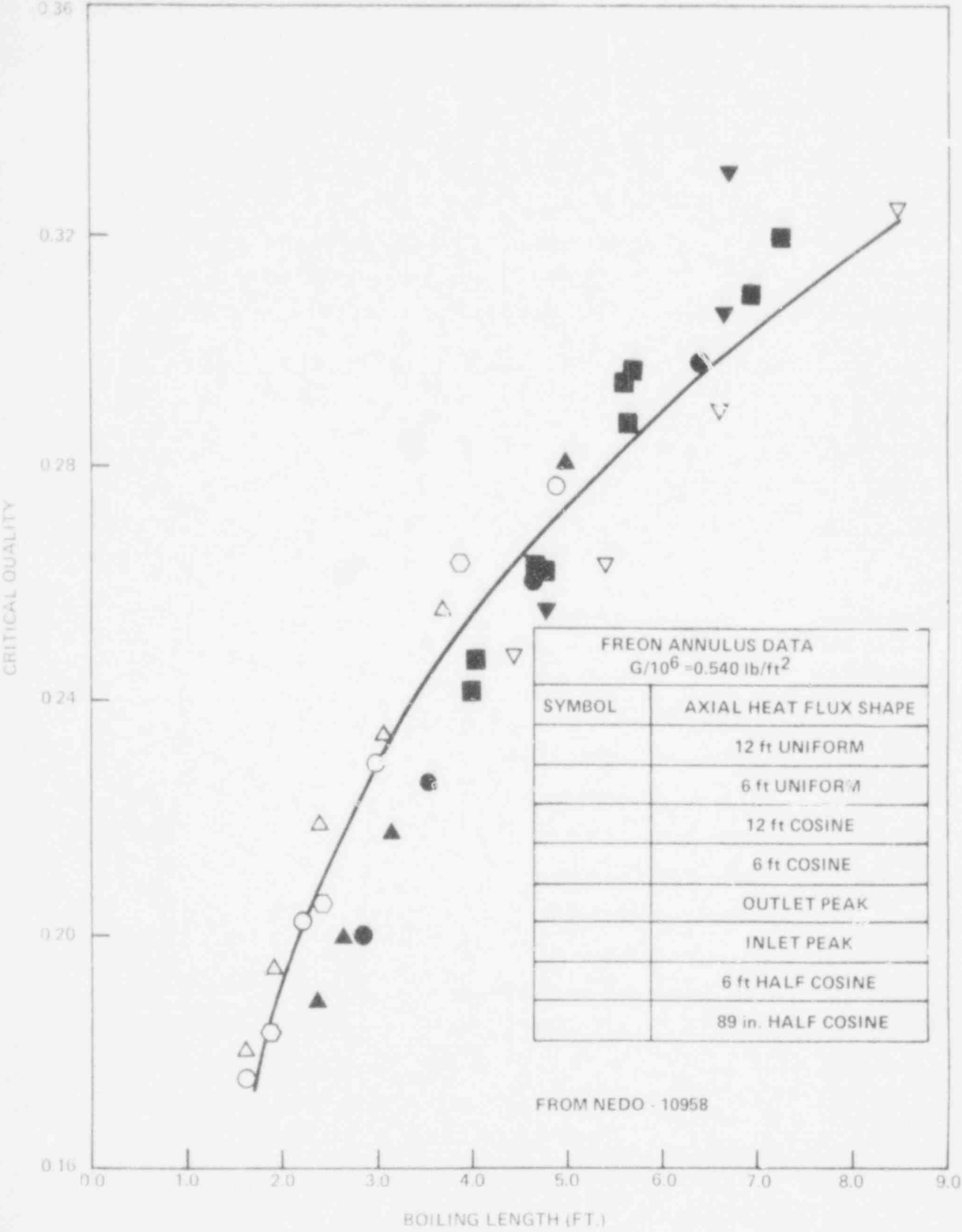

Figure 2 CRITICAL QUALITY VS. BOILING LENGTH, FREON-114 ANNULUS DATA

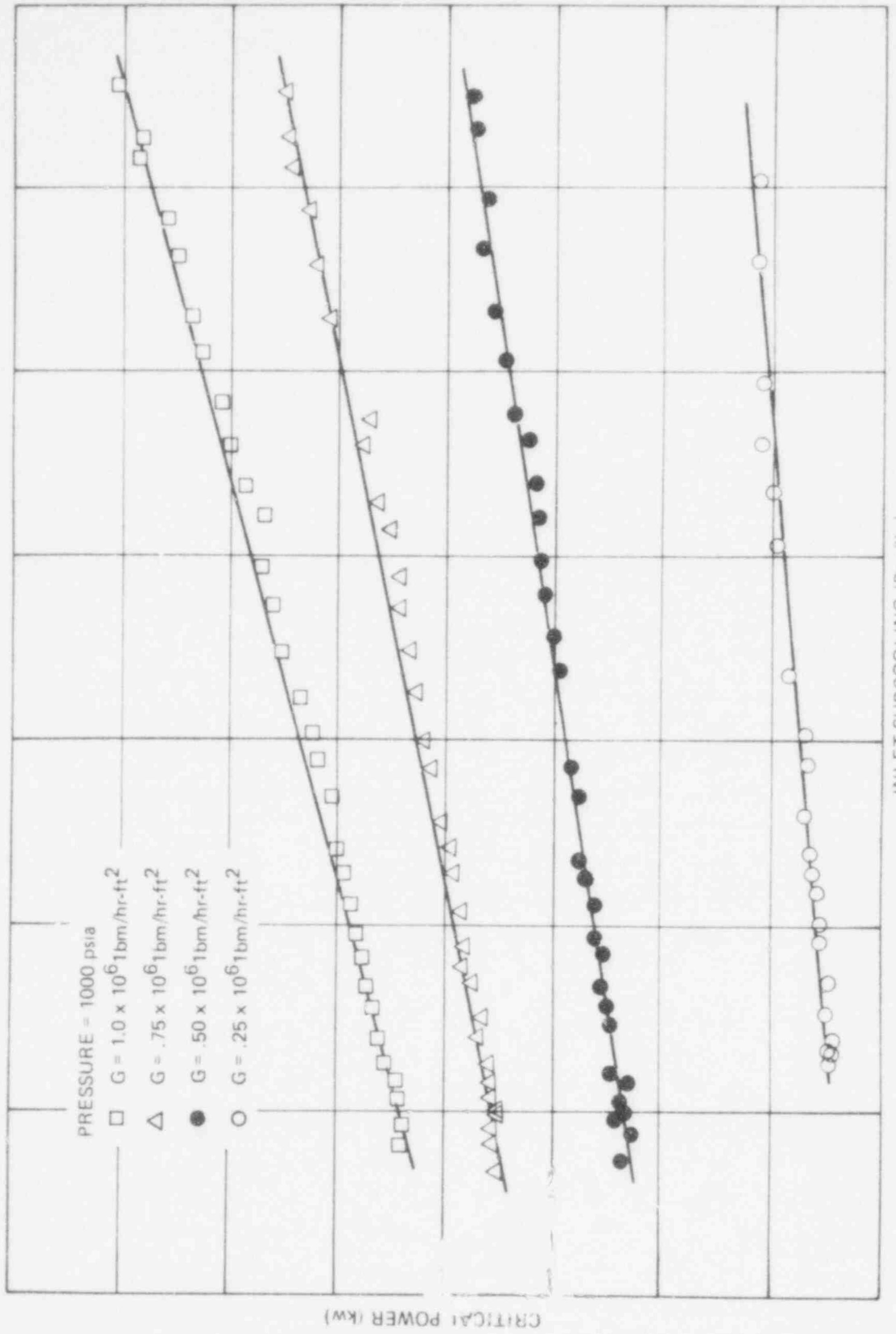

Figure 3 CRITICAL POWER VS. INLET SUBCOOLING FOR ATLAS TEST ASSEMBLY 30B

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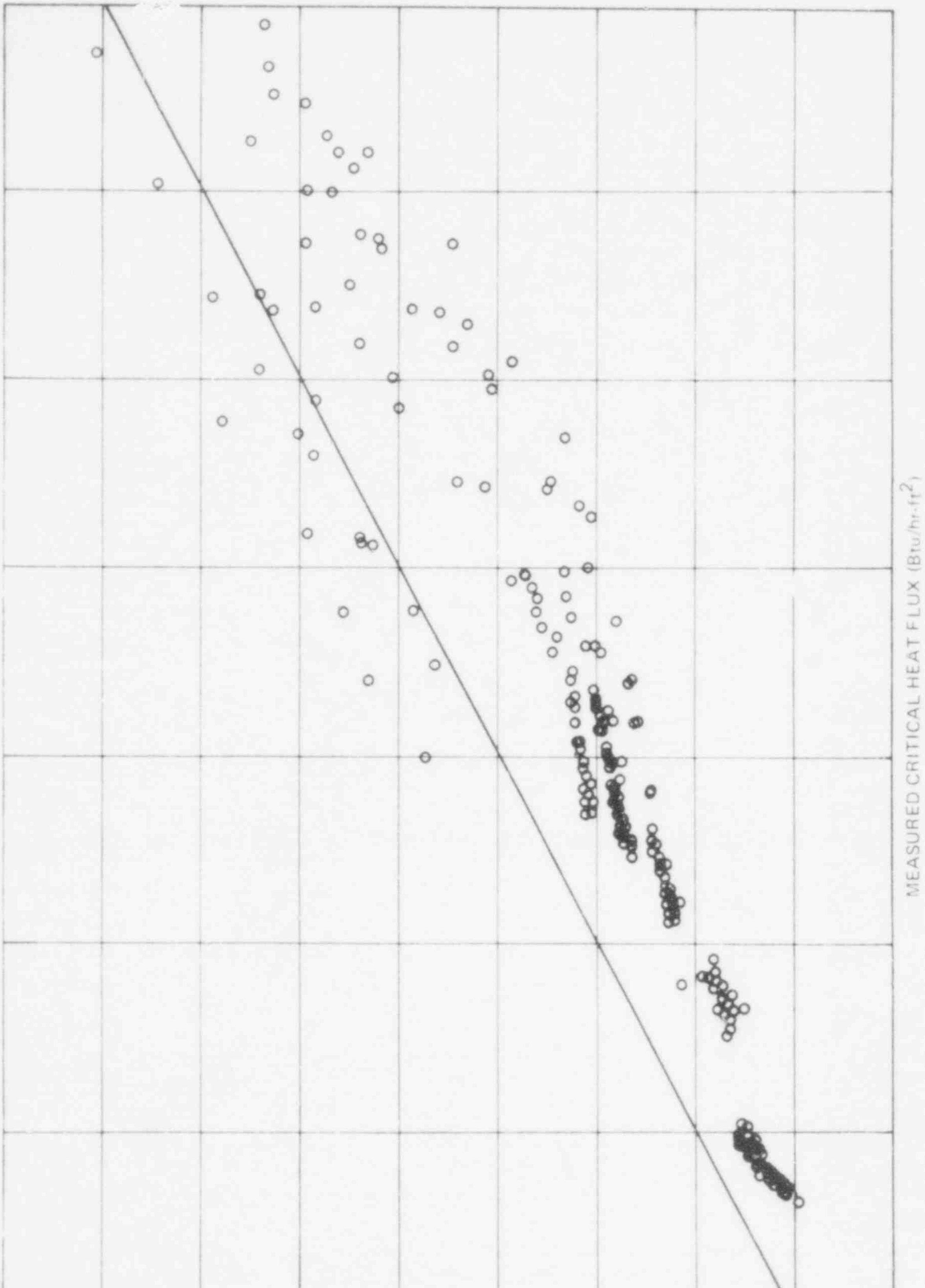

Figure 4 MEASURED VS. PREDICTED CRITICAL HEAT FLUX FOR ATLAS DATA WITH UNIFORM AXIAL AND UNIFORM RADIAL POWER PROFILES

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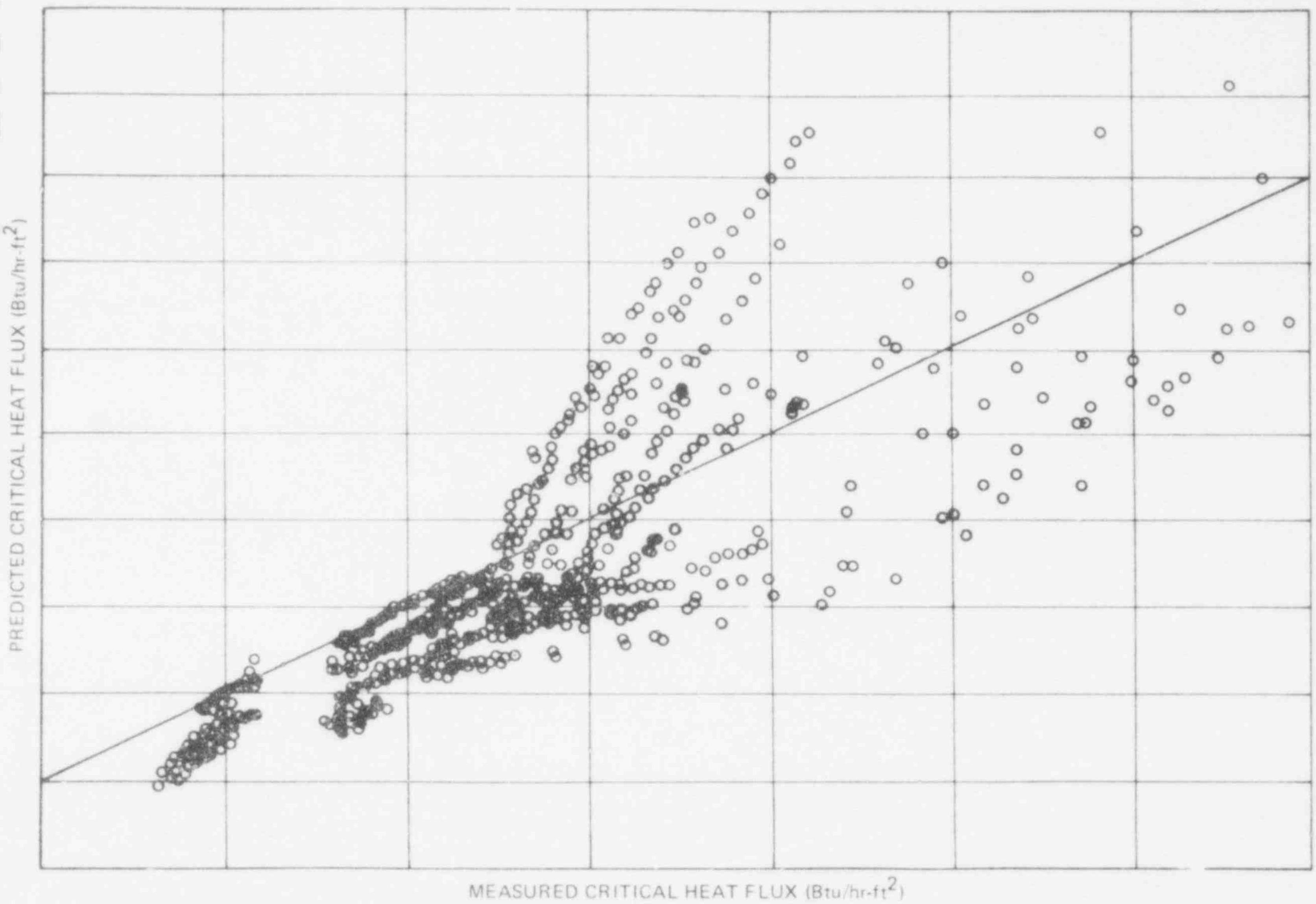

Figure 5 MEASURED VS. PREDICTED CRITICAL HEAT FLUX FOR ATLAS DATA WITH UNIFORM AXIAL POWER PROFILE

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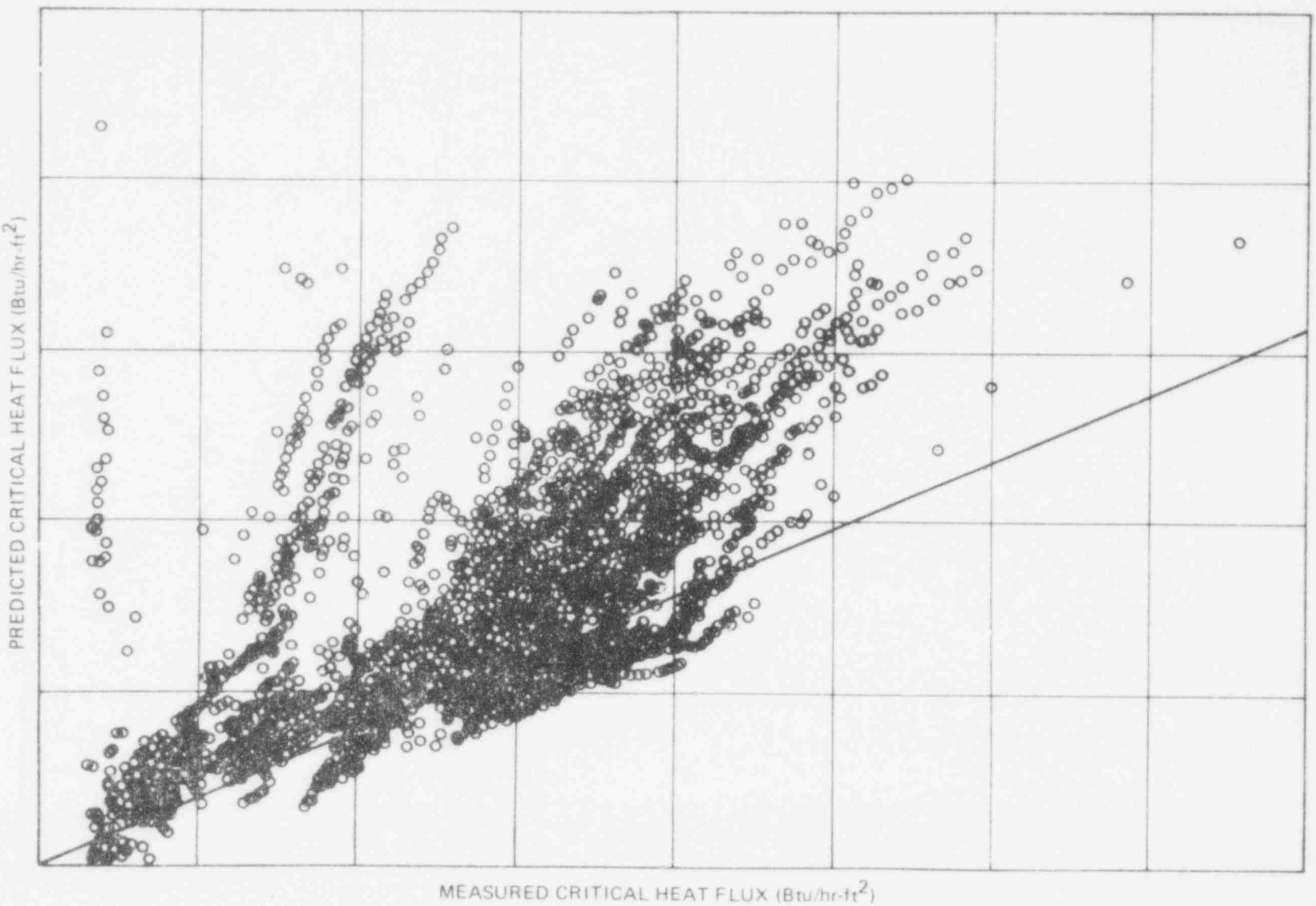

Figure 6 MEASURED VS. PREDICTED CRITICAL HEAT FLUX FOR ATLAS DATA WITH CORNER PEAKED RADIAL POWER PROFILE

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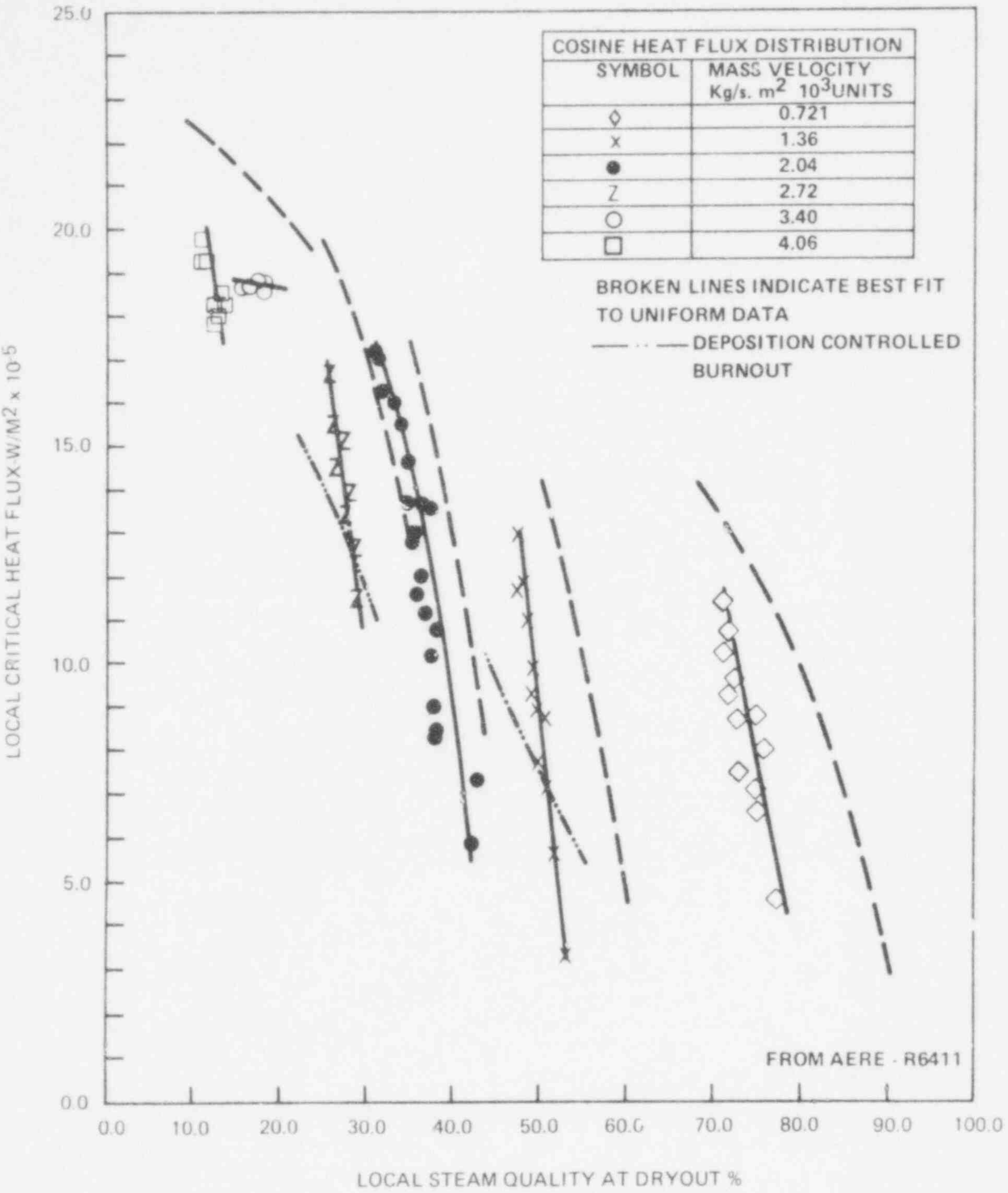

Figure 7 COMPARISON OF BURNOUT CONDITIONS ON A LOCAL CONDITIONS BASIS FOR UNIFORM AND COSINE HEAT FLUX DISTRIBUTIONS

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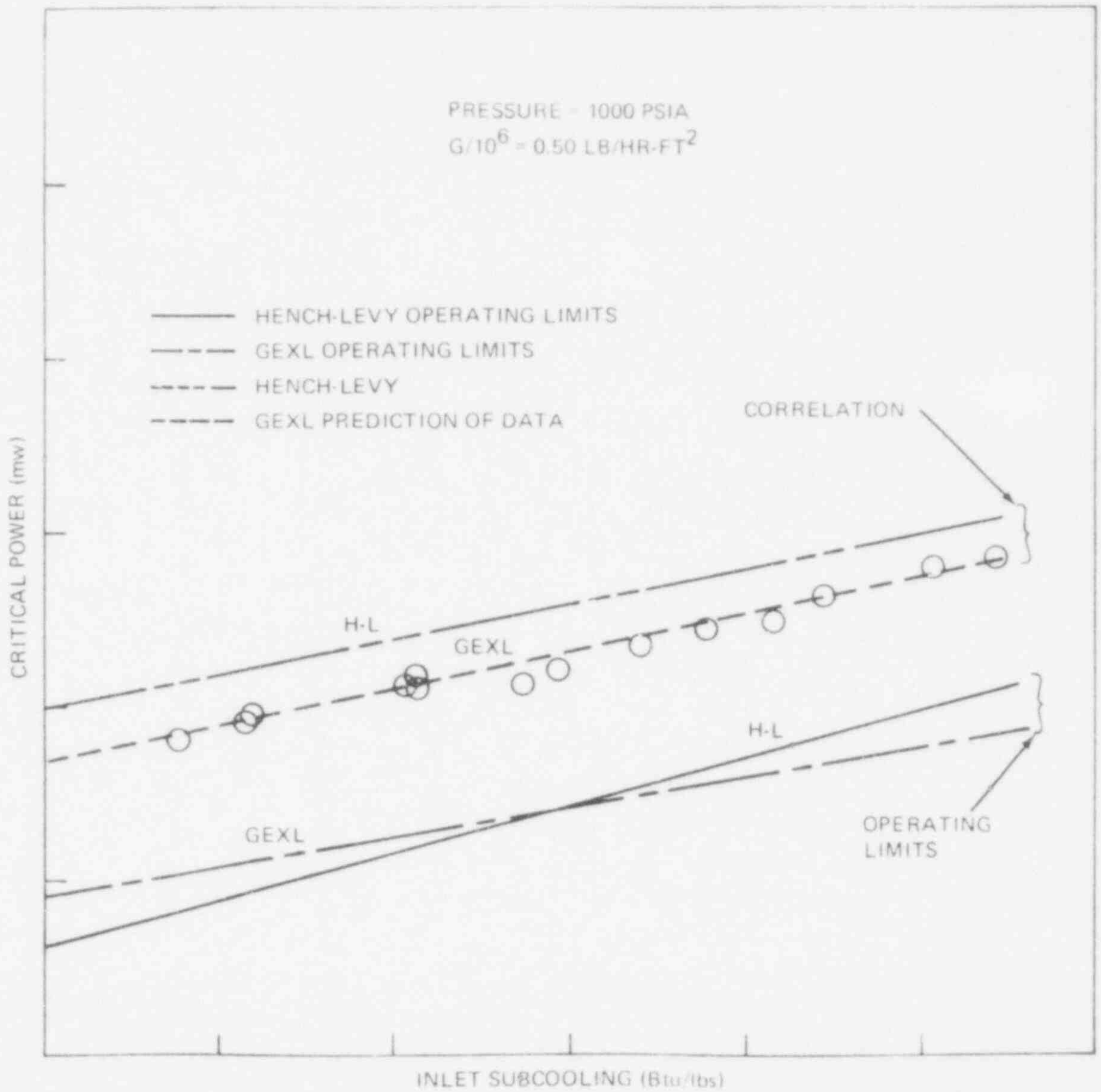

Figure 8 COMPARISON OF HENCH-LEVY AND GEXL WITH 49 ROD ATLAS DATA (1.23 CORNER PEAKING, COSINE)

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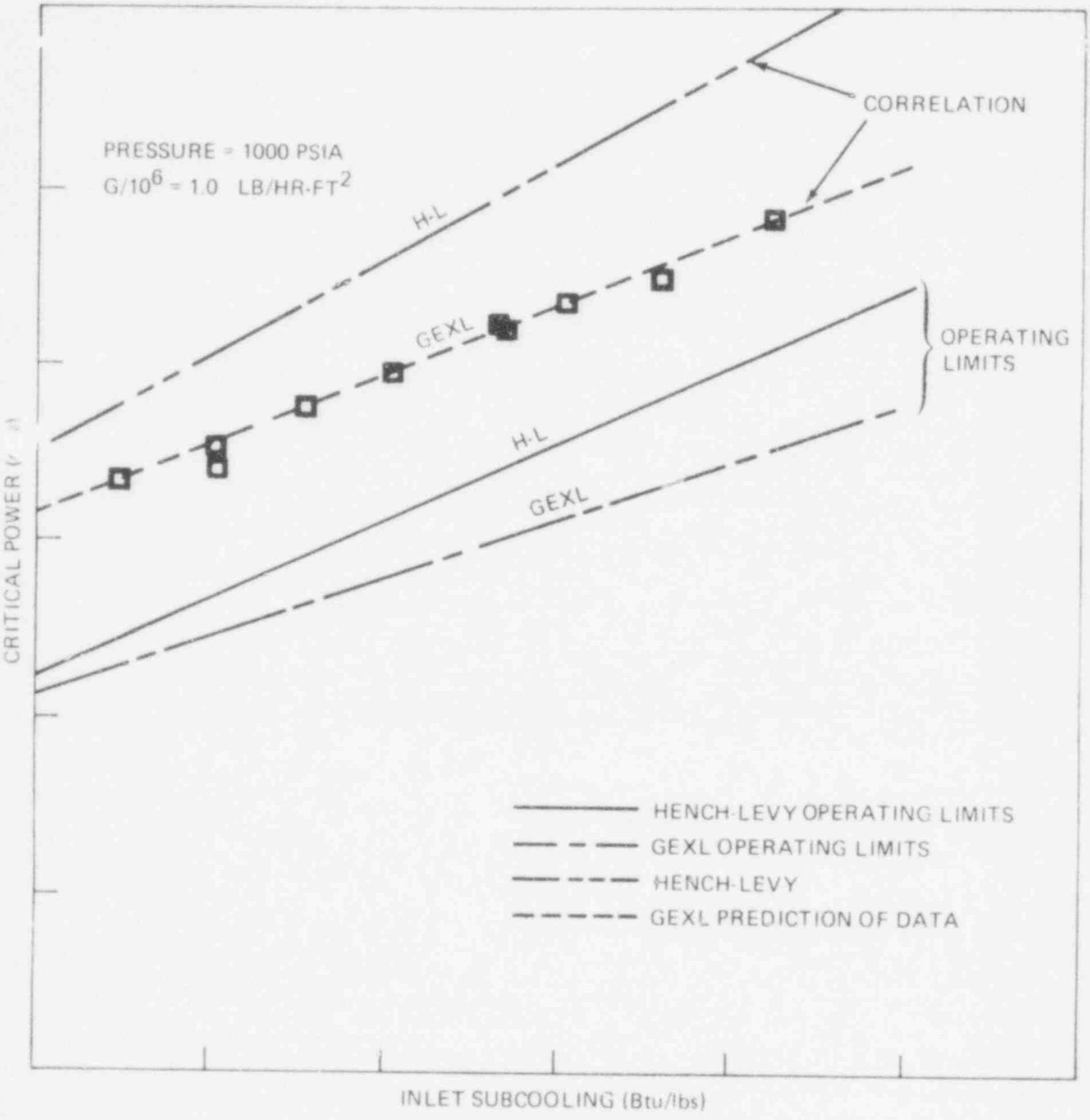

Figure 9 COMPARISON OF HENCH-LEVY AND GEXL WITH 49 ROD ATLAS DATA (1.23 CORNER PEAKING, COSINE)

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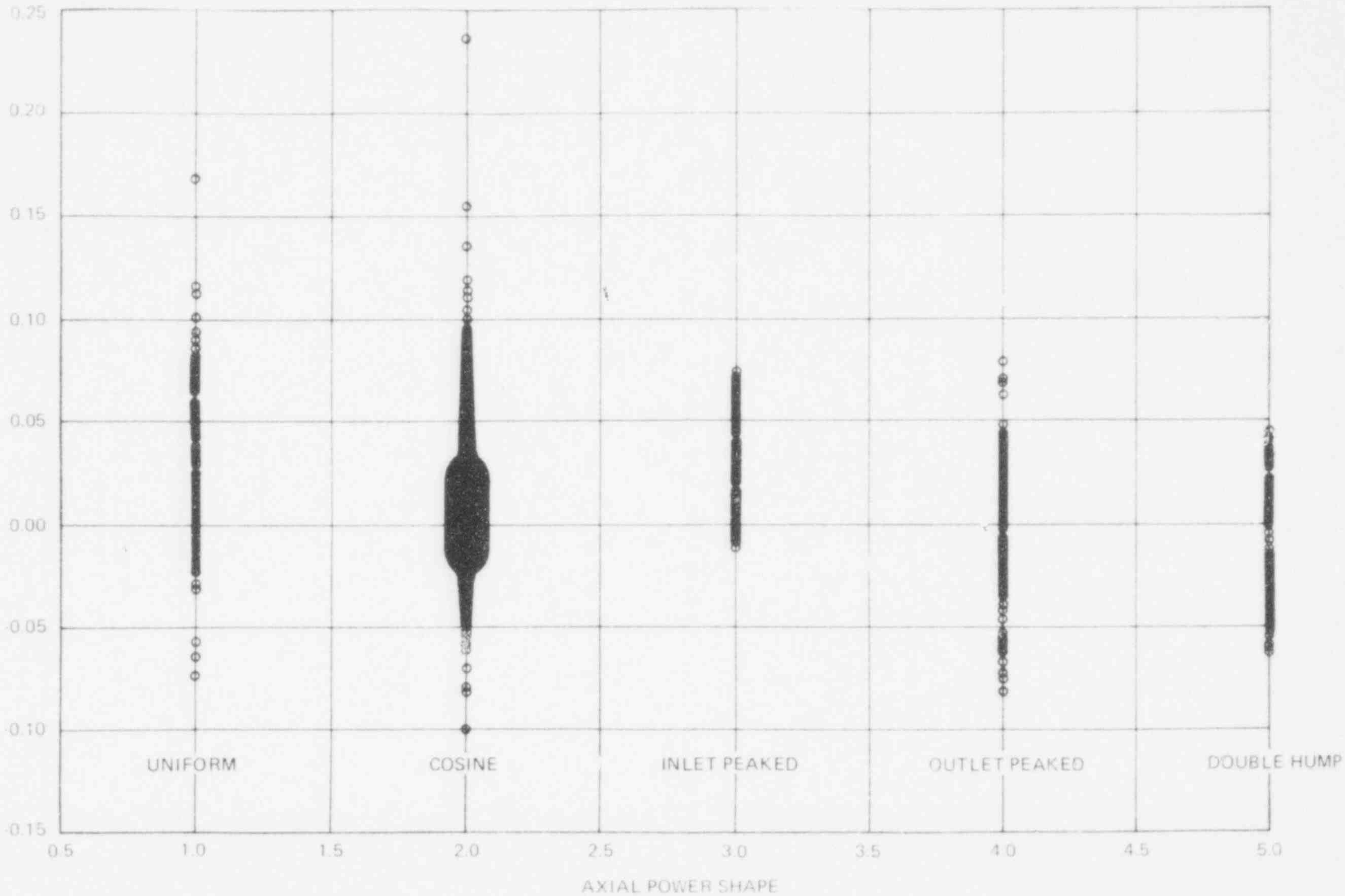

Figure 10 RESIDUAL VS. AXIAL POWER SHAPE FOR GEXL CORRELATION OF ATLAS DATA

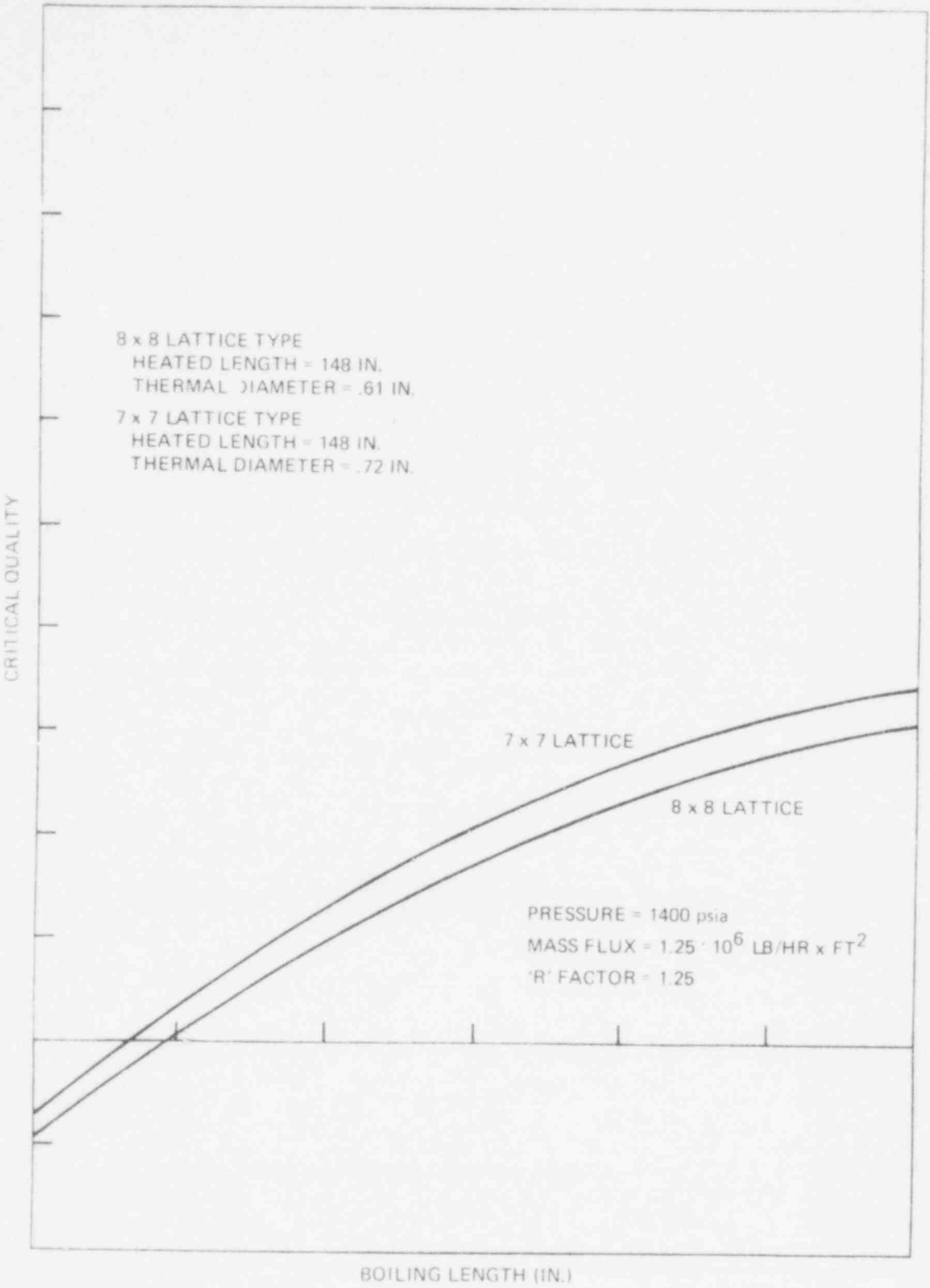

Figure 11 EFFECT OF LATTICE TYPE FOR GEXL CORRELATION

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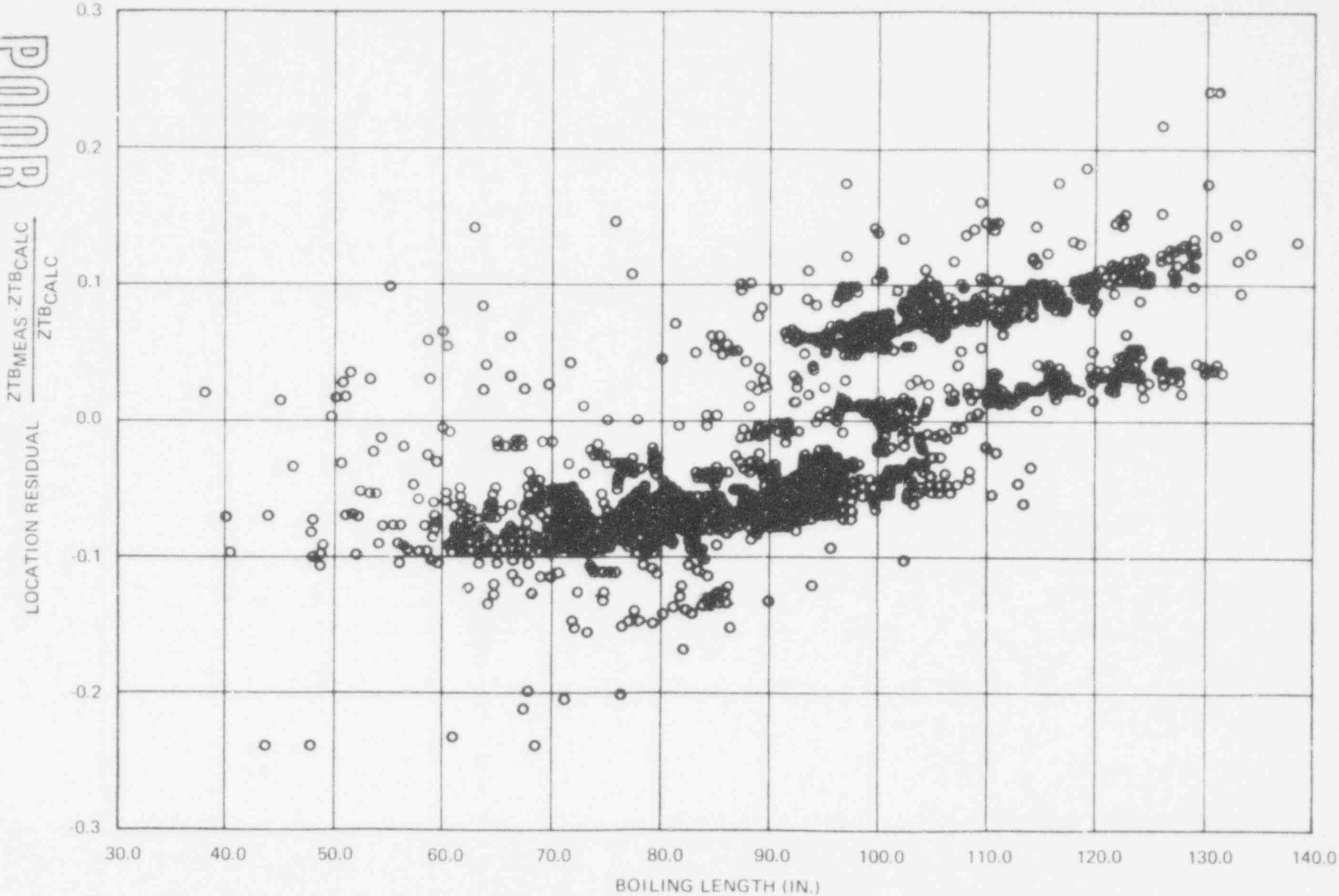

Figure 12 LOCATION RESIDUAL VS. BOILING LENGTH FOR GEXL CORRELATION OF ATLAS DATA

718 178

718 179

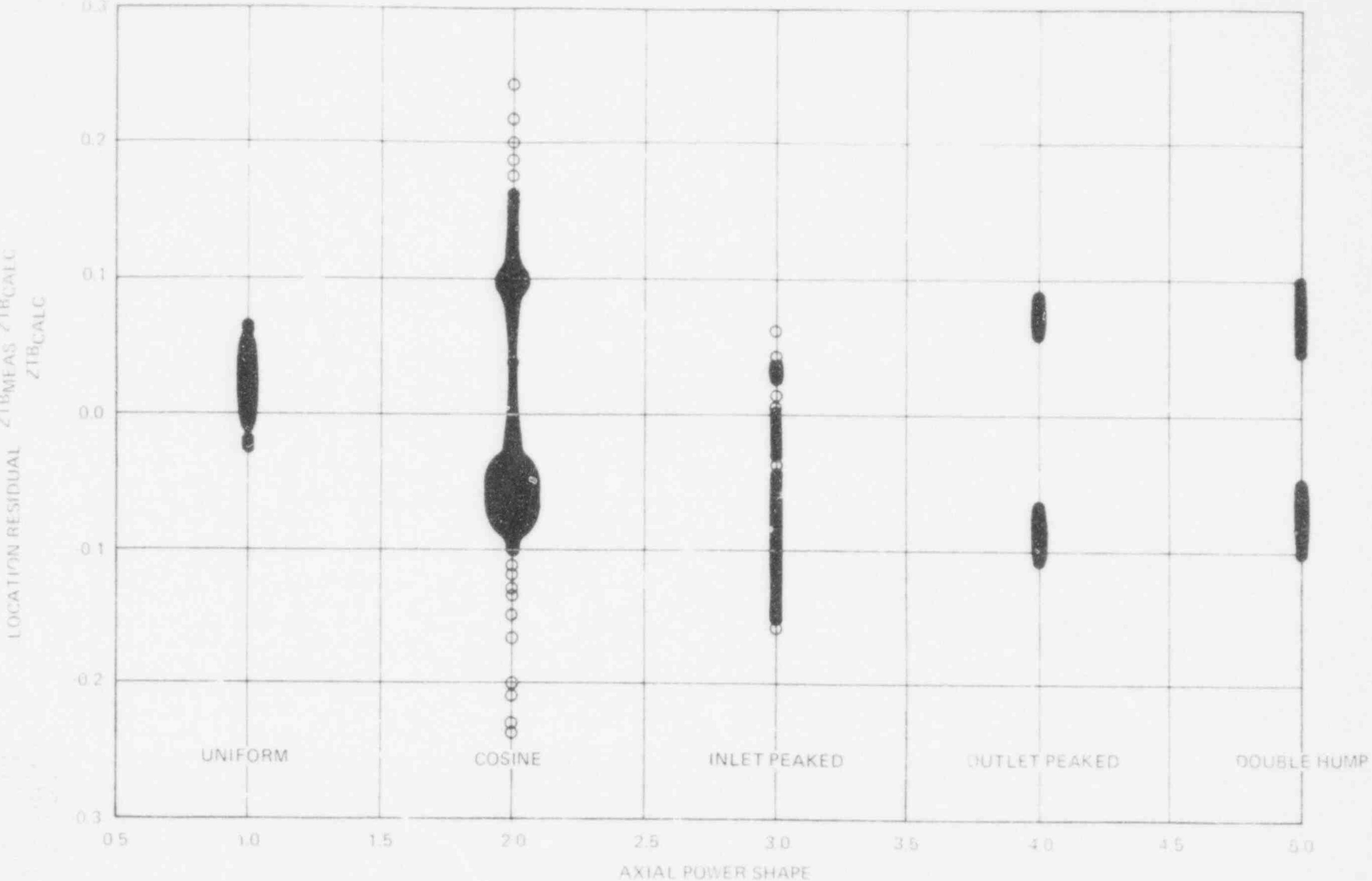

Figure 13 LOCATION RESIDUAL VS. AXIAL POWER SHAPE FOR GEXL CORRELATION OF ATLAS DATA

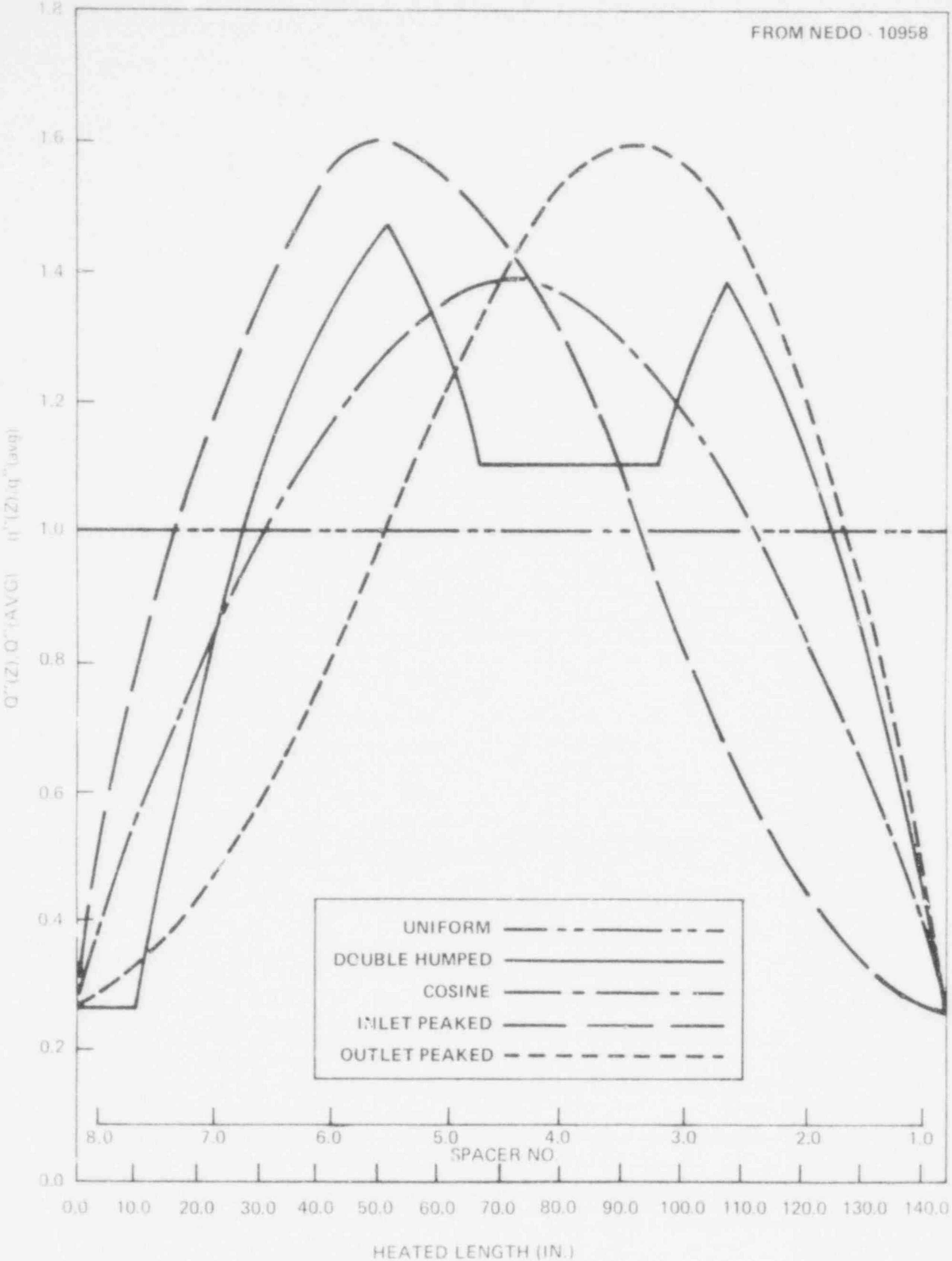

Figure 14 AXIAL FLUX SHAPES FOR 7 X 7 LATTICE

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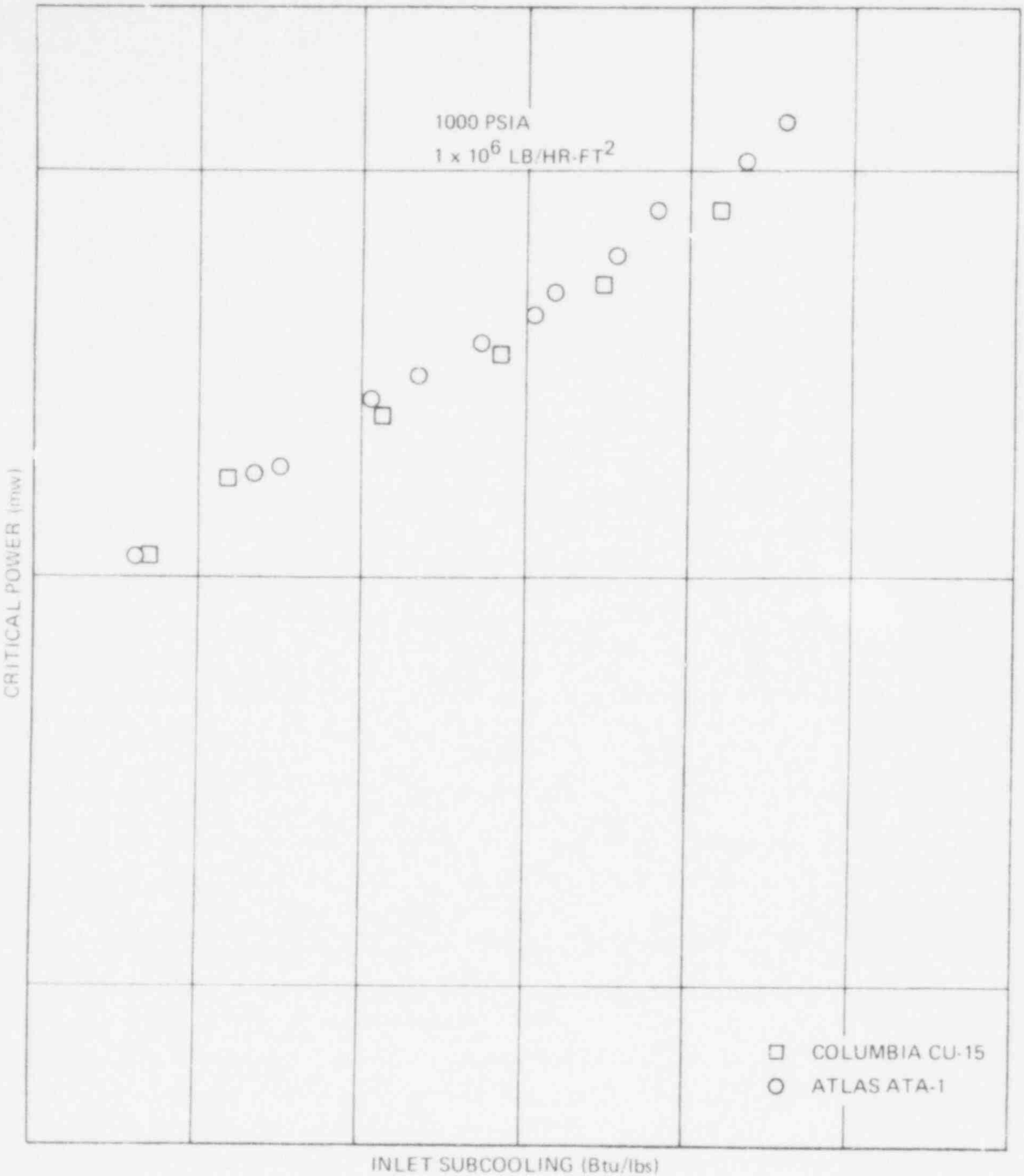

Figure 15 ATLAS VS. COLUMBIA DATA

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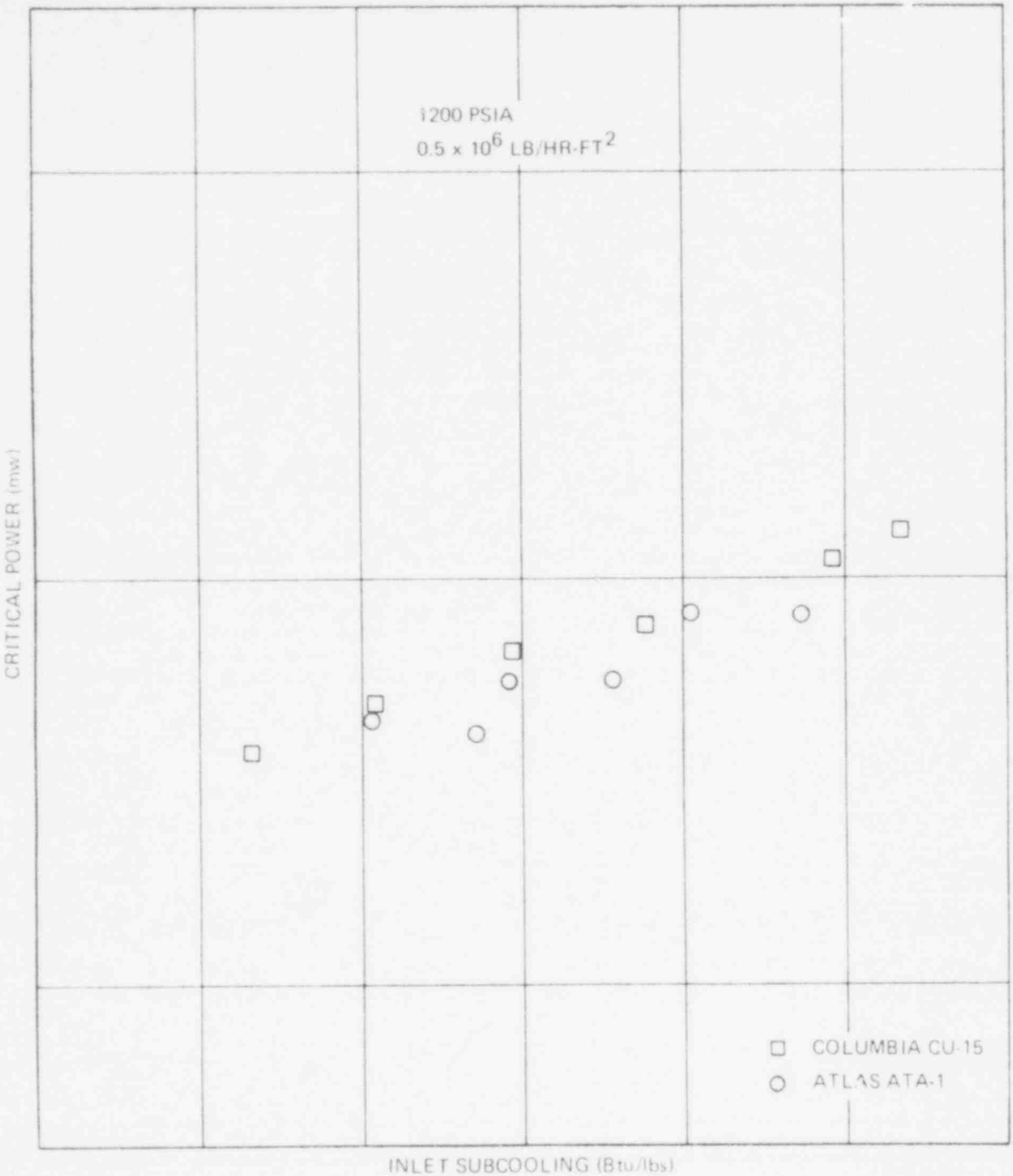

Figure 16 ATLAS VS. COLUMBIA DATA

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Figure 17 ATLAS VS. COLUMBIA DATA

718 183

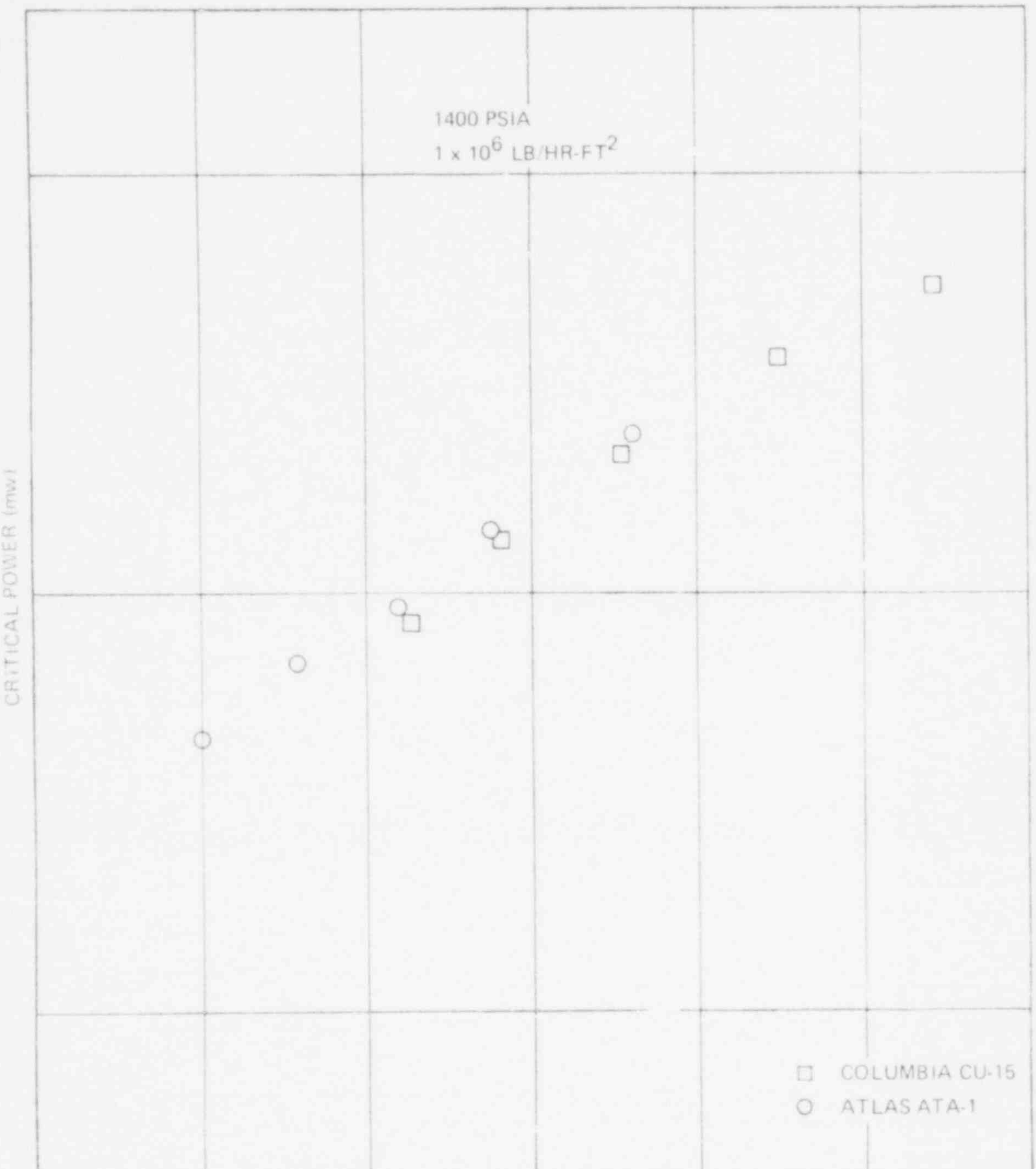

Figure 18 ATLAS VS. COLUMBIA DATA

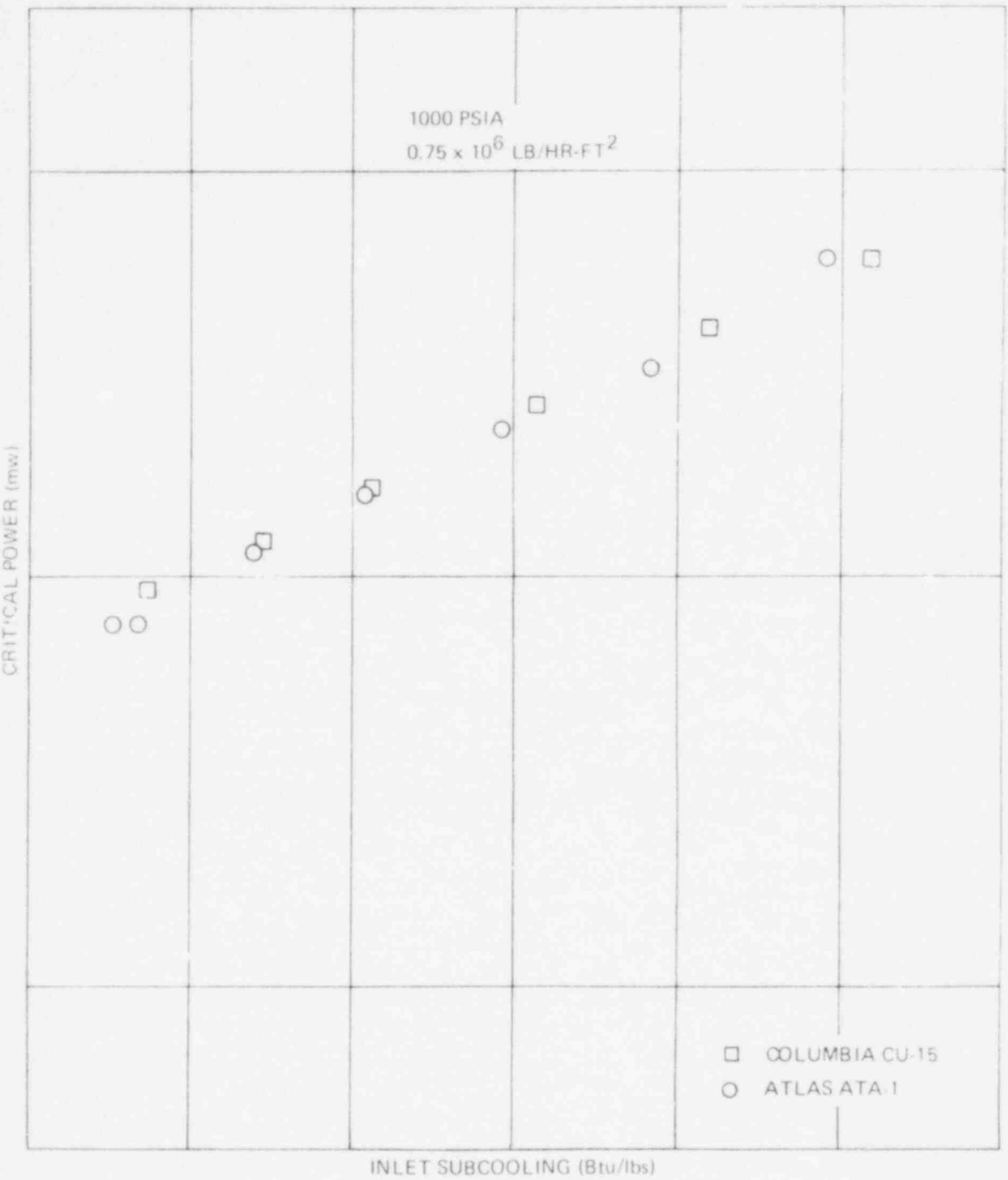

Figure 19 ATLAS VS COLUMBIA DATA

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APPENDIX F

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

March 14, 1975

Honorable William A. Anders
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT (GESSAR-238)

Dear Mr. Anders:

At its 179th Meeting, March 6-8, 1975, the Advisory Committee on Reactor Safeguards completed a review of the General Electric Standard Safety Analysis Report (GESSAR). GESSAR-238 provides the safety information for a reference system consisting of a single BWR-6/Mark III nuclear system, with a rated core thermal power of 3579 MW(t), and of the associated systems including the reactor building (the shield building and containment), fuel building, auxiliary building, diesel generator buildings, control building, radwaste building,, and the off-gas system. Subcommittee meetings were held with representatives of the General Electric Company and the Nuclear Regulatory Commission (NRC) Staff on July 1, 1974, and September 11, 1974, in Washington, D. C., on November 9, 1974, in Bloomington, Minnesota, and on January 18, 1975, in Washington, D.C. The Committee also had the benefit of the documents listed below.

Site envelope parameters are included in GESSAR and application of GESSAR will require that specific site evaluations be made to confirm the acceptability of the site within the GESSAR design. The use of GESSAR for multiple reactor units at a single station will also require review of the safety-related components of plant duplication and layout.

Safety-related interfaces between the reference system and the balance of plant are specified in GESSAR. Since the utility-applicant is responsible for instituting the quality assurance programs necessary to assure that all safety-related interfaces have been identified and that all safety-related requirements are being fulfilled, the Committee will review these matters in more detail with the Applicants on a case-by-case basis. The Committee recommends that, during the design, procurement, construction and startup, timely and appropriate interdisciplinary system analyses be carried out to assure complete functional compatibility across each interface for an entire spectrum of anticipated operations and postulated design basis accident conditions.

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The NRC Staff has identified 13 items requiring resolution prior to issuing their Preliminary Design Approval (PDA). The Committee believes that all of these matters should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed regarding the resolution of the following items:

1. Seismic capability of the offgas system.

2. Provisions to satisfy the single-failure criterion for the RHR system.

3. Additional requirements to be imposed if continuous venting of the containment is used.

4. Evaluation of the performance of the emergency core cooling systems using evaluation models meeting the requirements of 10 CFR 50.46, Appendix K.

The latest ACRS reports on nuclear generating stations utilizing the BWR-6/Mark III systems were the December 12, 1974 reports on the Allens Creek Nuclear Generating Station, Units 1 and 2, and the Perry Nuclear Power Plant, Units 1 and 2. In these reports, the ACRS has recommended that the ongoing R&D programs be used to fully resolve issues involving the Mark III containment design prior to completion of the affected portions of the plant. Further, additional generic matters, which include anticipated transients without scram (ATWS) and possible pump overspeed during a loss of coolant accident, should be dealt with appropriately by the NRC Staff. It is expected, that these items will be resolved in a manner satisfactory to the NRC Staff following Preliminary Design Approval (PDA) of GESSAR and prior to Final Design Approval (FDA). During this interim period, the Committee will continue to review these items on a case-by-case basis as well as through other appropriate ACRS Subcommittee meetings and full Committee meetings.

The Committee has not reviewed modifications which are expected to be made in the BWR/6 8x8 fuel. Such modifications and any other proposed changes will be reviewed when the appropriate documentation has been submitted and the improvements sought can be evaluated.

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The introduction of new features in the instrumentation and control systems has been submitted through the specification of functional designs and design criteria which the NRC Staff has found to be adequate for the PDA. As in previous reports on related matters the Committee recommends that the NRC Staff determine the necessary environmental and reliability tests, including <u>in situ</u> tests where desirable for qualification of the new systems. In another matter relating to a periodic testing provision, the General Electric Company has committed to a study of the improvement of the testability of the automatic depressurization system. On all these issues involving instrumentation and control, the Committee will use the case-by-case basis to ascertain progress of the work until the GESSAR design has progressed to the stage where Final Design Approval is achieved.

The Committee will need to review the development and proof testing of the fast scram system, and the implementation of the proposed Reactor Manual Control System along with the provisions for ganged rod withdrawal.

The Committee believes that the General Electric Company and the NRC Staff should continue to review GESSAR for design changes that would further improve industrial security features.

The GESSAR design should include provisions which anticipate the maintenance, inspection, and operational needs of the plant throughout its service life, including cleaning and decontamination of the primary coolant system, and eventual decommissioning. In particular, the Committee believes that the NRC Staff and the General Electric Company should review methods and procedures for removing accumulations of radioactive contamination whereby maintenance and inspection programs can be more effectively and safely carried out.

The Committee believes that methods that seek to develop reference systems through standardization and through replication need to be coupled with ongoing programs that will permit changes which improve safety and which, when justified, would be implemented in a timely manner. Use of reference systems should lead to more efficient and effective licensing reviews. Programs such as GESSAR will contribute to this process. A transition period will be required in which the Committee would still give considerable attention to the items noted, on a case-by-case basis.

The Committee believes that, subject to the above comments and to successful completion of the R&D prgrams, GESSAR-238 can be successfully engineered to serve as a reference system.

Sincerely yours,

WKerr

William Kerr
Chairman

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References attached

References

1. BWR/6 Standard Safety Analysis Report, Volume 1 through 7.

2. Amendments 1 through 28 to the Standard Safety Analysis Report.

3. General Electric Company letters and reports:

   a. July 31, 1973 letter forwarding proprietary information in support of the information made public in the safety analysis report.

   b. August 31, 1973 letter forwarding proprietary fuel data.

   c. September 28, 1973 letter forwarding proprietary information regarding core power distribution.

   d. December 28, 1973 letter regarding interfaces and electrical systems.

   e. November 6, 1974 letter regarding physics verification and number of safety/relief valves.

   f. February 19, 1974 letter regarding ATWS.

4. AEC/NRC Staff letters and reports:

   a. October 11, 1974 draft Safety Evaluation Report.

   b. November 13, 1974 Safety Evaluation Report.

   c. December 7, 1974 Supplement No. 1 to the Safety Evaluaion Report.

   d. January 30, 1975 letter regarding reevaluation of the high pressure drywell test.

   e. February 21, 1975 Supplement No. 2 to the Safety Evaluation Report.

   f. March 4, 1975 Supplement No. 3 to the Safety Evaluation Report.

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1. NRC Staff letter to J. C. McKinley dated February 25, 1975 regarding ACRS consultant comments on BWR reactor protection system response time and performance.

2. Letter from E.P. Epler to R.F. Fraley dated February 2, 1975 regarding the review of GESSAR.

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APPENDIX G

BIBLIOGRAPHY

This appendix includes those documents referenced within or used to prepare this SER for the General Electric Standard Safety Analysis Report (GESSAR). Those documents incorporated by the applicant as referenced in GESSAR are not repeated in this appendix. SERs for other plants mentioned in the text of the GESSAR SER are available upon request from the NRC. (Fee required)

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1. General Electric Standard Safety Analysis Report (GESSAR), General Electric Company, Amendments 1 through 36.

CHAPTER 2

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CHAPTER 3

4. ASME Boiler and Pressure Vessel Code, 1971 Edition, Section VIII, Nuclear Power Plant Components.

5. ASME Boiler and Pressure Vessel Code, 1971 Edition, Section VIII, Division 1, Pressure Vessels.

6. API Standard 620, Fourth Edition, February 197?, Recommended Rules for Design and Construction of Large, Welded, Low Pressure Storage Tanks.

7. API Standard 650, Third Edition, July 1966, Welded Steel Tanks for Oil Storage.

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9. American Concrete Institute, "Building Code Requirements for Reinforced Concrete (ACI 318-1971), P. O. Box 4754, Redford Station, Detroit, Michigan 48219.

10. American Institute of Steel Construction, "Specification for Design, Fabrication & Erection of Structural Steel for Buildings," 101 Park Avenue, New York, N. Y., 10017, Sixth Edition, 1969.

11. U. S. A. Standard B 16.5-1968, "Steel Pipe Flanges and Flanged Fittings," The American Society of Mechanical Engineers, New York, 1968.

12. U. S. A. Standard B 16.9-1964, "Wrought Steel Buttwelding Fittings," The American Society of Mechanical Engineers, New York, 1964.

13. U. S. A. Standard B 31.1.0-1967, "Power Piping," The American Society of Mechanical Engineers, New York, 1967.

14. U. S. A. Standard B 31.7-1969, "Nuclear Power Piping," The American Society of Mechanical Engineers, New York, 1969.

15. U. S. A. Standard B 36.10-1959, "Wrought-Steel and Wrought-Iron Pipe," The American Society of Mechanical Engineers, New York, 1959.

16. MSS Standard Practice SP-58, "Pipe Hangers and Supports-Materials and Design," Manufacturers Standardization Society, Arlington, Va., 1959.

17. MSS Standard Practice SP-66, "Pressure-Temperature Ratings for Steel Butt-Welding End Valves," Manufacturers Standardization Society, New York, 1959.

18. ANSI N176 Draft, "Design Basis for Protection of Nuclear Power Plants Against the Effect of Postulated Pipe Rupture," April 5, 1974.

19. IEEE Draft Standard 344, "IEEE Guide for Seismic Qualification of Class IE Electric Equipment for Nuclear Power Generating Stations," Revision 3, February 15, 1974, American National Standard Institute N 41.7.

CHAPTER 4

20. "Technical Report on Densification of Light Water Reactor Fuels," Regulatory Staff, U. S. Atomic Energy Commission, November 14, 1972.

21. GEGAP-III, "A Model for the Prediction of Pellet Clad Thermal Conductance in BWR Fuel Rods," NEDE-20181, December 3, 1973. Supplement 1, (Proprietary).

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23. Tong, L. S., "Boiling Heat Transfer and Two Phase Flow," John Wiley & Sons, Inc., New York, 1967.

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CHAPTER 7

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32. IEEE Standard 308-1971 - "Class IE Electric Systems for Nuclear Power Generating Stations."

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38. IEEE Standard 379-1972 - "IEEE Trial-Use for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems."

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CHAPTER 8

40. IEEE Standard 387-1972 - "IEEE Trail-Use Standard: Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations."

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APPENDIX B

Docket No. STN 50-447

DEC 22 1975

General Electric Company
ATTN: Mr. Ivan F. Stuart, Manager
Safety and Licensing Section
Nuclear Energy Division
175 Curtner Avenue
San Jose, California 95114

Gentlemen:

Based on the Nuclear Regulatory Commission's (NRC) staff and the Advisory Committee on Reactor Safeguards' review and evaluation of the preliminary design of the nuclear island portion of a BWR/6-Mark III boiling water type nuclear power plant as described in your General Electric Standard Safety Analysis Report (GESSAR-238), we have issued a Safety Evaluation Report (SER), NUREG-75/110, dated December 1975, and Preliminary Design Approval No. PDA-1, dated December 22, 1975. The SER integrates and updates the SER dated November 1974, and Supplements 1, 2 and 3 thereto, dated December 1974, February 1975, and March 1975, respectively.

In general, changes to the GESSAR-238 nuclear island design will be accepted for staff review if the NRC staff can conclude that the changes will result in important improvements in plant safety or important improvements in plant performance or design without adversely affecting the health and safety of the public. The number of changes should be held to a minimum. We plan to amend WASH-1341 to clarify the NRC staff procedures that will be followed for acceptance and subsequent approval of future design changes that may be necessary in the GESSAR-238 design.

The Commission has been informed of the issuance of PDA-1 by the staff and will be kept fully informed of any changes in PDA-1 in the future.

Copies of the Safety Evaluation Report, Preliminary Design Approval No. PDA-1 and a related notice, which is being forwarded to the Office

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General Electric Company DEC 2 2 1975

of the Federal Register for filing and publication, are enclosed for your information and use.

Sincerely,

Roger S. Boyd, Acting Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Enclosures:
1. Safety Evaluation Report dtd December 1975
2. Preliminary Design Approval No. PDA-1 dtd 12/22/75 w/Attachment A - Conditions of PDA-1
3. Federal Register Notice

cc: Mr. W. Gilbert, Manager
Safety and Standards
General Electric Company
175 Curtner Avenue
San Jose, California 95114

Mr. L. Gifford, Manager
Regulatory Operations Unit
General Electric Company
4720 Montgomery Land
Bethesda, Maryland 20014

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

GENERAL ELECTRIC COMPANY

DOCKET NO. STN 50-447

GESSAR-238 NUCLEAR ISLAND STANDARD DESIGN

PRELIMINARY DESIGN APPROVAL (PDA)

Preliminary Design Approval No. PDA-1

(1) The General Electric Company has submitted to the Nuclear Regulatory Commission's (NRC) Staff for its review a proposed standardized preliminary design for major portions of a nuclear power reactor of the type described in 10 CFR 50.22. The preliminary design is described in the General Electric Standard Safety Analysis Report, GESSAR-238 including Amendments 1 through 39 thereto, and the design information incorporated by letters from the General Electric Company dated November 21, 1975. (from W. Gilbert to NRC), and December 16, 1975 (from I. Stuart to NRC), (hereinafter collectively referred to as GESSAR-238).

(2) GESSAR-238 contains preliminary design information in accordance with 10 CFR Part 50 Appendix O, paragraph 3, for the nuclear island portion of a BWR-6/Mark III boiling water reactor nuclear power plant, which encompasses the nuclear steam supply system (NSSS), engineered safety feature systems, containment structure, auxiliary building, control building, radwaste building, fuel handling building, and related systems and structures. The GESSAR-238 reference design is designed to operate at a core thermal power level of 3579 megawatts (1220 megawatts electrical, nominal net).

(3) The GESSAR-238 reference design has been reviewed by the NRC Staff and by the Advisory Committee on Reactor Safeguards (ACRS). The results of NRC Staff evaluation of the GESSAR-238 reference design are presented in the Safety Evaluation Report (SER), NUREG-75/110, dated December 1975, which integrates and updates the information contained in the earlier Safety Evaluation Report dated November 1974 and three Supplements thereto dated December 1974, February 1975, and March 1975. The ACRS comments, including identification of items which the ACRS will review or a case-by-case basis, are set forth in its letter of March 14, 1975 (Appendix F of the SER).

(4) Based on its review, and the findings set forth in Section 19 of the SER, the NRC Staff has concluded that subject to the conditions set forth herein, the information provided in GESSAR-238 with

respect to the major portions of the preliminary design encompassed by GESSAR 238, as described in paragraph 2 above, complies with the requirements of 10 CFR Part 50, Appendix O and is acceptable for incorporation by reference in applications for construction permits. In accordance with 10 CFR Part 50, Appendix O, subject to the conditions set forth herein and in Appendix O, the approved design shall be utilized and relied upon by the Staff and the ACRS in their review of any individual facility license application which incorporates by reference the approved design, unless there exists significant new information which substantially affects the determination set forth in this Preliminary Design Approval or other good cause.

(5) GESSAR-238 and the GESSAR-238 reference design are acceptable for use as a reference design for construction permit applications:

(a) for facilities to be located at sites whose characteristics conform to the envelope of site parameters postulated for the preliminary design of the GESSAR-238 reference design, which are set forth in GESSAR-238; provided that (b) the design of portions of the balance of plant which interface with the approved design, shall conform to the safety-related interface requirements set forth in GESSAR-238.

(6) (a) GESSAR-238 and the GESSAR-238 reference design are acceptable for use as a reference design for construction permit applications, on the condition that the four design features specified in Attachment A hereto are modified to conform to the NRC Staff design requirements described in the NRC Staff SER in the section specified in Attachment A.

(b) To the extent that the General Electric Company subsequently modifies the GESSAR-238 reference design and modifies GESSAR-238 to conform to the Staff requirements, this condition shall be modified accordingly. If all four design features are modified to conform to the Staff requirements, this condition will be deleted.

(7) (a) This Preliminary Design Approval is subject to the satisfactory and timely completion of the development and verification test programs described in Table 1-2 of the Safety Evaluation Report and is subject to the satisfactory and timely submission of further information concerning the items listed in Table 1-3, "Post-PDA Items", of the SER.

(b) The status of such further information shall be addressed in applications for construction permits referencing GESSAR-238 and will be considered in the Staff review of such applications.

(8) This Preliminary Design Approval and all applications for construction permits incorporating it by reference, are subject to all applicable provisions of the Atomic Energy Act, as amended, and the rules and regulations and Orders of the Commission now or hereafter in effect.

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(9) This Preliminary Design Approval does not constitute a commitment to issue a permit or license or in any way affect the authority of the Commission, Atomic Safety and Licensing Appeal Board, Atomic Safety and Licensing Boards and other presiding officers in any proceeding under Subpart G of 10 CFR Part 2.

(10) This Preliminary Design Approval is effective as of its date of issuance and shall expire on December 22, 1978, unless earlier superseded by the issuance of a Final Design Approval for the GESSAR-238 reference design, or unless extended by the NRC staff. The expiration of this PDA on December 22, 1978, shall not affect the use of this PDA for reference in any construction permit application docketed prior to such date.

FOR THE NUCLEAR REGULATORY COMMISSION STAF

Original signed by:
Roger S. Boyd

Roger S. Boyd, Acting Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Attachment A - Conditions of PDA-1

Date of Issuance: DEC 22 1975

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ATTACHMENT A

GESSAR-238 NUCLEAR ISLAND STANDARD DESIGN

DOCKET NO. STN 50-447

CONDITIONS OF PDA-1 WHICH INCORPORATE NUCLEAR REGULATORY STAFF POSITIONS THAT IDENTIFY REQUIREMENTS DIFFERENT FROM THOSE NOTED IN GESSAR-238 NUCLEAR ISLAND STANDARD SAFETY ANALYSIS REPORT ARE LISTED BELOW

Condition No.	Design Feature	GESSAR-238 Section Which Describes Design Feature	SER Section Where Staff Positions are Discussed
1	Tornado Missile	3.5	3.5
2	Main Steam Line Isolation Valve Leakage Control System	9.3.6	9.3.1 & 15.3.1
3	Containment Purge	11.3	6.2.4
4	Mark III Containment Dynamic Load Criteria	6.2	6.2.1.9

Dated: DEC 2 2 1975

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September 1976

APPENDIX C

SUPPLEMENT NO. 1

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

GENERAL ELECTRIC

STANDARD SAFETY ANALYSIS REPORT

(GESSAR-238 NUCLEAR ISLAND)

DOCKET NO. STN 50-447

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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 Introduction

On December 22, 1975, the United States Nuclear Regulatory Commission issued the Safety Evaluation Report (NUREG 75/110) and the Preliminary Design Approval for the General Electric Standard Safety Analysis Report (GESSAR-238 Nuclear Island) design (Docket Number STN 50-447). In our Safety Evaluation Report on the GESSAR-238 Nuclear Island design, we identified nineteen items (in Table 1-3) that we indicated would require continued review after the issuance of the GESSAR-238 Nuclear Island Preliminary Design Approval and four staff design requirements imposed as conditions to the GESSAR-238 Nuclear Island Preliminary Design Approval.

Since the issuance of the Safety Evaluation Report, the General Electric Company has submitted five amendments (Amendments 40, 41, 42, 43 and 44) to the GESSAR-238 Nuclear Island Safety Analysis Report. The purpose of Supplement Number 1 to the Safety Evaluation Report is to update the Safety Evaluation Report by providing the staff's evaluation of the additional information received since the issuance of the Safety Evaluation Report. Each of the following sections in this supplement is numbered the same as the section of the Safety Evaluation Report that is being updated.

Appendix A to this supplement is a continuation of the chronology section of the Safety Evaluation Report.

1.7 Facility Modifications as a Result of Regulatory Staff Review

1.7.2 Facility Modifications Required by the Staff

In the Safety Evaluation Report, we identified four staff requirements which were imposed as conditions to the Preliminary Design Approval for the GESSAR-238 Nuclear Island design. Since that time, the General Electric Company has provided us with acceptable commitments to the staff positions on two of these conditions, and an acceptable resolution to a third condition.

The status of the four conditions and the section in this supplement where each condition is discussed are:

(1) Tornado missile velocities - Resolved (See Section 3.5).

(2) Containment pool dynamics - Under review (See Section 6.2.1.9).

(3) Continuous purging of containment - Resolved (See Sections 6.2.4 and 11.3).

(4) Main steam isolation valve leakage control system - Resolved (See Sections 6.2.3 and 9.3.1).

1.8 Requirements for Future Technical Information

1.8.2 Post-PDA Review

Table 1-3 of the Safety Evaluation Report for the GESSAR-238 Nuclear Island contained a list of items which we planned to continue to review after the issuance of the Preliminary Design Approval. Table 1-3 contained several items which are not normally reviewed at the construction permit stage of review along with those items which would normally be reviewed. However, at that time, no distinction was made between these two types of items.

Since the issuance of the Safety Evaluation Report, we have been working with the General Electric Company in an effort to complete the review of these nineteen items. During the course of our review, it became apparent that the list presented in Table 1-3 was inappropriate to identify the work being done to satisfy the requirements for the issuance of a construction permit to a referencing plant since:

(1) Table 1-3 identifies several items which we plan to continue to review on the GESSAR-238 Nuclear Island application although sufficient information exists at the present time for the issuance of a construction permit.

(2) Item 14 in Table 1-3 is in actuality a group of issues concerning a particular area of review. In order to clearly understand the extent of the outstanding safety matters, item 14 should be separated into individual items requiring resolution.

For improved ease of use, we have decided to separate Table 1-3 in the Safety Evaluation Report into two different lists. The first, Table 1-1, identifies the same items as Table 1-3 of the Safety Evaluation Report and shows the present status of each item. The status for thirteen of the items in Table 1-1 is given as "Acceptable for construction permit stage of review" and a section of this supplement is referenced where a discussion is provided on whether the review is completed or whether resolution of the item is not required for the issuance of a construction permit. For the remaining six items, the current review status is listed.

In Table 1-2, we have provided a detailed list of those specific items which remain outstanding, as of the date of this supplement, and which require resolution prior to the issuance of a construction permit to a referencing plant. Following each item, a reference to a section in this supplement is provided where the item is discussed in detail.

TABLE 1-1
POST-PDA ITEMS

	Item	Discussed in Section	Status
1.	Leakage characteristics of primary coolant pump seals	3.2.1	Analysis of consequences provided July 1976; under staff review
2.	Description of combined effects of safe shutdown earthquake and steam line break	3.9.1.4	Acceptable for construction permit stage of review
3.	a. List of specific equipment to be seismically qualified b. The qualification procedures to be used	3.10	Acceptable for construction permit stage of review
4.	a. List of specific equipment to be environmentally qualified b. The qualification procedures to be used	3.11	Acceptable for construction permit stage of review
5.	Preliminary design of drywell penetrations	3.11.1	Acceptable for construction permit stage of review
6.	Procedures and methods to be used to qualify the shield building, containment and drywell penetrations	3.11.1	Acceptable for construction permit stage of reveiw
7.	Implementation methods of separation criteria for safety-related electrical equipment	3.12	Staff review to be accomplished coincident with review of item 14
8.	Detailed information on: a. Lattice physics methods. b. Boiling water reactor simulation code. c. Verification of core calculational methods.	4.3.7	Acceptable for construction permit stage of review
9.	Confirming data from large scale Mark III tests for short-term containment response.	6.2.1.6	Acceptable for construction permit stage of review

TABLE 1-1 (Continued)

Item	Discussed in Section	Status
10. Assumptions used to size containment vacuum breakers	6.2.1.5	Response to staff concerns provided July 1976; under staff review
11. Environmental design criteria for isolation valves and other safety-related equipment in the drywell	6.2.4	Acceptable for construction permit stage of review
12. Address question 6.125 (manual operator action on emergency core cooling systems following a loss-of-coolant accident)	6.3.1	Acceptable for construction permit stage of review
13. Proprietary version of 8 x 8 zirconium spray cooling test	6.3.1	Acceptable for construction permit stage of review
14. New modified instrumentation and control systems preliminary design review for:	7.0	Areas a through f are acceptable for the construction permit stage of review with the exception of sub-items 3 through 12 in Table 1-2; The July 1976 submittal on these items is under review
a. Reactor trip system		
b. Engineered safety features actuation system		
c. Safe shutdown system		
d. Safety related display instrumentation		
e. All other instrumentation required for safety		
f. Control systems - reactor manual control system, recirculation flow control, gaseous and liquid radwaste control, feedwater flow control and interaction between safety and non-safety control systems		
15. Scope of onsite electrical system	8.0	July 1976 submittal under review
16. Review of high pressure core spray power system	8.0	Being reviewed separately as Topical Report NEDO-10905, "High Pressure Core Spray Power Supply Unit"

TABLE 1-1 (Continued)

	Item	Discussed in Section	Status
17.	Fast scram system and Chapter 15 transients	4.2.3 and 15.2	Acceptable for construction permit stage of review
18.	Preliminary design of systems to control bypass containment leakage	6.2.3	Acceptable for construction permit stage of review
19.	Anticipated transients without scram	15.4	Acceptable for construction permit stage of review

TABLE 1-2

SAFETY MATTERS THAT REMAIN TO BE RESOLVED PRIOR TO A DECISION ON THE ISSUANCE OF A CONSTRUCTION PERMIT FOR A REFERENCING PLANT

(The number in parentheses following each item denotes the item in Table 1-1 under which this area was previously classified)

Item	Discussed in Section
1. Leakage characteristics of primary coolant pump seals (1)	3.2.1
2. Containment vacuum breakers (10)	6.2.1.5
3. Automatic depressurization system testability (14)	7.3.2.2
4. High water level trip (14)	7.2
5. Two pump trip (14)	7.6.3
6. Low pressure core spray and low pressure (14) coolant injection system interlocks (14)	7.3.2.3
7. Electrical review of essential service water system (7, 14)	7.3.4
8. Electrical review of flammability control system (7, 14)	7.3.5
9. Electrical review of standby gas treatment system (7, 14)	7.3.6
10. Electrical review of suppression pool makeup system (7, 14)	7.3.7
11. Electrical review of containment spray system (7, 14)	7.3.8
12. Review of Nuclear Island/balance-of-plant electrical (14) interfaces	7.8
13. Review of onsite power systems (15)	8.0

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1.11 Conclusion

Based on our review of the information submitted by the General Electric Company since the issuance of the Preliminary Design Approval, we conclude that, except for those items identified in Table 1-2 of this supplement, the General Electric Company has supplied sufficient information on the post-PDA items identified in Table 1-3 of the Safety Evaluation Report to provide a suitable basis for the issuance of a construction permit to a referencing plant. In addition, we conclude that the commitments provided by the General Electric Company, in Amendment 43 to the GESSAR-238 Nuclear Island Safety Analysis Report, provide an acceptable resolution to the Preliminary Design Approval conuitions with the exception of Condition Two identified in Section 1.7.2.

We will address the resolution of Condition Two to the Preliminary Design Approval and those items in Table 1-2 of this supplement in a future supplement to the Safety Evaluation Report prior to the issuance of a construction permit to a referencing plant.

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3.0 DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

3.2 Classification of Structures, Systems and Components

3.2.1 Seismic Classification

In Section 3.2.1 of the Safety Evaluation Report, we provided a discussion on two systems which we felt did not have the proper seismic classification. These were the primary coolant pump seals' cooling water system and the offgas system. We stated that we would continue to review the leakage characteristics of the primary coolant through the primary coolant pump seals as a result of assumed loss of seal water or cooling water to the primary coolant pump seals. Specifically, we requested the General Electric Company to provide an analysis to demonstrate that the consequences of recirculation pump seal failure are acceptable.

The General Electric Company has since informed us that the consequences of total failure of the recirculation pump seals' cooling water supply would be a gradual deterioration of the seals over a period of several hours finally resulting in total seal failure. They further stated that seal failure would result in a loss of primary coolant at a rate of less than 50 gallons per minute.

We found the General Electric response unacceptable as it was not supported by either tests or analyses. General Electric has since supplied us with the pump vendor's analysis of the consequences of pump seal cooling water failure. We are reviewing the analysis supplied by General Electric and will report on our conclusions in a future supplement to the Safety Evaluation Report.

In the Safety Evaluation Report, we stated that the General Electric Company had proposed to design the offgas system delay tank supports to the seismic design criteria listed in Effluent Treatment Systems Branch Technical Position 11-1, "Design Guidance for Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants." This statement requires further clarification. The General Electric Company, in Amendment 31 to the GESSAR-238 Nuclear Island Safety Analysis Report, modified their application to include the following design criteria: "The support elements, including the skirts, legs and anchor bolting for the charcoal absorber tanks of the offgas system shall be designed as follows:

(1) The fundamental frequency of the charcoal absorber tanks, including the support elements, is greater than 33 Hertz.

(2) The charcoal absorber tanks are mounted on the base mat of the building housing the tanks.

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(3) The charcoal absorber tanks including the support elements are designed with a horizontal static coefficient of 0.15g.

(4) The stress levels in the support elements of the charcoal absorber tanks shall not exceed 1.33 times the allowable stress levels permitted by the AISC Manual of Steel Construction, Seventh Edition, 1970."

Items 1 and 2 of the General Electric Company's design criteria assure that the ground level acceleration resulting from a seismic event will not be magnified by the absorber tank supports since the amplification factor associated with structures having fundamental frequencies exceeding 33 Hertz is unity. Since the static coefficient specified for the absorber tanks (item 3) is compatible to the operating bases earthquake intensity specified for the GESSAR-238 Nuclear Island design, and since item 4 of the General Electric Company's design criteria is identical to the stress criteria specified in Effluent Treatment Systems Branch Technical Position 11-1, we conclude that the General Electric Company's design criteria are as conservative as those specified in Effluent Treatment Systems Branch Technical Position 11-1, and are therefore acceptable. Since the issuance of the Safety Evaluation Report, we have recalculated the offsite dose which could result from a failure of the offgas tank supports (assuming a relative concentration value of 1 x 10^{-3} seconds per cubic meter which is consistent with the relative concentration value used for the 0-2 hour accident analysis calculations) and found it to be less than two rem.

Since the offgas tank supports are being designed with a seismic capability to meet the operating basis earthquake requirements, we would postulate a failure of the offgas tank supports only for seismic events exceeding the intensity of the operating basis earthquake--events which have a low probability of occurrence. Since the probability of the failure of these supports is low, and since the consequences of failure is a relatively small fraction of 10 CFR Part 100 guidelines on whole body doses, we conclude that the intermediate level of seismic design is acceptable for the offgas tank supports.

3.5 Missile Protection Criteria

3.5.3 Tornado Missiles

In the Safety Evaluation Report, we stated that the tornado missile velocity spectrum proposed by the General Electric Company was unacceptable as it was unconservatively low and unsupported by data. We therefore required the General Electric Company to adopt our tornado missile velocity spectrum as a design basis for the GESSAR-238 Nuclear Island design. This requirement was made a condition of the Preliminary Design Approval issued to the General Electric Company.

The General Electric Company has since modified their application with Amendment [illegible] to the GESSAR-238 Nuclear Island Safety Analysis Report, to incorporate our

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position on tornado missile velocities as a design basis for the GESSAR-238 Nuclear Island design. We therefore consider this condition resolved.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

3.9.1.4 Analysis Methods for Loss-of-Coolant Accident Loadings

In the Safety Evaluation Report, we indicated that General Electric was to provide us with a dynamic analysis of the combined effects of a safe shutdown earthquake plus a steam line break on the reactor internals to confirm the structural adequacy of the internals. The results of this analysis will be used to demonstrate the conservatism present in the design loads for the internals. The methods and models for determining these loads have, however, already been submitted and approved. That provides an acceptable basis for the issuance of the Preliminary Design Approval to General Electric and is, therefore, acceptable for use by a referencing applicant in a construction permit application. We will report on the results of the dynamic analysis during our review of the final design.

3.10 Seismic Design of Category I Instrumentation and Electrical Equipment

In the Safety Evaluation Report, we identified the following as open items to be resolved as part of our review following the issuance of the Preliminary Design Approval:

(1) The list of equipment to be seismically qualified.

(2) The qualification procedures to be used.

The General Electric Company has resolved these items in the following manner. It has been agreed between the General Electric Company and the staff that, for the final design stage of review, we will conduct a detailed review of the methods and procedures used in implementing the seismic design criteria and of the scope of the seismic qualification program.

Pursuant to this detailed review:

(1) The General Electric Company has submitted a typical list of equipment which will be seismically qualified.

(2) The General Electric Company will submit, for review and approval, a topical report which will describe the detailed qualification procedures which will be used in implementing the seismic design qualification criteria.

(3) The General Electric Company will submit to an audit, by the staff, of the qualification of selected instrumentation and electrical equipment.

Acceptable implementation of criteria described in the detailed procedures of item 2 will be verified during this audit.

We have concluded that these commitments provide an acceptable basis for the Preliminary Design Approval.

3.11 Environmental Design of Mechanical and Electrical Equipment

In the Safety Evaluation Report, we identified the following as open items to be resolved as part of our review following the issuance of the Preliminary Design Approval:

(1) The list of electrical equipment to be environmentally qualified.

(2) The qualification procedures to be used.

In Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report, the General Electric Company responded to all of our outstanding requests for additional information concerning environmental qualification of electrical equipment. In Tables 3.11.1 through 3.11.4 of GESSAR-238 Nuclear Island Safety Analysis Report, they have provided the expected post-accident thermal and radiological conditions for safety-related equipment within the drywell, containment and auxiliary building.

With regard to the environmental qualification of Class IE equipment, the General Electric Company has committed to conformance to the requirements of the Institute of Electrical and Electronic Engineers Standards 323-1974, "Qualifying Class IE Electrical Equipment for Nuclear Power Generating Stations," and has supplied their qualification test program which describes their proposed method of conforming to this standard.

The program outlined by the General Electric Company includes temperature-humidity testing, seismic testing and life testing (aging) in a sequence which is in conformance with the requirements. The justification for excluding any environmental parameter in the testing will be provided in the qualification report for the specific equipment involved.

The above commitments and outlined program are suitable bases for the development of the more detailed qualification program. We have concluded that this is acceptable for this stage of review and that there is reasonable assurance that General Electric can develop a detailed qualification program acceptable to the staff. We will report on the results of our evaluation of the detailed qualification program at the final design stage of review.

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3.11.1 Electrical Penetrations in the Shield Building, Containment and Drywell Walls

In the Safety Evaluation Report, we identified two items in which the General Electric Company had not provided adequate information for the review to be completed. These were the procedures and methods to be used to qualify the shield building, containment and drywell penetrations, and the preliminary design for the drywell penetrations.

With regard to the qualification of the shield building and containment penetrations, the General Electric Company meets the requirements of the Institute of Electrical and Electronics Engineers Standard 317-1974, "Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," for the containment penetrations and for the electrical considerations associated with the shield building penetrations. We find this acceptable.

With respect to the drywell penetrations, the General Electric Company has provided us with a preliminary design for the drywell penetrations and a description of their proposed qualification program.

The qualification program for the drywell penetrations includes results from previously conducted tests to demonstrate the sealing capability of the potting material used in the penetrations. Once the penetrations are constructed, the drywell structural proof test and the periodic drywell leak tests will provide continued assurance that the penetrations are intact.

Based on the previous discussion, we conclude that the design and testing provisions for the electrical penetration in the shield building, containment and drywell are acceptable.

3.12 Separation Criteria for Safety-Related Mechanical and Electrical Equipment

In the Safety Evaluation Report, we concluded that the General Electric Company's proposed design criteria for the separation of safety-related equipment met our requirements. However, the review of the General Electric Company's implementation of these criteria had to await the submission of the preliminary design of many of the instrumentation systems. The General Electric Company has since submitted this information in Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report.

We have utilized Regulatory Guide 1.75, "Physical Independence of Electrical Systems," in our review of the preliminary designs (elementary wiring diagrams) for the instrumentation and control systems. The specific conclusions for these systems are reported in the applicable portions of Section 7.0 of this supplement.

All the instrumentation and control preliminary designs utilized isolation devices where signals are transmitted between redundant divisions of equipment

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and where signals are transmitted from division equipment to non-divisional equipment. Though isolation devices have been employed in the manner on other plant designs, we required the General Electric Company to distinctly identify the type of devices they intend to use and to define the test program which will be utilized to demonstrate the isolation capabilities of these devices.

The General Electric Company has since provided us with the test program which they intend to use to qualify optical isolators, which is the isolation device they will utilize to isolate signals within their proposed solid state protection system. (An optical isolator is basically a light emitting diode and a light sensitive diode separated by a transparent membrane.) We have reviewed the proposed test program and conclude that it provides a suitable basis for qualifying the isolation devices. We will report on the results of this test program for isolation devices at the final design stage of review.

We have not completed our review of all the systems which are outside the GESSAR-238 Nuclear Steam Supply System scope of design yet inside the GESSAR-238 Nuclear Island scope of design. The systems involved are the essential service water system, the standby gas treatment system, the flammability control system, the containment spray system, the suppression pool makeup system and the onsite power system. We will report on our conclusions concerning the physical independence of these systems in the applicable portions of Section 7.0 of a future supplement to the Safety Evaluation Report.

We conclude that the electrical, instrumentation and control systems within the GESSAR-238 Nuclear Steam Supply System portion of the GESSAR-238 Nuclear Island design have adequate separation of redundant safety-related equipment.

4.0 REACTOR

4.2 Mechanical Design

4.2.3 Fast Scram

In the Safety Evaluation Report we stated that we would require General Electric to provide a detailed description of (and commitment to) a test program for the fast scram system.

The General Electric Company has committed to a test program for the fast scram system to verify the design bases used in the analyses in Section 15.0 of the GESSAR-238 Nuclear Island Safety Analysis Report. The test program will consist of four phases:

(1) Development Testing - This phase of the program is complete and was conducted to identify and optimize those performance objectives for the fast scram system.

(2) Design Acceptance Testing - This phase of the program is scheduled for completion in 1976 and is intended to verify the final design using, as close as possible, actual production control rod drive components.

(3) Production Qualification Testing - This phase of the program is scheduled for completion in 1977 and is intended to establish a firm statistical base on control rod drive performance by testing preproduction control rod drives to the design requirements.

(4) Production Verification Testing - This phase of the program is scheduled for completion in 1978 and is intended to verify that the procedures and techniques used in manufacturing produce control rod drives that meet all of the requirements.

We have reviewed the test program and conclude that, if adequately implemented, it will verify the design objectives for the fast scram system. We will review the various phases of the test program and report the results of our evaluation at the final design stage of review. We conclude that the proposed test program is acceptable for the Preliminary Design Approval.

4.3 Nuclear Design

4.3.7 Analytical Methods

In the Safety Evaluation Report we stated that General Electric had committed to provide us with a series of topical reports to address lattice physics methods, the boiling water reactor simulation code, and verification of lattice physics and core calculational methods.

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These topical reports deal with the final design details of the core configuration and were not needed to reach the conclusion that an appropriate core design can be developed for the final design. We will report on our conclusions concerning these reports during the final design stage of review.

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5.0 REACTOR COOLANT SYSTEM

5.4 Component and Subsystem Design

5.4.5 Residual Heat Removal System

During our review of the GESSAR-238 Nuclear Island design, we noted that the residual heat removal system could be disabled by a single failure in the shutdown cooling mode. General Electric agreed to provide an alternate method of achieving cold shutdown which employed safety-grade equipment and which was not susceptible to the same single failure which could disable the residual heat removal system. This commitment formed the basis for the issuance of the Preliminary Design Approval to General Electric.

Since that time, General Electric has described how they plan to achieve cold shutdown in the event of failure of the residual heat removal system. This involves the use of an alternate heat removal path through the automatic depressurization system valves to the suppression pool. We have reviewed the proposed method and conclude the following:

(1) The alternate path employs safety-grade equipment.

(2) No single failure will disable both the residual heat removal system and the alternate heat removal path.

(3) Either the residual heat removal system or the alternate shutdown path is capable of bringing the plant to cold shutdown assuming the loss of either onsite or offsite power and taking credit only for those actions capable of being performed from the control room.

We therefore conclude that the proposed residual heat removal system design combined with the alternate shutdown path is acceptable.

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6.0 ENGINEERED SAFETY FEATURES

6.2 Containme t Systems

6.2.1 Containment Functional Design

6.2.1.5 External Pressure Design

In the Safety Evaluation Report we stated that certain of General Electric's assumptions used in the sizing analyses for the containment vacuum breakers may not be sufficiently conservative. We have continued to review this item with General Electric and they have since revised their analysis. We have this information under review and will report our findings in a future supplement to the Safety Evaluation Report.

6.2.1.6 Test Program

In the Safety Evaluation Report we stated that we would require the General Electric Company . complete a series of confirmatory tests on the Mark III design prior to the issuance of an operating license for any Mark III plant. These tests do not represent design governing conditions nor are they expected to escalate into design basis considerations. The design criteria for the GESSAR-238 Nuclear Island containment were previously established by tests and analyses as discussed in Section 6.2 of the Safety Evaluation Report.

We will review the results of these tests, when submitted, and will report our findings at the final design stage of review.

6.2.1.9 Pool Dynamics

In the Safety Evaluation Report we stated that the General Electric Company had agreed to comply with the staff's load criteria for structures located within and above the suppression pool with the exception of two areas:

(1) The dynamic loads generated during the clearing of the safety relief valve discharge lines.

(2) The impact loads on pipes at elevations between 17 and 19.5 feet.

These two exceptions were made conditions of the Preliminary Design Approval issued to the General Electric Company.

The General Electric Company has since provided us with additional information, in Amendment 43 to the GESSAR-238 Nuclear Island Safety Analysis Report, in an effort to provide a basis for the use of a different type of safety/relief valve discharge device (quencher) which results in lower design loads than the device evaluated by the staff (ramshead). As a result of our review of Amendment 43, we have concluded that the statistical method proposed by the General Electric Company and the load criteria shown on Table 6-1 are acceptable. This conclusion is based on the following:

(1) The method has properly treated all available test data and is based essentially on the large-scale data with correction terms that take into account the influence of non-large-scale variables. Since the large-scale tests were performed in an actual reactor with a suppression containment conceptually similar to the GESSAR-238 Nuclear Island containment, extrapolation from the large-scale by statistical technique is appropriate and acceptable.

(2) The method has been conducted in a conservative manner. The primary conservatisms are:

(a) The calculation is based on the most severe parameters. For example, the maximum air volume initially stored in the line, the maximum initial pool temperature and the highest primary system pressure were selected to establish quencher load criteria.

(b) For the cases of multiple valve actuation, the load criteria are based on the assumption that the maximum pressures resulting from each valve will occur simultaneously. We believe that the assumption is conservative since different lengths of line and safety/relief valve pressure set points will result in the occurrence of maximum pressures at different times and, consequently, lower loads.

(3) The proposed load criteria in Table 6-1 are acceptable. The criteria were established by using a 95-95 percent confidence limit. Our consultant, the Brookhaven National Laboratory, has performed an analysis of the effect of this confidence limit. The result of this analysis indicates that for a 95-95 percent confidence limit, approximately one percent of the number of safety/relief valve actuations may result in containment loads above the design value. We believe that this low probability is acceptable considering the conservatism of the method of prediction. The actual loads should not exceed the design value.

(4) With regard to the subsequent actuation, the load criteria are based upon a single safety/relief value actuation. The General Electric Company has established this basis by regrouping the safety/relief valves in each group of pressure set points. As indicated in Amendment 43, there are three groups of pressure set points for the 19 safety/relief valves--namely, one safety/relief valve at a pressure set point of 1103 pounds per square inch gauge, 9 safety/relief valves at 1113 pounds per square inch gauge, and the remaining 9 safety/relief valves at 1123 pounds per square inch gauge. Only one safety/relief valve is now set at the lowest pressure set point. Based on this pressure set point arrangement for the 19 safety/relief valves, the General Electric Comoany has analyzed the most severe primary pres[illegible]nsient--a turbine trip without bypass. Results of the analysis show th[illegible]ion of reactor isolation will

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TABLE 6-1

QUENCHER BUBBLE PRESSURE FOR THE GESSAR-238 NUCLEAR ISLAND DESIGN

95-95 PERCENT CONFIDENCE LEVEL

	Case Description	Design Value Maximum Pressure (pounds per square inch differential) Positive pressure	Negative pressure
1.	Single valve first actuation at 100 degree Fahrenheit pool temperature	13.5	-8.1
2.	Single valve subsequent actuation at 120 degree Fahrenheit pool temperature	28.2	-12.0
3.	Two adjacent valves first actuation at 100 degree Fahrenheit temperature	13.5	-8.1
4.	10 valves (one low set and nine next level low set) first actuation at 100 degree Fahrenheit pool temperature	16.7	-9.3
5.	19 valves (all valve case) first actuation at 100 degree Fahrenheit pool temperature	18.6	-9.9
6.	8 automatic depressurization system valves first actuation at 120 degree Fahrenheit pool temperature	17.4	-10.4

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activate all or a portion of the 19 safety/relief valves which will release the stored energy in the primary system. Following the initial blowdown, the energy generated in the primary system consists primarily of decay heat which will cause the lowest set safety/relief valve to reopen and reclose (subsequent actuation). The time duration between subsequent actuation was calculated to be a minimum of 62 seconds and increasing with each actuation. The time duration of each blowdown decreases from 51 seconds for the initial blowdown to three seconds at the end of the period of subsequent actuations, which is 30 minutes after initiation of reactor isolation.

We find the result of the General Electric Company analysis reasonable. Therefore, the assumption of only the lowest set safety/relief valve operating in subsequent actuation is justified and acceptable.

We note, however, that the tests performed to date lack complete dynamic or geometric similarity with the Mark III quencher system. Therefore, the test results had to be extrapolated. Though the resulting loads are conservative in comparison with the test data, we feel that these loads should be verified through in-plant testing. Therefore, we will require that an in-plant test be conducted, by either General Electric or the utility applicant referencing the GESSAR-238 Nuclear Island design, at the first reference plant to complete construction. If tests are conducted at other Mark III design plants prior to the completion of construction of the first reference plant, those test results may be submitted in lieu of the test requirements on the reference plant, provided the tests had been conducted at a plant having similar containment parameters.

With regard to the exception on pipe loads between 17 and 19.5 feet, General Electric provided us with additional information in Amendment 40 to the GESSAR-238 Nuclear Island Safety Analysis Report to justify lower loads on these pipes. The staff is reviewing this information to determine the applicability of the scaling relation of the small scale tests to a full scale system. We will report on our conclusions in a future supplement to the Safety Evaluation Report.

6.2.3 Secondary Containment Functional Design

In the Safety Evaluation Report we stated that the General Electric Company committed to provide design provisions to eliminate bypass containment leakage. This was to be accomplished through the use of positive leakage control systems, upgrading some piping systems to seismic Category I to achieve credit for closed loops or water seals, and identifying those water legs or loop seals which could presently perform a sealing function. This commitment and functional description formed the basis for the issuance of the Preliminary Design Approval to the General Electric Company.

The General Electric Company has since provided, in an April 13, 1976 letter from I. Stuart of the General Electric Company to B. Rusche of the Nuclear Regulatory Commission, specific information on the functioning of the systems used to eliminate bypass containment leakage.

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The April 13, 1976 letter also contained a proposal to replace the present leakage control system on the main steam lines with a positive sealing system. The main steam line positive sealing system will be designed in a similar fashion to the sealing system on the other lines penetrating containment in that the space between the inboard and outboard isolation valves will be pressurized, via a safety-grade air system, to a pressure exceeding the peak calculated pressure that the inboard isolation valve will experience during the accident. Thus, the inward pressure on the isolation valves will always exceed the outward pressure, thereby precluding outward leakage. We conducted a preliminary evaluation of this new system and concluded that it could result in a significant decrease in the radiological doses associated with the loss-of-coolant accident and therefore warranted additional review. The General Electric Company was informed of our position at a July 1, 1976 meeting, and of our decision to review the systems designed to preclude bypass containment leakage and the proposed positive sealing system for the main steam lines, concurrently.

In their submittal, the General Electric Company addressed each of the lines penetrating the primary containment and the system proposed to preclude leakage through each line following a loss-of-coolant accident. However, specific information regarding system actuation times, system designs, and the justification for taking credit for certain closed loops to perform a sealing function was not provided.

The General Electric Company has since agreed to provide us with a topical report to address these matters. We will report on the results of our review of this report at the final design stage of review.

The General Electric Company has, however, provided us with sufficient information on the containment bypass sealing systems and the main steam line positive sealing system to enable us to conclude that the concept is technically feasible, using state-of-the-art technology, and that there is reasonable assurance that the General Electric Company can develop a final design acceptable to the staff. We conclude that these designs are acceptable for the Preliminary Design Approval and are, therefore, acceptable for use by a referencing applicant in a construction permit application.

We will require that the results of our review of the General Electric Company topical report be incorporated into the accident analysis calculations performed at the final design stage of review for any plant referencing the GESSAR-238 Nuclear Island design.

6.2.4 Containment Isolation System

In the Safety Evaluation Report, we stated that we would require the General Electric Company to qualify isolation valves and other safety-related equipment in the drywell for a negative drywell pressure representative of limiting post-loss-of-coolant accident conditions. We consider this to be a part of the overall detailed qualification program. We did not require the General Electric Company to specify a test pressure for the issuance of the Preliminary Design Approval, nor will it be required for the issuance of a construction permit to a referencing plant. We will evaluate

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the test pressure which the General Electric Company proposes against the environmental conditions expected in the drywell and report on our conclusions during the final design stage of review.

In the Safety Evaluation Report, we noted that it was our position that continuous purging of the containment through large penetrations was undesirable, but that we could find continuous purging acceptable if General Electric would adopt Containment Systems Branch Technical Position 6-4, "Containment Purging During Normal Plant Operations," as a design basis for their purge system. This position was also made a condition of the Preliminary Design Approval issued to the General Electric Company.

The General Electric Company has since modified their application to incorporate the branch position as a design basis for the GESSAR-238 Nuclear Island design. We now consider this condition resolved. We will review General Electric's manner of implementing the branch position and report on our conclusions at the final design stage of review.

6.3 Emergency Core Cooling System

6.3.1 System Description

In the Safety Evaluation Report we stated that we would continue to review the following areas related to the emergency core cooling system:

(1) Overall role of manual actions required to mitigate the consequences of a loss-of-coolant accident.

(2) Results of the spray distribution tests for the GESSAR-238 Nuclear Island reactor.

(3) The final proprietary version of the topical report on the 8 x 8 zirconium spray cooling test.

With respect to the first item, General Electric had already specified the general functions required of the reactor operator. This was found to be an acceptable basis for the issuance of the Preliminary Design Approval to the General Electric Company and is an acceptable basis for the issuance of a construction permit to a referencing plant. We will review the list of specific actions required of the operator when the design is finalized and will report on our conclusion during the review of the final design.

With respect to item two, General Electric has provided us with the results of spray distribution tests for an earlier reactor design. However, the present design has a slightly different configuration. General Electric has agreed to conduct confirmatory spray distribution tests with the new configuration to verify the predicted distribution. Since we consider these tests to be confirmatory in nature, we will not require that the tests be completed prior to the issuance of a construction

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permit to a referencing plant. We will report on the results of the tests during the review of the final design.

With respect to item three, the General Electric Company has provided the final proprietary version of the topical report on the 8 x 8 zirconium spray cooling test, Topical Report NEDE 20231, "Emergency Core Cooling Tests of an Internally Pressurized Zircaloy-Clad, 8 x 8 Simulated BWR Fuel Bundle." We found the topical report acceptable, and our evaluation of the report has been incorporated into the acceptance version of the report.

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7.0 INSTRUMENTATION AND CONTROLS

7.1 Introduction

In the Safety Evaluation Report, we concluded that the design bases and criteria for the instrumentation and control systems were acceptable for the Preliminary Design Approval, but the review of the preliminary designs had to await the submission of additional information by the General Electric Company.

The review of the preliminary design of the instrumentation and control systems has been conducted on the GESSAR-251 Nuclear Steam Supply System docket (Docket Number STN 50-531). However, with Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report, the General Electric Company updated the GESSAR-238 Nuclear Island Safety Analysis Report to the same detail as found in the GESSAR-251 Nuclear Steam Supply System Safety Analysis Report. The results of our review efforts are summarized in the following sections.

7.2 Reactor Trip System

In the Safety Evaluation Report, we concluded that the design bases and criteria of the reactor trip system were acceptable for the Preliminary Design Approval but the review of the preliminary design could not proceed until the General Electric Company supplied us with additional information. Therefore, the discussion in the Safety Evaluation Report was related mainly to the functional design. In addition, we indentified two design changes which the General Electric Company had proposed in Amendment 24 to the GESSAR-238 Nuclear Island Safety Analysis Report which appeared to result in undesirable design features.

The General Electric Company has since submitted the preliminary design of the reactor trip system in Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report which incorporates the design bases and addresses the staff concerns discussed in the Safety Evaluation Report. The following discussion details the staff review of the preliminary design of the reactor trip system and how the design bases and criteria have been implemented by the preliminary design.

The preliminary design of the reactor trip system (consisting of the system elementary wiring diagrams) was reviewed to determine conformance to the design criteria approved by the staff during the initial GESSAR-238 Nuclear Island review. In addition, the preliminary design was analyzed for operational capabilities with regard to the safety design bases. Both the equipment and the logic implementation are vastly different from previous boiling water reactor designs. Figures 7-1 and 7-2 show these design differences.

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Figure 7-1
Previous General Electric Company Reactor Trip System Logic

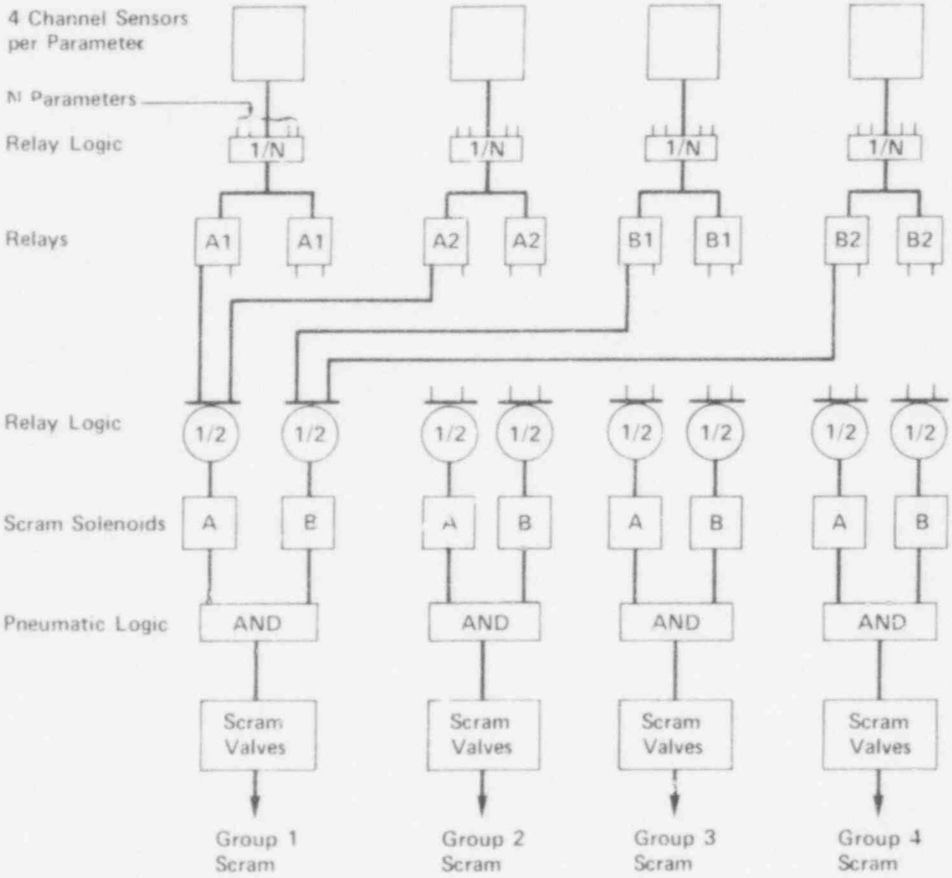

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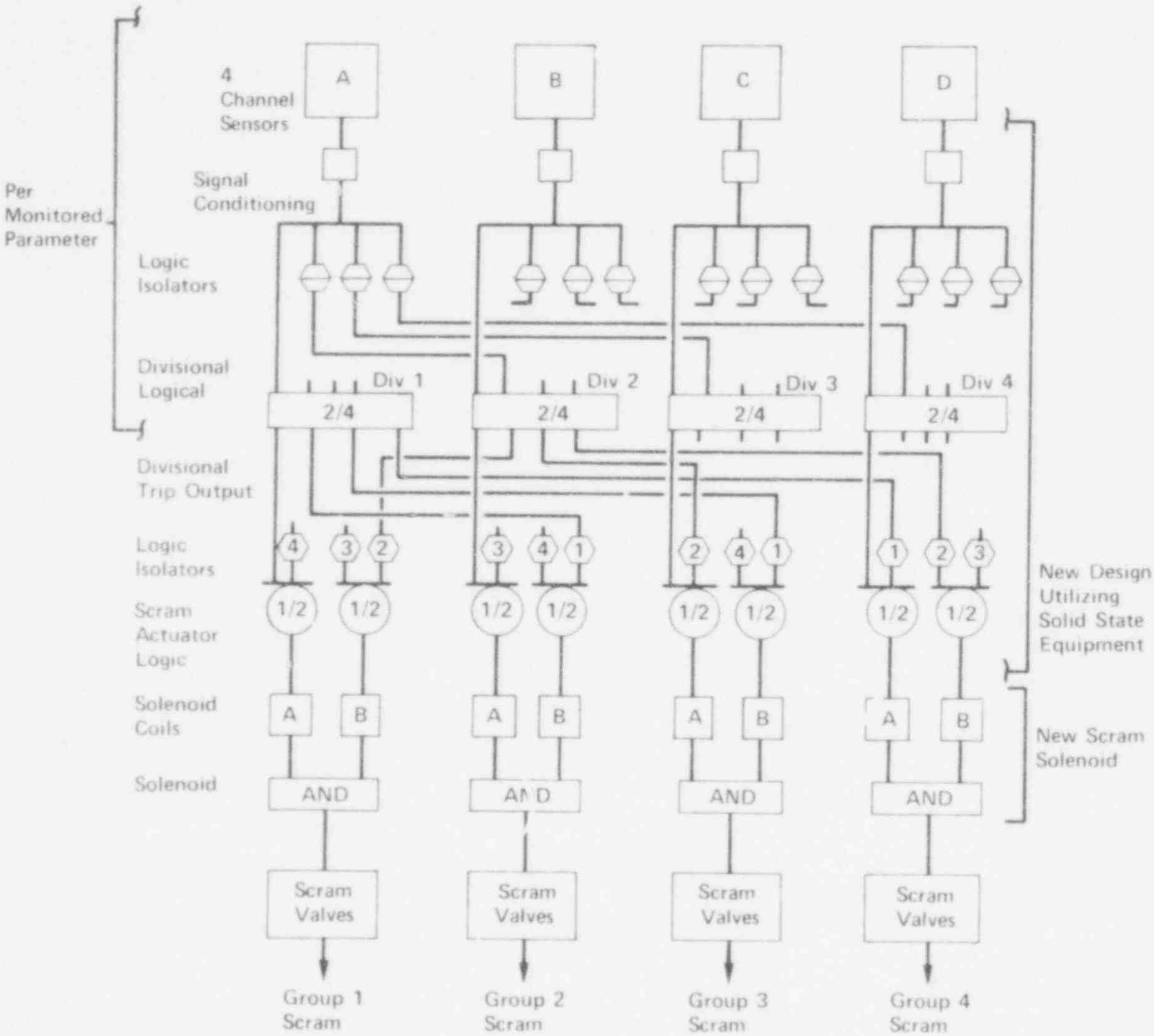
Figure 7–2
GESSAR-238 Reactor Trip System Logic



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The system utilizes solid state circuitry which is divided into four separate and independent divisional logic channels. The four divisional and independent divisional logics (located in the four divisional output cabinets) receive inputs from each of the four instrumentation channels which monitor each critical plant parameter. Both analog and digital instrumentation channels are used. In general, an analog instrumentation channel includes the sensor/transmitter, signal conditioning circuitry, and the trip units and logic isolation devices located in the divisional input cabinets. A digital instrumentation channel includes the sensor/switch and the signal conditioning circuitry and logic isolation devices located in the divisional input cabinets.

The following parameters are monitored to provide inputs to the reactor trip system:

PARAMETER	TYPE INSTRUMENT
(1) Scram discharge volume high water level.	Analog.
(2) Reactor vessel water level (low and high).	Analog.
(3) Main steam line high radiation.	Analog.
(4) Drywell high pressure.	Analog.
(5) Nuclear system high pressure.	Analog.
(6) Turbine control valve fast closure.	Digital.
(7) Main steamline isolation (closure of main steamline isolation valves).	Digital.
(8) Turbine stop valve closure.	Digital.
(9) Neutron monitoring system inputs (startup range monitoring, average power range monitoring and intermediate range monitoring).	Upscale and inoperative trips via neutron monitoring system.

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The outputs of the four divisional input cabinets are transmitted to the four divisional logics--directly to the same division and through isolation devices to the other three divisions. In this manner, adverse division interaction is prevented. The suitability of these devices is contingent on successful demonstrations by tests of their effectiveness, as discussed in Section 3.12 of this supplement. Each of the four divisional logics (located in the division output cabinets) performs a two-out-of-four coincident check of the logic level signals from each set of four instrument inputs. In addition, certain interlock and manual functions are performed in the divisional logics.

The outputs of each of the four divisional logics provide an input directly to its associated scram actuator logic and through an isolator to each of the other three divisional scram actuator logics. Thus, divisional separation is again maintained through isolation devices.

In each of the four scram actuator logics, the four input signals are combined into two one-out-of-two logic configurations. The division 1 or 4 logic configuration produces a scram actuator logic trip (de-energizes a load driver) for the "A" solenoid coils of rod groups 1 and 4 and for the "B" solenoid coils of rod groups 2 and 3. The division 2 or 3 logic configuration produces a similar trip for the "A" solenoid coils or rod groups 2 and 3 and for the "B" solenoid coils of rod groups 1 and 4. The de-energizing of both the "A" and "B" solenoid coils of each rod produces a scram. Therefore, the overall scram logic combining the divisional logic outputs to produce a reactor scram is one-out-of-two-taken-twice.

The power for the "A" solenoid coils of each of the four rod groups is supplied respectively from one of the four independent Class IE reactor trip system power supply busses. The power for the "B" solenoid coils of all four rod groups is supplied from one common non-Class IE bus. This power supply configuration provides the necessary redundancy for the reactor trip system safety functions and in addition provides protection against inadvertent scrams on loss of any one power supply.

The manual scram action for the new design includes four manual scram switches which are arranged in a one-out-of-two-taken-twice logic. By actuating one switch of either the "A" channel (division 1) or the "D" channel (division 4) and one switch of either the "B" channel (division 2) or the "C" channel (division 3), the manual scram is initiated through the load drivers. Additional manual scram capability is provided through the mode switch which provides a scram when placed in the shutdown mode.

Another portion of the design for manual scram utilizes the same four manual scram switches, but each switch is used to open the main circuit breakers which feed the load drivers and solenoid coils. By actuating any one switch (division 1, 2, 3 or 4) the associated group of rods (group 1, 2, 3, or 4) can be inserted. Therefore, this method of manual scram relies on a minimum of equipment common to the automatic scram action.

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The testing and maintenance capabilities for the reactor trip system have been improved in the sense that now instrumentation channels can be completely bypassed (removed from service if necessary) one at a time and the logic will become two-out-of-three and still meet the single failure criterion during the bypassed time interval. Operationally, this is an improvement in the sense that a failure (spurious signal) during the bypassed interval will not cause an unnecessary scram.

The results of our review of specific areas of the reactor trip system are presented below:

(1) The bypass circuitry provided in the preliminary designs includes provisions for indication of the bypassed condition and provisions for preventing simultaneous bypass of more than one instrument channel. The circuitry provisions include a channel bypass switch with a key lock (one for each of the four channels) which when placed in the bypass position sends a signal (through an isolator) to each of the other channels to prevent bypass of other channels and provides indication of the channel bypass condition.

    The bypass circuity is acceptable under the condition that the isolation devices utilized are acceptable as outlined in Section 3.12 of this supplement.

(2) The testing provisions for the reactor trip system consist of six overlapping tests that can check the operation of the reactor trip system through the solenoid coils during reactor operation. These include manual scram, simulated input calibration tests, single rod scram test, analog instrument channel test, sensor check for digital channels and a system pulse test. All these tests, except the pulse test, are similar to tests used on previous designs. The pulse test provisions are utilized for all the solid state safety systems which include the General Electric Company reactor trip system and engineered safety features systems.

    The system introduces a short duration pulse into the solid state logic at the instrument input cabinets for the purpose of checking the logic circuitry. To preclude an inadvertent scram during testing, the test equipment will deliver a pulse long enough to propagate through the logic but not long enough to actuate the equipment. The pulse output is monitored at the output of the signal conditioning equipment (analog comparator unit or bistable trip unit) in the input cabinets to the output of the load driver. The load drivers can then be tested separately to verify their ability to energize and/or de-energize the respective connected loads.

    We have concluded that the General Electric Company has defined the necessary design criteria (which include the Institute of Electrical and Electronics Engineers Standards 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," 379-1972, "Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Stations Protection Systems," and 338-1971, "Criteria for the Periodic Testing of Nuclear Power Generating Stations Protection Systems") and has provided sufficient preliminary design information for the testing of the solid state safety system. We find this acceptable for the Preliminary Design Approval.

During the final design review, we will review the pulse test portion of the final design, including the qualification of the isolation devices used in this system (see Section 3.12 of this supplement), in order to determine that this new design feature will not degrade the overall safety function of the solid state safety system.

(3) We have reviewed the distribution scheme of the 120 Volt-alternating current power for the scram solenoid valve coils. During the review, we noted a possible concern regarding a common failure mode of the load drivers and their ability to de-energize when required. The General Electric Company has addressed this concern with a description of the isolation capability of the load drivers. We believe that this is an acceptable approach. The isolation capability of the load drivers will be a part of the qualification program required under Section 3.12 of this supplement.

(4) The General Electric Company has made a change in the areas of power supply voltage for the load drivers. Previously the logic voltage for the load drivers in a divisional output cabinet was supplied by the same divisional power supply. The General Electric Company has changed this such that there is interconnection for this power between division cabinets. (Division 1 supplied a portion of the d-c logic voltage for division 4 and vice versa; division 2 supplied a portion of the d-c voltage for division 3 and vice versa.)

We required the General Electric Company to justify this change. The General Electric Company has documented that, by this design modification, an operational advantage is achieved while maintaining the channel independence. The operational advantage is that failure of the power supply voltage of one logic division will not cause an unnecessary scram of one group of rods as would have previously occurred. The channel independence is maintained by conformance to Regulatory Guide 1.75, "Physical Independence of Electrical Systems," recommendations in this area of the modified design and therefore is acceptable.

(5) During our review of the reactor trip system, we noted that certain inputs to the reactor trip system originated from the turbine building area which is not normally categorized as a seismic Category I structure. The inputs provide a signal to scram the reactor on a turbine trip event and include the turbine stop valve closure signal and the turbine control valve fast closure signal (and associated turbine first stage pressure permissives).

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Because our position with regard to reactor trip system inputs has maintained that all the inputs must meet the Institute of Electrical and Electronics Engineers Standard 279-1971 (including the seismic qualification requirements), the inputs from the turbine building area were unacceptable as proposed.

Therefore, we required the General Electric Company to demonstrate that safety grade backup trip inputs to the reactor trip system did exist, which could provide a reactor scram when required in order to prevent unacceptable consequences. The General Electric Company has performed an analysis to demonstrate that safety-grade trips are available to prevent unacceptable consequences. We have reviewed the analysis and the results are presented in Section 15.5 of this supplement.

Based on our conclusion on the General Electric Company's analysis of the failure of these turbine trip inputs and the General Electric Company's commitment that these inputs satisfy all the requirements of the Institute of Electrical and Electronics Engineers Standard 279-1971, with the exception of seismic qualification, we conclude that these inputs to the reactor trip system from the turbine building area are acceptable.

(6) The General Electric Company has stated that the reactor vessel high water level input to the reactor trip system will be designed in a manner identical to other safety-grade inputs to the reactor trip system. We find this commitment acceptable but will require formal documentation of the high water level trip prior to the issuance of a construction permit to a referencing plant. We will report on this documentation in a future supplement to the Safety Evaluation Report.

We have concluded that, except for the previously mentioned item 6, the preliminary design of the reactor trip system satisfies the Commission's regulations and is acceptable for the Preliminary Design Approval.

7.2.1 Fast Scram (New Section)

The General Electric Company has removed all reference to the prompt relief trip system and has proposed to utilize a "fast scram" system. The General Electric Company has stated that the "fast scram" system does not affect any of the instrumentation and control systems in the plant except for the control rod drive system's hydraulic lines.

However, we noted in Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report that the scram solenoid arrangement for each control rod drive has changed from two individual solenoid valves to a single dual coil solenoid valve.

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We conclude that this change does not alter our conclusions with regard to the plant instrumentation and control systems since there are no electrical functional changes from previous plant designs, as power must be removed from both coils to scram as in previous designs. We find that the instrumentation and control systems aspects of the "fast scram" system are acceptable for the Preliminary Design Approval.

7.3 Engineered Safety Features Systems

7.3.1 Introduction

In the Safety Evaluation Report, we concluded that the design bases and criteria for the engineered safety features systems were acceptable for the Preliminary Design Approval, with the exception of those specified for the standby gas treatment system. We could not make any conclusion as to the preliminary design of the engineered safety features systems at that time, for the designs had not been submitted for review.

The General Electric Company has since submitted the preliminary design of these systems in Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report.

The preliminary design of the engineered safety features systems (consisting of the elementary wiring diagrams) was reviewed to determine conformance to the design criteria established during the initial GESSAR-238 Nuclear Island review. In addition, the preliminary designs were analyzed for operational capabilities with regard to the system's safety design bases.

Although the functional performance requirements of the engineered safety features systems are fundamentally the same as previous boiling water reactor plant designs, the instrumentation and control systems (that is, the sensors, logic and actuators) for the engineered safety features systems are not similar to any previous boiling water reactor designs. The engineered safety features instrumentation and control systems utilize solid state equipment and, coupled with the reactor trip system, form what the General Electric Company refers to as the solid state safety system. The testing for the engineered safety features includes the same pulse test provisions as the reactor trip system described in Section 7.2, item 2 of this supplement.

As in the case of the reactor trip system, the preliminary designs were reviewed based on the criteria established during the review of the conceptual designs. The results of our review of the preliminary designs are summarized in the following sections.

7.3.2 Emergency Core Cooling System

7.3.2.1 High Pressure Core Spray

In the Safety Evaluation Report on the GESSAR-238 Nuclear Island design, we concluded that the design bases and criteria for the high pressure core spray system were

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acceptable for the Preliminary Design Approval but the review of the preliminary design could not be completed without additional information from the General Electric Company.

The General Electric Company has since submitted the preliminary design for the system in Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report. We have reviewed this information and conclude that all aspects of the design bases and criteria, which have been previously approved, have been appropriately incorporated into the preliminary design. We conclude that the instrumentation and controls for the high pressure core spray are acceptable for the Preliminary Design Approval.

7.3.2.2 Automatic Depressurization System

In the Safety Evaluation Report, we identified two generic issues of long standing which we required the General Electric Company to address. These were the capability for on-line testing of the solenoid valves and the consequences of inadvertent actuation of the automatic depressurization system resulting from a single failure.

The General Electric Company has included the single failure design criterion as part of the design bases for the inadvertent actuation of the automatic depressurization system. The preliminary design of the actuation system for the automatic depressurization system includes two logic trains within each division. A coincidence of the two logic trains of a division must be satisfied to achieve actuation. Each logic train is identical except that one train of each division contains a delay timer to delay automatic depressurization system actuation for 120 seconds after the loss-of-coolant accident signals are present.

We identified the 120-second timer as a potential single equipment failure that could cause inadvertent actuation during the first 120 seconds of a loss-of-coolant accident. The General Electric Company has verified that this inadvertent actuation (during the 120-second time interval) has acceptable consequences and concludes that multiple failures must occur to cause inadvertent actuation of the automatic depressurization system during plant operation. We find this acceptable.

In Amendment 24 to the GESSAR-238 Nuclear Island Safety Analysis Report, the General Electric Company committed to performing a study of methods to improve the testability of the automatic depressurization system. We found this commitment acceptable for the issuance of the Preliminary Design Approval to the General Electric Company, subject to our review and approval of the preliminary design for the automatic depressurization system, including the design provisions for testing.

In conjunction with this program, the General Electric Company performed reliability analyses for various alternate conceptual designs for the automatic depressurization system which included on-line testing capability. Based on these analyses, the General Electric Company concluded that the existing system design is more reliable than any of the alternate designs that were evaluated.

We are reviewing this information and are not able to concur with the General Electric Company's conclusion because of concerns with some of the assumptions used in the analyses. For example, we do not believe that the General Electric Company assumption for the solenoid valve failure rate has been adequately justified.

We are currently pursuing this aspect of the automatic depressurization system design with the General Electric Company and will report on our conclusions in a future supplement to the Safety Evaluation Report.

7.3.2.3 Low Pressure Core Spray and Low Pressure Coolant Injection System

In the Safety Evaluation Report, we concluded that the design bases and criteria for the low pressure core spray and low pressure coolant injection systems were acceptable for the Preliminary Design Approval, but the review of the preliminary designs of these systems required additional information from the General Electric Company. Part of the basis for the acceptance of the design bases and criteria was the General Electric Company commitment, in Amendment 24 to the GESSAR-238 Nuclear Island Safety Analysis Report, to modify the designs to provide diverse initiation signals that were not dependent on a non-diverse interlock and to evaluate the designs to assure that adequate protection against high pressure is provided for the low pressure portions of these systems.

The design of the emergency core cooling system injection lines was reviewed to confirm that the isolation provisions at the interface with the reactor coolant system were adequate. The number and type of valves used to form the interface between low pressure portions of the emergency core cooling system and the reactor coolant system must provide adequate assurance that the emergency core cooling system will not be subjected to a pressure greater than its design pressure. This may be accomplished by any of the following provisions:

(1) One or more check valves in series with a normally closed motor-operated valve. The motor-operated valve is to be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the emergency core cooling system design pressure.

(2) Three check valves in series.

(3) Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leaktightness and that testing is performed at least annually.

The General Electric Company has utilized a check valve in series with a normally closed motor-operated valve. The motor operated valve is to be opened upon receipt of an accident signal once the reactor coolant pressure has decreased below the design

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design pressure of the low pressure core spray system and the low pressure coolant injection system. The General Electric Company has also agreed to provide a pressure interlock on the motor-operated valve to preclude the opening of the valve until the reactor system pressure is sufficiently low. The General Electric Company has agreed to make the interlock diverse in order that the diverse accident initiation signals (low reactor water level and high drywell pressure) are not dependent on a non-diverse interlock. However, the General Electric Company has not supplied the preliminary design information to demonstrate diversity. We will report on this matter in a future supplement to the Safety Evaluation Report prior to the issuance of a construction permit to a referencing plant.

7.3.3 Containment and Reactor Vessel Isolation Control System

In the Safety Evaluation Report, we concluded that the design bases and criteria for the containment and reactor vessel isolation control system were acceptable for the Preliminary Design Approval, subject to the staff review of the preliminary design. We also agreed with the General Electric Company that the review of the control arrangement for the main steam isolation valves would be conducted through the review of the General Electric Company Topical Report APED-5790, "Design and Performance of General Electric Boiling Water Reactor Main Steam Isolation Valves."

The applicable staff positions resulting from the review of that topical report are as follows:

(1) The solenoid valves must be qualified to the requirements of the Institute of Electrical and Electronics Engineers Standard 323-1974, "Qualifying Class I Electrical Equipment for Nuclear Power Generating Stations," for the specific plant in which they will be used.

(2) The solenoid valves must be physically separated and electrically isolated in order to preserve the electrical independence of the initiating logic and satisfy the requirements of the Institute of Electrical and Electronics Engineers Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

(3) A method of testing the valves for proper response time which satisfies the requirements of General Design Criterion 21 (as clarified by Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions") must be provided. Testing shall be conducted on a routine basis during reactor operation.

With regard to item 1 on solenoid valve qualification, the General Electric Company has submitted their qualification program which will include this equipment. A discussion of the review of this qualification program is included in Sections 3.10 and 3.11 of this supplement.

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With regard to item 2 on separation and electrical isolation, physical separation between solenoids on the same main steam isolation valve is not required to maintain the independence of the initiating logic circuitry since these circuits are kept independent through isolation devices (load drivers).

We find this an acceptable approach and therefore it is concluded that the isolation capability of the load drivers has to be established as part of the qualification program required under Section 3.12 of this supplement.

With regard to item 3 on the response time testing of the valves during reactor operation, the GESSAR-238 Nuclear Island design incorporates provisions for each main steam isolation valve to be response-time tested during reactor operation.

Other portions of the containment and reactor vessel isolation control system will function similar to previous designs, but the equipment (sensors, actuation and logic) are not similar to previous plants and were included in the review of the preliminary design of the overall solid state safety system.

We conclude that the preliminary design of the containment and reactor vessel isolation control system is acceptable for the Preliminary Design Approval.

7.3.4 Essential Service Water System

The preliminary design review of the essential service water system has not as yet been completed. The General Electric Company provided preliminary design information on this system in Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report.

We have reviewed this information and have issued requests for additional information to the General Electric Company. When complete responses to these requests for additional information are provided, we will complete our review and report our conclusions regarding this system in a future supplement to the Safety Evaluation Report.

7.3.5 Flammability Control System

We have reviewed the preliminary design information which the General Electric Company has supplied on the flammability control system through Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report and have issued requests for additional information to the General Electric Company. When complete responses to these requests for additional information are provided, we will complete our review and report our conclusions regarding this system in a future supplement to the Safety Evaluation Report.

7.3.6 Standby Gas Treatment System

We have reviewed the information supplied to date by the General Electric Company and have issued requests for additional information on the design criteria of the system and on certain aspects of the preliminary design. When complete responses to these requests for additional information are provided, we will complete our review and report our conclusions regarding this system in a future supplement to the Safety Evaluation Report.

7.3.7 Suppression Pool Makeup System

The General Electric Company has not as yet provided us with that portion of the logic and control for the suppression pool makeup system describing the specific source of the loss-of-coolant accident or emergency core cooling system operating signals. Information demonstrating channel independence for the logics and controls for the suppression pool makeup system is similarly lacking. When this information is provided, we will complete our review and report our conclusions in a future supplement to the Safety Evaluation Report.

7.3.8 Containment Spray System

The General Electric Company has submitted preliminary design information on the containment spray system in Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report. We have reviewed this information and issued requests for additional information to the General Electric Company. Once complete responses to our requests for additional information are provided, we will complete our review and report our conclusions in a future supplement to the Safety Evaluation Report.

7.3.9 Indication of Bypasses

The review of the preliminary designs included a review of the provisions for the indication of bypassed conditions as outlined in Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

The General Electric Company has made provisions for this indication to satisfy the requirements of Regulatory Guide 1.47. We conclude that this is acceptable for the Preliminary Design Approval.

7.4 Safe Shutdown System

7.4.1 Reactor Core Isolation Cooling System

The preliminary design (elementary wiring diagram) of the reactor core isolation cooling system was reviewed to determine the conformance to the design criteria and staff requirements established during our initial review, as reported in the Safety Evaluation Report.

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In that review, we determined that the reactor core isolation cooling system would be required to be classified as an engineered safety feature because, together with the high pressure core spray system, it provided the protection necessary in the event of a rod drop accident. Therefore, the review of the preliminary design concentrated on the conformance to the established design criteria and the same requirements of other engineered safety features such as the high pressure core spray system.

The preliminary design, as proposed by the General Electric Company, did not provide a feature to transfer the suction of the reactor core isolation cooling system from the condensate storage tank to the suppression pool.

The General Electric Company has revised the preliminary design to include the automatic transfer feature and has designed it similar to the high pressure core spray system feature. With the above modification, we conclude that the preliminary design of the reactor core isolation cooling system is acceptable for the Preliminary Design Approval.

7.4.2 Standby Liquid Control System

We reviewed the preliminary design (elementary wiring diagram) of the standby liquid control system to determine the conformance to the design criteria and staff requirements established during our initial review as reported in the Safety Evaluation Report.

In accordance with the review, the General Electric Company has submitted the preliminary design which provides for redundant actuation of the standby liquid control system pumps. In addition, they have submitted the results of an analysis which demonstrate that failure of the reactor water cleanup system to isolate will not prevent the standby liquid control system from performing its function.

We conclude that the preliminary design of the standby liquid control system conforms to the design bases approved during our initial review and is therefore acceptable for the Preliminary Design Approval.

7.5 Safety-Related Display Instrumentation

7.5.1 General

In the Safety Evaluation Report, we concluded that the agreements made with and commitments made by the General Electric Company on the design criteria for safety-related display instrumentation were acceptable for the Preliminary Design Approval. We also reported that the General Electric Company had agreed to provide specific information such as lists of the indications and controls to be provided and the physical arrangements of the control board panels. This information was to be reviewed after the issuance of the Preliminary Design Approval to the General Electric Company. The General Electric Company has since provided us with a list of the indication and controls within their scope of supply, the preliminary design and controls within their scope of supply, and the preliminary control board arrangement drawings. We have reviewed this information and consider it acceptable for the Preliminary Design Approval.

The GESSAR-238 Nuclear Island design for the control room incorporates both the Nuclenet and Power Generation Control Complex design packages which are discussed in the following section.

7.5.2 Nuclenet

We have reviewed the Nuclenet design criteria in the past through individual case reviews and through Topical Report NEDO-10939, "Design Criteria and Technical Description of Plant/Operator Interface of the Nuclenet 1000 Control Complex." In those reviews, we have determined that Nuclenet is basically an operator-to-plant system interface aid which utilizes cathode ray tube displays. The General Electric Company has submitted the preliminary control board arrangement drawings as part of the GESSAR-238 Nuclear Island Safety Analysis Report and has stated their commitment to Regulatory Guide 1.75, "Physical Independence of Electrical Systems." Based on the General Electric Company's statement that all instrumentation and control equipment required for plant safety in the area of the main control room are unaffected by utilizing Nuclenet (since it remains hardwired in Nuclenet) and the review of the preliminary design information in the GESSAR-238 Nuclear Island Safety Analysis Report, we conclude that the Nuclenet design is acceptable for the Preliminary Design Approval.

7.5.3 Power Generation Control Complex

We have reviewed the Power Generation Control Complex concept in Topical Report NEDO-10466, "Power Generation Control Complex," and have been participating in on-going discussions in this area with the General Electric Company. The main subjects involve the provisions for satisfying the recommendations of Regulatory Guide 1.75, "Physical Independence of Electrical Systems," and the acceptability of large quantities and large concentrations of cables in a small area immediately beneath the control room floor.

The General Electric Company will design the Power Generation Control Complex to Regulatory Guide 1.75. As a result of our generic study in the area of fire protection criteria, further requirements may be imposed on this system so that unacceptable damage will not result from a fire. This area of the control room design is still a subject of on-going discussions between the staff and the General Electric Company in our evaluation of NEDO-10466. We will provide the results of our review of NEDO-10466 at the final design stage of review.

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7.6 All Other Systems Required for Safety

7.6.1 General

The systems reviewed for this section were:

(1) Refueling interlocks.

(2) Reactor vessel instrumentation.

(3) Process radiation monitoring system.

(4) Area radiation monitoring system.

(5) Reactor water clean-up system.

(6) Leak detection system.

(7) Process computer system.

(8) Containment atmosphere monitoring system.

(9) Neutron monitoring system.

(10) Fuel pool cooling and cleanup system.

The review of the preliminary designs concentrated on the safety design bases of these systems and the interaction of these systems with the plant systems required for safety.

In the case of the process computer system, this system has no stated safety design basis and, therefore, we have not required a submittal of the preliminary design for our review. The process computer system will be reviewed at the final design stage of review to verify that the process computer system is not relied upon to perform a safety function.

For all other systems, we have reviewed the preliminary design submitted by the General Electric Company to determine that the systems are designed in accordance with the safety design bases and to verify that the system design will not compromise the safety systems with which they interact.

The neutron monitoring system supplies inputs to the reactor trip system and receives power from the reactor trip system safety-related power supplies buses. The GESSAR-238 Nuclear Island design will utilize the reactor trip system for only divisional and associated equipment in accordance with the recommendations of Regulatory Guide 1.75, "Physical Independence of Electrical Systems."

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We conclude that the preliminary designs of these systems are acceptable for the Preliminary Design Approval.

7.6.2 Reactor Pressure Relief Instrumentation

In the Safety Evaluation Report we stated that the General Electric Company had committed to upgrading the reactor pressure relief instrumentation to provide redundancy and independence equal to that required for protection systems. This commitment formed the basis of the Preliminary Design Approval issued to the General Electric Company.

The review of the preliminary design concentrated on verifying that the design requirements, as stated, were implemented in a satisfactory manner. The preliminary design originally proposed by the General Electric Company included only a single division (energized to actuate) for the pressure relief function. We concluded that the system did not meet the single failure criterion and was, therefore, unacceptable. In a subsequent revision to the preliminary design, the General Electric Company provided a two division system similar to the automatic depressurization system function. Now either of the two divisions can actuate all of the valves as required. As similiarly noted in the automatic depressurization system evaluation, this type of design allows for the remote possibility that all the safety-related valves could be opened due to a spurious division signal.

In order to help prevent such a spurious signal, the General Electric Company has included the single failure criterion as part of the design bases for inadvertent actuation of the safety-relief valves. As in the design of the automatic depressurization system, the preliminary design of the actuation system for the safety-relief valves includes dual logic trains within each division. A coincidence of the two logic trains (within the same division) must be satisfied to achieve actuation. Therefore, multiple failures must occur to cause inadvertent actuation.

Since the actuation of the safety-relief valves via the pressure relief instrumentation and control system is identical to the actuation of the automatic depressurization system, and since the design criteria of the reactor pressure relief instrumentation will be equivalent to that of protection systems, we will require that the resolution achieved for the testing of the solenoid valves on the automatic depressurization system also be applied to the solenoid valves on the reactor pressure relief system.

We will report on the resolution of this item in a future supplement to the Safety Evaluation Report.

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7.6.3 Recirculation Pump Trip System

The recirculation pump trip system performs the function of disconnecting (via redundant circuit breakers in series in the motor feeders) the main power to both reactor recirculation pumps when the turbine trips. The equipment of this system is all safety-grade equipment and is connected to the reactor trip system circuitry (see Figure 7-3). When the turbine stop valves close and/or the turbine control valves fast close, indicating a turbine trip, the four recirculation pump trip divisional logics receive signals from the two-out-of four divisional logics of the reactor trip system, which produces a reactor scram for this event. Each recirculation pump trip divisional logic then energizes a load driver which in turn opens the associated circuit breaker. Two circuit breakers are arranged in series in the power feed circuit of each of the two recirculation pump motors and, therefore, a one-out-of-two system level actuation is produced.

This actuation logic prevents a single failure from either disabling the safety function when required or inadvertently tripping more than one recirculation pump during operation.

We have concluded that the design criteria and design description provided for the recirculation pump trip circuitry are acceptable. We will require the General Electric Company to submit the preliminary design information (elementary diagrams) for this system prior to the issuance of a construction permit to a referencing plant.

7.7 Control Systems

7.7.1 Rod Control and Information Systems and Instrumentation (Formerly referred to as Reactor Manual Control System

The rod control and information system is included in Section 7.7 in "Control System"; however, portions of the system perform functions related to safety.

The General Electric Company has provided preliminary design information for the rod control and information system which includes functions related to ganged rod motion, rod information monitoring, and rod pattern control.

We have been reviewing the reactor manual control system design for the General Electric Company's earlier plant designs as part of a generic review of the General Electric Company's reactor control systems. Although the system title has been changed from reactor manual control system to rod control and information system for the GESSAR-238 Nuclear Island design, these designs are similar.

Therefore, we required the General Electric Company to provide a detailed comparison between these designs in order that the previous and on-going generic review effort can be appropriately applied to the GESSAR-238 Nuclear Island design.

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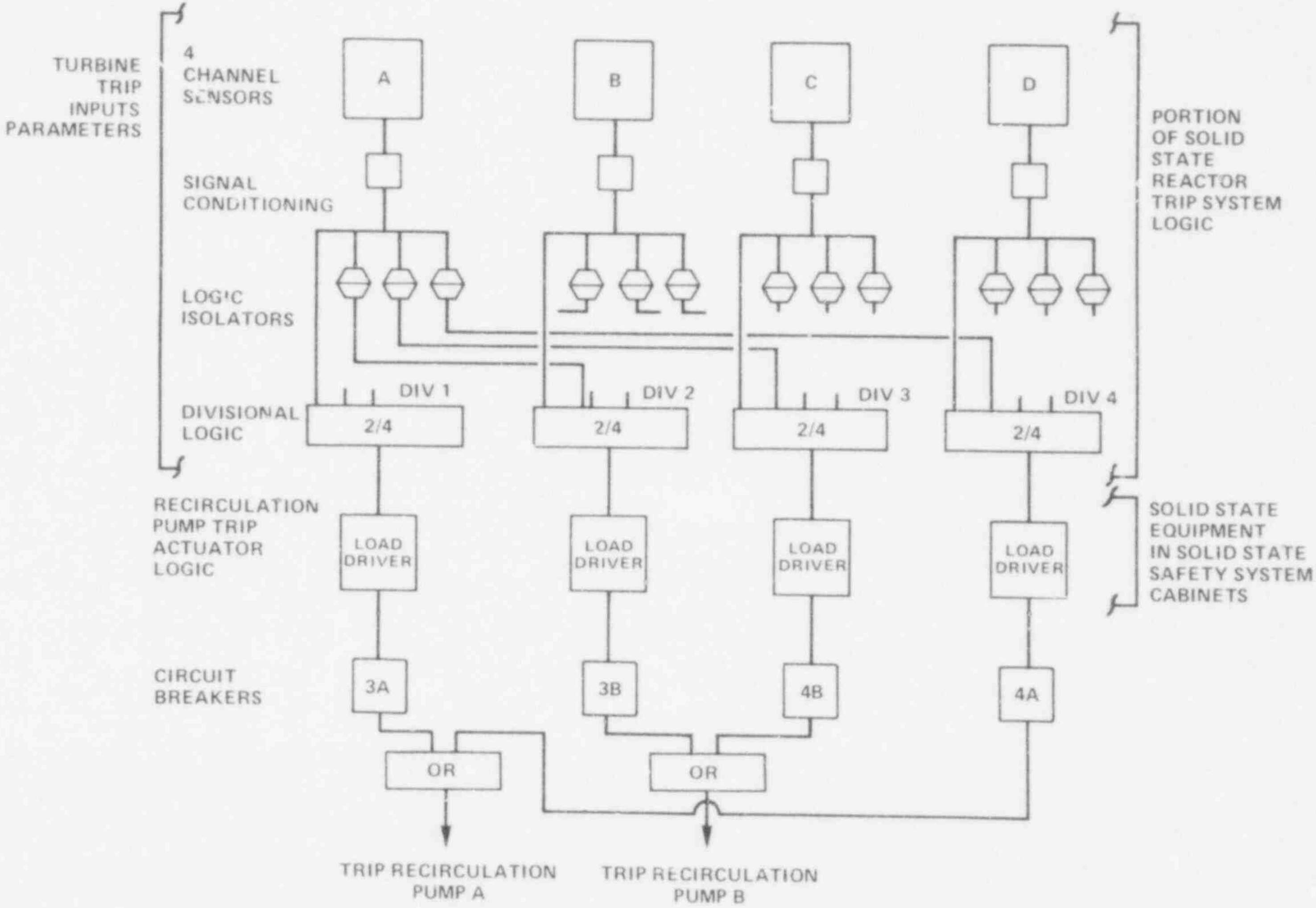
Figure 7-3

Recirculation Pump Trip System Logic



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The General Electric Company has provided information which verifies that areas of the proposed rod control and information system and instrumentation is similar to previous reactor manual control system designs. Both systems use multiplexed signals to transponders on the control rod hydraulic control units and both systems use similar solid state rod inhibit logic and rod motion timers with automatic fault-finding circuitry.

The differences as enumerated by the General Electric Company include new provisions for satisfying the design criteria established during the GESSAR-238 Nuclear Island review. Specific differences noted include dual rod position information and redundant rod action control to satisfy the safety design bases. Other differences include provisions for ganged rod motion and equipment utilized for operator control and display functions.

Based on the above comparative design information, coupled with the on-going design review of the reactor manual control system of the previous General Electric Company designs and the review of the preliminary design information submitted for the rod control and information system, we conclude that this area is acceptable for the Preliminary Design Approval. We will require that the results of the on-going generic review be applied to the GESSAR-238 Nuclear Island when they become available.

7.7.2 Other Control Systems

The systems reviewed for this section of this supplement were:

(1) Recirculation flow control system.

(2) Feedwater control system.

(3) Pressure regulator and turbine generator control system.

(4) Gaseous radwaste control system.

(5) Liquid radwaste control system.

We have reviewed the controls for these systems to determine the effects of failures or malfunction of the controls on the reactor protection system and other plant safety-related systems. We conclude that failures or malfunctions of the controls would not be expected to degrade the capabilities of plant safety systems in any significant degree, or to lead to plant conditions more severe than those for which the safety systems are designed and are, therefore, acceptable.

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7.8 Instrumentation Interfaces with Balance-of-Plant Systems

In the Safety Evaluation Report, we stated that we could not proceed with the review of instrumentation interfaces with balance-of-plant systems until the General Electric Company provided the preliminary designs of the instrumentation and control systems within their scope of design.

Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report provided this preliminary design information, and we have issued requests for additional information to the General Electric Company on electrical interfaces. When complete responses to our requests for additional information are provided, we will complete our review and report on this area in a future supplement to the Safety Evaluation Report.

8.0 ELECTRICAL POWER SYSTEMS

In the Safety Evaluation Report, we concluded that the proposed design criteria for the standby power instrumentation and control systems form a generally acceptable basis for developing a design for the electrical power systems.

Since the issuance of the Safety Evaluation Report, the General Electric Company has submitted additional information on the electrical power system in Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report. We have reviewed this information and have issued requests for additional information to the General Electric Company.

Our main areas of concern relate to the load shedding and sequencing logic for the onsite power system, interfaces between the onsite power system and the balance-of-plant, and the exceptions which the General Electric Company is taking to Topical Report NEDO-10905, "High Pressure Core Spray System Power Supply Unit," which is referenced by the GESSAR-238 Nuclear Island. We will report on these areas in a future supplement to the Safety Evaluation Report.

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9.0 AUXILIARY SYSTEMS

9.3 Process Auxiliaries

9.3.1 Main Steam Isolation Valve Leakage Control System

In the Safety Evaluation Report, we concluded that the design of the main steam isolation valve leakage control system was acceptable, subject to two requirements. We required an interlock to preclude the operation of any inboard leakage control system if the inboard main steam isolation valve associated with that system was not fully closed. In addition, we required that the setpoint of the flow element timers for the inboard leakage control system be set at 11.5 standard cubic feet per hour. Both of these requirements were made a condition to the Preliminary Design Approval issued to the General Electric Company.

In an April 13, 1976 letter from I. Stuart of the General Electric Company to B. Rusche of the Nuclear Regulatory Commission, the General Electric Company proposed a new positive sealing system to replace the present leakage control system on the main steam lines. As discussed in Section 6.2.3 of this supplement, we have reviewed the General Electric Company proposal and conclude that it is technically feasible using state-of-the-art technology and that there is reasonable assurance that the General Electric Company can develop a final design which we could find acceptable.

The General Electric Company has committed to provide us with a topical report to address the remaining areas of concern associated with the review of the main steam line positive sealing system. We will report on the results of our review of that report at the final design stage of review and will require that the results of our review be incorporated into the accident analysis calculations performed for any plant referencing the GESSAR-238 Nuclear Island design.

Based on the information and commitments provided, we conclude that the proposed positive sealing system for the main steam lines is acceptable for the Preliminary Design Approval and is therefore acceptable for use by a referencing applicant in a construction permit application.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.3 Gaseous Waste Treatment System

11.3.3 Conclusions

In the Safety Evaluation Report we stated that continuous purging of the containment directly to the environment without treatment was unacceptable and purge filtration was made a condition of the Preliminary Design Approval issued to the General Electric Company.

In Amendment 43 to the GESSAR-238 Nuclear Island Safety Analysis Report, the General Electric Company modified the GESSAR-238 Nuclear Island design to incorporate a non-safety-grade filtration system connected to the containment purge system. A safety-grade filtration system is not required since the containment purge line is isolated in the event of an accident. We consider this modification acceptable and the condition resolved.

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15.0 ACCIDENT ANALYSIS

15.2 Abnormal Operational Transients

In Amendment 41 to the GESSAR-238 Nuclear Island Safety Analysis Report, the transient analyses have been revised to incorporate fast scram. The types of typical transients evaluated were losses of flow, increases in pressure and power, decreases in coolant temperature, and increases in coolant flow. The most limiting transients in these categories were a two-pump trip, a generator load rejection without bypass, a loss of feedwater heating, and flow controller failure (increasing flow). Of these, the most limiting core-wide transient was generator load rejection without bypass resulting in a change in the minimum critical power ratio of 0.16. Addition of the change in minimum critical power ratio to the safety limit minimum critical power ratio gives the operating limit minimum critical power ratio required to avoid violation of the safety limit, should this limiting transient occur. Thus, the operating limit minimum critical power ratio is 1.23.

The transient analyses were evaluated with the scram reactivity insertion rates shown in Figure 15.1.1-1 of the GESSAR-238 Nuclear Island Safety Analysis Report. The initial condition parameters used for the analyses are acceptable. The initial minimum critical power ratio assumed in the transient analyses was equal to the established operating limit minimum critical power ratio of 1.23.

The analyses were performed using a computer-simulated analytical model of a generic direct-cycle boiling water reactor, as described in Topical Report NEDO-10802, "Analytical Methods of Plant Transient Evaluations for the GE BWR." Our review of this topical report will be completed prior to the final design review of the GESSAR-238 Nuclear Island design. Changes on limitations resulting from our review will be reflected in the final design.

The rod withdrawal error (limiting local event) transient is discussed in the GESSAR-238 Nuclear Island Safety Analysis Report in terms of worst case conditions. The discussion indicates that if the peak linear power design limits are exceeded, the nearest local power range monitor subsystems will detect this phenomenon and initiate an alarm.

However, if the operator ignores the alarm, the analysis considers the continuous withdrawal of the maximum worth control rod or control rod gang (at maximum drive speed) to the full out position when operating between 20 percent and 70 percent power, and the withdrawal of a control rod or rod gang over two feet beyond the original rod pattern when operating at greater than 70 percent power. The results of the analysis show that the minimum critical power ratio will remain above 1.07 and the cladding will remain under the one percent plastic strain limit. The rod withdrawal error transient is not limiting for the GESSAR-238 Nuclear Island reactor above 70

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percent power at rated flow since the rod pattern control system prevents the motion of any control rod or control rod gang in excess of two feet beyond the existing rod pattern.

A two-pump trip system has been added to the GESSAR-238 Nuclear Island design. This system trips the recirculation pumps upon sensing either a fast closure of the turbine control valves or excess coolant inventory in the reactor vessel. We have reviewed these two transients allowing credit for the two-pump trip feature. As indicated in Section 7.6.3 of the supplement, we have reviewed the markup drawings of the two-pump trip feature but require that this design change be formally documented.

In addition to the criterion on fuel cladding integrity, there is a requirement that pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures during transients. For the GESSAR-238 Nuclear Island reactor this limit is 1375 pounds per square inch gauge, and it is not exceeded during the transients analyzed.

We conclude that the plant design is acceptable with regard to transients that are expected to occur during the life of the plant (minimum critical power ratio will not exceed 1.07 and reactor coolant pressures will not exceed 1375 pounds per square inch gauge).

15.4 Anticipated Transients Without Scram

In September 1973, the Atomic Energy Commission published WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," establishing acceptance criteria for anticipated transients without scram. In conformance with the requirements of Section II.B of Appendix A to WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," the General Electric Company submitted Topical Report NEDO-20626, "Studies of BWR Designs for Mitigation of Anticipated Transients Without Scram." This report is referenced in the GESSAR-238 Nuclear Island Safety Analysis Report.

We have completed our review of Topical Report NEDO-20626, "Studies of BWR Designs for Mitigation of Anticipated Transients Without Scram," and published our findings in December 1975 as the report "Status Report on the General Electric Analyses of Anticipated Transients Without Scram." This report lists our concerns with the General Electric Company submittal, and requires the General Electric Company to update their anticipated transients without scram analyses. For example, the General Electric Company has not addressed the reliability of systems which they assume function in their present analyses (such as the relief valves or the high pressure core spray). If the General Electric Company cannot demonstrate that the combined unreliability of these systems is low enough to meet the WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," safety objective, the anticipated transient without scram analyses must be revised assuming these systems fail to operate.

In our letter of April 7, 1976 (R. E. Heineman to I. Stuart of General Electric), we required the General Electric Company to provide the following information by June 30, 1976:

(1) The results of additional analyses and the further justification of the General Electric Company analysis model identified in our report, "Status Report on the General Electric Analyses of Anticipated Transients Without Scram."

(2) Based on these analyses, identification of the design changes needed to assure that the limits specified in WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," will not be violated following an anticipated transient without scram event.

The General Electric Company has since provided a response to our request by letter dated July 2, 1976. We are continuing our generic review of this matter and will require that any changes which may be required be incorporated into the design in a timely manner. We conclude that appropriate measures to mitigate the consequences of such events are technically feasible and are within the state-of-the-art. We find this acceptable for the Preliminary Design Approval.

15.5 Failure of Inputs From Turbine Building to Reactor Protection System (New Section)

In our review of the reactor protection system we noted that certain inputs to the reactor protection system did not meet all of the safety design criteria required by the Institute of Electrical and Electronics Engineers Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." The specific inputs identified were those originating from the turbine building. The turbine building in the case of most boiling water reactor plants is not seismically qualified. For these reasons we requested that the General Electric Company investigate the probability and consequences of losing these inputs to the reactor protection system.

In their response the General Electric Company stated that the probability of losing these inputs to the reactor protection system was extremely small--on the order of 10^{-6} per year. The General Electric Company also investigated the percentage of fuel rods in the core which are subject to boiling transition prior to termination of the transient by the backup scram (high flux). The most limiting case was that for turbine trip with a backup scram in which the turbine bypass system failed to operate. For this case, the General Electric Company indicated that seven percent of the fuel rods in the core would experience boiling transition.

We have evaluated the General Electric Company response and conclude that they have adequately established that this is an extremely low probability event and that they have conservatively calculated the effects of the transient. In our evaluation of this event we have used the assumptions of the control rod drop accident to provide a very conservative estimate of the offsite dose consequences. We have also assumed that those fuel rods which experience boiling transition perforate. The quantity and behavior of fission products which are released are treated in the same manner as for the control rod drop accident.

Based on the above, we have concluded that the two hour doses will not exceed 95 rem to the thyroid and 4.5 rem whole body, assuming a relative concentration value of 1×10^{-3} seconds per cubic meter. We have also concluded that the doses for the course of the accident will be less than 55 rem to the thyroid and 4.5 rem whole body, assuming a relative concentration value of 1×10^{-4} seconds per cubic meter. Based on our review we conclude that the probability of the event is low enough to permit us to use the conservative accident assumptions and compare the offsite doses to 10 CFR Part 100. We also conclude that the offsite dose consequences are well within 10 CFR Part 100 limits.

February 1977

APPENDIX D

SUPPLEMENT NO. 2
TO THE
SAFETY EVALUATION REPORT
BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
U. S. NUCLEAR REGULATORY COMMISSION
IN THE MATTER OF
GENERAL ELECTRIC
STANDARD SAFETY ANALYSIS REPORT
(GESSAR-238 NUCLEAR ISLAND)
DOCKET NO. STN. 50-447

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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 Introduction

On December 22, 1975, the United States Nuclear Regulatory Commission issued the Safety Evaluation Report (NUREG 75/110) and the Preliminary Design Approval for the General Electric Standard Safety Analysis Report (GESSAR-238 Nuclear Island) design (Docket Number STN 50-447). In our Safety Evaluation Report on the GESSAR-238 Nuclear Island design, we identified nineteen items (in Table 1-3) that we indicated would require continued review after the issuance of the GESSAR-238 Nuclear Island Preliminary Design Approval and four staff design requirements imposed as conditions to the GESSAR-238 Nuclear Island Preliminary Design Approval.

Since the issuance of the Safety Evaluation Report, the General Electric Company has submitted five amendments (Amendments 40, 41, 42, 43 and 44) to the GESSAR-238 Nuclear Island Safety Analysis Report. The purpose of Supplement Number 1 to the Safety Evaluation Report is to update the Safety Evaluation Report by providing the staff's evaluation of the additional information received since the issuance of the Safety Evaluation Report. Each of the following sections in this supplement is numbered the same as the section of the Safety Evaluation Report that is being updated.

Appendix A to this supplement is a continuation of the chronology section of the Safety Evaluation Report.

1.7 Facility Modifications as a Result of Regulatory Staff Review

1.7.2 Facility Modifications Required by the Staff

In the Safety Evaluation Report, we identified four staff requirements which were imposed as conditions to the Preliminary Design Approval for the GESSAR-238 Nuclear Island design. Since that time, the General Electric Company has provided us with acceptable commitments to the staff positions on two of these conditions, and an acceptable resolution to a third condition.

The status of the four conditions and the section in this supplement where each condition is discussed are:

(1) Tornado missile velocities - Resolved (See Section 3.5).

(2) Containment pool dynamics - Under review (See Section 6.2.1.9).

(3) Continuous purging of containment - Resolved (See Sections 6.2.4 and 11.3).

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(2) A letter from W. Gilbert of the General Electric Company to R. Boyd of the Nuclear Regulatory Commission dated November 9, 1976,

(3) A letter from W. Gilbert to D. Vassallo of the Nuclear Regulatory Commission dated November 9, 1976, and

(4) A letter from W. Gilbert to S. Varga of the Nuclear Regulatory Commission dated November 24, 1976.

We have reviewed this additional information and have concluded that it is sufficient to resolve these issues. Table 1-1 of this supplement lists each of these 13 issues and the section of this supplement where the resolution of the issue is discussed.

1.11 Conclusion

In Section 1.11 of Supplement No. 1 to the Safety Evaluation Report, we concluded that, except for those items identified in Table 1-2 of Supplement No. 1, the General Electric Company had supplied sufficient information on the post-PDA items identified in Table 1-3 of the Safety Evaluation Report to provide a suitable basis for the issuance of a construction permit to a referencing plant.

Based on our review of the additional information provided on the GESSAR-238 Nuclear Island docket since the issuance of Supplement No. 1 to the Safety Evaluation Report, we conclude that the General Electric Company has supplied sufficient information on all the post-PDA items (Table 1-3 of the Safety Evaluation Report) to provide a suitable basis for the issuance of a construction permit to a referencing plant. We therefore conclude that all the post-PDA items are acceptable for the Preliminary Design Approval stage of review.

TABLE 1-1

SAFETY MATTERS RESOLVED SINCE THE ISSUANCE
OF SUPPLEMENT NO. 1 TO THE SAFETY EVALUATION REPORT

	Item	Discussed in Sections
1.	Leakage characteristics of primary coolant pump seals	3.2.1
2.	Containment vacuum breaker	6.2.1.5
3.	Automatic depressurization system testability	7.3.2.2
4.	High water level trip	7.2
5.	Two pump trip	7.6.3
6.	Low pressure core spray and low pressure coolant injection systems interlocks	7.3.2.3
7.	Electrical review of essential service water system	7.3.4
8.	Electrical review of flammability control system	7.3.5
9.	Electrical review of standby gas treatment system	7.3.6
10.	Electrical review of suppression pool makeup system	7.3.7
11.	Electrical review of containment spray system	7.3.8
12.	Review of Nuclear Island/balance-of-plant electrical interfaces	7.8
13.	Review of onsite power systems	8.0

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3.0 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

In the Safety Evaluation Report, we stated that we would continue to review the matter of leakage of primary coolant through the primary coolant pump seals as a result of an assumed loss of seal water and cooling water to the primary coolant pump seals.

The General Electric Company has since committed to modifying their design to incorporate a safety-grade recirculation pump trip feature to trip the recirculation pumps upon sensing a loss of pump seal cooling water. We find this design commitment acceptable.

The General Electric Company has indicated that additional studies are underway to demonstrate that the consequences of total cooling water failure are acceptable. Should the results of these studies verify that the consequences of pump seal cooling water failure are acceptable, or an alternate acceptable design be proposed by the General Electric Company, the staff will reconsider the requirement for incorporating the recirculation pump trip feature.

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5.0 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary Components

Several boiling water reactor plants have reported finding cracks on the reactor vessel feedwater nozzle blend radii and on the nozzle bore surface. In some plants, corrective measures have been required which involved grinding out a portion of the nozzle inner corner radius or bore surface. During our review of the GESSAR-251 Safety Analysis Report (Docket No. STN 50-531), we asked the General Electric Company to describe any design modifications to be made to eliminate this problem on GESSAR-251 plants and to demonstrate the adequacy of the GESSAR-251 reactor vessel feedwater nozzle design to withstand the imposed service conditions without the cracking that has been experienced in operating plants.

In response to our request the General Electric Company has stated that the GESSAR-251 reactor vessels will have two features incorporated into their design to alleviate the cracking problem:

(1) The vessels will have thermal sleeves welded into the feedwater nozzle to reduce rapid thermal cycling which the General Electric Company believes is responsible for the cracking.

(2) The vessel nozzles will not be clad thus eliminating the possibility of cracking in the cladding.

Additionally, the General Electric Company described the results of analyses that have been performed which show that the nozzles meet all American Society of Mechanical Engineers Boiler and Pressure Vessel Code design criteria in the absence of rapid thermal cycling. The General Electric Company has committed to perform out-of-reactor design verification tests to verify that the welded thermal sleeve design will reduce thermal cycling to acceptable limits and has additionally committed to instrument the first reactor vessel employing these design changes to confirm the adequacy of these modifications.

The General Electric Company has further stated that elimination of the cladding from the nozzles greatly enhances the sensitivity of ultrasonic examination from the outside surface of the vessel thus ensuring that ultrasonic inspection may be used for inservice inspection to ensure safe operation of the reactor.

In response to our request the General Electric Company, in Amendment 45 to the GESSAR-238 Nuclear Island Safety Analysis Report, committed to the same design modification for the GESSAR-238 Nuclear Island reactor vessels as were being made to the GESSAR-251 reactor vessels.

We are reviewing this problem with the General Electric Company on a generic basis. At this stage of our review, without the availability of test data, we can not conclude that incorporation of the referenced design modifications will eliminate the cracking in GESSAR-238 Nuclear Island reactor vessel feedwater nozzles. We consider the commitments to perform the out-of-reactor test program and to instrument the first of a kind reactor vessel sufficient for the Preliminary Design Approval stage of review. However, at the final design stage of review, we will require that the General Electric Company furnish test data and analyses that establish that the design modifications incorporated into the GESSAR-238 Nuclear Island design plants have eliminated the nozzle cracking problem. Additionally, we will require that the feedwater nozzle blend radii and nozzle bore surface areas of the GESSAR-238 Nuclear Island reactor vessels be amenable to inservice inspection by a non-destructive inspection technique with a sensitivity demonstrated to be sufficient to permit detection of the type of cracking that has been experienced in the operating plants as of the date of this report.

On May 7, 1975 we were informed by a licensee of a pressurized water reactor, Virginia Electric and Power Company, that the loading due to a transient asymmetric pressure distribution over the reactor core barrel, resulting from a postulated pipe rupture at a particular location in the reactor coolant loop, had not been taken into account in the original design analysis of the reactor vessel support system for North Anna Units 1 and 2 (Docket Nos. 50-338 and 50-339). We subsequently determined that the question of the adequacy of reactor vessel support systems was applicable to all reactor designs, and we requested that the General Electric Company demonstrate that these types of loads were adequately accounted for in the design of the reactor vessel support system.

The General Electric Company has provided a response to this concern in Amendment 45 to the GESSAR-238 Nuclear Island Safety Analysis Report. The response states that asymmetric differential pressure loadings due to postulated pipe ruptures are accounted for in the design of the reactor supports and pedestal and the design loads have been modified to account for these new loadings. The General Electric Company has also provided a description of the method used for calculating pressure transients in the volume between the reactor pressure vessel and the core shroud which results from the assumption of an instantaneous and complete severance of a primary coolant pipe at the reactor nozzle. The method used to calculate these pressure transients is dependent on the use of a computer code entitled "WHAM".

The General Electric Company has stated that it uses the WHAM computer code to calculate the transient pressure distribution on the core shroud in the region of the recirculation suction line nozzle for the initial decompression wave following a postulated pipe rupture. Although the WHAM code was originally written for calculating pressure transients in one-dimensional networks, the General Electric Company has stated that the WHAM code can also be used to calculate pressure transients in two or three dimensional fluid regions. The General Electric Company has committed to submit a topical report to describe the modeling of the asymmetric pressure loads

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across the reactor internals using the WHAM code. We will report the results of our review of this topical report at the final design stage of review. We will require that any modifications to the design loads, if required, resulting from our review of this topical report be factored into the final design of the GESSAR-238 Nuclear Island reactor vessel support system.

Based on the General Electric Company's commitment to provide a topical report on the WHAM code methodology and our review of the modified design loads, we conclude that there is reasonable assurance that any required modifications to the reactor vessel support system, resulting from our review of the topical report, could be incorporated into the final design and that this matter is therefore acceptable for the Preliminary Design Approval stage of review.

5.2.5 Austenitic Stainless Steel

In September 1974, cracking was experienced in the stainless steel piping at Dresden Nuclear Power Station, Unit 2, Docket No. 50-237. This was the first of a series of incidents of intergranular stress corrosion cracking that occurred in eight General Electric Company boiling water reactors. The cracking occurred in weld heat-affected zones in Type 304 stainless steel recirculation system bypass piping systems and core spray lines. As a result of these incidents, a special task group within the Nuclear Regulatory Commission was formed to investigate the causes of the cracking. The results and conclusions of the task group are given in the staff technical report, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," NUREG 75/067, October 1975.

The task group found that austenitic stainless steel piping in the reactor coolant pressure boundary of boiling water reactors is susceptible to stress corrosion cracking due to the presence of oxygen in the coolant, high residual stresses and some sensitization of metal adjacent to welds. They found that such cracks were expected to be in the heat-affected zones adjacent to welds and not to occur outside these zones where sensitization has not taken place, provided the pipe material is properly annealed. They concluded that the most promising solution to intergranular stress corrosion cracking of austenitic stainless steel is to replace susceptible piping with material that will be less adversely affected by oxygenated water.

The General Electric Company formed its own task group to investigate the causes of stainless steel pipe cracking. They have concluded that our task group findings are in general agreement with the results of their own findings. Further, the General Electric Company has made the commitment that all areas where intergranular stress corrosion cracking has been found to occur in the past 10 to 15 years of boiling water reactor operation will be replaced with materials which are not susceptible to intergranular stress corrosion or qualified fabrication or processing techniques will be used to prevent sensitized unstabilized wrought austenitic stainless steel from being exposed to the reactor coolant.

In the short term, the General Electric Company proposes to employ service proven materials and metallurgical structures made nonsusceptible to intergranular stress corrosion cracking through special processing steps, thereby avoiding any risks associated with immediate changes to new materials. They propose the following processes which have been proven by full size pipe tests:

(1) Solution heat treatment.

(2) Corrosic resistant cladding.

(3) Welding heat input control.

(4) Reduced inside (diameter) temperature welding.

These processes will be applied to newly fabricated Type 304 stainless steel weld joints going into service on service sensitive lines - those pipe runs with a previous history of stress corrosion cracking (core spray, recirculation system bypass and control rod drive return lines). In addition, they propose to apply the same process improvements to other Type 304 stainless steel lines that have not experienced problems in the past (such as recirculation system lines).

Solution heat treatment of stainless steel welds places them in a metallurgical condition that is nonsusceptible to stress corrosion cracking. Solution heat treatment is applied . the pipe fabricator's shop in accordance with procedures that include rapid cooling through the 1800 to 800 degrees Fahrenheit sensitization range. This prevents carbide precipitation.

Where solution heat treatment is not practical, weld joint inside surfaces will be protected with corrosion resistant cladding prior to making the final weld. Tests now in progress indicate that a minimum of eight percent ferrite is effective in preventing corrosion.

Where neither of the above processes is possible, the General Electric Company proposes to apply welding heat input control for both shop and field welds. A special process is being developed that reduces the inside surface temperature of the pipe subsequent to the root pass. This will minimize sensitization and reduce residual stresses in the inside diameter of piping. Accelerated stress corrosion tests on actual pipes are being performed to qualify these processes.

The General Electric Company has taken steps to solution heat treat all piping in the recirculation lines where the spools have not yet been fabricated. This program is limited by the current pipe fabricator's furnace and quench tank capabilities. They intend to implement other process steps as they are proven and become practical. Systematic qualification will precede the introduction of any changes.

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The General Electric Company has identified changes in materials as their long term solution to the austenitic stainless steel pipe cracking problem. The long term pipe replacement material qualification program includes laboratory experiments, testing of welded sections, vendor qualification, American Society of Mechanical Engineers Boiler and Pressure Vessel Code qualification and trial introduction into boiling water reactor service. As successful performance of a new material is demonstrated, use of it for pipe applications will be expanded with the ultimate goal of complete commitment for boiling water reactors.

We concur with the objectives of the General Electric Company programs and we find the progress and commitments of the General Electric Company acceptable. We will report on the progress of the short term and long term programs at the final design stage of review.

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6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.5 External Pressure Design

In the Safety Evaluation Report, we stated that the containment vacuum relief system contained four, 36-inch lines connecting the containment to the shield building annulus and that each line had a check valve in series with a motor-operated globe valve. The design was revised to incorporate two, 24-inch lines with each line having a check valve in series with an air-operated butterfly valve.

In Supplement No. 1 to the Safety Evaluation Report we stated that the General Electric Company had revised their vacuum breaker sizing analysis and that we had this revision under review. We have since completed our review of the vacuum breaker analysis. The limiting containment operating condition of 90 degrees Fahrenheit with an associated relative humidity of 20 percent was selected by the General Electric Company to establish the vacuum breaker sizing. Based on these operating conditions, the General Electric Company has calculated that an external pressure differential of 0.72 pounds per square inch differential will not be exceeded during postulated accidents or transients. Using the CONTEMPT LT 26 computer program, we have performed confirmatory calculations which are in reasonable agreement with the General Electric Company results. The CONTEMPT calculation assumes instantaneous vaporization of the containment spray water (inadvertent containment spray actuation event) until the containment reaches 100 percent relative humidity.

Based on our review and our confirmatory calculations, we find the external containment design pressure and the containment vacuum breaker sizing acceptable provided that, during normal plant operation, the containment temperature and relative humidity are maintained within the containment temperature and relative humidity limits used in the General Electric Company's vacuum breaker sizing analysis. (These limits were a relative humidity of 20 percent at 90 degrees Fahrenheit and a relative humidity of 30 percent at 105 degrees Fahrenheit, with the relative humidity varying linearly between these two temperatures). At the final design stage of review, we will develop appropriate requirements for the Technical Specifications for Operation for plants referencing the GESSAR-238 Nuclear Island design to assure that these limits on containment temperature and relative humidity are maintained.

6.3 Emergency Core Cooling System

6.3.2 Performance Evaluation

In the Safety Evaluation Report we stated that the emergency core cooling system for the GESSAR-238 Nuclear Island design meets all the criteria of paragraph 50.46 and the requirements of Appendix K to 10 CFR 50, and is acceptable.

Subsequent to the issuance of Supplement No 1 to the Safety Evaluation Report, the General Electric Company informed us that certain calculational errors had been discovered which may affect the performance evaluation of the GESSAR-238 Nuclear Island emergency core cooling system. We have been advised by the General Electric Company that the overall effect of these errors is small. We are pursuing further review of this new information obtained from the General Electric Company and will issue a supplement reporting our conclusions on this matter prior to a decision on the issuance of a construction permit for any application referencing the GESSAR-238 Nuclear Island design.

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7.0 INSTRUMENTATION AND CONTROLS

7.1 Introduction

In Supplement No. 1 to the Safety Evaluation Report on the GESSAR-238 Nuclear Island design, we identified 10 items which remained outstanding in our review of instrumentation and controls systems and for which we required resolution prior to a decision on the issuance of a construction permit for an application referencing the GESSAR-238 Nuclear Island design.

Since the issuance of Supplement No. 1 to the Safety Evaluation Report, the General Electric Company has provided us with sufficient information to enable us to complete our review of these items. The results of our review efforts are summarized in the following sections.

7.2 Reactor Trip System

In Supplement No. 1 to the Safety Evaluation Report we stated that we would require formal documentation of the reactor vessel high water level input to the reactor trip system prior to a decision on the issuance of a construction permit for an application referencing the GESSAR-238 Nuclear Island design.

The General Electric Company has since provided the preliminary design information for the reactor vessel high water level trip. We have reviewed this information and concluded that this input is implemented in a manner identical to the other safety-grade trip inputs to the reactor trip system and is therefore acceptable.

7.3 Engineered Safety Features Systems

7.3.2 Emergency Core Cooling System

7.3.2.2 Automatic Depressurization System

In Supplement No. 1 to the Safety Evaluation Report, we stated that the General Electric Company had performed reliability analyses for alternate conceptual designs for the automatic depressurization system which included on-line testing capabilities of the pilot solenoid actuation valves. We further stated that we were unable to concur with their conclusion because we felt that some of the assumptions used in their analyses had not been adequately justified.

We therefore required that the General Electric Company develop a program to establish the reliability of the pilot solenoid valves that will be used in the automatic depressurization system and the pressure relief system of the GESSAR-238 Nuclear Island design.

The General Electric Company has committed to such a program and agreed that the program is to be submitted to the staff for review and approval prior to its implementation. The results of the program are to be submitted at the final design phase of review. The results of the program will be utilized to demonstrate that the test interval of the solenoids is compatible with the longest anticipated refueling outage for the GESSAR-238 Nuclear Island design.

We conclude that a testing program coupled with the aforementioned General Electric Company commitments can be utilized to demonstrate the failure rates assumed in the General Electric Company analyses and thereby establish the test interval for the pilot solenoid valves. We therefore consider these commitments acceptable.

7.3.2.3 Low Pressure Core Spray and Low Pressure Coolant Injection Systems

In Supplement No. 1 to the Safety Evaluation Report we stated that the General Electric Company had agreed to provide a pressure interlock on the motor-operated valves in the low pressure core spray and low pressure coolant injection systems and that the interlock would be diverse, although the General Electric Company had not provided preliminary design information to demonstrate diversity.

The General Electric Company has since demonstrated diversity in design with the preliminary design information provided in Amendment 45 to the GESSAR-238 Nuclear Island Safety Analysis Report. We therefore conclude that the preliminary designs of the low pressure core spray and low pressure coolant injection systems are acceptable.

7.3.4 Essential Service Water System

The essential service water system is separated into three divisions to provide the required support for the engineered safety features of a plant utilizing the GESSAR-238 Nuclear Island design (Division 1 and 2, and High Pressure Core Spray). The instrumentation and control for this system is part of the solid state protection system supplied by the General Electric Company. The General Electric Company has committed to designing all the auxiliary supporting systems, such as the essential service water system, in accordance with the same criteria applied in the design of the supported safety system.

As required in our previous evaluation of this area of the design, the General Electric Company has submitted preliminary design information, including elementary wiring diagrams for the instrumentation and control system for the GESSAR-238 Nuclear Island scope of the essential service water system. We have reviewed this information in conjunction with the design criteria and design bases established during the Preliminary Design Approval review. Based on our review of this information, we conclude that the preliminary design of the essential service water system appropriately implements the previously approved bases and criteria and is therefore acceptable.

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7.3.5 Flammability Control System

During our initial Preliminary Design Approval review, we concluded that the bases and criteria for the instrumentation and control system associated with the flammability control system were acceptable and we would complete our review of the preliminary design as a post-PDA item.

The General Electric Company has since provided the preliminary design information for the flammability control system. Based on our review of this information, we conclude that the preliminary design of the flammability control system appropriately implements the previously approved bases and criteria and is therefore acceptable.

7.3.6 Standby Gas Treatment System

During our initial Preliminary Design Approval review, we reviewed the description of the proposed design of the instrumentation and controls for the standby gas treatment system, the simplified functional control diagram provided in Figure 7.3-20 of the GESSAR-238 Nuclear Island Safety Analysis Report, and the proposed design criteria identified in Figure 7.1-2 of the GESSAR-238 Nuclear Island Safety Analysis Report. We concluded that the proposed design criteria were unacceptable because the Institute of Electrical and Electronics Engineers Standard 308-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations" (as modified by Regulatory Guide 1.32, "Use of IEEE Std. 308-1971, 'Criteria for Class IE Electric Systems for Nuclear Power Generating Stations'"), Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," and Criteria 17 and 18 of the General Design Criteria were not included. These criteria had been identified by the General Electric Company as applicable to other engineered safety feature systems and we concluded that they were equally applicable to the standby gas treatment system.

The General Electric Company has since identified these aforementioned design criteria as applicable to the standby gas treatment system. We have reviewed the proposed logic diagram for this system and concluded that the bases and criteria were appropriately implemented. We therefore conclude that the instrumentation and control system associated with the standby gas treatment system is acceptable.

7.3.7 Suppression Pool Makeup System

During our initial Preliminary Design Approval review, we reviewed the proposed conceptual design, the proposed design criteria and the General Electric Company's evaluation of the effects of inadvertent dump of the upper pool. We have also reviewed the proposed logic diagrams for the suppression pool makeup system. We have concluded that the proposed design is <u>not</u> susceptible to single failures that could cause inadvertent dumping of the upper pool. Based on the above evaluation, we conclude that the preliminary design of the suppression pool makeup system appropriately implements the previously approved bases and criteria and is therefore acceptable.

7.3.8 Containment Spray System

During our initial Preliminary Design Approval review, we concluded that the proposed design criteria for the containment spray system were generally acceptable for proceeding with the development of a design and required the submittal of the preliminary design as a post-PDA review item.

The General Electric Company has submitted the elementary wiring diagrams for the residual heat removal system which included the containment spray mode of operation. We have reviewed this preliminary design information in conjunction with the design criteria and design bases established during the Preliminary Design Approval review. We have also reviewed the General Electric Company's evaluation on the acceptability of the effects of simultaneous actuation of both the spray systems.

Based on the above review, we conclude that the preliminary design of the containment spray system appropriately implements the previously approved bases and criteria and is therefore acceptable.

7.6 All Other Systems Required for Safety

7.6.2 Reactor Pressure Relief Instrumentation

In Supplement No. 1 to the Safety Evaluation Report, we stated that the resolution achieved for the testing of the solenoid valves on the automatic depressurization system would also be applied to the solenoid valves on the reactor pressure relief system.

In Section 7.3.2.2 of this supplement, we described the program commitment which resolved this issue for the automatic depressurization system. The General Electric Company has agreed to extend this program to cover the solenoid valves on the reactor pressure relief system. We consider this commitment acceptable.

7.6.3 Recirculation Pump Trip System

In Supplement No. 1 to the Safety Evaluation Report, we stated that we would require the General Electric Company to submit the preliminary design information (elementary diagrams) for the recirculation pump trip system prior to a decision on the issuance of a construction permit for an application referencing the GESSAR-238 Nuclear Island design.

The General Electric Company has since provided this preliminary design information as a part of the elementary wiring diagrams for the reactor trip system. We have reviewed this information and conclude that the preliminary design appropriately implements the approved bases and criteria and is therefore acceptable.

7.8 Instrumentation and Control Interfaces

As stated in our Safety Evaluation Report, we were unable at that time to complete our review of the Nuclear Island/blanace-of-plant interfaces because not all of the preliminary design information had been submitted. During our review of the preliminary design information submitted after the issuance of the Preliminary Design Approval, we reviewed the Nuclear Island/balance-of-plant interfaces which were necessary to insure that the GESSAR-238 Nuclear Island instrumentation and control systems perform, as required, under normal, transient and accident conditions. Our review relied heavily on the previous interface review conducted on the GESSAR-251 Nuclear Steam Supply System application (Docket No. STN 50-531).

We required the General Electric Company to provide, in Section 1.10 of the GESSAR-238 Nuclear Island Safety Analysis Report, references to the preliminary design information which contain the required interface information. The General Electric Company has provided this information on a system by system basis in Section 1.10 of the GESSAR-238 Nuclear Island Safety Analysis Report, as amended.

We have reviewed this information and the referenced design information and determined that the General Electric Company has sufficiently identified interfaces between the GESSAR-238 Nuclear Island and balance-of-plant instrumentation and control systems. We have supplemented this information with the additional requirements specified in Table 7-1 of this supplement. Table 7-1 specifies the requirements that an applicant utilizing the GESSAR-238 Nuclear Island design must address in order to demonstrate that the total plant design (Nuclear Island coupled with balance-of-plant) conforms to the Nuclear Regulatory Commission's requirements and those requirements specified in the GESSAR-238 Nuclear Island Safety Analysis Report.

We have found that the interface information provided in the GESSAR-238 Nuclear Island Safety Analysis Report, as supplemented with the criteria identified in Table 7-1 of this supplement, provide reasonable assurance that the instrumentation and control for the total plant design can be accomplished in a manner that will validate the assumptions made in the GESSAR-238 Nuclear Island review. Based on the above, we conclude that the instrumentation and control systems interfaces specified for the GESSAR-238 Nuclear Island design can be implemented in a satisfactory manner and hence are acceptable. We will review the acceptability of the implementation of the above interface requirements in each application referencing the GESSAR-238 Nuclear Island design.

TABLE 7-1

GESSAR-238 NUCLEAR ISLAND
INTERFACE ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS

CRITERIA	TITLE	REACTOR PROTECTION SYSTEM	ENGINEERED SAFETY FEATURES AND AUXILIARY SUPPORT SYSTEMS
10 CFR PART 50	CONTENT OF APPLICATION		
	TECHNICAL INFORMATION	X	X
	TECHNICAL SPECIFICATIONS	X	X
	CODES AND STANDARDS	X	X
GENERAL DESIGN CRITERIA (GDC APPENDIX A TO 10 CFR PART 50)	(SEE STANDARD REVIEW PLAN TABLE 7-1 FOR SPECIFIC GDC & TITLE)	X	X
INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS STANDARDS			
IEEE STD 279-1971 (ANSI N42.7-1972)		X	X
IEEE STD 308-1971			X
IEEE STD 317-1972		X	X
IEEE STD 323-1974		X	X
IEEE STD 336-1971		X	X
IEEE STD 338-1971		X	X
IEEE STD 344-1971		X*	X
IEEE STD 379-1972		X	X
IEEE STD 384-1974		X	X
REGULATORY GUIDES (RGs)			
RG 1.6			X
RG 1.11			X

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TABLE 7-1 (Continued)

GESSAR-238 NUCLEAR ISLAND
INTERFACE ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS

CRITERIA	TITLE	REACTOR PROTECTION SYSTEM	ENGINEERED SAFETY FEATURES AND AUXILIARY SUPPORT SYSTEMS
REGULATORY GUIDES (Cont'd)			
RG 1.22		X	X
RG 1.29		X*	X
RG 1.30		X	X
RG 1.32			X
RG 1.47		X	X
RG 1.53		X	X
RG 1.62			X
RG 1.63			X
RG 1.70		X	X
RG 1.75		X	X
BRANCH TECHNICAL POSITIONS			
BTP EICSB 1		X	X
BTP EICSB 9		X	X
BTP EICSB 10			X
BTP EICSB 18			X
BTP EICSB 19			
BTP EICSB 21			
BTP EICSB 22		X	X
BTP EICSB 23			
BTP EICSB 24		X	X
BTP EICSB 25			X
BTP EICSB 26		X	
BTP EICSB 27			X

*Turbine trip inputs to reactor protection system are exempted from seismic requirements as outlined in Section 7.2 of Supplement No. 1 to Safety Evaluation Report.

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8.0 ELECTRIC POWER SYSTEMS

8.1 Introduction

During our early Preliminary Design Approval review effort, we believed that a major portion of the information contained in Section 8.0 of GESSAR-238 Nuclear Island Safety Analysis Report was only typical and outside the GESSAR-238 Nuclear Island scope. Subsequently, but prior to the issuance of the Preliminary Design Approval, the General Electric Company informed us that Section 8.3, "Onsite Power System" was part of the GESSAR-238 Nuclear Island design scope. Therefore, as part of the post-PDA review, we required the General Electric Company to provide additional preliminary design information on all of the areas in their scope of design and to provide adequate interface information for all of the areas outside the GESSAR-238 Nuclear Island scope, where design compatibility was necessary to demonstrate the acceptability of the GESSAR-238 Nuclear Island design.

8.2 Offsite Power System

The offsite power system is totally the responsibility of an applicant referencing the GESSAR-238 Nuclear Island design as part of their construction permit application. Therefore, our review of the offsite electrical power system design involved only the identification of the interfaces for this system to be satisfied by the applicant referencing the GESSAR-238 Nuclear Island design. The General Electric Company has identified, within the GESSAR-238 Nuclear Island scope, provisions for feeders from the offsite system to each of the engineered safety features Division 1 and Division 2 onsite alternating current distribution systems. In addition, the General Electric Company has identified the provision for the feeder from the offsite system to the Division 3/high pressure core spray system onsite alternating current distribution system.

8.3 Onsite Power System

8.3.1 Onsite Alternating Current System

8.3.1.1 Division 1 and Division 2 Standby Alternating Current Power System

The onsite standby alternating current power system for the GESSAR-238 Nuclear Island will consist of two independent distribution systems and associated onsite power sources to support the engineered safety features and auxiliary support systems of the Nuclear Island and for the necessary auxiliary supporting features of an application utilizing the GESSAR-238 Nuclear Island design. Normal power will be supplied by the offsite power system (see Section 8.2 of this supplement) and, under a condition of a loss of the offsite power source, each of the two distribution systems will receive power from its associated onsite diesel generator unit.

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Each distribution system (Division 1 and 2) will include 6900, 480, and 120 volt alternating current load centers to provide power to the associated divisional safety related loads. Each standby diesel generator will be operated independently and located in separated seismic Category I structures with separate ventilation air intake and diesel engine exhausts.

The fuel oil storage and transfer system for the diesel generator is partially in the scope of an applicant utilizing the GESSAR-238 Nuclear Island design; therefore, interface information was required to complete our review. The General Electric Company has identified the interface information in the GESSAR-238 Nuclear Island Safety Analysis Report, Sections 1.10 and 8.0, as amended.

With regard to the proposed Division 1 and 2 standby power sources, we required the General Electric Company to conform to our position outlined in Standard Review Plan Appendix 7-A BTP EICSB 2, "Diesel-Generator Reliability Qualification Testing." The General Electric Company has stated their commitment to a qualification program in conformance with this position.

The General Electric Company has stated that as the detailed design of the GESSAR-238 Nuclear Island progresses, the diesel generator rating may be revised. The final sizing of the diesel generator will be based on a rating that will be consistent with the recommendation of Regulatory Guide 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies." Since functional requirements of the onsite alternating current power system are within the design responsibility of the General Electric Company and the overall implementation of this system requires interfaces to include preliminary design information including logic diagrams, to demonstrate that the design requirements of the total onsite alternating current power system and the attendant interfaces (such as fuel storage and transfer system, and applicant supplied loads) can be satisfactorily implemented by an applicant utilizing the GESSAR-238 Nuclear Island design.

We have reviewed the proposed design for the Division 1 and 2 onsite alternating current power systems, including the interface information (see Section 8.5 of this supplement) and conclude that they are acceptable.

8.3.1.2 Division 3/High Pressure Core Spray System Power System

The GESSAR-238 Nuclear Island Safety Analysis Report referenced Topical Report NEDO-10905, "High Pressure Core Spray Power Supply Unit," for the design of the third division onsite alternating current and direct current power systems. The power system described in Topical Report NEDO-10905, which is presently under review by the staff, is dedicated solely for the operation of the high pressure core spray system and associated support systems. We will require that applicants who reference the GESSAR-238 Nuclear Island implement the resolution of that topical report on their specific application.

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During our review of the GESSAR-238 Nuclear Island, the General Electric Company proposed to utilize the power supply described in Topical Report NEDO-10905 to provide power to other Division 3 instrumentation and control systems.

We were concerned that the power system that was proposed as solely dedicated to the high pressure core spray function was now being utilized for other plant safety functions. Our main concern was that the exceptions to Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Systems," in the topical report for the power supply source (high pressure core spray) diesel generator may not remain a valid approach for the GESSAR-238 Nuclear Island electrical system. Therefore, the General Electric Company was requested to identify in the GESSAR-238 Nuclear Island Safety Analysis Report all deviations (such as that noted above) in the proposed GESSAR-238 Nuclear Island design from the design information presented in the topical report. In addition, the General Electric Company was requested to analyze the effects on plant safety resulting from these deviations.

The General Electric Company stated that the Division 3 logic power supply (120 volt vital alternating current) is the only plant function connected to the high pressure core spray power system and that this was desirable in order to maintain divisional assignments within the plant, consistent with the design change to a four division vital alternating current power system. In the proposed design there are no other plant loads connected to the high pressure core spray power system which have functions other than those dedicated to the high pressure core spray function.

With regard to the Division 3 logic power supply, the General Electric Company has stated that the normal power supply to the Division 3 vital alternating current power system will be supplied from the 125 volt direct current bus through a direct current to alternating current inverter. In addition, the Division 3 battery source for the Nuclear Island has been sized to supply all the Division 3 loads (including high pressure core spray functions and other Division 3 instrumentation and control functions) for a period of two hours without any aid from the battery charger. Therefore, the Division 3/high pressure core spray diesel generator is not totally depended upon to supply the power for the Division 3 vital alternating current system during this period. From the results of their analyses, the General Electric Company has concluded that the voltage and frequency dips during the starting of the high pressure core spray motor or Division 3 diesel generator will have no adverse effect on the Division 3 vital alternating current power system.

In Amendment 45 to the GESSAR-238 Nuclear Island Safety Analysis Report, the General Electric Company demonstrated a design modification which would disconnect the vital alternating current power system from the high pressure core spray alternating current system during the period when the above referred voltage and frequency dips occur on the high pressure core spray alternating current power system. We will require that the validity of their analysis regarding this aspect of the design be demonstrated as part of the pre-operational testing of the Division 3/high pressure core spray power

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system in an application referencing the GESSAR-238 Nuclear Island Design. If the results of the General Electric Company analyses cannot be satisfactorily demonstrated, we will require that the General Electric Company institute the above cited design modification and provide the details of the design modification for our evaluation at the final design stage of review.

We have evaluated the information submitted by the General Electric Company for this area of the GESSAR-238 Nuclear Island power system and concluded that the utilization of high pressure core spray system alternating current power supply for other Division 3 instrumentation and control functions identified in the proposed design will not adversely affect the intended safety function of the Division 3 vital alternating current power system.

Based on the above evaluation and requirements stipulated herein, and the commitment to the topical report resolution, we conclude that the GESSAR-238 Nuclear Island Division 3/high pressure core spray power system is acceptable.

8.3.2 Direct Current Power System

The onsite direct current power system of the GESSAR-238 Nuclear Island will consist of four redundant and independent 125 volt direct current supplies, each consisting of a battery with an associated charger and direct current distribution system. The ampere-hour capacity of each 125 volt direct current battery will be suitable for supplying all safety related loads for a minimum of two hours without the aid of the battery charger. In addition each battery charger will have sufficient capacity for steady state operation of the connected loads while maintaining the battery in a fully charged state.

The alternating current source for the battery charger of each direct current division will be supplied by the associated alternating current division with the exception of the Division 4 direct current system. The source for the Division 4 direct current system battery charger is the Division 2 alternating current system. This interdependence had previously been evaluated in conjunction with the GESSAR-251 Nuclear Steam Supply System review (Docket No. STN 50-531) and found acceptable.

Non-safety related loads will not be supplied by the Class IE 125 volt direct current system.

The proposed 125 volt direct current power system is in conformance with the requirements of the Institute of Electrical and Electronics Engineers Standard 308-1971, "Criteria for Class IE Electric Systems for Nuclear Generating Stations" and with the recommendation of Regulatory Guides 1.6, "Independence Between Redundant Standby (On-Site) Power Sources and Between Their Distribution Systems," and 1.32, "Use of IEEE Std. 308-1971; Criteria for Class IE Electric Systems for Nuclear Generating Stations."

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Four redundant 120 volt alternating current vital bus systems will be provided to supply power to the four channel solid state protection system (which includes the reactor trip system and enginnered safety features actuation system). The system is designated the nuclear system protection system power system and will be designed in accordance with the Institute of Electrical and Electronics Engineers Standard 308-1971. Each vital bus will be fed from an associated inverter which, in turn, will be supplied by the corresponding direct current system. Also, each bus will be fed through a 480/120 volt transformer as an alternate source in case of inverter failure.

We have reviewed the design description, design criteria, design bases, and single line diagrams for the direct current onsite power system and the associated 120 volt alternating current vital bus system. We have concluded that the proposed design for the direct current system and 120 volt vital alternating current system satisfies the Nuclear Regulatory Commission's requirements, as described above, and is acceptable.

8.4 Physical Independence of Electrical Systems

The General Electric Company has committed to Regulatory Guide 1.75, "Physical Independence of Electrical Systems," as the primary design objective for the physical independence of the plant's electrical system. We have reviewed the preliminary design of the electrical system to the guidelines of this Regulatory Guide and concluded that it is acceptable.

8.5 Interface Requirements for the Electric Power Systems

During the review of the preliminary design information on the electrical power system, we noted that the General Electric Company had not provided a clear definition for the electrical power interfaces in the GESSAR-238 Nuclear Island Safety Analysis Report. We therefore required that they amend Section 1.10 to include a reference to all applicable interface information in Section 8.0.

The General Electric Company has provided additional interface information including a system by system reference to all applicable preliminary design interface information. We have supplemented this information with the additional interface requirements specified in Table 8-1 of this supplement. Table 8-1 specifies the requirements that an applicant utilizing the GESSAR-238 Nuclear Island design must address in order to demonstrate that the total plant design (Nuclear Island coupled with balance-of-plant) conforms to the Nuclear Regulatory Commission's requirements and those requirements specified in the GESSAR-238 Nuclear Island Safety Analysis Report for electrical power systems.

We have found that the interface information provided by the General Electric Company, as supplemented with the criteria identified in Table 8-1, provide reasonable assurance that the electrical power system of the total plant design can be accomplished in a manner that will validate the assumptions made in the GESSAR-238 Nuclear Island review. Based on the above, we conclude that the electrical power system interface specified

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TABLE 8-1

INTERFACE ACCEPTANCE CRITERIA FOR ELECTRIC POWER SYSTEMS

CRITERIA	TITLE	OFFSITE POWER SYSTEM	ONSITE ALTERNATING CURRENT POWER SYSTEM	ONSITE DIRECT CURRENT POWER SYSTEM
	CONTENT OF APPLICATION			
	TECHNICAL INFORMATION	X	X	X
	TECHNICAL SPECIFICATIONS	X	X	X
10 CFR Part 50	CODES AND STANDARDS	X	X	X
GENERAL DESIGN CRITERIA (GDC), APPENDIX A TO 10 CFR Part 50	(SEE STANDARD REVIEW PLAN TABLE 8-1 FOR SPECIFIC GDC & TITLE)	X	X	X
IEEE (INSTITUTE OF ELECTRICAL AND ELECTRONIC ENGINEERS) STANDARDS				
IEEE STD 279-1971			X	X
IEEE STD 308-1971		X	X	X
IEEE STD 317-1972			X	X
IEEE STD 323-1974			X	X
IEEE STD 334-1974			X	
IEEE STD 336-1974		X	X	X
IEEE STD 338-1971			X	X
IEEE STD 344-1971			X	X
IEEE STD 379-1972			X	X
IEEE STD 382-1972			X	X
IEEE STD 383-1974			X	X
IEEE STD 384-1974			X	X
IEEE STD 387-1972			X	X
IEEE STD 450-1972				X
REGULATORY GUIDES (RGs)				
RG 1.6			X	X
RG 1.9			X	
RG 1.22		X	X	X
RG 1.29			X	X
RG 1.30		X	X	X
RG 1.32		X		X
RG 1.40			X	
RG 1.41		X	X	X
RG 1.47		X	X	X
RG 1.53			X	X
RG 1.62			X	
RG 1.63		X	X	X
RG 1.68		X	X	X
RG 1.70		X	X	X
RG 1.73			X	X
RG 1.75			X	X
BRANCH TECHNICAL POSITIONS (BTPs)				
BTP EICSB 1			X	X
BTP EICSB 2			X	
BTP EICSB 6				X
BTP EICSB 7			X	X
BTP EICSB 8			X	
BTP EICSB 10			X	X
BTP EICSB 11		X		
EICSB 17				
EICSB 21		X		X
BTP EICSB 27			X	X

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in the GESSAR-238 Nuclear Island Safety Analysis Report can be implemented in a satisfactory manner and hence are acceptable. We will review the satisfactory implementation of the above interface requirements in each application referencing the GESSAR-238 Nuclear Island design.

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15.0 ACCIDENT ANALYSIS

15.3 Design Basis Accidents

In the Safety Evaluation Report we noted that the calculated doses resulting from a postulated fuel handling accident in the spent fuel pool area in the fuel building (outside containment) were well below the exposure guidelines of 10 CFR Part 100. It was noted by the General Electric Company in the GESSAR-238 Nuclear Island Safety Analysis Report that fuel handling accidents can also be postulated inside containment, but that, because the containment is isolated, accidents in the fuel building would result in higher offsite radiological doses.

Effective containment isolation requires that radiation monitors promptly detect an accidental release and that there is timely closure of isolation valves.

Based on our review, we conclude that for the GESSAR-238 Nuclear Island design a timely isolation can be effected to limit the activity released to acceptable levels. We will review the system layout in detail in the Final Safety Analysis Report review stage to assure that the provisions for mitigating the consequences of refueling accidents meet out acceptance criteria in Standard Review Plan Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," (well within the guideline values of 10 CFR Part 100) by appropriate technical specifications and radiation monitor placement, if necessary.

Since the General Electric Company proposes a system which can acceptably control the doses from this postulated accident and since any changes that may be required at the Final Safety Analysis Report stage are expected to be of a minor nature, we conclude that the proposed design is acceptable.

15.5 Failure of Inputs from Turbine Building to Reactor Protection System

In Supplement No. 1 to the Safety Evaluation Report, we stated that the failure of the reactor protection system inputs located in the turbine building coupled with a turbine bypass system failure would result in two-hour doses of 95 rem to the thyroid and 4.5 rem whole body, assuming a relative concentration value of 1×10^{-3} seconds per cubic meter, and course of accident doses of 55 rem to the thyroid and 4.5 rem whole body, assuming a relative concentration of 1.0×10^{-4} seconds per cubic meter. These dose calculations represented a very conservative upper bound estimate of the doses associated with this event since we arbitrarily assumed, in the calculations, that all of the fuel which experienced boiling transition also melted.

Subsequent to the issuance of Supplement No. 1 to the Safety Evaluation Report, the General Electric Company informed us that they had calculated that no fuel melting

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would occur during this event. We have reanalyzed this transient and calculated a margin to melt for the fuel during this transient of over 1200 degrees Fahrenheit and therefore agree with the General Electric Company's assessment.

Accordingly, we have reanalyzed the radiological consequences of this event utilizing the more appropriate assumptions contained in the Appendix to Standard Review Plan 15.4.9, "Radiological Consequences of Control Rod Drop Accident." (We conservatively assume that all fuel rods which experience boiling transition will perforate and release the activity contained in the fuel-clad gaps.)

The resulting two hour doses were 7.6 rem to the thyroid and 0.7 rem whole body, assuming a relative concentration value of 1×10^{-3} seconds per cubic meter, and a dose for the course of the event of 5.1 rem to the thyroid and 0.2 rem whole body, assuming a relative concentration value of 1×10^{-4} seconds per cubic meter.

Based on our review we conclude that the probability of this event is low enough to permit us to use the conservative accident analysis assumption and compare the offsite doses to 10 CFR Part 100 limits. We also conclude that the offsite dose consequences are a relatively small fraction of 10 CFR Part 100 limits.

This, therefore, reaffirms our original conclusions presented in Section 7.2 of Supplement No. 1 to the Safety Evaluation Report regarding the acceptability of those reactor protection system inputs originating from the turbine building.

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APPENDIX A

CONTINUATION OF CHRONOLOGY OF
RADIOLOGICAL REVIEW

September 20, 1976	We requested additional information from the General Electric Company on bypass containment leakage.
September 21-22, 1976	We met with representatives from the General Electric Company and Tennessee Valley Authority regarding the remaining GESSAR-238 Nuclear Island open items.
September 30, 1976	The Nuclear Regulatory Commission issued Supplement No. 1 to the GESSAR-238 Nuclear Island Safety Evaluation Report.
October 8, 1976	The General Electric Company provided us with additional information requested during our September 21-22, 1976 meeting.
October 18, 1976	The minutes of the September 21-22, 1976 meeting with the General Electric Company and Tennessee Valley Authority were issued.
October 18, 1976	The General Electric Company filed Amendment 45.
November 1, 1976	We requested additional information from the General Electric Company on fire protection criteria.
November 3, 1976	The General Electric Company responsed to our September 20, 1976 request for information on bypass containment leakage.
November 9, 1976	The General Electric Company committed to the staff's position on automatic depressurization system testability and recirculation pump seal leakage.
November 9, 1976	The General Electric Company provided us with additional information on the high pressure core spray power system.
November 24, 1976	The General Electric Company provided us with additional information on the containment vaccum breaker sizing analysis.

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APPENDIX E

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS-GENERIC MATTERS

The Advisory Committee on Reactor Safeguards (Committee) periodically issues a report listing various generic matters applicable to large light-water reactors. These are items which the Committee and the Commission's staff, while finding present plant design acceptable, believe have the potential of adding to the overall safety margin of nuclear power plants, and as such should be considered for application to the extent reasonable and practicable as solutions are found, recognizing that such solutions may occur after completion of the plant. This is consistent with our continuing efforts toward reducing still further the already small risk to the public health and safety from nuclear power plants. The most recent such report concerning these generic items was issued to Commission Chairman Rowden on April 16, 1976 in a letter from Committee Chairman D. Moeller.

The status of staff efforts leading to resolution of all these generic matters is contained in our Status Report on Generic Items periodically transmitted to the Committee. The latest such Status Report is contained in a letter from B. Rusche to M. Bender dated January 31, 1977.

For many of the items we have provided in this report specific discussions particularizing for the proposed facility the generic status in the Status Report. These items are listed below with the appropriate section numbers of this report where such discussions are to be found. The group numbering corresponds to that in the April 16, 1976 report of the Committee.

For those items applicable to the proposed facility which have not progressed to where specific action can be initiated relevant to individual plants, our Status Report on Generic Items referred to above provides the appropriate information.

Group II

1. Turbine Missiles - Section 3.5.3.

2. Monitoring for Excessive or Loose Parts Inside the Pressure Vessel - Section 5.2.

3. Common Mode Failures - Section 15.4 and Section 15.4 of Appendix A.

4. BWR Recirculation Pump Overspeed During a Loss-of-Coolant Accident - Section 5.4.1 of Appendix A.

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5. Instrumentation to Follow the Course of an Accident - Section 7.4 of Appendix A.

Group 11-A

1. Pressure in Containment Following a Loss-of-Coolant Accident - Section 6.2.1.7 of Appendix A.

2. Rupture of High Pressure Lines Outside Containment - Section 3.6.2.

3. Isolating Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary - Section 7.3.2.3 of Appendix A.

Group 11-B

1. Qualification of New Fuel Geometries - Section 1.8.1 of Appendix A.

2. Behavior of BWR Mark III Containment - Section 6.2.2 and Section 6.2.1 of Appendix A.

3. Stress Corrosion Cracking in BWR Piping - Section 5.2.1 of Appendix D.

Group 11-C

1. Fire Protection - Section 9.4.1.

2. Design Features to Control Sabotage - Section 13.6.

3. Reactor Vessel Supports - Section 5.2.1 of Appendix D.

4. Maintenance and Inspection of Plants - Section 12.2.

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APPENDIX F

BIBLIOGRAPHY

NOTE: Documents referenced in or use` to prepare this Safety Evaluation Report, excluding those listed in the Preliminary Safety Analysis Report, may be obtained at the source stated in the Bibliography or, where no specific source is given, at most major public librarie‹. Correspondence between the Commission and the applicant (Preliminary Safety Analysis Report, Environmental Report, and applic‹tion) and Commission Rules and Regulations and Regulatory Guides may be inspected at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. Correspondence between the Commission and the applicant may also be inspected at the Public Document Room identified in Section 1.1 of this report. Specific documents relied upon by the Commission's staff and referenced in this Safety Evaluation Report are listed as follows:

METEOROLOGY

1. American Meteorological Society: Hurricane Season Summaries from Weatherwise, published through February 1975; Weather‹ise, Inc., Princeton, N.J.

2. Cry, G. W., 1965: Tropical Cyclones of the North Atlantic Ocean. Technical Paper No. 55, U.S. Department of Commerce, Weather Bureau, Washington D.C.

3. Dunlap, D. V.: Probabilities of Extreme Snowfalls and Snow Depths, Northeast Regional Research Publication, Bulletin 821, New Jersey Agricultural Experiment Station, Rutgers University, New Brunswick, New Jersey.

4. Gross, E., 1970: The National Air Pollution Potential Forecast Program. ESSA Technical Memorandum WBTM NMC 47, National Meteorological Center, Washington, D.C.

5. Korshover, J., "Climatology of Stagnating Anticyclones East of the Rocky Mountains, 1936-1970," NOAA Technical Memorandum ERL ARL-34, Air Resources Laboratories, Silver Spring, Maryland, 1971.

6. Marshall, J. L., "Lightning Protection," John Wiley and Sons, Inc., New York, 190 pp., 1973.

7. National Severe Storms Forecast Centers, 1975: Listing of Tornadoes for the Period 1953 - 1974. National Oceanic & Atmospheric Administration. Kansas City, Mo. (Unpublished).

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GEOLOGY AND SEISMOLOGY

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18. Eardley, A. J., 1973, Tectonic Division of North America, in Gravity and Tectonics, edited by K. DeJong and R. Scholten, John Wiley Publishing Company.

19. King, P. B., 1959, The Evolution of North America, Princeton University Press, New Jersey, 189 p.

20. Rodgers, J., 1970, The Tectonics of the Appalachians, Interscience Publishers, New York, 271 p.

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APPENDIX G

CHRONOLOGY-REGULATORY REVIEW OF TENNESSEE VALLEY AUTHORITY'S
APPLICATION FOR CONSTRUCTION PERMITS FOR
PHIPPS BEND NUCLEAR PLANT UNITS NO. 1 AND NO. 2
DOCKET NO. STN 50-553 AND STN 50-554

October 1, 1975	Tennessee Valley Authority submits a tendered application consisting of general information, Preliminary Safety Analysis Report, and Antitrust Information.
October 6, 1975	Nuclear Regulatory Commission letter advising that the tendered application is under review and the review should be completed by November 3, 1975.
November 4, 1975	Nuclear Regulatory Commission letter advising that the application and Preliminary Safety Analysis Report tendered on October 1, 1975 are sufficiently complete to docket.
November 7, 1975	Tennessee Valley Authority application and Preliminary Safety Analysis Report for Phipps Bend Nuclear Power Plant, Units No. 1 & No. 2 docketed and assigned the numbers Docket STN 50-553 and STN 50-554. Incoming letter from Tennessee Valley Authority transmitting the above was dated November 6, 1975.
November 18, 1975	Nuclear Regulatory Commission letter advising of docket numbers and enclosing a copy of Federal Register Notice - Receipt of Application.
November 28, 1975	Nuclear Regulatory Commission letter to Tennessee Valley Authority regarding the review schedule for Phipps Bend.
December 5, 1975	Nuclear Regulatory Commission letter to Tennessee Valley Authority transmitting a portion of Round 1 Questions. (Effluent Treatment Systems Branch, Auxiliary and Power Conversion Systems Branch, Structural Engineering Branch, Accident Analysis Branch, Site Analysis Branch, Geology/Seismology Branch).
December 12, 1975	Tennessee Valley Authority letter transmitting Amendment No. 1 to the Preliminary Safety Analysis Report consisting of responses to acceptance review questions and the results of slope stability analyses and miscellaneous text changes.

January 14, 1976	Tennessee Valley Authority letter transmitting distribution list for correspondence.
January 19, 1976	Nuclear Regulatory Commission letter to Tennessee Valley Authority requesting additional information.
January 28, 1976	Nuclear Regulatory Commission letter to Tennessee Valley Authority requesting additional information.
January 30, 1976	Tennessee Valley Authority letter transmitting Amendment No. 2 to Preliminary Safety Analysis Report.
February 19, 1976	Meeting with Tennessee Valley Authority to discuss hydrology and meteorology.
February 23, 1976	Letter from Tennessee Valley Authority transmitting Amendment No. 3 to Preliminary Safety Analysis Report consisting of proprietary information concerning cargo carried by Southern Railway.
March 1, 1976	Issued summary of meeting held February 19, 1976.
March 8, 1976	Tennessee Valley Authority letter transmitting Amendment No. 4 to the Preliminary Safety Analysis Report.
March 1976	Tennessee Valley Authority letter transmitting working copies needed for transient flow routing.
March 23, 1976	Tennessee Valley Authority letter transmitting Amendment No. 5 to Preliminary Safety Analysis Report.
March 17/18, 1976	Meeting with Tennessee Valley Authority at site to discuss soil and foundation engineering.
March 24, 1976	Tennessee Valley Authority letter furnishing Nuclear Regulatory Commission staff with new addressee.
March 30, 1976	Tennessee Valley Authority letter submitting Environmental Report.
April 7, 1976	Issued summary of meeting held March 17-18, 1976.
April 14, 1976	Meeting with Tennessee Valley Authority to discuss the antecedents storms and hydrometeorological Report No. 45.
April 15, 1976	Meeting with Tennessee Valley Authority to discuss location of the permanent meteorological tower.

April 16, 1976	Tennessee Valley Authority letter transmitting Amendment No. 6 to the Preliminary Safety Analysis Report consisting of proprietary information concerning cargo carried by Southern Railway.
April 21, 1976	Nuclear Regulatory Commission letter to Tennessee Valley Authority transmitting copies of display ad and hearing notice.
April 21, 1976	Tennessee Valley Authority letter transmitting Amendment No. 7 to the Preliminary Safety Analysis Report consisting of text changes, responses to miscellaneous round-one questions and the results of the seismic stability evaluation of Watauga Dam.
April 26, 1976	Notice of Hearing published in the Federal Register.
April 29, 1976	Tennessee Valley Authority letter transmitting Amendment No. 8 to the Preliminary Safety Analysis Report consisting of revised responses to miscellaneous round-one questions.
May 4, 1976	Issued summary of meeting held on April 14, 1976 to discuss hydrology.
May 10, 1976	Nuclear Regulatory Commission letter to Tennessee Valley Authority advising that due to change in Parts 2, 50 and 51 applicant must mail amendments directly to Federal, State and local officials.
May 17, 1976	Nuclear Regulatory Commission letter to Tennessee Valley Authority requesting additional information.
May 19, 1976	Nuclear Regulatory Commission letter to Tennessee Valley Authority requesting additional information (2nd round questions).
May 25, 1976	Nuclear Regulatory Commission letter to Tennessee Valley Authority concerning anticipated transients without scram.
May 28, 1976	Nuclear Regulatory Commission letter advising that Amendments 3 and 6 to the Preliminary Safety Analysis Report have been withheld from public disclosure as proprietary.
June 3, 1976	Tennessee Valley Authority letter enclosing a list of activities they plan to undertake under a Limited Work Authorization.
June 7, 1976	Notice issued by the Atomic Safety and Licensing Board designating Dr. David R. Schink to serve on the Licensing Board for Phipps Bend instead of Dr. Richard F. Cole.

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June 11, 1976	Tennessee Valley Authority transmits a report concerning the 50-mile regional air flow description which fulfills its response to Appendix I.
June 21, 1976	Tennessee Valley Authority letter transmitting Amendment No. 9 to the Preliminary Safety Analysis Report consisting of revised responses to NRC questions and miscellaneous text changes.
June 23, 1976	Meeing with Tennessee Valley Authority to discuss the safe shutdown earthquake for the Phipps Bend Site.
July 6, 1976	Tennessee Valley Authority transmits Amendment No. 10 to the Preliminary Safety Analysis Report consisting of revised responses to miscellaneous round-one questions and responses to round-two questions.
July 14, 1976	Tennessee Valley Authority submits their 1975 Power Annual Report.
July 19, 1976	Issued summary of meeting held June 23, 1976.
August 2, 1976	Nuclear Regulatory Commission letter to Tennessee Valley Authority requesting a commitment to installing a loose parts monitor in each Phipps Bend unit.
August 9, 1976	Nuclear Regulatory Commission letter to Tennessee Valley Authority advising of change in regulations for submittal of required numbers of copies of Preliminary Safety Analysis Report, Amendments, Environmental Report and Application,
August 10, 1976	Appeals meeting with Tennessee Valley Authority to discuss seismology matters.
August 10, 1976	Draft Environmental Statement issued.
August 11, 1976	Appeals meeting with Tennessee Valley Authority to discuss hydrology matters.
August 19, 1976	Tennessee Valley Authority letter transmitting Amendment No. 11 concerning of physical protection of the Phipps Bend plant. Withholding from public disclosure as proprietary information was requested by applicant.
August 23, 1976	Tennessee Valley Authority letter concerning anticipated transient without scram for the Phipps Bend and Hartsville Plants.
August 26, 1976	Issued summary of appeals meeting on seismic matters held on August 10, 1976.

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September 1, 1976	Order for Special Prehearing Conference issued by Atomic Safety and Licensing Board. Conference to be held September 10, 1976.
September 1, 1976	Issued summary of appeals meeting on hydrology held August 11, 1976.
September 14, 1976	Tennessee Valley Authority letter transmitting Amendment No. 12 to the Preliminary Safety Analysis Report consisting of miscellaneous text changes, revised responses to Nuclear Regulatory Commission questions.
September 16, 1976	Tennessee Valley Authority letter transmitting a revised response to round-two question 323.17 regarding Safe Shutdown Earthquake design acceleration value to be used at Phipps Bend.
September 21, 1976	Order Following Special Prehearing Conference issued by the Atomic Safety and Licensing Board.
September 22, 1976	Tennessee Valley Authority letter advising of address changes for individuals on the service list.
September 29, 1976	Nuclear Regulatory Commission letter transmitting a request for Additional Information concerning Quality Assurance.
September 30, 1976	Nuclear Regulatory Commission letter concerning Fire Protection Evaluation.
October 21, 1976	Tennessee Valley Authority letter providing a response to question 413.1 regarding the qualification of initial test program employees.
October 27, 1976	Tennessee Valley Authority letter concerning a schedule for answering the fire protection program for Tennessee Valley Authority plants.
November 1976	Nuclear Regulatory Commission letter to all utilities concerning Industrial Security Plan.
November 15, 1976	Tennessee Valley Authority letter transmitting the 1976 Power Annual Report for Tennesse Valley Authority.
November 18, 1976	Nuclear Regulatory Commission letter requesting additional information - Accident Analysis Branch.
December 8, 1976	Nuclear Regulatory Commission letter regarding the Phipps Bend Site Seismology.

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December 8, 1976	Tennessee Valley Authority letter concerning the review of turbine missile risk assessment.
December 17, 1976	Nuclear Regulatory Commission letter concerning Fire Protection Evaluation.
December 23, 1976	Nuclear Regulatory Commission letter concerning the Phipps Bend Site Hydrology.
December 27, 1976	Tennessee Valley Authority letter transmitting Amendment No. 13 to the Preliminary Safety Analysis Report. This amendment consists of miscellaneous text changes, revised responses to NRC questions and the information submitted by Tennessee Valley Authority letters, dated 9/16/ 10/21/76, 10/26/76 and 11/1/76.
December 29, 1976	Tennessee Valley Authority letter concerning Safe Shutdown Earthquake design acceleration value for the Phipps Bend site.
December 30, 1976	Tennessee Valley Authority letter concerning the Phipps Bend Site Hydrology.
January 6, 1977	Tennessee Valley Authority letter advising that two items were erroneously included under the Construction Plant heading in their June 3, 1976 letter.
March 4, 1977	Tennessee Valley Authority letter providing additional information on the central service facility substructure, post-preliminary design approval items, and fire stops and seals.
March 15, 1977	Tennessee Valley Authority letter concerning site hydrology.
March 16, 1977	Nuclear Regulatory Commission letter concerning evaluation of fuel handling accident,
March 25, 1977	Tennessee Valley Authority letter concerning the operating basis earthquake.
March 29, 1977	Tennessee Valley Authority letter concerning flood design.
March 29, 1977	Tennessee Valley Authority letter concerning control of exclusion area.
March 31, 1977	Tennessee Valley Authority letter concerning Mark III containment dynamic loads.

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