Safety Evaluation Report

related to operation of

Wm. H. Zimmer Nuclear Power Station, Unit 1

Cincinnati Gas and Electric Company

NUREG-0528

U. S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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NUREG-0528
January 31,1979

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

CINCINNATI GAS AND ELECTRIC COMPANY

WILLIAM H. ZIMMER NUCLEAR POWER STATION

UNIT 1

DOCKET NO. 50-358

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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This report is a safety evaluation report of the application for an operating license for the William H. Zimmer Nuclear Power Station, Unit No. 1, and was prepared by the United States Nuclear Regulatory Commission staff. The application was filed by the Cincinnati Gas and Electric Company on behalf of itself, the Columbus and Southern Ohio Electric Company and the Dayton Power and Light Company. The three companies (applicant or applicants) share undivided interest in the station as owners. The Cincinnati Gas and Electric Company is authorized to act as agent for the other two and is primarily responsible for design, construction and operation of the station.

The application for a construction permit was docketed on April 7, 1970, and the construction permit was issued on October 27, 1972. The application for an operating license was docketed on September 10, 1975. The applicant has scheduled fuel loading of Unit 1 for June 1979.

A Final Safety Analysis Report was submitted as part of the application for an operating license. The facility will employ a nuclear steam supply system of the General Electric Company BWR/5 design housed in a Mark II type boiling water reactor containment. During the review, we requested that the applicant modify the application to meet certain requirements where appropriate to safety. The additional information and modifications are provided in Amendments 23 through 82 and various supplements to the Final Safety Analysis Report. The Final Safety Analysis Report, its amendments and supplements, are available for public examination at the Nuclear Regulatory Commission Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Clermont County Library, Third and Broadway Streets, Batavia, Ohio.

This report summarizes the results of the technical evaluation of the Zimmer Station which was performed by us and our consultants. The design of the station was reviewed against the construction permit criteria and our Standard Review Plan, NUREG-75/087, September 1975. In situations where the construction permit criteria did not fully meet current criteria specified in the Standard Review Plan, the differences were justified and the bases for acceptance stated in the appropriate subsections of this report. Specific Standard Review Plan sections are frequently referenced throughout the text as the basis for our acceptance. This report covers the scope of the technical matters considered by us and the generic concerns of the Advisory Committee on Reactor Safeguards (see Appendix B) in evaluating the project. The generic concerns defined by us in NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," are not discussed in this report. We have concluded that although resolution of some of these issues may provide a significant increase in assurance of the health and safety of the public, resolution is not required for licensing. We will discuss the bases for this conclusion relative to this application in future documentation. Our review of the environmental aspects of the project is presented in the Final Environmental Statement which we issued in June 1977. The statement complies with 10 CFR, Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection."

Subject to favorable resolution of the outstanding matters described in this report, and a finding by our Office of Inspection and Enforcement that the facility has been completed in conformity with the provisions of the construction permit, the application, the Act, and the rules and regulations of the Nuclear Regulatory Commission, we conclude that the Zimmer Station can be operated without endangering the health and safety of the public. The detailed conclusions are presented in Section 22.0 of this report.

The review and evaluation reported herein are only part of our continuing review of this project. We will survey the operating activities of the applicant to assure that all of our requirements are met. The station may be operated only in accordance with the terms of the operating license and the applicable regulations under our continued surveillance.

1.2 General Plant Description

1.2.1 Reactor System

The Zimmer Nuclear Power Station has a nuclear steam supply system which uses a BWR/5 class of boiling water reactor. The fission reaction which will produce the heat to power the station will take place within the reactor core. The reactor core is a vertical cylinder about 14 feet across and 12 feet high. It contains 560 Zircaloy-4 fuel channels. Each fuel channel holds 63 fuel rods and one water rod in an 8 x 8 array. This fuel rod bundle is called a fuel assembly. Each fuel rod is composed of slightly enriched uranium dioxide in the form of sintered ceramic pellets enclosed in Zircaloy-2 tubes. Some of the fuel rods have pellets with gadolinium mixed with the uranium dioxide. The gadolinium is a "burnable poison" designed to flatten the power distribution and limit the core reactivity variations throughout the core lifetime. The cladding has been evacuated, backfilled with helium and sealed by welding Zircaloy-2 end-plugs on each end during fabrication. The cladding is the first barrier to radioactive fission products produced during reactor operation.

1.2.2 Reactor Coolant System

Water flowing through the core serves as both a neutron moderator and as a coolant. Water and a water-steam mixture move upward through the core. The moving force is thermal convection (natural circulation) in combination with the force provided by the action of 20 jet pumps. Two recirculation loops motivate 10 jet pumps each. The recirculation loops, each with its recirculation pump and flow control valve, are located outside the reactor vessel. Steam generated within the core is separated from the water and dried by steam separators and dryers located in the upper region of the reactor vessel. Four main steam lines direct the steam from the reactor vessel to the turbine generator system where the energy is converted to electricity. The steam, after passing through the turbine, is condensed in the main steam condenser located beneath the turbine. The condensate is collected and, after passing through a cleanup system, is recycled by the feedwater system to the reactor vessel. The main steam condenser is cooled by recirculating water which is cooled in turn by a large natural draft cooling tower. The large cooling tower system receives makeup water from the Ohio River. The cooling tower blowdown will be returned to the Ohio River after holdup for settling and dilution with raw water. The ultimate heat sink is the Markland Pool located on the Ohio River. The Markland Pool will dissipate the maximum heat load due to the total integrated decay heat, station auxiliary system heat released and sensible heat removed from the containment and primary system following a design basis loss-of-coolant accident.

1.2.3 Reactor Coolant Pressure Boundary and Containment System

The reactor coolant pressure boundary includes the reactor vessel and branch lines out to their outermost isolation valves. It is the second barrier to the fission products produced in the core.

The containment system is a dual barrier system with a primary and secondary containment structure. The primary containment structure is a steel-lined post-tensioned concrete structure which is divided into a drywell and pressure suppression chamber by a membrane floor. The secondary containment is called the reactor building and is the gas control boundary which encloses the primary containment. The containment system is the third barrier to the fission products produced in the core.

The upper portion of the primary containment (the drywell) encloses the reactor system and is shaped like the frustum of a cone. The lower portion (the suppression chamber or wetwell) contains a suppression pool and gas space and is shaped like a cylinder. The primary containment is designed for the maximum temperature and pressure conditions that can result from a loss-of-coolant accident.

The gas control boundary is the boundary of the reactor building and also encompasses the fuel building. It is designed as a fission product barrier only and not for the pressure and temperature generated by the loss-of-coolant accident inside containment. Gas releases from the reactor building are treated by the standby gas treatment system when appropriate.

The function of the drywell is to force the steam released from a break in the reactor system pressure boundary to pass into the suppression pool to be quenched, thus reducing the resultant primary containment overpressure. Direct leakage of gas from the primary containment during the course of the loss-of-coolant accident is held up and mixed with air in the secondary containment and is then treated by the standby gas treatment system prior to being released to the outside atmosphere.

Piping restraints are installed in the containment to limit pipe whip in the event of a pipe rupture to protect safety related components and systems. Concentrations of hydrogen are controlled within an acceptable range during accident conditions to reduce the potential for hydrogen burning or explosion. Thermal recombiners reduce hydrogen concentrations by recombining the hydrogen with oxygen. The containment is isolated whenever there is a potential for the uncontrolled release of radioactivity to the outside environment.

1.2.4 Reactor Protection System

The reactor protection system is used to protect against conditions that may cause fuel failure or reactor coolant pressure boundary failure. It initiates a reactor scram in the event of an abnormal operational transient, a pressure pulse, a gross failure of fuel or a failure of the nuclear system process barrier.

1.2.5 Reactor Internal Components

The major reactor internal components are the core which includes the fuel assemblies, control rods and instrumentation; the core support structure which includes the core shroud, top guide, and core plate, the shroud head and steam separator assembly; the steam dryer assembly; and the jet pumps. Except for the Zircaloy in the reactor core, these reactor internals are fabricated of stainless steel or other corrosion resistant materials. All major internal components can be removed except the jet pump diffusers, the core shroud, the core spray spargers, and the jet pump inlet piping. The removal of the steam separators and dryers, shroud head, fuel assemblies, and control rods can be accomplished on a routine basis.

1.2.6 Reactor Control System Components

Bottom entry cruciform shaped control rods are used for normal reactivity control and scram. These control rods move vertically in the space between the fuel assemblies. They are actuated by a hydraulic mechanism which uses water as the driving fluid. Nitrogen under pressure in an accumulator is the normal driving force for rapid insertion. Each control rod is mechanically independent of the others and has its own hydraulic control system. In addition, a standby liquid control system is available to inject a boron solution into the reactor for emergency long term reactivity control.

Each control rod is in the shape of a cruciform and contains 76 stainless steel tubes (19 in each wing of the cruciform) filled with vibratory compacted boron carbide powder. The tubes are seal welded with end plugs on either end. Stainless steel balls are used to separate the tubes into individual compartments. The stainless steel balls are held in position by a slight crimp in the tube. The tubes are maintained in a cruciform array by a stainless steel sheath extending the full length of the tubes. A top handle aligns the tubes and provides structural rigidity at the top of the control rods. A bottom casting is used to provide structural rigidity and contains a parachute shaped velocity limiter. The velocity limiter is designed to limit the free fall velocity and thereby the reactivity insertion rate of a control rod in the event of a rod drop accident.

The control rod drive mechanisms used for positioning the control rods in the reactor core are mounted on the bottom head of the reactor vessel. Each control rod drive is a double acting, mechanically latched hydraulic cylinder using water as the operating fluid. The control rod drives are capable of inserting or withdrawing the control rods at a slow controlled rate (shim), as well as providing rapid insertion (scram) when required. A mechanism on each control rod drive locks the control rod in 6-inch increments of stroke over the length of the core. The control rod drive holds the control rod in distinct latch positions until the hydraulic system actuates movement to a new position.

The control rod drive hydraulic system supplies and controls the pressure and flow to and from the control rod drives through a hydraulic control unit. The water discharged from a control rod drive during a scram flows through the hydraulic control units to the scram discharge volume. The water discharged from a control rod drive during normal operation enters the reactor vessel. There is one hydraulic control unit for each control rod drive. Each hydraulic control unit imparts its pressurized water on signal to a control rod drive unit. The control rod drive then positions its control rod as required.

1.2.7 Reactor Coolant System Components

The reactor coolant system includes the reactor vessel, recirculation pumps, recirculation piping, feedwater piping out to and including the second isolation valve, main steam line piping out to and including the second isolation valve, safety relief valves, design provision to accommodate inservice inspection, and equipment supports.

The reactor vessel is a vertical cylindrical pressure vessel of welded construction which has been designed, fabricated, tested, inspected, and stamped in accordance with American Society of Mechanical Engineers Code, Section III, "Nuclear Vessels." The reactor vessel contains the core and supporting structures; the steam separators and dryers; jet pumps; control rod guide tubes; distribution lines for the feedwater, core sprays, and liquid control; incore instrumentation; and other components. The main connections to the reactor vessel include steam lines, coolant recirculation lines, feedwater lines, control rod drive and incore nuclear instrument housings, high and low pressure core spray lines, residual heat removal lines, standby liquid control line, core differential pressure line, jet pump pressure line, jet pump pressure sensing lines, water level instrumentation lines, and control rod drive system return lines. The reactor vessel is designed for a pressure of 1250 pounds per square inch gauge. The normal operating pressure is 1020 pounds per square inch gauge. The reactor vessel is fabricated of low alloy steel and is clad internally with stainless steel except for the top head and feedwater nozzles.

The reactor recirculation system consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the jet pumps located in the reactor vessel. Each external loop contains one high capacity motor driven recirculation pump, a flow control valve and two motor operated gate valves for pump maintenance.

Each of the four main steam lines includes a flow restrictor which is a venturi type nozzle insert welded into the main steam line. The flow restrictor limits the coolant blowdown rate from the reactor vessel in the event of a main steam line break outside the containment. Two main steam line isolation valves are welded in a horizontal run of each of the main steam lines. One valve is located as close as possible to the inside of the drywell wall while the other is located just outside of containment. The main steam isolation valves are designed to close within 5.5 seconds to prevent damage to the fuel by limiting the loss of reactor coolant for a major steam piping leak outside the containment.

A total of 13 safety relief valves are located on the main steam lines between the reactor vessel and the inboard main steam isolation valve. These valves protect against overpressurization of the primary system. The safety relief valves will discharge through piping to the suppression pool.

All areas of the reactor coolant system are designed to accommodate inservice inspection requirements. Engineered safety features to shut down the reactor, isolate the containment, condense steam in the containment, reflood the reactor core, remove heat for long term core cooling and restrict radioactive releases are available in the plant.

1.2.8 Emergency Core Cooling Systems

Four emergency core cooling systems are provided to maintain fuel cladding below prescribed peak temperatures in the event of a breach in the reactor coolant pressure boundary that results in a loss of reactor coolant. These systems are the high pressure core spray system, the automatic depressurization system, the low pressure core spray system, and the low pressure coolant injection system.

The high pressure core spray system consists of one independent pump and the valves and piping to deliver cooling water to a spray sparger over the core. This system provides and maintains an adequate coolant inventory inside the reactor vessel in the event of small breaks in the reactor coolant pressure boundary. The system operates independently of all other systems over the entire range of pressure differentials from greater than normal operating pressure to atmospheric. The system is initiated by high drywell pressure or low reactor water level.

The automatic depressurization system reduces reactor vessel pressure rapidly in a loss-of-coolant accident situation in which the high pressure core spray system fails to maintain the reactor vessel water level. The depressurization provided by the system enables the low pressure emergency core cooling systems to deliver cooling water to the reactor vessel. The automatic depressurization system uses some of the safety relief valves that are a part of the nuclear system pressure relief system. The safety relief valves associated with the automatic depressurization system are arranged to open during conditions indicating both that a break in the reactor coolant pressure boundary has occurred and that the high pressure core spray system is not delivering sufficient cooling water to the reactor vessel to maintain the water level above a preselected value. The automatic depressurization system is interlocked to prevent actuation unless one of the low pressure emergency core cooling systems is operating in order to ensure that adequate coolant be available to maintain reactor water level following depressurization.

The low pressure core spray system consists of one independent pump and the valves and piping to deliver cooling water to a spray sparger over the core. The system is actuated by conditions indicating that a breach exists in the reactor coolant pressure boundary but water is delivered to the core only after reactor vessel pressure is reduced. The system provides the capability to cool the fuel by spraying water over the active fuel.

The low pressure coolant injection system is an operating mode of the residual heat removal system. The low pressure coolant injection system uses the pump loops of the residual heat removal system to inject cooling water directly into the reactor vessel. The system is actuated by conditions indicating a breach in the reactor coolant pressure boundary but water is delivered to the core only after reactor vessel pressure is reduced.

1.2.9 Instrumentation and Control Systems

The instrumentation and control systems for the station include the main control room panel board including all integral equipment, instrumentation and control equipment, racks and panels, reactor control and protection systems including actuation systems, nuclear instrumentation system, and process instrumentation and control including control valves.

The reactor trip system instrumentation and control initiate an automatic reactor shutdown (scram) if monitored system variables exceed preestablished limits. This action prevents fuel damage, limits system pressure and thereby restricts the release of radioactive material.

The containment and reactor vessel isolation control system instrumentation and control initiate closure of various automatic isolation valves if monitored system variables exceed preestablished limits. This action limits the loss of coolant from the reactor vessel and minimizes the release of radioactive materials from either the reactor vessel or the containment.

The emergency core cooling systems instrumentation and control provide initiation and control of specific core cooling systems such as the high pressure core spray system, automatic depressurization system, low pressure core spray system, and the low pressure coolant injection system.

The neutron monitoring system instrumentation and control use incore neutron detectors to monitor core neutron flux. The neutron monitoring system provides signals to the reactor protection system to shut down the reactor when an overpower condition is detected. High average neutron flux change with time is used as the overpower indicator during startup and shutdown. The neutron monitoring system also provides power level indication during normal operation.

The refueling interlocks instrumentation and control serve as a backup to procedural core reactivity control during refueling operations.

The rod control and information system instrumentation and control permit the operator to manipulate control rods and determine their positions. Various interlocks are provided in the control circuitry to prevent single or multiple operator errors or equipment malfunctions from requiring the action of the reactor protection system.

The reactor vessel instrumentation monitors and transmits information concerning key reactor vessel operating parameters.

The recirculation flow control system instrumentation and control regulate the reactor recirculation pumps and valves to vary the reactor coolant flow rate through the core. Manual or automatic control of the recirculation flow control system is permitted. Recirculation pump trip circuitry also is provided for a number of anticipated transient events.

The process radiation monitoring system instrumentation and control for lines containing process liquid and gases provide for control of radioactive material releases. The main steam line radiation monitors detect gross release of fission products from the fuel and provide a trip signal resulting in reactor scram and containment isolation if monitored variables exceed preestablished limits.

The reactor core isolation cooling system instrumentation and control initiate the flow of makeup water to the reactor vessel in the event the reactor becomes isolated from the main condenser during plant operation due to closure of the main steam isolation valves.

The standby liquid control system instrumentation and control provide manual initiation of a redundant reactivity control system which can shut down the reactor from rated power to the cold condition in the event that all withdrawn control rods cannot be inserted to achieve reactor shutdown.

The reactor water cleanup system instrumentation and control provide operation of system equipment to maintain high water purity and to reduce concentration of fission products in the reactor coolant.

The leak detection system instrumentation and control use various temperature, pressure, water level and flow sensors to detect, annunciate, and isolate, if required, water and steam leakages in selected reactor systems.

The reactor shutdown cooling system instrumentation and control provide for manual initiation of cooling to remove the decay and sensible heat from the reactor vessel during normal shutdown.

1.2.10 Auxiliary Systems

The auxiliary systems for the nuclear steam supply system include special handling equipment for the fuel and reactor vessel internals, the reactor core isolation cooling system, the main steam line isolation valve leakage control system, the standby liquid control system, the reactor water cleanup system, the residual heat removal system, the pressure relief system, the pressure regulation system, and the leak detection systems.

The reactor core isolation cooling system provides makeup water to the reactor vessel when the vessel is isolated from the main condenser. The reactor core isolation cooling system uses a steam driven turbine pump unit and operates automatically to maintain adequate water level in the reactor vessel.

The main steam line isolation valve leakage control system controls the release of fission products which could leak through the closed main steam isolation valves following a postulated loss-of-coolant accident. Any leakage is drawn from the space between the inner and outer main steam isolation valves or between the outer main steam line isolation valve and the downstream block valve and vented to an area served by the standby gas treatment system.

The reactor water cleanup system recirculates a portion of the reactor coolant to remove particulate and dissolved impurities. The reactor water cleanup system consists of pumps, regenerative and non-regenerative heat exchangers, and a filter demineralizer. The system also is used to reduce the reactor water inventory as required by plant operation.

The standby liquid control system provides a redundant and diverse means to attain and maintain the reactor in a subcritical condition from any power operation to cold shutdown by injecting a solution of sodium pentaborate into the reactor vessel.

The residual heat removal system is designed for two principal modes of operation besides the safety related modes. For normal usage, the residual heat removal system functions to remove reactor decay and sensible heat during either a normal shutdown or following isolation of the reactor. In one safety related mode of operation the residual heat removal system provides heat removal capability, restores and maintains coolant inventory in the reactor vessel following the postulated loss-of-coolant accident. In the other safety related mode of operation, the residual heat removal system provides heat removal capability in the balance-of-plant containment during the post-loss-of-coolant accident period. The residual heat removal system consists of two heat exchangers, three main system pumps and associated valves, piping, controls and instrumentation.

The pressure relief system prevents overpressurization of the reactor coolant pressure boundary under the most severe operational transients. The pressure relief system consists of 13 dual purpose safety relief valves. All are mounted on the main steam lines between the reactor vessel and the innermost main steam isolation valve. All discharge through piping goes directly to the suppression pool. The valves contain auxiliary pneumatic actuators and can be operated either by automatic or remote manual controls at any pressure above atmospheric.

The pressure regulation system provides main turbine control valve and bypass valve position demands in order to maintain a nearly constant reactor pressure during normal plant operation.

1.2.11 Containment Systems and Other Structures

The containment houses the reactor, reactor coolant pressure boundary and the pressure suppression pool. The reactor building houses the engineered safety features auxiliary equipment, the fuel storage and shipping areas. Operation of the standby gas treatment system maintains a negative internal pressure so that the atmosphere within the reactor building is filtered before release to the outside environment. The standby gas treatment system exhausts to the plant vent during an accident or other abnormal operating conditions. Other structures such as the turbine building, diesel generator building, auxiliary building, radwaste building and the circulating water systems structures are described in the appropriate sections of this report.

1.2.12 Waste Management Systems

The radioactive waste management system collects, treats and disposes of radioactive waste in a controlled and safe manner. Gaseous waste disposal systems collect, monitor, purify and hold up noncondensible radioactive gases and suspended radioactive materials. Liquid radioactive wastes are collected, monitored and processed. Solid wastes are collected, drummed and shipped to licensed burial grounds.

1.3 Comparison With Similar Facility Designs

Many of the features of the Zimmer Nuclear Power Station are similar to boiling water reactor plants previously reviewed and approved by us. Some of these plants are now under construction or in operation. Use has been made of previous evaluations where feasible and appropriate in the evaluation of this station. However, the Zimmer station is the first of the BWR/5, Mark II generation of plants to be reviewed for an operating license, and therefore, we consider the Zimmer station as a lead plant with many features which have not been reviewed at the final design stage previously. Where the plant features are substantially the same as features reviewed and evaluated before, this report identifies the other facilities in the appropriate subsections. The Safety Evaluation Reports for these other facilities are published and are available for public inspection at the Nuclear Regulatory Commission Public Document Room at 1717 H Street, N.W., Washington, D.C.

A listing of comparable principal parameters and features of the Zimmer Station and other facilities of similar design are shown in Table 1-1. These facilities were selected for comparison because they have similar nuclear steam supply systems and/or containment systems.

1.4 Identification of Agents and Contractors

The General Electric Company furnished the nuclear steam supply system including the initial core. Westinghouse Electric Corporation designed, fabricated and delivered the turbine generator. The Sargent and Lundy Company performed engineering services for the balance of the nuclear station. Kaiser Engineers Incorporated was designated as the prime constructor for the station. Construction and operation responsibilities belong to the Cincinnati Gas and Electric Company, on behalf of itself and coapplicants.

1.5 Summary of Principal Review Matters

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

TABLE 1-1

COMPARISON OF PRINCIPAL DESIGN FEATURES
OF ZIMMER AND SIMILAR FACILITIES

Design Feature	Hatch Unit 2	Hatch Unit 1	Zimmer Unit 1
Rated Thermal Power, megawatts	2436	2436	2436
Gross Electrical Output, megawatts	822	813	839
Net Electrical Output, megawatts	795	786	797
Main Steam Flow Rate, pounds per hour	10,470,000	10,030,000	10,470,000
Total Reactor Core Flow Rate, pounds per hour	77,000,000	78,500,000	78,500,000
Feedwater Temperature, degrees Fahrenheit	420	387.4	420
Reactor Operating Pressure, pounds per square inch gauge	1005	1005	1020
Fuel Lattice	8x8	7x7	8x8
Number of Fuel Assemblies	560	560	560
Number of Control Rods	137	137	137
Reactor Vessel Inside Diameter, inches	218	218	218
Reactor Vessel Inside Height, feet	69.3	69.3	69.3
Reactor Vessel Design Pressure, pounds per square inch gauge	1250	1250	1250
Reactor Vessel Wall Thickness, inches	5-17/32	5-17/32	5-3/8

TABLE 1-1 (Continued)

Design Feature	Hatch Unit 2	Hatch Unit 1	Zimmer Unit 1
Number of Recirculation Loops	2	2	2
Recirculation Loop Inside Diameter, inches	28	28	20
Recirculation Pump Capacity, gallons per minute	45,200	45,200	33,880
Number of Jet Pumps	20	20	20
Number of High Pressure Coolant Injection Pumps	1	1	1#
Number of Core Spray Loops	2	2	2##
Number of Low Pressure Coolant Injection Pumps	4	4	3
Number of Containment Spray Loops	2	2	2
Maximum Heat Flux, British thermal units per square foot per hour	361,594	428,300	354,000
Average Heat Flux, British thermal units per square foot per hour	145,528	164,410	143,900
Maximum Power per Fuel Rod Length, kilowatts per foot	13.4	18.5	13.4
Average Power per Fuel Rod Length, kilowatts per foot	5.39	7.11	5.45
Maximum Fuel Temperature, degrees Fahrenheit	3435	4380	3325
Minimum Critical Power Ratio	1.30	1.32	1.24
Total Peaking Factor	2.49	2.60	2.43

#High pressure core spray used on Zimmer.

##One low pressure core spray used on Zimmer.

(1) The population density and land use characteristics of the site environs and the physical characteristics of the site (including seismology, meteorology, geology, and hydrology) to establish that 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 in 10 CFR, Part 100, taking into consideration the design of the facilities, including the engineered safety features provided.

(2) 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) The expected response of the facility 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 and 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) The Cincinnati Gas and Electric Company'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 industry security, and the planning for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that the applicant is technically qualified to safely operate the facilities.

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

(6) The Cincinnati Gas and Electric Company's quality assurance program for the operation of the facilities to assure that the program complies with the Commission's regulations in 10 CFR, Part 50, and that the applicant will have proper

controls over the facility operations such that there is a high degree of assurance that the facilities can be operated safely and reliably.

(7) The financial data and information supplied by the Cincinnati Gas and Electric Company and its coapplicants as required by the Commission's regulations (Section 50.33(f) of 10 CFR, Part 50, and Appendix C to Part 50) to determine that the applicants are financially qualified to operate the proposed facilities.

1.6 Modifications to the Facility During the Course of Our Review

During the review, we met a number of times (see Appendix A to this report) with the applicant's representatives, contractors and consultants to discuss various technical matters related to the facility. Also, we made a number of site visits to assess specific safety matters related to the station. The applicant made a number of changes to the facility design as a result of our review. We reviewed these design changes also. Special details concerning these changes are included in amendments to the Final Safety Analysis Report and in appropriate subsections of this report.

The nuclear steam supply system contains new and significantly modified features that are different than previous boiling water reactor designs that had been evaluated by us at the final design stage. The General Electric Company provided new designs in instrument and control areas. These changes include: 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; and a revised rod pattern control system. These are discussed in section 7.0 of this report.

1.7 Requirements for Future Technical Information

The Cincinnati Gas and Electric Company identified the generic Mark II containment long term test program as a program applicable to the Zimmer Station. This program is aimed at verifying the containment designs and confirming the design margins to accommodate pool dynamic loads. The objective, schedule for completion, and current results are summarized in subsection 6.2.1 of this report. We have issued conservative acceptance criteria for the Mark II containment, which if met by the Zimmer design, will provide reasonable assurance that the health and safety of the public will be protected against the effects of pool dynamic loads.

1.8 Summary of Outstanding Issues

As a result of our review, several items remain outstanding at the time of issuance of this report. Since we have not completed our review and reached our final positions in these areas, we consider the issues to be open. Our review will be completed prior to a decision on issuance of an operating license and will be reported in a supplement to this report. The items, with appropriate references to subsections of this report, are stated below.

3.8.1, 3.8.2, 3.9.2, 6.2.1 - We have issued our acceptance criteria for the Mark II containment design to accommodate pool dynamic loads. The applicant has reported those criteria from which he wishes to deviate and has presented his plans for justifying the deviations. We will complete our review of the Zimmer Design Assessment Report and Closure Report and report our conclusions with respect to the acceptability of the Zimmer design to accommodate pool dynamic loads when all needed information is available.

6.3.4 - We will not be able to make a final conclusion reporting the ability of the plant to meet 10 CFR 50.46 acceptance criteria for the emergency core cooling system until we have completed our evaluation of the Two Loop Test Apparatus test results.

1.9 Summary of Confirmatory Items

As a result of our review, there are a number of items for which we have completed our review and have reached positions. The applicant will implement our positions. However, we are continuing our review of the application to permit us to confirm the implementation of these positions. These items will be completed to our satisfaction prior to decision on issuance of an operating license and will be reported in a supplement to this report. These items, with appropriate references to subsections of this report, are stated below.

2.2, 6.4.2 - We have not been able to define the hazard, if any, to the control room operators resulting from the transport of toxic chemicals along U.S. Route 52 past the Zimmer site. We are pursuing the needed information.

3.9.1, 3.9.2, 3.10, 5.2.1, 7.6, 8.3 - We have not completed our audit of the seismic qualification of mechanical and electrical equipment and the results of the applicant's component operability assurance program.

4.4.1, 4.4.2, 15.1 - We are evaluating a new calculational basis (ODYN code) for the load rejection without turbine bypass. We expect to complete this evaluation prior to Zimmer licensing.

4.4.1 - We are continuing our review of Appendix H to the Final Safety Analysis Report which describes the reactor flow control system. We will report on our safety evaluation this system prior to reactor startup.

4.5.1 - We are investigating evidence of cracking in control rod drive tubes and have requested additional information on this matter from the applicant.

5.2.1, 5.2.3, 5.2.4, 5.3.3 - We have completed our review of the applicant's preservice inspection program. We are reviewing the applicant's justification for exceptions taken to the specific provisions of 10 CFR, Part 50, 50.55a(g)(2). The applicant will provided an acceptable augmented inservice inspection method which will assure early detection of feedwater and control rod drive nozzle and blend radii cracking which has been experienced in earlier operating boiling water reactors. The applicant will also provide information regarding implementation of the positions stated in NUREG-0313 related to stainless steel cracking. The applicant plans to submit his inservice inspection program (in accordance with 10 CFR, Part 50, 50.55a(g)(4)) six months prior to commercial operation.

5.2.2 - The applicant did not include the effects of the anticipated transients without scram recirculation pump trip in his overpressurization analyses. We have requested this information and have also requested documentation of information on the qualification of safety/relief valves.

5.3.1, 5.3.2, 6.2.6 - We are reviewing information to determine whether or not we may grant certain exemptions to 10 CFR, Part 50 Appendixes G, H and J for some Zimmer equipment.

6.2.1, 6.3.4 - The Zimmer design diverts low pressure core injection water to the wetwell spray in order to increase allowable suppression pool steam bypass. We asked the applicant to show that the consequences of diverting low pressure core injection water are acceptable with respect to emergency core cooling and long term pool cooling performance. The applicant will provided the information.

7.1, 7.2, 7.3, 7.4, 7.5, 7.6, 7.7, 8.3 - During our electrical review site visit and drawing reviews a number of deficiencies with regard to physical separation and electrical independence were disclosed. The applicant committed to making field modifications to systems wiring in order to meet his construction permit commitment with respect to physical separation and electrical independent. He also has committed to perform qualification testing of electric isolators used between redundant safety systems circuits and nonsafety grade and safety grade circuits. We will audit these modifications and review the qualification testing results. We will also resolve the issues regarding range, setpoint drift, response time testing and test frequencies prior to issuance of the technical specifications with the operating license.

7.2.3, 7.2.4, 7.5.2 - During the Hatch Unit 2 review we became concerned about the protection of motor/generator sets which provide power to the reactor scram system. We are waiting for detailed information on the applicants' proposed resolution of this safety matter.

7.3.3, 7.5, 7.6 - We have not completed our review of instrumentation required for the automatic actuation of wetw[illegible] sprays and monitoring of essential drywell and wetwell parameters in the event of a small break in the primary system.

7.5.3, 7.5.4 - The applicant has proposed some modification to the rod display system power source in order to make the system acceptable to us with respect to availability of rod scram indication in the event of power source failure during an event requiring rod scram. We are waiting for detailed information on the applicant's proposed resolution of this safety matter.

7.6.3, 7.6.4, 7.7.3, 7.7.4, 15.1 - We have completed our review of all other instruments required for safety including systems not required for safety. We are continuing our review of the use of nonsafety grade equipment to mitigate the consequences of some abnormal operational transients in order to determine appropriate surveillance and testing requirements for that equipment.

9.5 - The applicant has responded to all of our positions with regard to fire protection. We have reviewed these responses for acceptability. The applicant has met all of our positions with some minor exceptions which require resolution.

13.7 - The applicant has responded to all of our positions with regard to industrial security and has filed a revised industrial security plan. We have reviewed the revised plan for acceptability. The applicant has met all of our positions with some minor exceptions which require resolution.

14.1 - The applicant has provided all the information needed for us to complete our review of the preoperational and startup test programs. The applicant will meet all of our positions in this area.

20.4 - We will update our financial conclusions when updated information is available from the applicant.

1.10 Items of Disagreement Between the Staff and Applicant

During the review a disagreement regarding the appropriate level for dewatering the compacted backfill under Category I structures developed between the applicant and us. We will resolve this matter with assistance from our consultant (see subsection 2.5.3 of this report).

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

The Wm. H. Zimmer Nuclear Power Station Unit No. 1 is located on a 632 acre tract of land on a floodplain of the Ohio River at river mile 443 in Washington Township of Clermont County, in southwestern Ohio. The site is located approximately 24 miles southeast of Cincinnati, Ohio and 1/2 mile north of Moscow, Ohio, as is shown in Figures 2-1 and 2-2. The topography of the western portion of the site is relatively level at about 500 feet above mean sea level and was primarily farm land. The surrounding area of the site is hilly with small meandering creeks flowing into the Ohio River, with elevations running up to 800 feet above mean sea level. The reactor plant grade is 520 feet above mean sea level.

2.1.2 Exclusion Area Authority and Control

The applicant specified a minimum exclusion area distance of 250 meters (820 feet) and a low population zone with a radius of 4827 meters (3 miles). Figures 2-3 and 2-1 show the exclusion area and low population zone, respectively. The applicant owns the exclusion area which is not traversed by any roadway, railroad or waterway, and we conclude that the applicant has the authority and control necessary to assure that activities within the exclusion area will not interfere with normal plant operation.

2.1.3 Population Distribution

The population within the three mile low population zone is about 1580. Moscow, Ohio is the largest nearby town and had a 1970 population of 348. The nearest population center with a population in excess of 25,000 is Newport, Kentucky (population 25,998), 20.8 miles northwest. Covington (population 52,535), Kentucky and Cincinnati (population 452,524), Ohio are located at 20.9 and 24.2 miles northeast, respectively. These distances are at least 1-1/3 times the low population zone distance.

The transient population within the low population zone is low and is composed of 145 students and teachers at the public schools, visitors to Camp Sunshine and Grant's Birthplace and 202 employees at industrial sites. The total 1974 resident population living within the low population zone was 1578.

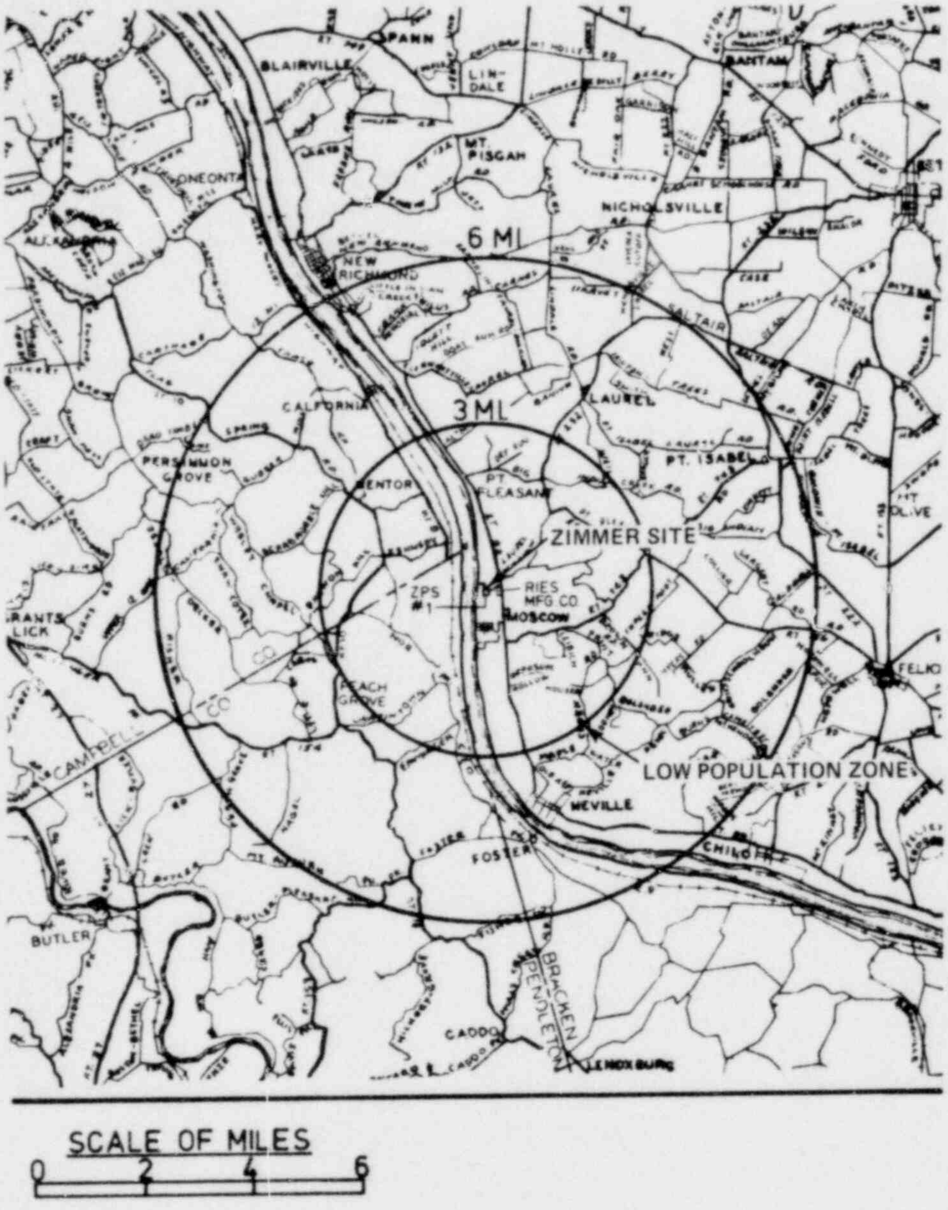

FIGURE 2-1 GEOGRAPHIC LOCATION OF PLANT SITE

FIGURE 2-2 PLANT SITE AREA

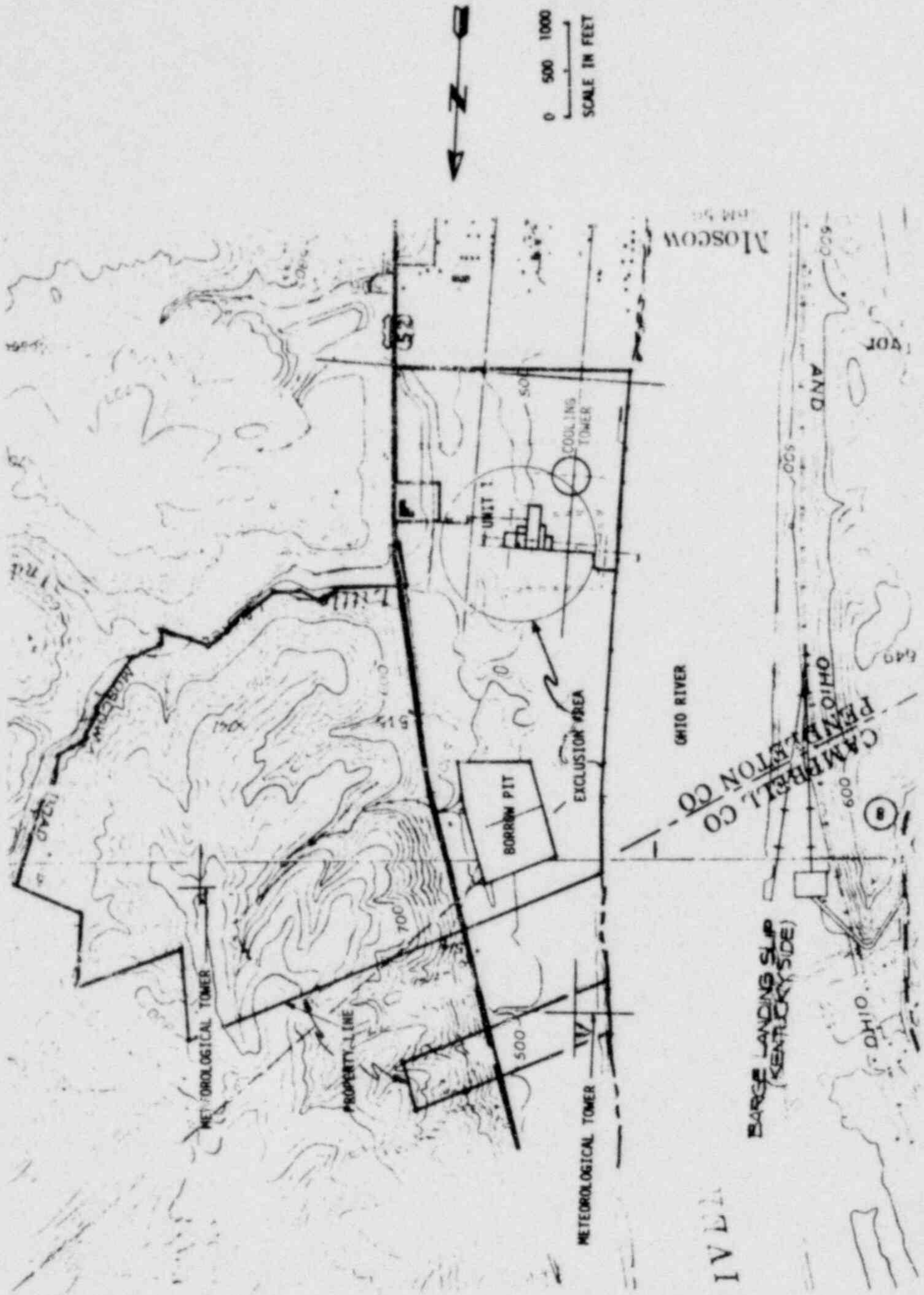

FIGURE 2-3 PLANT SITE PROPERTY

On the basis of our rev' described above, our analysis of the onsite meteorological data (subse ... 2.3 of this report) and our calculated potential radiological dose consequences of design basis accidents (subsection 15.0 of this report), we conclude that the exclusion area, low population zone and population center distance meet the guidelines of 10 CFR, Part 100 and are acceptable.

2.2 Nearby Industrial, Transportation and Military Facilities

There are no military facilities or airports located within 10 miles of the Zimmer site. The nearest military base is located more than 50 miles from the Zimmer site. The nearest airport is the Clermont Airport located approximately 15 miles north of the site. The nearest commercial airport is Lunken Field located 19.5 miles north-northwest of the site. The greater Cincinnati Airport is located approximately 27 miles northwest of the site in Covington, Kentucky. Our evaluation of aircraft hazards is contained in subsection 3.5.1 of this report.

The nearest industry, the Ries Manufacturing Company Incorporated, manufactures surgical supplies. Two quarry operations, the Black River Mining Company and the Pickney Brewer Company located two and 2.5 miles, respectively, from the reactor site store and use explosives in their operations. We evaluated the use of explosives at distances greater than two miles from the Zimmer site and conclude that they will not affect plant safety.

The Ohio River is used for commercial barge traffic and small pleasure boats. The applicant designed a narrow and staggered intake channel to protect the service water intake structure from the effects of a barge impact or explosion. We concur that this design is acceptable. Barge, rail and road shipments of toxic materials near the site were reviewed to determine their potential effect on control room personnel, and subsection 6.4.2 of this report provides additional details. There are no oil pipelines or tank farms within five miles of the site. The closest gas pipeline is approximately 2.2 miles west of the site.

The nearest highway to the site is U.S. Route 52 which runs adjacent to the eastern border of the site. Kentucky Route No. 8 passes parallel on the opposite side of the river at a distance of approximately 4500 feet from the reactor building. The Chesapeake and Ohio Railroad line is located on the Kentucky side of the Ohio River 3100 feet west of the reactor building and the shipment of toxic materials is discussed in subsection 6.4.2 of this report. We conclude that the distances from the plant to potential carriers of hazardous cargo meet the guidelines described in Regulatory Guide 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites."

On the basis of the present use of these nearby facilities and the separation distances involved, and with the exception noted in subsection 6.4.2 of this report, we conclude that they will not affect the safe operation of the Zimmer Nuclear Power Station.

2.3 Meteorology

2.3.1 Regional Climatology

The Zimmer plant is located in southwestern Ohio, 24 miles southeast of Cincinnati. This area is under the influence of a continental type climate, which means it is subject to warm, humid summers and cold winters.

Long-term meteorological measurements made in the vicinity of the plant include those at Abbe Observatory in Cincinnati and at the Greater Cincinnati Airport. These observations identify the large range of temperature to be expected in the area. Extremes have been observed from -19 degrees Fahrenheit to 109 degrees Fahrenheit, although average daily temperatures range from maximums in the mid-60's to minimums in the mid-40's.

Precipitation is distributed throughout the year and totals about 40 inches of liquid water per year. Part of this precipitation is in the form of snow; about 20 inches of snow falls annually during the period from October through May.

Prevailing winds are from the southwest quadrant and average less than 10 miles per hour. The highest wind observed in the region was 49 miles per hour at the Abbe Observatory.

Severe weather phenomena affecting the site include thunderstorms, which may occur 40 to 50 days per year, and on occasion tornadoes and hail resulting from the thunderstorms. During the period 1953 - 1974, 90 tornadoes were observed in the 10,000 square mile area including the site. This would result in a probability of one tornado nearly every 800 years. Other severe type weather phenomena which can affect the site include freezing rain that produces glaze on surfaces and occasionally the passage of the remnants of tropical storms moving from the Gulf of Mexico. Air stagnation episodes affected the region 140 days during the period 1936-1970.

We conclude that the regional climatology, including severe weather phenomena, has been described by the applicant in an acceptable manner to identify parameters affecting the siting of the plant.

2.3.2 Local Meteorology

The Cincinnati observations provide a basis for defining expected meteorological conditions that may be observed locally in the Zimmer area. Some of these observed meteorological parameters have been identified in subsection 2.3.1 above. Although the airport experiences an average of 26 days with heavy fog each year, the proximity of the site to the Ohio River should cause a greater frequency of fog at the site. Proximity to the river will also tend to moderate temperature extremes at the site. This temperature moderation could modify the total annual snowfall; the site may receive less than the 20 inch average at Cincinnati, which is somewhat removed from the influence of the river.

Nearby precipitation measurements have been made along the Ohio River at Chilo Dam #34 and generally show good relationship to measurements made at the Cincinnati locations of Abbe Observatory and Greater Cincinnati Airport.

We conclude that the applicant provided adequate information on local meteorological and air quality conditions that are of importance to the safe design and siting of the plant.

2.3.3 Onsite Meteorological Measurements Program

Meteorological monitoring at the Zimmer site began in the fall of 1969 using two mechanical weather stations. These were replaced in 1971 by a 200 foot meteorological tower situated north of the plant structures in the valley, and a 50 foot tower (which was subsequently increased to 150 foot) on a hilltop northeast of the plant, 5500 feet away.

The parameters measured on each tower are shown in Table 2-1. The accuracies of the equipment used meet the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs."

We conclude that the onsite meteorological measurements program has produced data which have been summarized to provide an adequate meteorological description of the site and its vicinity for the purpose of making atmospheric estimates for accidental and routine airborne releases of effluents from this nuclear facility.

2.3.4 Short-term (Accident) Diffusion Estimates

Short-term accidental gaseous releases at ground level were evaluated. Two years of onsite meteorological data from 1972-1974 as measured on the valley tower were used for the evaluation. These data were comprised of joint frequency of stability, as determined by vertical temperature differences, and wind speed and direction as measured at 10 meters above grade level. The determination of relative concentration (X/Q) allowed credit for building wake mixing, and in accordance with the May 16, 1978 Interim Branch Technical Position, Hydrology and Meteorology Branch -2, utilized a modification of Standard Review Plan 2.3.4 methods for the determination of X/Q. This modification, outlined below, is considered to be appropriate for this site due to the long narrow nature of the site. Another factor affecting the site is the predominant up and down valley wind circulation which is reflected in the method of analysis applied to this evaluation. The evaluation was done for the 250 meter exclusion radius and the 4826 meter distance to the outer boundary of the low population zone.

The modification to the Standard Review Plan 2.3.4 method reflects combined credit for building wake and increased lateral plume meander, direction dependent variation of dispersion conditions, wind frequencies, and exclusion area boundary distances.

TABLE 2-1

METEOROLOGICAL INSTRUMENTATION

Valley tower 200 foot.

Parameter	Height
Wind speed and direction	30-feet, 200-feet
Temperature	30-feet
Temperature difference	30-200 feet
Relative humidity	30-feet
Turbulence	30-feet

Hilltop tower 150-foot

Parameter	Height
Wind speed and direction	50-feet, 150-feet
Temperature	150-feet
Relative Humidity	150-feet

Detailed information on the model and the basis for its acceptability are contained in Draft Regulatory Guide 1.XXX, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (9/23/77).

Use of the modified procedure resulted in a 0-2 hour X/Q that would be expected to be exceeded no more than five percent of the total time of 7.1×10^{-3} seconds per cubic meter at the 250 meter exclusion distance. This limiting value occurred northwest of the plant and is expected to occur less than a total of five percent of the time around the entire 250 meter circular exclusion area boundary. Values of relative concentration at actual site boundary distances would be lower than this 250 meter relative concentration.

Values of relative concentration for the outer boundary of the low population zone for selected post-accident time periods at the adjusted five percent probability level are given in Table 2-2.

2.3.5 Long-Term Diffusion Estimates

Reasonable estimates of average atmospheric dispersion conditions at the Zimmer site were made, with corrections for down valley and up valley air flows, using methods described in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors." The model evaluates routine atmospheric releases at various points of interest as identified in Table 2-3. The highest undecayed, undepleted relative concentration (X/Q) and relative deposition (D/Q) without correction factors are shown. Values are given for three types of releases, as shown in the footnote to Table 2-3.

We made reasonable estimates of average atmospheric diffusion conditions using the applicant's onsite meteorological data and the diffusion model described in Regulatory Guide 1.111.

2.3.6 Conclusions

We conclude that the applicant provided sufficient meteorological information in the Final Safety Analysis Report to allow our determination of the plant's suitability for operation under the meteorological regime affecting this site.

2.4 Hydrology

2.4.1 Hydrologic Description

The plant is located on the east (Ohio) side of the Ohio River about a half mile north of the village of Moscow; 25 miles southeast of Cincinnati, Ohio; 443 river miles downstream from Pittsburgh, Pennsylvania.

TABLE 2-2

RELATIVE CONCENTRATIONS

Time Period	(X/Q seconds per cubic meter)
0-8 hrs at the low population zone*	9.2×10^{-5}
8-24 hrs at the low populaton zone	6.4×10^{-5}
1-4 days at the low population zone	2.8×10^{-5}
4-30 days at the low population zone	9.5×10^{-6}

*Low population zone 4826 meters

TABLE 2-3

SUMMARY OF ATMOSPHERIC DISPERSION FACTORS AND DEPOSITION VALUES FOR SELECTED LOCATIONS NEAR THE ZIMMER NUCLEAR POWER STATION

Location*	Source	X/Q (seconds per cubic meter)	Relative Deposition per square meter
Nearest site land boundary (northeast 1.05 miles)	A	5.95 E-07**	3.34 E-09
	B	3.59 E-06	2.01 E-06
	C	4.73 E-06	2.65 E-08
Nearest residence and garden (north 0.82 miles)	A	3.72 E-08	2.54 E-09
	B	3.14 E-07	2.14 E-08
	C	4.34 E-07	2.96 E-08
Nearest farm and milk animal (southeast 0.75 miles)	A	3.01 E-08	1.93 E-09
	B	2.77 E-07	1.77 E-08
	C	3.90 E-07	2.50 E-08

*"Nearest" refers to that type of location where the highest radiation dose is expected to occur from all appropriate pathways.

**5.95 E-07 = 5.95×10^{-7}

Source A is continuous plant vent.
Source B is mechanical vacuum pump - 4 times/year for 24 hours
Source C is drywell purge - 24 times/year for 2 hours
#uncorrected for valley flows

The drainage area of the Ohio River above the site is about 70 000 square miles. The area consists mostly of glacial till plains in Ohio and the Appalachian Plateau in most of the rest of the area. Elevation in the basin ranges from 5720 feet above mean sea level in Virginia to 445 feet above mean sea level at the site.

The river near the site is an old entrenched stream with a flood plain slightly less than a mile wide entrenched about 400 feet below the elevation of the uplands. The plant is located on the flood plain of the river about 40 feet above the normal reservior level of Markland Lock and Dam. Plant grade is elevation 520 feet above mean sea level; this elevation is three feet above the stage corresponding to the flood of record (January - February, 1937).

The only perennial stream on the site is Little Indian Creek. It has a drainage area of 5.5 square miles. The creek flows from the east and empties in the Ohio River downstream of the site.

2.4.2 Flood Potential

We reviewed the information submitted in the Final Safety Analysis Report with respect to flooding and found it to be essentially the same as that submitted in support of the application for a construction permit and reported in that Safety Evaluation Report. The procedures used in the determination of the probable maximum flood discharge were also reviewed and found to be unchanged and to meet the present criteria. Accordingly, the discharge of 1,980,000 cubic feet per second previously determined meets the current criteria set forth in Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," Revision 1, April, 1976. This flood produces a design basis still water level of 545.4 feet above mean sea level at the plant. Coincident wind waves produce runup of about 4 feet against a vertical surface. This yields a design basis flood level of 549.4 feet above mean sea level.

The applicant designed the seismic Category I structures to withstand this flood level. All openings on these buildings, including openings through exterior walls such as air intakes and vents, below this level are provided with watertight seals or bulkhead doors designed to resist the pressure resulting from the probable maximum flood and coincident wind generated waves and runup. All seismic Category I structures required for safe shutdown will remain accessible during all flood conditions.

The effects of a probable maximum precipitation event in the immediate vicinity of the plant were also investigated and would not flood the plant. The applicant also reevaluated the effects of the potential failure of upstream dams. They concluded, and we concur, that those located along the main stream of the Ohio River are too low to cause a higher flood level at the site than the probable maximum flood. Their investigation of the many tributary reservoirs indicates the relative location and size would allow flood waves caused by failures to be almost fully attenuated before reaching the site, and in no case cause a flood level greater than the probable maximum flood. The applicant's flood protection procedures consist of various action levels based on predicted river elevations.

Early forecasts of flood conditions are provided by the National Weather Service. The actual crest estimates by magnitude and date are a joint effort of the National Weather Service and the U.S. Corps of Engineers; however, the Weather Service is the official issuing agent. The station will be in direct contact with the Weather Service on a routine basis during normal working days to ascertain river and weather conditions and predictions. In addition, emergency plan implementing procedures provide a method for contacting the National Weather Service any time there is concern for river and weather conditions and/or predictions. In addition to obtaining river predictions and conditions from the Weather Service, station personnel will monitor river level and conditions periodically from the service water pump structure.

When the National Weather Service predicts river elevation crests in excess of 487 feet (52 feet gage), the following flood-protection preparations will be initiated and completed as soon as possible, but before the river elevation exceeds 496 feet (61 feet gage).

(1) Visually inspect the seals on watertight flood-control doors and flood-control ventilation dampers.

(2) Observe the proper operation of the locking mechanisms of the watertight bulkhead doors.

The visual inspection of the seals and observation of proper operation of the locking mechanisms of the watertight flood-control doors and flood-control ventilation dampers are precautionary measures which can be completed in four hours.

When the actual river elevation reaches 496 feet (61 feet gage), the station operations personnel shall increase river surveillance by monitoring the level and the rate at which the river level increases.

The station superintendent will declare an emergency alert due to river flood when the river elevation exceeds 496 feet (61 feet gage) and crest predictions exceed 510 feet (75 feet gage) or river level rises greater than 0.50 feet in less than one hour.

Upon emergency alert, the following flood-protection precautionary measures will be initiated:

(1) Test run the emergency diesel generators.

(2) Order necessary consumable items.

(3) Test telephone and radio communication links.

(4) Verify proper operation of the sump pumps.

The above precautionary measures can be initiated and completed within four hours. When river elevation reaches 510 (75 feet gage) feet and flood predictions are greater than 517 (82 feet gage) feet, the watertight outside access doors to the reactor building and the service water structure, located at or above the 520-foot elevation, are closed or verified closed.

At a river elevation of 517 (82 feet gage) feet the unit will be shut down and will be brought to the hot shutdown condition within the next six hours and to the cold shutdown condition within the following thirty hours. The remaining watertight flood control doors are closed when the river reaches the 517-foot elevation. After the main shutdown is complete, all watertight ventilation flood-control dampers are closed. Within four hours of reaching hot shutdown condition, all watertight flood-control ventilation dampers shall be closed. In addition, the circulating water pump structure shall be electrically isolated and all power feeds to the structure safety-tagged. Supplementary action includes placing the unit in the cold shutdown mode, shifting the entire heat load to one residual heat removal heat exchanger when the decay heat load has decreased sufficiently. The demineralized water, well water, and cycled condensate storage tanks shall be filled to levels specified by the operations supervisor based upon flood predictions and future anticipated operation. Radwaste systems shall be isolated, with the radwaste tanks filled to levels specified by the rad/chem engineer.

Should the river level reach 520 feet (85 feet gage), the auxiliary electrical load shall be shifted to the diesel generators. After the river returns to less than 517 feet (82 feet gage), cleanup and testing requirements are specified.

Based on the above, we conclude that the applicant's proposed emergency operation requirements are adequate to safeguard the plant from all floods up to and including the probable maximum flood. These emergency operation requirements will be set forth in greater detail in the technical specifications.

2.4.3 <u>Water Supply</u>

Condenser cooling requirements for the plant are satisfied by a natural draft cooling tower. Make-up water is pumped from the Ohio River and the blowdown, after being discharged to a settling pond, is returned to the river. Safety-related water supply is a once through system. The applicant stated that the essential cooling water requirements are 28 cubic feet per second. The historical minimum instantaneous low flow of 2100 cubic feet per second in the area was recorded at Louisville, Kentucky, on August 20, 1930. Since then dams have been constructed which can be counted on to augment low flow for all but the most severe drought conditions. The applicant concluded, and we concur, that sufficient flow exists in the Ohio River for emergency requirements. We also evaluated the effects of zero flow condition and concluded that sufficient storage exists in Markland Reservoir above the service water intake to provide sufficient water supply storage for safe shutdown. The applicant also evaluated the extreme low flow condition in the Ohio River (equivalent to the local minimum flow of record in the area) coincident with complete loss of Markland Dam. His analysis indicates that the resulting water level would be of sufficient depth to assure adequate suction head on the pumps to maintain the safe shutdown conditions.

Accordingly, we conclude that the hydrologic design for the ultimate heat sink meets the requirements set forth in Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Revision 1, dated March 1974 and is, therefore, acceptable.

2.4.4 Groundwater

The principal aquifers in the region are alluvial and outwash sand and gravel deposits occurring along major stream courses. Bedrock aquifers, although of wide areal extent, have limited potential for the development of groundwater supplies. Groundwater levels at the site slope toward the Ohio River. Water levels in the alluvial materials throughout the site are essentially horizontal.

Groundwater levels near the river will fluctuate in response to changes in river stage. Under normal river stage conditions, the hydraulic gradient is toward the river with a downstream component, but high water conditions will temporarily change the direction of the gradient. The water table near the site is at a depth of about 40 feet, approximately the same as normal river stage.

The plant will use groundwater at an average rate of about 38 gallons per minute with a short-term peak requirement of about 500 gallons per minute for make-up for the demineralizer trains. This water will be supplied by two wells, each capable of developing 500 gallons per minute from the alluvial aquifer.

The applicant and we 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 a 25,000 gallon waste tank in the radwaste building.

Upon a postulated rupture of the waste storage tank and the building, the radioactive liquids would mix and travel with the groundwater according to the hydraulic gradient of the groundwater. Even under flood conditions the direction of travel would be towards the Ohio River. The nearest, and therefore most conservative, discharge point would be 700 feet down-gradient at the river. Any spill would be diluted first by the groundwater and further by the Ohio River. As surface wave survey was performed to interpret the shear wave velocity characteristics of the surficial soils. Micromotion measurements were made at three locations at the site to measure ambient ground motions.

Foundation Conditions

Subsurface investigations revealed the site to be underlain by 80 to 90 feet of soil consisting of 14 to 50 feet of Recent and Pleistocene alluvium underlain by 52 to 75 feet of Pleistocene glaciofluvial deposits. The alluvium consists of floodplain deposits of the Ohio River and Little Indian Creek. The upper unit of alluvium consists of 3 to 15 feet of clayey and fine sandy silt, and silty sand with root material. The lower alluvial unit consists of 12 to 35 feet of interlayered fine silty sand, fine sand, clayey sand, and silty clay. The alluvium is highly variable in composition both laterally and vertically.

The Pleistocene glaciofluvial deposits which underlie the flood plain alluvium were deposited from glacial meltwater and consist of fine to medium sand grading to a fine to coarse sand with depth. These deposits are highly variable both laterally and vertically and contain local lenses of silty sand and gravelly sand with occasional gravel layers.

The bedrock below the site consists of interbedded limestones and shales of Middle Ordovician age. The bedrock surface is relatively flat at an approximate elevation of 410 feet above mean sea level. These rocks have closely to widely spaced fractures and some irregularly spaced shale partings. Core recovery ranged from a minimum of 46 percent near the top of rock to 100 percent approximately 10 feet below top of rock. No evidence of solution cavities in the limestone was reported in the Final Safety Analysis Report. These rocks are underlain by older Paleozoic sedimentary strata which in turn are underlain by igneous and metamorphic rocks of the Precambrian complex.

No deformational zones such as shear zones, joints, fractures, folds or any other feature which could produce instability of the foundations were reported in the Final Safety Analysis Report.

Groundwater levels at the plant site fluctuate with the stage of the Ohio River and with recharge due to precipitation. An increase in the Ohio River stage is reflected relatively quickly (within 24 hours) in the elevation of the groundwater table at the site.

The normal river stage at the site is at elevation 455 feet above mean sea level. The static groundwater level during normal river stage ranges from approximately elevation 457 to 465 feet above mean sea level. The maximum groundwater level measured at the site between August 1972 and November 1976 was 474.3 feet above mean sea level. Soils below the groundwater table had to be dewatered in order to construct the foundation. This was accomplished with 15 deep wells designed to dewater soils above elevation 440 feet above mean sea level.

Foundation Preparation

Based upon laboratory liquefaction analysis, it was determined that alluvial soils and the upper glaciofluvial soils above about elevation 450 feet above mean sea level did not have a sufficient margin of safety against liquefaction during the safe shutdown earthquake (0.20 times the acceleration of gravity horizontal), while glaciofluvial soils below about elevation 450 feet above mean sea level indicated an acceptable margin of safety. A suitable foundation base was provided for the major structures by excavating all soils above elevation 450 feet above mean sea level and backfilling with sand compacted to 85 percent minimum relative density. Seismic Category I structures founded on compacted backfill include the reactor, diesel generator, and auxiliary buildings. Other major structures founded on compacted backfill include the turbine, radwaste, and heater bay buildings.

During liquefaction, pore water pressures in sands increase resulting in a decrease in shearing resistance. In the event of liquefaction of the natural sands adjacent to the compacted backfill, resultant high pore water pressures could be hydraulically transferred to the pore water of the compacted backfill. This would decrease the shearing resistance of the compacted backfill as well as reduce the effective stress on the base of the foundations.

To prevent the transfer of high pore water pressures to the compacted backfill in the event of liquefaction of the adjacent soils, the compacted backfill was enclosed in a relatively impervious clay envelope.

Prior to excavation below the groundwater table, the area was dewatered using a deep well system. Excavations for structures were extended a minimum lateral distance beyond foundation lines equal to the vertical distance between elevation 450 feet above mean sea level and the foundation base elevation. For the reactor building the excavations were extended laterally 30 feet greater than the distance indicated above to ensure adequate foundation support within the zone of major stress concentrations. All excavation faces were cut with a 1.5 horizontal to 1 vertical slope yielding a reported minimum factor of safety against slope failure of 1.4.

The bottom of the excavation, composed of glaciofluvial sands, was densified to a relative density of 85 percent prior to backfilling, as documented by 71 in-place density tests performed at elevation 449 feet above mean sea level, before installation of the clay blanket and compacted backfill materials.

outlined in subsection 15.3.6 of this report we 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 analysis, we conclude that the plant is protected for all floods up to and including the probable maximum flood, that an adequate water supply can be assured for safety-related purposes, and that postulated accidental spills of radioactive liquids will result in radionuclide concentrations that are a small fraction of 10 CFR, Part 20, limits at unrestricted areas. Conditions for emergency operations during severe floods up to and including a probable maximum flood will be included in the technical specifications.

2.5 Geology, Seismology and Geotechnical Engineering

2.5.1 Geology

The geological aspects of the Zimmer 1 site as presented in the Preliminary Safety Analysis Report were reviewed by us and our advisor, the U.S. Geological Survey. We and the U.S. Geological Survey concluded at the time that the analysis performed by the applicant appeared to be carefully derived and to have considered the geologic conditions pertinent to an engineering evaluation of the site. We completed our review of the geology portion of the Final Safety Analysis Report. As a result of that review, we find no information was presented to alter our conclusions stated in the Safety Evaluation Report which followed our review of the Preliminary Safety Analysis Report.

2.5.2 Vibratory Ground Motion

The Seismological Investigations Group of the Earth Sciences Laboratories, National Ocean and Atmospheric Administration, which is now a part of the U.S.Geological Survey acting as our advisor reviewed the seismology of the Zimmer 1 site at the construction permit stage. Their report concluded:

> "As a result of this review of the seismological and geological characteristics of the area around this proposed lower terrace site, the Seismological Investigations Group is in agreement with the applicant's values of acceleration, as stated in Amendment 5, of 0.10g to be adequate for representing earthquake disturbances likely to occur within the lifetime of this facility. Also, the Group agrees that an acceleration of 0.20g, as stated in Amendment 5, is adequate to represent the ground motion from the maximum earthquake likely to affect this site. It is believed that these values would be adequate for designing protection against the loss of function of components important to safety."

The acceptance of the aforementioned design acceleration values as recommended by the Seismologic Group was conditional on the following commitment by the applicant:

"Because of the possibility of liquefaction in the upper level of the flood plain terrace at this site, the applicant has indicated that for all Class I structures, the foundation materials will be excavated to a level of 450 feet and replaced with compacted fill. It is with this understanding that the design values are recommended."

During our present review, no new information which would alter National Ocean and Atmospheric Administration's conclusions has come to our attention. The applicant has complied with the Seismological group's recommendation to remove existing foundation materials to the 450 foot level and replace them with compacted fill. Therefore, we conclude that the earlier evaluation findings remain valid. In implementing plant design, the ground motion for the safe shutdown earthquake was to be applied at the free field foundation level. The applicant has met this requirement.

2.5.3 Stability of Subsurface Materials and Foundations

The site lies on a 0.5-mile-wide alluvial plain on the east side of the Ohio River. The approximate elevation of the flood plain is 500 feet above mean sea level; the level of the Ohio River ranges from 455 to 490 feet above mean sea level. The essentially flat alluvial plain is bordered on the east by moderately sloping uplands which rise to an elevation ranging between 700 and 850 feet mean sea level. Plant structures are to be located on the alluvial plain at an average final plant grade of 520 feet mean sea level.

Surface Explorations

Subsurface conditions were explored with 75 borings to determine the stratification and properties of subsurface materials, and groundwater conditions. Representative soil and rock samples were obtained for field and laboratory testing to establish static and dynamic engineering properties of the in situ materials. Standard penetration tests were performed in conjunction with soil sampling and the percent recovery of rock core was calculated for borings which extended into bedrock.

Piezometers were installed in 21 of the borings to measure groundwater levels in the overburden soils. Four piezometers were also placed in the underlying rock. Geophysical explorations were conducted in the area of the plant site to measure wave propagation velocities and to estimate the dynamic properties of the soil and rock for comparison with laboratory data. Three seismic refraction profiles were performed to evaluate the compressional wave velocities of the soil and rock. A

Documents submitted by the applicant indicate the clay blanket was constructed to conform to required density specifications, i.e., a minimum dry density of 90 percent of American Society for Testing Materials Standard D698 maximum dry density. The remainder of the clay envelope was then constructed concurrently with the placement of compacted and general backfill. Documented results of in-place tests indicate that minimum dry density (clays) and relative density (sands) requirements have been achieved.

When structural fills reached specified structural bearings elevations, mud mats were installed over completed areas to prevent subsequent damage to the sand bearing surface.

Figure 2.5-50 of the Final Safety Analysis Report shows that the water level within the clay envelope (Piezometer B) occasionally rises above the groundwater level and above the base of the reactor containment and other seismic Category I foundations.

It is our position that the compacted backfill materials within the clay envelope be dewatered during plant operation, when necessary as intended at the construction permit stage and as described in Section 2.5.4.5.1.3 and Fig. 2.5-49 of the Final Safety Analysis Report. Such measures will assure stability of the foundation backfill in the event liquefaction occurs at the site, provide a means for collecting and discharging infiltrated water which may be trapped in the encapsuled fill, assure adequate effective stresses between foundations and fill, and provide a means for determining the effectiveness of the clay blanket and for detecting anomalous conditions. Water levels in the encapsulated backfill shall be maintained at or below elevation 457 feet above mean sea level measured at the backfill dewatering well. The applicant agreed to implement this position in accordance with the construction permit stage commitment but has taken issue with the dewatering level of 457 feet above mean sea level. The applicant believes that the construction permit stage commitment was 480 feet above mean sea level and that 480 feet provides adequate protection against excessive pore pressure in the compacted backfill. We will continue discussions with the applicant and try to resolve this detail prior to reactor operations. Until the matter is resolved, we will require maintenance of the 457 foot level during operation. The applicant provided us with the description and location of the dewatering system. We find the applicant's commitment and implementation acceptable provided agreement is reached on the dewatering level to be maintained. The resolution of this matter will be provided in a supplement to this report and the required dewatering level will be specified in the technical specifications.

The service water pump structure was built into the east bank of the Ohio River. The structure is founded on a concrete caisson extending to rock. A reinforced concrete shell was sunk to bedrock by excavating soil within the perimeter of the shell and pouring successive concrete lifts. A 15-foot thick tremie concrete plug was poured over the rock and a 10-foot thick structural slab was poured over the plug. Both the plug and the slab are keyed into the shell wall to insure sound bearing on rock over the entire area of the structure.

The area surrounding the service water pump structures was backfilled to elevation 500 feet above mean sea level with sands compacted to a minimum relative density of 85 percent. The final backfill to elevation 520 feet above mean sea level will be composed of either granular materials compacted to a minimum relative density of 75 percent or cohesive soils compacted to a minimum dry density of 95 percent of the standard Proctor dry density test (American Society for Testing Materials Standard D-698).

Two seismic Category I pipelines and one duct run, each approximately 520 feet long, were run together from the service water pump structure to the radwaste building. The tops of the 30-inch pipes and duct runs were buried in sandy soils about four feet below the surface to protect them from damaging surface loads and from frost action. Each pipeline and duct unit length of 25 to 30 feet is supported on six 20-inch diameter piles approximately 80 feet long. All piles were pre-drilled to a depth of 32 to 80 feet using a 16-inch diameter auger. The piles were driven to complete refusal using a vibratory hammer.

The pile bents have been designed to withstand the lateral drag forces that might be imposed on the structure in the event of liquefaction of alluvial soils.

The intake flume consists of a 30-foot wide by about 120-foot long channel forming a 45 degree angle with the upstream bank of the Ohio River. The walls of this channel are formed by sheet piling driven to bedrock and supported by three tiers of horizontal struts spanning the channel and two tiers of tie rods. A cellular cofferdam was formed on the upstream sides of the flume by driving two additional lines of sheet piling forming a triangular area that was back-filled with granular fill and a top 10-foot layer of rip-rap. The base of the flume is at elevation 437 feet above mean sea level.

Foundation Stability

Bearing Capacity

Static foundation pressures on compacted backfill range from 4340 pounds per square foot for the radwaste building to 9200 pounds per square foot for the reactor containment. The minimum factor of safety computed against bearing type failure under maximum static loading is 12 for the diesel generator building, which is founded on compacted backfill. Bearing stability under dynamic loading was addressed by sizing and proportioning foundations such that the peak foundation pressures during seismic loading conditions will not exceed 150 percent of the static foundation pressures. See subsection 2.5.4 below (Slope Stability) for safety factors under earthquake effects.

The factor of safety against bearing failure in the rock supporting the service water pump structure is conservatively estimated to be about 70 for static conditions and approximately 40 for combined static and dynamic loading. The applicant based these factors on a gross static bearing pressure of 9.75 kilopounds per square foot, a maximum combined static and dynamic gross bearing pressure of 16.8 kilopounds per square foot, and an ultimate bearing capacity of limestone bedrock of 680 kilopounds per square foot.

Settlement

Settlement points were established in the buildings soon after the base mats were poured. The applicant reported that the maximum measured total settlement for seismic Category I structures was 0.96 inches for the reactor building. This was measured approximately 25 months after the base mat was poured. The circulating water pump structure had settled 1.8 inches in 17 months. Other structures reportedly had measured total settlement less than the estimated values. Structures founded on bedrock are expected to settle less than 0.06 inches immediately upon application of the load.

It is our position that settlement points should be observed regularly until it is evident that movement has ceased. Reports that include settlement and differential settlement plots versus time and a comparison of allowable and actual settlement should be provided to us on a semi-annual schedule until settlement of safety-related structures has ceased. The applicant committed to meet the provisions of this position.

Liquefaction

Laboratory liquefaction analysis indicated that soils above elevation 450 feet above mean sea level did not have a sufficient margin of safety against liquefaction during the safe shutdown earthquake, while soils below elevation 450 feet above mean sea level indicated an acceptable margin of safety. As a result, all soils above 450 feet above mean sea level were removed and replaced with sand compacted to 85 percent relative density to provide a suitable foundation for plant structures.

The resistance of the compacted backfill to liquefaction was evaluated using a two-dimensional plane-strain finite element analytical model. Factors of safety against liquefaction were computed based on laboratory cyclic triaxial tests using both 10 percent double amplitude strain and initial liquefaction to define soil failure. The factor of safety was then defined as the ratio of cyclic shear strength to the average cyclic shear stress for five cycles induced in a soil element by the safe shutdown earthquake. The safe shutdown earthquake used in the analysis was 0.20 times the acceleration of gravity horizontal. The water level at the site was assumed to be 508.6 feet above mean sea level or about 45 feet above normal groundwater level. The higher value was used because of the large fluctuations in river stages, the plant's proximity to the river and the quickness of the groundwater level to river stages. The minimum factors of safety with respect to 10 percent double amplitude strain were found to be 2.91 for the recompacted fill and 1.87 for the in situ soils below 450 feet above mean sea level. Similarly, the minimum factors of safety against liquefaction with respect to initial liquefaction were determined to be 3.04 for recompacted fill and 1.96 for in situ soils below 450 feet above mean sea level.

Based on the results of the liquefaction analysis, we conclude that there is adequate safety against liquefaction for ground acceleration levels up to 0.20 times the acceleration of gravity provided the compacted backfill materials are kept in a dewatered condition as stated in the Foundation Preparation subsection above.

2.5.4 Slope Stability

Slopes of compacted backfill beneath the reactor building and the diesel generator building were analyzed for stability in the event of liquefaction of adjacent soils. These buildings were chosen as the most critical slope stability cases because of the high foundation load of the reactor building (9.2 kilopounds per square foot) and the high slope beneath the diesel generator building (50 feet). Slopes were analyzed using finite element programs in a pseudo-dynamic approach as described in the Final Safety Analysis Report. For the reactor building under 1.5 times foundation load and 0.2 times the acceleration of gravity horizontal acceleration, the minimum seismic safety factor calculated was 1.49. For the diesel generator building under the same conditions, the minimum seismic safety factor is 1.30.

The intake flume has been designed to protect the intake area from blockage in the event soil liquefaction occurs by forming a sheet pile walled channel extending about 80 feet from the bank and anchored by a sheet pile cellular section at the end. The sheeting was driven to rock or below the zone of potentially liquefiable soil.

2.5.5 Summary

We conclude that sufficiently conservative measures have been taken to mitigate the effects of potential instability of the slopes along the river bank. Also, based on our review of the information available and on the implementation of the above positions, we conclude that the site and plant foundations are acceptable for the safe operation of Unit 1 of the W. H. Zimmer Nuclear Power Station.

3.0 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Conformance with Nuclear Regulatory Commission General Design Criteria

In Section 3.0 of the Final Safety Analysis Report, the applicant presented an evaluation of the design bases for the William H. Zimmer Nuclear Power Station against the Nuclear Regulatory Commission's General Design Criteria listed in Appendix A of 10 CFR, Part 50. We evaluated the final design and the design criteria and conclude, subject to the applicant's adoption of the additional requirements made by us as discussed in this report, that the Zimmer Station has been designed to meet the requirements of the General Design Criteria.

3.1.1 Conformance With Industry Codes and Standards

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

3.2 Classification of Structures, Components and Systems

3.2.1 Seismic Classification

Criterion 2 of the General Design Criteria requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety 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 10 CFR, Part 100 guideline exposures.

Structures, systems, and components important to safety that are required to be designed to withstand the effects of a safe shutdown earthquake and remain functional have in general been properly classified as seismic Category I items, 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 have been designed to other than seismic Category I requirements including those portions of seismic Category I systems such as vent, drain and test lines on the downstream side of isolation valves which are not required to perform a safety function. Structures, systems, and components important to safety that have been designed to withstand the effects of a safe shutdown earthquake and remain functional are identified in an acceptable manner in Table 3.2-1 of the Final Safety Analysis Report. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria and design bases for structures, systems, and components important to safety with the Commission's regulations as set forth in General Design Criterion 2 and to Regulatory Guide 1.29, "Seismic Design Classification," our technical positions and industry codes and standards.

We conclude that structures, systems, and components important to safety that are designed to withstand the effects of a safe shutdown earthquake and remain functional have been properly classified as seismic Category I items in conformance with the Commission's regulations, the applicable regulatory guides, and industry codes and standards and are acceptable. Design of these items in accordance with seismic Category I requirements provides reasonable assurance that in the event of a safe shutdown earthquake, the plant will perform in a manner providing adequate safeguards to the health and safety of the public.

3.2.2 System Quality Group Classification

Criterion 1 of the General Design Criteria requires that the nuclear power plant systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Fluid system pressure-retaining components important to safety have been designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed. The applicant identified those fluid-containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety where reliance is placed on these systems: (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintain it in a 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 of the Final Safety Analysis Report and on system piping and instrumentation diagrams in the Final Safety Analysis Report based on conformance with Regulatory Guide 1.26, "Quality Group Classification and Standards."

The applicant has applied Quality Groups A, B, C, and D in Regulatory Guide 1.26, "Quality Group Classifications and Standards," to the fluid system pressure-retaining components important to safety. These components that are classified Quality Group A, B, C, or D have been constructed to the codes and standards identified in Table 3.2-2 of the Final Safety Analysis Report.

3.3 Wind and Tornado Loadings

3.3.1 Wind Design Criteria

All seismic Category I structures exposed to wind forces are designed to withstand the effects of the design wind. The design wind specified has a velocity of 90 miles per hour based on a recurrence of 100 years.

The procedures that are used to transform the wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gusts factors are in accordance with American Society of Civil Engineers Standard Paper Number 3269. This document is acceptable to us.

The procedures that are utilized, based on the above paper, to determine the loadings on seismic Category I structures induced by the design wind 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 provide reasonable assurances that in the event of design basis winds, the structural integrity of the plant siesmic Category I structures will not be impaired and, in consequence, seismic Category I systems and components located within these structures are 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 General Design Criterion 2.

3.3.2 Tornado Design Criteria

All seismic Category I structures exposed to tornado forces and needed for the safe shutdown of the plant are designed to resist a tornado of 300 miles per hour tangential wind velocity and a 60 miles per hour translational wind velocity. The simultaneous atmospheric pressure drop was assumed to be 3 pounds per square inch in 2 seconds.

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

The procedures utilized, based on the American Society of Civil Engineers paper cited above, to determine the loadings on structures induced by the design basis tornado specified for the plant are acceptable since the 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 basis tornado, the structural integrity of the plant structures that have to be designed for tornadoes will not be impaired and, in consequence, safety-related systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions as required. Conformance with these procedures is an acceptable basis for satisfying the applicable requirements of General Design Criterion 2.

3.4 Water Level Design

3.4.1 Flood Protection

The external walls of safety-related structures that are below the plant grade elevation are protected from flood waters by waterproofing materials. The probable maximum flood level is 545.4 feet above mean sea level and, in addition, wind wave runup was assumed to be an additional four feet. The safety-related components at the elevations between the plant grade and the probable maximum flood levels are protected since the applicant designed the walls of the structure in which the safety related equipment are housed of sufficient thickness to withstand the hydrostatic pressure resulting from the probable maximum flood.

The piping penetrations into safety-related structures that are below the probable maxmum flood level are provided with flood seals. Access openings to safety-related structures that are below the probable maximum flood level are provided with water tight doors to protect equipment from flood waters.

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

3.4.2 Water Level (Flood) Design Procedures

The design flood level resulting from the most unfavorable condition or combination of conditions that produce the maximum water level at the site is discussed in subsection 2.4, Hydrology, of this report. The hydrostatic effect of the flood was considered in the design of all seismic Category I structures exposed to the water head.

The use of the loads from the design flood or highest groundwater level provides reasonable assurance that in the event of floods or high groundwater, the structural integrity of the plant seismic Category I structures will not be impaired and, in consequence, seismic Category I systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions, as required. Conformance with these design procedures is an acceptable basis for satisfying the applicable requirements of General Design Criterion 2.

3.5 Missile Protection

Criteria 2 and 4 of the General Design Criteria require that structures, systems and components important to safety be protected against or designed to withstand the effects of missiles that may result from equipment failures both within and outside the containment and from events and conditions outside the plant such as tornadoes.

3.5.1 Internally Generated Missiles

The plant is designed so that missiles from plant sources outside of containment do not cause or increase the severity of an accident.

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

We reviewed the adequacy of the applicant's design necessary to maintain the capability for a safe plant shutdown in the event of any missile generated outside containment. We conclude that the design is in conformance with General Design Criterion 4 as it relates to structures housing essential systems and to the systems being capable to withstand the effects of plant generated missiles, Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," as it relates to protection of spent fuel pool systems and spent fuel assemblies from plant generated missiles, and is, therefore, acceptable.

Inside containment the applicant protects all seismic Category I structures, systems, and safety-related components from missiles through component arrangement or through suitable barriers which isolate the missile or shield the component. The applicant identified the following missiles from pressurized equipment: valve bonnets, valve stems, thermowells, retaining bolts, and control rod drive mechanisms.

Missiles may also result from destructive overspeeding of a recirculation pump and motor following a postulated full double-ended pipe break in either the suction or discharge line of a recirculation loop. The applicant stated that parts from the fractured impeller are not capable of causing damage because pipe restraints minimize displacement of the ends of the broken pipe. Use of a decoupler between the pump and motor to prevent destructive overspeed of the motor has been studied by the General Electric Company and discussed with us. The use of such decouplers and the generation of missiles from overspeed of both the motor and impeller of the recirculation pump is a generic problem which is being reviewed by us under Task Action Plan B-68, "Pump Overspeed During a LOCA." At this time, we believe that the probability of such an event that would result in damage to safety-related equipment is acceptably low. We will require that Zimmer modify the plant in accordance with the requirements resulting from the generic study.

We conclude that the applicant's design provides adequate protection for essential structures and system inside of containment and that they are protected from internally generated missiles as required by Criterion 4 of the General Design Criteria and is acceptable.

3.5.2 Turbine Missiles

Portions of the Zimmer Nuclear Power Station safety-related structures are within one low trajectory turbine missile strike zone. This includes the auxiliary building, the diesel generator building, and the reactor building. The diesel generators are situated at an elevation which is below the turbine operating floor. This location rules out the potential for turbine missiles damaging the diesel generators, since the missiles would impact the turbine building floor with a large angle of incidence (about 80 degrees), producing a grazing impact rather than penetration.

Our preliminary review of low trajectory turbine missiles with respect to safety-related systems within the auxiliary building and reactor building indicated the potential for damaging some safety-related systems. We requested the applicant to provide a detailed missile strike and damage analysis with respect to these plant areas. The applicant, using a Monte Carlo type computer program for missile simulation, calculated the probability of striking and damaging a safety-related system. His results indicate that the probability of striking and damaging a safety-related system (taking redundancy and separation into account) is 3×10^{-3} per turbine failure at destructive overspeed.

We reviewed the applicant's turbine missile transport model and found that there are several assumptions within the model which are not sufficiently conservative. Specifically, the applicant uses (a) a dimensionless point representation of the missile along its flight (b) the Petry equation for calculating penetrations into concrete, and neglects (c) the effects of secondary missiles due to concrete spalling. Consequently we performed an independent analysis wherein the above items are considered conservatively. Our analysis is based on non-point missile representation, the use of the modified National Defense Research Committee equations for concrete penetration, and the inclusion of secondary missiles due to spalling. Using these assumptions, we reviewed the detailed plant layout drawings and estimated that the strike and damage probability for safety-related systems (taking redundancy and separation into account) is about 0.043 per destructive overspeed. Using 4×10^{-5} as the annual frequency for destructive turbine overspeed, we find that the total probability for turbine missile damage is approximately 10^{-6} per turbine year and so meets the acceptance criteria outlined in Standard Review Plan Section 2.2.3. Thus, we conclude that the turbine missile hazard is sufficiently low for the Zimmer Unit No. 1 plant design, and that the plant layout is acceptable in this respect.

3.5.3 Tornado Missiles

The applicant's assessment of hazards due to missiles generated by natural phenomena at the site was reviewed by us. The applicant showed that barriers are provided against tornado missiles for all plant structures, systems, and components requiring protection. These barriers were designed in accordance with the procedures discussed below and are acceptable.

In accordance with proposed Revision 1 of the Standard Review Plan 3.5.1.4, operating license applicants, who were not required to design to the missile spectrum described in Revision 0 of the Standard Review Plan 3.5.1.4, should provide sufficient protection at least against the following missiles:

Missile C.a. Steel Rod, 1-inch diameter, 3 feet long, weight 8 pounds, travelling horizontally at 316 feet per second and vertically at 252 feet per second at all elevations.

Missile F.a. Utility Pole, 13 1/2-inches diameter, 35 feet long, weight 1490 pounds, travelling horizontally at 211 feet per second and vertically at 169 feet per second, at all elevations below 30 feet above all grade levels within 1/2 mile of the facility structures.

We have reviewed the applicant's design and find that the applicant has provided barriers against these missiles with respect to all plant structures, syst , and components requiring tornado missile protection. Specifically, the applicant indicates that, with the exception of portions of the auxiliary building and diesel generator building, all structures have at least two feet of reinforced concrete of 4,000 pounds per square inch minimum compressive strength. The auxiliary building and diesel generator building roof openings are protected by concrete cubicles with a minimum wall and ceiling thickness of 20 inches.

The results of the Electric Power Research Institute tornado missile tests have demonstrated that walls and roofs consisting of steel reinforced concrete having a concrete strength of 3000 pounds per square inch or greater and thicknesses of 18 inches or greater will provide adequate protection against tornado missiles C and F. We have previously agreed with the results of the tornado protection criteria used by the applicant in a letter from B. Rusche to Kennedy at Stone and Webster, dated October 18, 1976. We therefore find that all of the above barriers are sufficient to provide adequate protection against the steel rod and utility pole missiles and thus find the plant design to be acceptable with respect to tornado missile protection.

3.5.4 Aircraft Hazards

The applicant's assessment of aircraft hazards at the site has been independently verified by us. There are no major airports within 10 miles of the site, and the nearest significant air traffic is Federal Airway V128, which is about three miles at the nearest approach to the site. Based on this information, we estimate that the probability for an accident having radiological consequences greater than the exposure guidelines of 10 CFR, Part 100 is less than about 10^{-7} per year. We conclude, therefore, that aircraft activity in the vicinity of the Zimmer plant will not unduly affect the plant operation and does not present an undue risk to the health and safety of the public.

3.5.5 Barrier Design Procedures

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

Information has been provided indicating that the procedures cited in Standard Review Plan 3.5.3, "Barrier Design Procedures," that were used in the design of the structures, shields and barriers to resist the effect of missiles are acceptable. The analysis of structures, shields and barriers to determine the effects of missile impact was accomplished in two steps. In the first step, the potential damage that could be done by the missile in the immediate vicinity of impact was investigated. This was accomplished by estimating the depth of penetration of the missile into the impacted structure. Furthermore, secondary missiles are prevented by fixing the target thickness well above that determined for penetration. In the second step of the analysis, the overall structural response of the target when impacted by a missile was determined using established methods of impactive analysis. The equivalent loads of missile impact, whether the missile is environmentally generated or accidentally generated within the plant, were combined with other applicable loads as is discussed in subsection 3.8 of this report.

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

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

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

Plant design criteria applied to the design of the facility are intended to accommodate the effects of postulated pipe breaks and cracks, including pipe whip, jet effect, and environmental effects. The means used to protect safety-related systems and components include physical separation, enclosure within suitably designed structures, pipe whip restraints and equipment shields. Protection against pipe failure outside containment is in accordance with A. Giambusso's letter (NRC), dated December 12, 1972, "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment," which is referenced in Standard Review Plan 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment." The applicant analyzed high energy piping systems for the effects of pipe whip, jet impingement and environmental effect on safety-related systems and structures.

The plant design basis includes the ability to sustain a high energy pipe break accident coincident with a single active failure and retain the capability for safe cold shutdown. For postulated pipe failures, the resulting environmental effect will not preclude the habitability of the control room, and will not cause a loss of function of electric power supplies, controls and instrumentation needed to complete a safety action.

The applicant also presented an analysis on the effect of the moderate energy line breaks outside containment on safety-related systems. The moderate energy systems are designed to meet the criteria set forth in Branch Technical Position, Auxiliary Power and Conversion Systems Branch 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment." We evaluated the analysis and conclude that a postulated pipe crack in a moderate energy line will not cause loss of function of a safety-related system.

Based on our review, we conclude that the applicant adequately designed and protected areas and systems required for safe plant shutdown following postulated events, including the combination of pipe failure and single active failure. The plant design meets the criteria set forth in A. Giambusso's letter dated December 12, 1972, "General Information Required for Consideration of the Effects of a Piping

System Break Outside Containment" as regards to protection of safety-related systems and components from a postulated high energy line break, and the Branch Technical Position, Auxiliary Power and Conversion Systems Branch 3-1 as regards to protection of safety-related systems and components from a postulated moderate energy line failure and is, therefore, acceptable.

We reviewed the applicant's criteria for classifying piping systems as moderate or high energy systems and selecting the locations where pipe breaks and leakage cracks are postulated to occur, both inside and outside the containment. The applicant provided a summary of results indicating the systems and postulated pipe break locations in the Zimmer Station systems consistent with the criteria. We find that the applicant's criteria provides a level of protection equivalent to that provided by Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment" and Branch Technical Position, Mechanical Engineering Branch 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," for postulating pipe breaks inside and outside containment. The analytical methods and procedures used to establish restraint locations and pipe and restraint interaction are based on generally acceptable methods in accordance with the above referenced criteria which have been demonstrated to provide realistic results. The pipe whip restraints are designed to withstand the resultant loads and remain intact to assure the protection of essential structures, systems and components.

The applicant provided for protection against the simultaneous occurrence of a safe shutdown earthquake and rupture of the largest pipe at any one of the design break locations and resulting coolant discharge. The applicant's program provides reasonable assurance that the following safety conditions and functions are accommodated and ensured:

(1) The magnitude of the design basis loss-of-coolant accident cannot be aggravated by potentially multiple failures of piping.

(2) The reactor emergency core cooling systems can be expected to perform their intended function.

(3) Structures, systems and components important to safety will be protected.

We conclude that the applicant's postulated pipe break criteria constitutes an acceptable design basis for meeting the applicable requirements of General Design Criteria Nos. 1, 2, 4 14, and 15.

3.7 Seismic Design

3.7.1 Seismic Input

The input seismic design response spectra, operating basis earthquake and safe shutdown earthquake applied in the design of seismic Category I structures, systems, and components and the specific percentage of critical damping values used in the seismic analysis of seismic Category I structures, systems and components are considered acceptable. We have not changed our conclusions in this regard from those arrived at and described in our Safety Evaluation Report dated February 18, 1972, for the construction permit review.

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

Conformance with these quirements provides reasonable assurance that for an earthquake whose intensity is 0.10 times the acceleration of gravity for the operating basis earthquake, and 0.20 times the acceleration of gravity for the safeshutdown earthquake, the inputs to seismic Category I structures, systems, and components are adequately defined to assure a conservative basis for the design of such structures, systems and components to withstand the consequent seismic loadings.

3.7.2 Seismic System and Subsystem Analysis

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

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

The finite element approach was used to evaluate soil-structure interaction and structure to structure interaction effects upon seismic responses. For the finite element analysis, appropriate nonlinear stress-strain and damping relationships for the soil were considered in this analysis.

We conclude that the procedures indicated by the applicant in the above paragraphs provide reasonable assurance that the design loads will not be exceeded during the design basis earthquake event.

3.7.3 Seismic Instrumentation Program

The type, number, location and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude and phase relationship of the seismic response of the containment structure comply with Regulatory Guide 1.12, "Instrumentation for Earthquakes." Supporting instrumentation is being installed on seismic Category I structures, systems and components in order to provide data for the verification of the seismic responses determined analytically for such seismic Category I items.

The installation of the specified seismic instrumentation in the reactor containment structure and at other seismic Category I structures, systems, and components, which complies with Regulatory Guide 1.12, constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical 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.

3.8 Design of Seismic Category I Structures

3.8.1 Concrete Containment

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

Static analyses for the containment shell, the base and the liner design for the containment were based on accepted industry methods in accordance with Standard Review Plan 3.8.1, "Concrete Containment."

Materials, construction methods, and quality control measures are covered in the Final Safety Analysis Report and, likewise, are in accordance with accepted industry practices as cited above.

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

The use of these criteria, codes, standards and specifications as defined in the Standard Review Plan 3.8.1 cited above, the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents, except as noted below, occuring within the containment, the structure will withstand the specified design conditions without impairment of structural integrity or safety function. Conformance with these criteria constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2, 4, 16, and 50.

The design of the containment and its interior structures to withstand pool dynamic loads such as hydrodynamic pipe break accident loads remains to be confirmed. The applicant has not completed all of the reanalysis taking into consideration pool dynamic loads nor have bounding pool dynamic loads been confirmed. The applicant plans to complete the remaining analysis six months prior to construction completion and the results will be compared against our criteria for acceptance (see subsection 6.2.1 of this report). In addition, we informed the applicant of additional concerns we have as a result of our review of the closure report. Resolution of these matters will be provided in a supplement to this report.

3.8.2 Concrete and Structural Steel Internal Structures

The containment interior structures consist of the shield wall around the reactor, a reactor pedestal and other interior walls, compartments and floors. The major code used in the design of concrete internal structures was American Concrete Institute Standard 318-71, "Building Code Requirements for Reinforced Concrete." For steel internal structures the American Institute of Steel Construction Specification, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," was used.

The containment concrete and steel internal structures are designed to resist various combinations of dead and live loads, accident-induced loads, including pressure and jet loads, and seismic loads. The load combinations used cover those cases likely to occur and include all loads which may act simultaneously. The design and analysis procedures that were used for the internal structures are in accordance with procedures delineated in the American Concrete Institute Standard 318-71 Code and in the American Institute of Steel Construction Specification for concrete and steel structures, respectively.

The containment internal structures are designed and proportioned to remain within limits established by us under the various load combinations. These limits are, in general, based on the American Concrete Institute Standard 318-71 Code and on the American Institute of Steel Construction Specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction, their fabrication, construction and installation, are in accordance with the American Concrete Institute Standard 318-71 Code and American Institute of Steel Construction Specification for concrete and steel structures, respectively.

The criteria that were used except as noted below 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, and with codes, standards, and specifications acceptable to us.

The use of these criteria, codes, standards and specificications, as defined in Standard Review Plan 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments;" the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents, except as noted below, occurring within the containment, the interior structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Conformance with these criteria constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4.

The design of concrete and structural steel structures interior to the wet well to withstand pool dynamic loads such as hydrodynamic pipe break accident loads remains to be confirmed as discussed in subsection 3.8.1 above.

3.8.3 Other Seismic Category I Structures

Seismic Category I structures other than containment and its interior structures are all of structural steel and concrete. The structural components consist of slabs, walls beams and columns. The major code used in the design of concrete seismic Category I structures is the American Concrete Institute Standard 318-71, "Building Code Requirements for Reinforced Concrete." For steel seismic Category I structures, the American Institute of Steel Construction Specification, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," was used.

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

The design and analysis procedures that were used for these seismic Category I structures are the same as those approved on previously licensed applications and, in general, are in accordance with procedures delineated in the American Concrete Institute 318-71 Code and in the American Institute of Steel Construction Specification for concrete and steel structures, respectively.

The various seismic Category I structures are designed and proportioned to remain within limits established by us under the various load combinations. These limits are, in general, based on the American Concrete Institute Standard 318-71 Code and on the American Institute of Steel Construction Specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction, their fabrication, construction and installation, are in accordance with the American Concrete Institute 318-71 Code and the American Institute of Steel Construction Specification for concrete and steel structures, respectively.

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

The use of these criteria, codes, standards and specifications as defined in the Standard Review Plan 3.8.4, "Other Seismic Category I Structures"; the loads; loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occuring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4.

3.8.4 Foundations

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

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

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

The use of these criteria, applicable codes, standards, and specifications as defined in Standard Review Plan 3.8.5, "Foundations;" the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated events, seismic Category I foundations will withstand the specified design conditions without impairment of structural integrity and stability or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

Piping Vibration Operational Test Program

The applicant agreed to perform a piping preoperational vibration dynamic effects test program to check the vibration performance of piping important to safety. The preoperational vibration dynamic effects test program that will be conducted on safety-related American Society of Mechanical Engineer Class 1, 2 and 3 piping systems and their restraints, components, and supports during startup and the initial operating conditions constitutes an acceptable program in accordance with guidance described in the Standard Review Plan 3.9.2.

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

Seismic Qualification of Mechanical Equipment

The applicant submitted procedures for dynamic testing and analysis techniques to confirm the adequacy of seismic Category I mechanical equipment, including their supports, to function during and after an earthquake of magnitude up to and including the safe shutdown earthquake.

In instances where components have been qualified by testing to other than current standards such as Institute of Electrical and Electronics Engineers Standard 344-75, such components, particularly those vital to the actuation and continued operation during and after an earthquake of magnitude up to and including the safe shutdown earthquake, may have to be retested. Our seismic qualification review team is reviewing the nuclear steam supply system and balance-of-plant equipment lists and has inspected the Zimmer Station balance of plant equipment already installed at the site. This review will evaluate the qualification testing to determine that the effects of the combination of seismic and hydrodynamic loads have been properly accounted for. On the basis of the review audit and site visit, the seismic qualification review team will ascertain whether any nuclear steam supply system or balance of plant equipment components have to be retested. We initiated discussions with the applicant to develop a mutually acceptable resolution of any problems arising in this area. We expect a timely resolution of this issue and will present the results in a supplement to this report.

Preoperational Vibration Assurance Program for Reactor Internals

With regard to flow-induced vibration testing of reactor internals the applicant proposed the Fitzpatrick Nuclear Power Plant reactor internals test programs as the established prototype for the Zimmer Station. The applicant further proposed to conduct a vibration testing program on reactor internals as confirmatory tests to full compliance with the guidelines established in Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," for other than prototype plants.

We reviewed the preoperational vibration test program proposed by the applicant for verifying the design adequacy of the reactor internals under loading conditions that will be comparable to those experienced during operation. The combination of tests, predictive analysis and post test inspection will provide adequate assurance that the reactor internals can be expected to withstand flow-induced vibrations without loss of structural integrity during their service lifetime. We conclude that the proposed preoperational vibration test program, which meets Regulatory Guide 1.20, constitutes an acceptable basis for demonstrating the design adequacy with respect to General Design Criteria Nos. 2 and 14.

3.9.2 American Society of Mechanical Engineers Code Class 2 and 3 Components Design, Load Combinations and Stress Limits

All seismic Category I pressure retaining systems, components, equipment and their supports outside of the reactor coolant pressure boundary, including active pumps and valves, are designed to sustain normal loads, anticipated transients, operating basis earthquake and the safe shutdown earthquake within stress limits which are consistent with those outlined in Regulatory Guide 1.48, "Design Limits and Loading Combinations." The specified design basis combinations of loading as applied to the design of the safety-related American Society of Mechanical Engineers Code Class 2 and 3 pressure-retaining components in systems classified as seismic Category I provide reasonable assurance that in the event (a) an earthquake should occur at the site, or (b) an upset, emergency or faulted plant transient should occur during normal plant operation, the resulting combined stresses imposed on the system components may be expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without gross loss of structural integrity. The applicant has agreed to reassess the existing structural margins of American Society of Mechanical Engineers Class 1, 2 and 3 components, equipment, and supports which were designed by combining the dynamic response of loads by the square root of the sum of the squares method. The reassessment will use the absolute sum method of combining these responses. When this reassessment is complete, we will evaluate the results on a component by component basis and will present our findings in a supplement to this report.

The applicant's operability assurance program for active American Society of Mechanical Engineers Class 2 and 3 seismic Category I pumps and valves includes component testing, or a combination of tests and predictive analysis. In instances where components were qualified by testing to standards other than our current standards, such components, particularly those vital to the actuation and continued operation of equipment, may have to be retested. Joint efforts by us and the applicant in conjunction with this issue are discussed in detail in subsection 3.9.1 of this report. The balance of the applicant's program provides assurance that such components can withstand postulated seismic loads in combination with other significant loads without loss of structural integrity, and can perform the "active" function (i.e., valve closure or opening or pump operation) when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated. Subject to satisfactory resolution of the testing procedure (see subsection 3.9.1 of this report) the applicant's component operability assurance program constitutes an acceptable basis for implementing the requirements of General Design Criterion No. 1 as related to operability of American Society of Mechanical Engineers Code Class 2 and 3 active pumps and valves.

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

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

3.10.1 Discussion

The supporting information is contained or referenced in Section 3.10 of the Final Safety Analysis Report. We review this information as detailed in the Standard Review Plan 3.10, "Seismic Qualification of Category I Instrumentation and Electrical Equipment," and also determine the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format for Safety Analysis Reports.

3.10.2 Findings

Our review of the results of the seismic testing and analysis of Class IE sensors and components, as listed in the Final Safety Analysis Report Tables 3.10-1 and 3.10-2, indicated that the seismic testing has not been completed.

The applicant agreed to complete the seismic qualification of the remaining instruments and panels and to complete the tables of Final Safety Analysis Report, Section 3.10, by early 1979. However, the applicant has not yet satisfied all of this commitment and some equipment remains to be qualified.

We are pursuing the seismic qualification program of the applicant as discussed below. An onsite seismic audit was conducted as described in subsection 3.9.1 of this report.

3.10.3 Qualification Program

We informed the applicant of our position pertaining to seismic qualification of the Zimmer Station seismic Category I electrical equipment and instrumentation. In instances where such equipment has been previously tested to standards not entirely in accord with our current requirements, it is our position that certain critical electrical components within both the nuclear steam supply system and the balance of plant scope of supply may have to be retested. Our seismic qualification review team will determine whether an original qualification finding based on single-frequency single-axis methods is valid in light of the multi-frequency multi-axis requirements of Standard Review Plans 3.9.2 and 3.10 and Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants." A subsidiary but equally important concern is whether hydrodynamic loads recently understood in new light for pressure suppression containments were properly included in the original component qualification findings. Identification and selection of such components will be based upon (1) review of information contained in Tables 3.10-1 and 3.10-2 of the Final Safety Analysis Report and (2) review by us of installed equipment at the Zimmer site.

Our seismic qualification review team is reviewing the nuclear steam supply system and balance-of-plant equipment lists and inspected the Zimmer Station balance of plant equipment already installed at the site. This review will evaluate the qualification testing to determine that the effect of the combination of seismic and hydrodyanmic loads have been properly accounted for. On the basis of the review audit, site visit the seismic qualification review team will ascertain whether any equipment or components have to be retested. We initiated discussions with the applicant to develop a mutually acceptable resolution of any problems arising in this area. We expect a timely resolution of this issue and will present the results in a supplement to this report.

3.10.4 Evaluation

We conclude that the seismic qualification testing program which has been implemented for seismic Category I instrumentation and electrical equipment as supplemented by the program described in subsection 3.10.3 above will provide adequate assurance that such equipment will function properly during the excitation from vibratory forces imposed by the safe shutdown earthquake and under the conditons of post-accident operation. We also conclude that this program constitutes an acceptable basis for satisfying the applicable requirements of General Design Criterion 2, when those items which remain to be qualified have been seismically qualified and the onsite seismic audit has been completed.

3.11 Environmental Design of Mechanical and Electrical Equipment

3.11.1 Discussion

We review this information as detailed in the Standard Review Plan, and also determine the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format for Safety Analysis Reports.

3.11.2 Findings

Section 3.11 of the Final Safety Analysis Report was reviewed to determine whether the required environmental capability of all safety-related equipment, i.e., the capability to perform design safety functions under normal and accident environments, and conformance with our criteria have been adequately demonstrated.

The Final Safety Analysis Report Tables 3.11-1, 3.11-2, 3.11-3 and 3.11-4 indicated that there were omissions of systems required for safety and some unqualified safety equipment in the Zimmer station design.

The applicant committed to complete the environmental qualification of the remaining Class IE instrumentation and equipment and to provide complete and correct tables in Final Safety Analysis Report Section 3.11.

The applicant satisfied this commitment in Final Safety Analysis Report Amendment 34. In addition, we reviewed the applicant's responses to Inspection and Enforcement Bulletins 77-05, 77.05A and 78-02 and find that our concerns which are expressed therein are not applicable to the Zimmer Station.

3.11.3 Variations in Energy Supply

Institute of Electrical and Electronic Engineers Standard 279-1971 requires that Class IE equipment be qualified for the range of transient and steady-state conditions of both the energy supply and the environments. Institute of Electrical and

Electronics Engineers Standard 323-1971 reflects this requirement in its definition of service conditions. Our audit of the test procedures and results for the flammability control system revealed that the equipment was tested only at nominal conditions of the energy supply. The procedures were changed and the equipment tested at the extremes of the expected variations in supply which were reviewed and found to satisfy our requirements. During the first site visit, we also reviewed and found acceptable the qualification of the 4.16 kilovolt switch groups and 480 volt engineered safety system substations.

3.11.4 Evaluation

The applicant identified all the safety-related mechanical and electrical equipment, defined the normal and postulated accident environments that this equipment may be subjected to, and described the environmental qualification program that has been performed to demonstrate its required environmental capability. We conclude from our review that there is assurance that all items of safety-related equipment will be capable of performing needed safety functions under normal and accident environmental conditions.

3.11.5 Post-Accident Chemical and Radiation Environment

The applicant addressed the chemical and radiation environmental conditions to be used in design of the engineered safety features mechanical and electrical equipment in the containment for the postulated design basis accidents. No chemical will be added to the spray injection for post-design basis accident mitigation. Therefore, the primary containment atmosphere is not expected to have a significant amount of harmful chemicals, which could be released from the reactor during normal operation or during and following a postulated accident.

The applicant stated that the engineered safety features equipment and components inside the primary containment were designed to operate in a post-accident environment of an integrated gamma dose of 2.6×10^7 rads. This design criterion was based on a calculation of radiation exposures integrated over the 40-year life of the plant plus a six-month design basis accident radiation exposure. This calculation is dependent on the size and type of containment. The value of 2.6×10^7 rads for an integrated gamma dose appears consistent with the source terms of Regulatory Guides 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and is therefore acceptable. The applicant has indicated that the radiation tolerance of the engineered safety features equipment is 10^8 rads. We conclude that the radiation environment of safety features equipment is adequately defined and that the equipment has suitable tolerance to operate in the postulated design basis accident environment.

4.0 REACTOR

4.1 Summary Description

The Wm. H. Zimmer Nuclear Power Station nuclear steam supply system is the BWR/5 nuclear steam supply system which includes a General Electric Company boiling water reactor to generate steam for direct use in the balance-of-plant steam driven turbine generator. The design of the BWR/5 reactor is in some respects different from other boiling water reactors which have been reviewed and approved by us at the final design stage. Therefore, we consider Zimmer the lead review for BWR/5 systems.

The fuel and heat source for the BWR/5 reactor consists of slightly enriched uranium dioxide pellets contained in sealed zirconium alloy tubes about one-half inch in diameter. These fuel rods, which are about 12 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. Five hundred and sixty of these fuel assemblies form a roughly cylindrical core.

The core is supported in a 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 core shroud and the reactor vessel. The high velocity water from the jet pump 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 initially by the steam separators and then by steam dryers. The steam flows to the turbine generator through four 24-inch diameter main steam lines. The heated condensate returns to the reactor through two 18-inch feedwater lines and is injected into the annulus between the core 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 are provided to insert the control rods axially within the core to any degree desired or to insert the control rods fully and to operate the flow control valves in the recirculation lines.

4.2 Fuel System Design

The fuel design for Zimmer Unit 1, is identical, except for the channel thickness, to the General Electric Company 8 x 8 fuel assembly design currently in operation in 14 boiling water reactors. The Zimmer fuel design is identical to that given in the General Electric Standard Safety Analysis Report (GESSAR), April 1973, and the generic reload report, NEDO-20360, Revision 1, "General Electric BWR Generic Reload Application for 8 x 8 Fuel," November 1974, except that the channel box is 0.020 inch thicker.

Mechanical and operating parameters for the 8 x 8 fuel assemblies are compared with the previously used 7 x 7 assemblies in Table 4-1. The smaller diameter rods, with lower linear heat generation rate and increased cladding thickness/ diameter ratio for the 8 x 8 fuel design compared with the 7 x 7 fuel assemblies, result in increased safety margins with respect to maximum design linear power and maximum fuel temperatures. In addition, the Zimmer 8 x 8 fuel assemblies have the following features: (a) finger springs for controlling moderator/coolant bypass flow at the interface of the channel and lower tie plate, and (b) bypass flow holes drilled in the lower tie plate to provide an alternate flow path. These features are currently found in most of the 8 x 8 fuel assemblies operating in boiling water reactors and have shown satisfactory performance.

Fuel performance calculations that account for the effects of fuel densification have been performed with a version of the General Electric Company analytical model GEGAP III contained in General Electric Company Topical Report, NEDO-20181, "GEGAP III: A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods," November 1973, which has been approved by us in a letter from V. A. Moore (USAEC) to I. S. Mitchell (GE), dated March 22, 1974. The effects of fuel densification on the fuel rod will increase the stored energy, increase the linear thermal output, and increase the probability of local power spikes from axial gaps. The primary effects of densification on the fuel rod mechanical design are manifested in calculations of fuel/cladding gap conductance and cladding collapse time. The approved analytical model incorporates time-dependent fuel densification, time-dependent gap closure and cladding creepdown for the calculation of gap conductance. Cladding collapse has not been observed in boiling water reactor fuel rods and is calculated with a code, SAFE-COLAPS contained in General Electric Company Topical Report, NEDO-20606A, "Creep Collapse Analysis of BWR Fuel Using SAFE-COLAPS Model," August 1976, approved by us in a letter from W. R. Butler (NRC) to I. Stuart (GE), dated April 4, 1975, to occur at core residence times in excess of five years, which is greater than the lifetime of the fuel.

Recently we questioned the validity of fission gas release calculations in most fuel performance codes including GEGAP-III for a burnup greater than 20,000 megawatt days per ton of Uranium. The General Electric Company was informed of this concern on November 23, 1976 and was provided with a method of correcting gas release

TABLE 4-1

COMPARISON OF PARAMETERS FOR 8 X 8 AND 7 X 7 FUEL ASSEMBLY DESIGN

Parameter	7 x 7	Zimmer 8 x 8
Fuel Rods/Assembly	49	63
Channel Thickness (inches)	0.080	0.100
Active Fuel Length (inches)	144	146
Uranium Weight/Assy. (pounds)	412.8	409.3
Rod-to-Rod Pitch (inches)	0.738	0.640
Water/Fuel Ratio (cold)	2.53	2.60
Cladding OD (inches)	0.563	0.493
Cladding Thickness (inches)	0.037	0.034
Thickness/Diameter Ratio	0.0657	0.0689
Fuel Pellet OD (inches)	0.477	0.416
Pellet/Clad Diametral Gap (mils)	12	9
Maximum Linear Heat Generation Rate (kilowatts per foot)	17.5	13.4
Maximum Fuel Temp. (degrees Fahrenheit)	4380	3325

calculations for burnups greater than 20,000 megawatt days per ton of Uranium (see NUREG-0418, "Fission Gas Release from Fuel at High Burnup," March 1978). Since there was no question of the adequacy of GEGAP-III for burnups below 20,000 megawatt days per ton of Uranium, the Zimmer Final Safety Analysis Report calculations are acceptable for operation early in life until the peak local burnup reaches 20,000 megawatt days per ton of Uranium. For burnups in excess of that value, GEGAP-III calculations (and other affected analyses) for Zimmer must be redone using the correction method mentioned above or such modified methods that might be submitted by the General Electric Company and approved by us.

The General Electric Company has provided (G. G. Sherwood (GE) letter to D. F. Ross (NRC), December 22, 1976) a generic reanalysis of fuel performance calculations using GEGAP-III with our fission gas correcton factor for BWR 2/3/4 plants with 7 x 7 and 8 x 8 fuel assemblies. The only affected safety analysis was the loss-of-coolant analysis. Although the calculations were not specifically performed for a BWR/5 plant, we conclude from our review that the 8 x 8 analysis performed for early reflooding plants will bound the BWR/5 case. The reanalysis results in less than an 85 degrees Fahrenheit increase in calculated peak cladding temperature at a target planar average exposure of 30,000 megawatt days per ton Uranium. For Zimmer, calculated peak cladding temperatures for the loss-of-coolant accident, as given in Table 6.3-1 of the Final Safety Analysis Report, are in the range of 1821 degrees Fahrenheit at 15,000 megawatt days per ton of Uranium to 1632 degrees Fahrenheit at 30,000 megawatt days per ton Uranium. Thus, a maximum increase of 85 degrees Fahrenheit in peak cladding temperature due to the fission gas effect will be insufficient to drive the cladding temperatures to or above the 2200 degrees Fahrenheit loss-of-coolant accident limit. On the basis that (a) only the loss-of-coolant accident is affected by the increased fission gas release above 20,000 megawatt days per ton Uranium, (b) loss-of-coolant accident limits appear not to be exceeded by this effect, and (c) confirmatory calculations will be provided prior to operation above 20,000 megawatt days per ton of Uranium, the operation of Zimmer fuel in the burnup range above 20,000 megawatt days per ton Uranium requires no licensing restrictions due to the revision in fission gas release.

Several uranium-235 enrichments are used within each fuel assembly to reduce the local power peaking factor. Gadolinium, a burnable poison, is also used to supplement the enrichment pattern and control rods in flattening the power distribution of the core. Gadolinium, in the form of gadolinia-urania pellets, is used in some of the interior rods. Gadolinium-bearing fuel was first incorporated as a regular component into the initial cores of Quad Cities, Units 1 and 2 with operation starting in 1971 and 1972, respectively. Since 1965, a substantial number of test and regular gadolinia-urania rods have been successfully irradiated to appreciable exposures.

To eliminate significant vibration of instrument tubes and source tubes and the resultant wear on channel box corners, Zimmer, Unit 1 incorporates plant modifications similar to those described in the General Electric Company report, NEDO-21156, "Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibration," January 1976. This modification eliminates the bypass flow holes in the lower core support plate and adds two holes in the lower tie plate of each assembly to provide an alternate flow path. Concomitant effects of this plant modification on the mechanical design are negligible and our approval for generic application is given in a letter from K. Goller (NRC) to G. G. Sherwood (GE), dated April 12, 1977.

A number of generic issues have been identified by us and considered in the Zimmer review. These generic issues include the following: pellet/cladding interaction, waterlogging failures, channel box deflection, flow blockage consequences and seismic and loss-of-coolant accident load effects. Each of these items will be treated in detail in the following discussion.

Pellet/cladding interaction is addressed in the Zimmer Final Safety Analysis Report. Since 1972, the General Electric Company has made changes in the fuel assembly design and in the mode of reactor operation to reduce the incidence of pellet/cladding interaction failures. To minimize the potential for pellet ridging, a shorter, chamfered pellet with no dishing will be used. The 8 x 8 design also includes a higher annealing temperature for the Zircaloy cladding to achieve better uniformity of the mechanical properties. In addition to these design changes, the General Electric Company recommended specific operating procedures identified as Preconditioning Interim Operating Management Recommendations. Under these procedures, the fuel is preconditioned for subsequent full-power operation and power cycling by being first taken to full power on a slow ramp.

We conclude on the basis of our generic review to date that the design changes and operating restrictions have been effective in reducing the potential for pellet/cladding interaction and reducing the incidence of related fuel failures during normal operation. While pellet/cladding interaction is being studied generically to determine if licensing criteria should be revised, current criteria are satified by the Zimmer reactor fuel. Should licensing criteria related to pellet/cladding interaction change in the future, the effects of such a change would be reviewed for all plants including Zimmer.

The potential and consequences of operating with waterlogged fuel rods is addressed adequately in the Zimmer Final Safety Analysis Report. We have independently reviewed the safety aspects of waterlogged fuel behavior (NUREG-0303, March 1978, "Evaluation of the Behavior of Water Logged Fuel Rod Failures in LWR's"). A survey of the available information, which includes (1) test results from the SPERT reactor, NSRR reactor in Japan and (2) observations of waterlogging failures in commercial reactors, indicates that rupture of a waterlogged fuel rod should not result in failure propagation or signifiant fuel assembly damage that would affect coolability of the fuel rod assembly. We thus agree that the evaluation of waterlogging failures as presented in the Zimmer Final Safety Analysis Report is correct.

Fuel assembly response to seismic and loss-of-coolant accident loads has been considered generically by the General Electric Company in a report NEDE-21175, "BWR 6 Fuel Assembly - Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings," July 1977. We have reviewed that report and, based on our review to date, we find that the analytical methods are acceptable and that in general the resultant loads are small compared with the fuel assembly component strengths. In our general review to date, we have not evaluated the asymmetric blowdown loads on the reactor fuel.

In accordance with our related Task Action Plan A-2 (see NUREG-0371, November 1978), General Electric Company has performed an analysis using the best available methodology and criteria and found the 8 x 8 fuel acceptable. Although we have not completed our review of this analysis, Task Action Plan A-2 stated the bases for determining that there is reasonable assurance that operation of boiling water reactor plants will not present an undue risk to the health and safety of the public pending the completion of the generic task related to this matter.

Boiling water reactor channel box deflection or distortion from irradiation growth during normal plant operation is treated generically in the General Electric Company report NEDE-21354-P, "BWR Fuel Channel Mechanical Design and Deflection," September 1976. This deflection is used as input to predict coolant bypass flow and to evaluate the possibility of control rod interference. Based on our review of the data presented in the report, we conclude that boiling water reactor channel deflections are applied in a conservative manner for bypass calculations. Further, based on the slow process of channel creep and growth and the test procedures applied to determine control rod interference, we conclude that the channel deflections raise no major safety concerns about the availability of control rods for maneuvering or scram.

The 8 x 8 fuel design is currently in operation in 14 boiling water reactors and a significant number of fuel bundles (250) are in their third irradiation cycle. A detailed post-irradiation examination has been performed at Quad Cities and Monticello on the lead test assemblies at the end of two cycles and the results indicate satisfactory performance (NEDM-23669, August 1977 and NEDM-21080, October 1975, respectively).

On the basis of our review of the Zimmer 8 x 8 fuel design analysis, technical specifications that will limit off-gas and effluent activity, and the confirmatory results from irradiated assemblies, we conclude that there is reasonable assurance that the cladding integrity of Zimmer, Unit 1 fuel will be maintained, that significant amounts of radioactivity will not be released, and that neither accidents nor earthquake-induced loads will result in either an inability to cool the fuel or interference with control rod insertion.

4.3 Nuclear Design

Our review of the nuclear design of the Wm. H. Zimmer Nuclear Power Station, Unit 1 was based on information supplied by the applicant in the Final Safety Analysis Report and amendments thereto. Our review was conducted in accordance with Section 4.3 of the Standard Review Plan.

4.3.1 Design Bases

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

We find the design bases presented in the Final Safety Analysis Report to be acceptable.

4.3.2 Design Description

Descriptions of the first cycle enrichment distribution, burnable poison loading, plutonium buildup, delayed neutron fraction, neutron lifetime, and core burnup have been provided. The values presented for these parameters are consistent with those reviewed and approved on previously approved boiling water reactor plants, now in operation, such as Hatch 2.

Power Distribution

We reviewed the methods used by the General Electric Company to predict power distributions during core lifetime (see subsection 4.3.3 of this report). These methods have been compared to measured power distribution in operating boiling water reactors in order to demonstrate their acceptability. Power distributions are controlled during reactor operation by adherence to predetermined control rod sequences so as to limit the maximum heat generation rate and minimum critical power ratio to values specified in the technical specifications. Power distributions will be monitored during reactor operation by the incore detector system. This system, described in the General Electric Company Topical Report APED 5706, "Incore Neutron Monitoring

System for General Electric Water Reactors," approved by us in a letter to General Electric Company, dated September 1, 1971, consists of a source range monitoring subsystem (up to 10^{-5} full power), an intermediate range monitoring subsystem (10^{-6} to 0.2 full power), and a local and average power range monitoring subsystem (N0.05 - 1.5 full power). In addition a traversing incore probe subsystem is used to calibrate the local power range monitors and to obtain detailed axial flux distributions.

A study of power distributions in boiling water reactors (BWR/4 and BWR/5) is given in NEDE-20944-P, "BWR/4 and BWR/5 Fuel Design." A comparison of calculated and measured power distributions is given in NEDO-20946, "BWR Simulator Verification Methods," approved by us in a letter to General Electric Company, dated September 22, 1976 (see subsection 4.3.3 of this report). This comparison demonstrates that the General Electric Company design methods are capable of adequately representing reactor operating states.

We conclude that discussions of power distributions in Section 4.3 of the Final Safety Analysis Report and in the other documents referenced above are acceptable. We further conclude that the information presented concerning monitoring of power distributions presented in the Final Safety Analysis Report and in Topical Report NEDO-20340, "Process Computer Performance Evaluation Accuracy," is acceptable.

Reactivity Coefficients

The most significant reactivity coefficients with respect to the stability and dynamic behavior of the reactor are the void coefficient and the Doppler coefficient. Of lesser significance is the moderator temperature coefficient. The fuel temperature, or Doppler, coefficient of reactivity will be negative at all operating conditions and times in life. The moderator void coefficient also will be always negative. The moderator temperature coefficient may become slightly positive for certain operating conditions but its effect is overshadowed by that of the other coefficients. The General Electric Company submitted a Topical Report, NEDO-20964, "Generation of Void and Doppler Reactivity Feedback for Application to BWR Design" which describes the methods used to obtain void and Doppler Reactivity coefficients. This report is currently under review (see subsection 4.3.3 of this report) and we conclude, based on the review to date, that predictions of the various reactivity coefficients are suitably performed (see Note 1 at end of subsection 4.3 of this report). We conclude that the multiplication by design conservatism factors (given in the Final Safety Analysis Report) ensures that the predicted values are suitably conservative for use in the point kinetics plant transient model.

Control Requirements

To allow for changes in reactivity due to reactor heatup (fuel and moderator temperature rise and void formation), load following (transient xenon), equilibrium xenon and samarium, and fuel burnup with consequent fission product buildup, a significant amount of excess reactivity is built into the core at beginning of life. In boiling water reactors this excess reactivity is accounted for by the control rods except for a portion of that needed to account for fuel burnup. That portion is accounted for by burnable poison located in the fuel assemblies. The burnable poison also functions to shape the radial and axial flux.

The applicant presented data to show that sufficent control exists to satisfy the above requirements with enough additional control to provide a cold xenon-free effective multiplication factor ≤ 0.99 at the most reactive point in the core lifetime with the most reactive rod stuck out of the core. NEDO-20946 (see subsection 4.3.2) provides comparison of calculated and measured cold critical states and provides a demonstration that calculation of shutdown margins is adequate. We conclude that suitably conservative assessments of reactivity control requirements have been made and that adequate reactivity control has been provided to assure shutdown capability, even with one rod stuck out of the core.

A standby liquid control system is provided which is completely independent of the control rod system and is capable of shutting down the reactor and maintaining it in the cold shutdown state at any time in core life (see subsection 4.6.3 of this report). This satisfies the requirement of General Design Criterion 26.

Control Rod Patterns and Reactivity Worth

Startup and operation of the reactor will be performed by manipulation of control rods and control of recirculation flow. The control rods will be withdrawn in sequence according to predetermined patterns. These patterns are established in such a way that the following design criteria are met:

(1) Control rod worths shall be limited so as to have acceptable consequences, as noted below, if a rod is dropped from the fully inserted to the fully withdrawn position (Rod Drop Accident).

(2) Control rod withdrawal increments and rod worths shall be limited so that withdrawal of a control rod by one notch does not produce a period that cannot be handled by the operator.

During operation the rod patterns are monitored by the rod worth minimizer and the rod sequence control system up to the preset power level (25 percent power). Above this power level, rod worths are not sufficient to violate the above criteria. The rods are withdrawn in the banked position withdrawal sequence which is described in the the General Electric Company Topical Report NEDO-21231, "Banked Position Withdrawal Sequence." This report was reviewed and approved by us in a letter to General Electric Company, dated January 17, 1978. Based on our review, we find use of the banked withdrawal sequence acceptable for Zimmer, Unit 1.

Above the preset power level the limits on heat generation rate and minimum critical power ratio are monitored by the process computer and by the rod block monitor, but no restrictions are placed on control rod sequence.

We conclude, based on our review, that the rod worth will be limited in the startup and low power range so that a dropped rod will not result in greater than 170 calories per gram enthalpy rise in the fuel (see introduction to section 15.0 of this report) and that reactivity insertion rates will be limited to those that may be controlled by the operator.

Stability

The stability of large boiling water reactors to xenon oscillations has been discussed in the the General Electric Company Topical Report APED-5640, "Xenon Considerations in Design of Large Boiling Water Reactors." These studies show that a boiling water reactor will be stable to any xenon-induced power oscillation because of the damping effect of the large, negative, spatially varying void coefficient. In addition attempts to induce undamped xenon oscillations in operating boiling water reactors confirm the presence of a large damping effect.

We reviewed the information provided by the applicant and conclude that large boiling water reactors such as Zimmer will be stable to xenon induced power oscillations.

Criticality of Fuel Assemblies

Our evaluation of this matter is discussed in subsections 9.1.1 and 9.1.2 of this report.

Vessel Irradiation

Neutron fluences at the reactor vessel were calculated for the Zimmer size plant using a one-dimensional discrete ordinates code. Continuous reactor operation at full power for 40 years was assumed and the reactor mid-plane flux was calculated. The calculations incorporated fission distributions prepared from core physics data as an initial fixed source distribution. Anisotropic scattering effects were included outside the core and the resultant vessel fluence was 1.5×10^{18} neutrons per square centimeter for neutrons having energies greater than one million electron volts.

We conclude from our review that the calculated value of vessel fluence noted above is acceptable.

4.3.3 Analytical Methods

We reviewed and evaluated the information presented on the analytical methods. The basic calculational procedures which are used by the General Electric Company 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 groups 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 through-out a fuel bundle. These calculations are performed for an extensive combination 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 General Electric Company submitted a licensing topical report NEDE-20913-P, "Lattice Physics Methods," approved by us in a letter to the General Electric Company, dated September 22, 1976, which describes in detail the procedures outlined above. We reviewed this report and conclude that the methods employed are state-of-the-art. We further conclude that the methods satisfy the provisions of the Standard Review Plan for core physics methods.

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 one energy group and can couple neutron and thermal hydraulic phenomena. These three-dimensional calculations are performed using 24 axial nodes and one radial node per fuel bundle resulting in about 14,000 to 20,000 spatial nodes. This three-dimensional calculation provides power distributions, void distributions, control rod positions, reactivities, eigenvalues, and average cross sections for use in the one-dimensional axial calculations. The three-dimensional calculations have been described in a licensing topical report - NEDO-20953, "Three-Dimensional BWR Core Simulator," approved by us in a letter to General Electric Company, dated January 17, 1978, which was submitted by the General Electric Company. We reviewed this report and reached similar conclusions to those reached for the Lattice Physics Methods report.

The one-dimensional calculation referred to above is a space-time diffusion calculation which is 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 the General Electric Company with results obtained using the industry standard code, WIGLE, and shown to be conservative. Our consultant, Brookhaven National

Laboratories, performed an extensive study of boiling water reactor scram reactivity behavior (BNL-NUREG-50584, "A Dynamic Analysis of BWR Scram Reactivity Characteristics") and concludes that the end of cycle all rods out configuration represents the limiting condition for boiling water reactor scram system effectiveness. Thus, we conclude that the method and assumptions used by the General Electric Company to obtain the scram reactivity curve are acceptable.

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 moderator void, Doppler, and moderator temperature reactivity coefficients.

The General Electric Company submitted a topical report - NEDO-20964, "Generation of Void and Doppler Reactivity Feedback for Application to BWR Design." We are currently reviewing this report. Based on the review to date, we conclude that the Doppler coefficient is suitably calculated. The void coefficient may not be conservative. However, any nonconservatism in the void coefficient may be accommodated by conservatisms in other portions of the transient calculations or by restrictions on the operating limits for Zimmer (see note at the end of subsection 4.3.4 of this report).

The effect of spatially varying xenon concentrations on the stability of a boiling water reactor is specifically discussed in the the General Electric Company Topical Report APED-5640, "Xenon Considerations in Design of Large Boiling Water Reactors (June 1968)." These studies show that a boiling water reactor will be stable to any xenon-induced power oscillations because of the damping effect of the large, negative, spatially varying void coefficient.

Comparisons between calculated and measured local and gross power distributions have been presented by the General Electrical Company in two topical reports - NEDO-20939, "Lattice Physics Methods Verification" and NEDO-20946, "BWR Simulator Methods Verification." Local (intrabundle) power distribution comparisons were made to data obtained from critical experiments and from gamma scans performed on operating plants. Gross radial and axial power distributions obtained from operating plants have been compared with values predicted by the boiling water reactor simulator code. These comparisons have yielded values for calculational uncertainties to be applied to power distributions. Comparisons have also been made of calculated values of cold, xenon-free reactivity and hot operating reactivity of a number of operating reactors as a function of cycle exposure. These comparisons have been used to establish shutdown reactivity requirements.

We reviewed the two topical reports, NEDO-20939 and NEDO-20946, and found them acceptable for reference in licensing actions. They were approved by us in letters to General Electric Company, both dated September 22, 1976.

4.3.4 Summary of Evaluation

The applicant described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and provided examples to demonstrate the ability of these methods to predict experimental results. We conclude that the information presented adequately demonstrates the ability of these analyses to predict reactivity and physics characteristics of the Zimmer, Unit 1 plant.

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is designed into the core. The applicant provided substantial information relating to core reactivity balances for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to shut down the reactor with at least a one percent fraction of reactivity subcritical margin in the cold xenon free condition at any time during the cycle with the most reactive control rod stuck in the fully withdrawn position.

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

Note - Recent calculations by our consultant, Brookhaven National Laboratories, have indicated that the void coefficient value used in the point kinetics model may not be conservative since spatial weighting has not been properly accounted for. In particular, the amount of reactivity inserted by the collapse of voids during overpressure transients at end-of-life may be underpredicted by the point kinetics model. We are pursuing this issue with the General Electric Company on a generic basis and will impose appropriate operating restrictions to account for lack of conservatism if necessary in the technical specifications (see subsection 4.4.1 of this report).

4.4 Thermal and Hydraulic Design

4.4.1 Evaluation

The thermal-hydraulic safety design bases for Zimmer follow:

(1) No fuel damage should occur as a result of abnormal operational occurrences. Specifically, the minimum critical power ratio operating limit is specified so that at least 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during the most severe abnormal operational transient event.

(2) The maximum linear heat generation rate should not permit fuel centerline melting for abnormal operational occurrences.

(3) There should be no undamped oscillations or other hydraulic instabilities.

A summary of the thermal-hydraulic parameters for Zimmer-1 is given in Table 4-2. A comparison with the parameters for Hatch-2 is given for reference. The Hatch-2 core design has been previously approved by us in the Safety Evaluation Report issued in June 1978.

The thermal-hydraulic parameters for Zimmer-1 and Hatch-2 are quite similar; both reactors have 2436 megawatt thermal, 560 bundle cores. The total core flow for Zimmer-1 is slightly higher than for Hatch-2, as is the inlet enthalpy. The primary difference between the two cores is that Hatch-2 has the new 8 x 8, two water rod fuel (150-inch heated length) whereas Zimmer has the 8 x 8, one water rod fuel (146-inch heated length). The bundle radial peaking pattern for the one water rod bundle is not as flat as for the two water rod bundle, but a more extensive critical power data base exists for the one water rod bundle than for the two water rod bundle.

The core and fuel design bases for steady-state operation, i.e., the minimum critical power ratio and maximum linear heat generation rate limits, have been defined to provide margin between the steady-state operating condition and any fuel damage to accommodate uncertainties and to assure that no fuel damage results even during the worst anticipated transient condition any time in life. GETAB, General Electric Thermal Analysis Basis, is used for Zimmer-1. The figure of merit chosen for reactor design and operation is the critical power ratio, defined as the ratio of the critical bundle power to the operating bundle power. In GETAB, the uncertainties associated with the parameters affecting steady-state bundle power are treated statistically in order to satisfy the criterion that, during a transient, 99.9 percent of the rods in the core will not experience boiling transition. This method is acceptable to us based on our review of NEDO-10958A, "General Electric BWR Thermal Analysis Basis (GETAB) Data Correlation and Design Application," which we approved in our letter to the General Electric Company, dated October 24, 1974. For Zimmer 1, the minimum critical power ratio ≥ 1.07 during normal operation and anticipated operational occurrences will meet this criterion.

During a reactor transient, fuel damage could occur if cooling were inadequate or if cladding fractured due to relative expansion of the pellet inside the cladding. The boiling transition limit assures adequate heat transfer and a one percent plastic strain limit for the zircaloy cladding assures no fuel damage due to overstraining. The linear heat generation rate required to cause one percent strain is approximately 25 kilowatts per foot in unirradiated fuel decreasing to approximately 20.5 kilowatts per foot at an exposure of 40,000 megawatt days per ton of Uranium.

TABLE 4-2

THERMAL-HYDRAULIC CHARACTERISTICS OF ZIMMER-1

	Zimmer-1 (218-560)	Hatch-2 (218-560)
Design thermal output, megawatts thermal	2436	2436
Steam flow rate at 420 degrees Fahrenheit Final feedwater temperature, 10^6 pounds per hour	10.477	10.47
Core coolant flow rate, 10^6 pounds per hour	78.5	77.0
Feedwater flow rate, 10^6 pounds per hour	10.477	10.44
System pressure, nominal in steam dome, pounds per square inch absolute	1020	1020
System pressure, nominal core design, pounds per square inch absolute	1035	1035
Average power density, kilowatts per liter	50.5	49.15
Maximum thermal output, kilowatts per foot	13.4	13.4
Average thermal output, kilowatts per foot	5.59	5.38
Core total heat transfer area, square feet	55,401	54,879
Fuel type	8 x 8	8 x 8
Number of water rods/bundle	1	2
Core inlet enthalpy at 420 degrees Fahrenheit, British thermal units per pounds	527.4	526.9
Core maximum exit void within assemblies, percent	76	76.3
Core average void fraction, active coolant, percent	41.9	42.2
Active coolant flow area/ass'y, square inches	15.50	15.82
Core average inlet velocity, feet per second	7.0	6.6
Total core pressure drop, pounds per square inch	24.8	23.9
Core support plate pressure drop, pounds per square inch	20.9	19.46
Average orifice pressure drop, pounds per square inch		
Central Region	8.32	8.0
Peripheral Region	17.23	16.52
Minimum critical power ration safety limit	1.07	1.06
Number of bundles	560	560
Number of fuel rods/bundle	63	62
Rod outside diameter, inches		
Fuel Rod	.493	.483
Water Rod	.493	.591
Active fuel length, inches	146	150
Rod pitch, inches	.640	.640

The applicant indicated that incipient center melting of the uranium dioxide pellet occurs in the range of 19 to 21 kilowatts per foot; this is higher than the peak linear heat generation rate of 17 kilowatts per foot during any abnormal operating transient. The operating limit for peak linear heat generation rate, 13.4 kilowatts per foot, is acceptable since it results in a sustained maximum linear heat generation rate less than 17 kilowatts per foot during transients. We conclude that the fuel limits stated by the applicant are acceptable and that the Zimmer plant has acceptable operating margins with these limits.

The steady-state operating limit for minimum critical power ratio is 1.24. This value is based on the assumption that the turbine trip without bypass event does not allow more than 0.1 percent of the fuel rods to experience boiling transition. The value is also based upon transient calculations using the REDY computer code described in NEDO-10802, "Analytical Methods of Plant Transient Evaluations for General Electric Boiling Water Reactors." The review of the General Electric Company analytical methods described in NEDO-10802 is still under consideration by us. In this regard, three turbine trip tests were performed at the Peach Bottom Unit 2 boiling water reactor. The purpose of the tests was to provide experimental data for code verification and to improve the understanding of integral plant behavior under transient conditions. These tests were jointly sponsored by the Philadelphia Electric Company, General Electric Company and the Electric Power Research Institute who planned and performed the test program on their own initiative. The results from the program have revealed that in certain cases the results predicted by REDY are nonconservative. We are reviewing this matter on a generic basis with the General Electric Company and evaluating a new calculation basis using the General Electric Company's new computer code ODYN for the load rejection without turbine bypass transient. We expect to complete our evaluation prior to licensing of the Zimmer plant and arrive at an acceptable operating minimum critical power ratio limit. The resolution of this matter must be resolved prior to power operation and will be reported in a supplement to this report.

We are performing a generic study of the hydrodynamic stability characteristics of light water reactors under normal operation, anticipated transients, and accident conditions under Task Action Plan B-19, "Thermal-Hydraulic Stability." The results of this study will be applied to our review and acceptance of stability analyses and analytical methods now in use by the reactor vendors. In the interim, we conclude that past operating experience, stability tests, and the inherent thermal-hydraulic characteristics of light water reactors provide a basis for accepting the Zimmer stability evaluation for normal operation and anticipated transient events. However, in order to provide additional margin to stability limits, natural circulation operation of Zimmer will be prohibited until our review of these conditions is complete. Any action resulting from our study will be applied to Zimmer.

Zimmer-1 uses valve flow control rather than pump speed flow control as has been used on boiling water plants using the BWR/4 design. The principal modes of normal operation with valve flow control low frequency motor generator sets are summarized in the following. The recirculation pumps are started on the 100 percent speed power source in order to unseat the pump bearings. Suction and block valves are fully open and the flow control valves are in the minimum position. When the pumps are near full speed, the main power source is tripped and the pumps allowed to coast down to near 25 percent speed, where the low frequency motor generator set will power the pumps and motors. The flow control valves are then opened to the maximum position, at which point the reactor heatup and pressurization can commence. When operating pressure has been established, reactor power can be increased. This power increase will follow a line within Region I of the power-flow map in Figure 4-1 of this report.

When reactor power is greater than approximately 30 percent of rated, the low feedwater flow interlock is cleared and the recirculation pumps can be switched to the 100 percent speed power source. The flow control valves are closed to the minimum position before the speed change to prevent large increases in core power and a potential flux scram. This operation occurs within Region II of the operating map. The system is then brought to the desired power-flow level within the normal operating area of the map (Region IV) by opening the flow control valves and withdrawing control rods.

An interlock has been installed for each pump to prevent system startup or transfer from 25 percent to 100 percent pump speed unless the flow control valve is in the minimum position. The primary safety concern which requires this interlock is a sudden reactivity insertion due to sweeping the voids from the core should the transfer to 100 percent speed occur with the flow control valve in the maximum position. Thaiconcern will be resolved in the safety evaluation of Appendix H of the Final Safety Analysis Report and will be reported in a supplement to this report.

In Amendment 76 to the Final Safety Analysis Report, the applicant committed to install a loose parts monitoring system, acceptable to us, which will be operational prior to startup. The system will employ a minimum of two sensors at each natural loose parts collection region in the primary system. These sensors will be capable of detecting acoustic disturbances equal to or greater than 0.5 foot-pounds within three feet of the sensor.

Crud deposition causes gradual flow reduction in some light water reactor cores. However, measurement of core flow by jet pump pressure drop and core plate pressure drop would provide adequate indication of such flow reduction, if such should occur. Technical specifications will require that the core flow be checked every 24 hours and the average power range monitor flow biased scram be recalibrated every month. This frequency is sufficient to detect crud deposition effects. The effects of crud buildup have been considered in design calculations and are not considered to cause significant problems (3.5 mils additional crud on fuel rods reduces the minimum critical power ratio by 0.009).

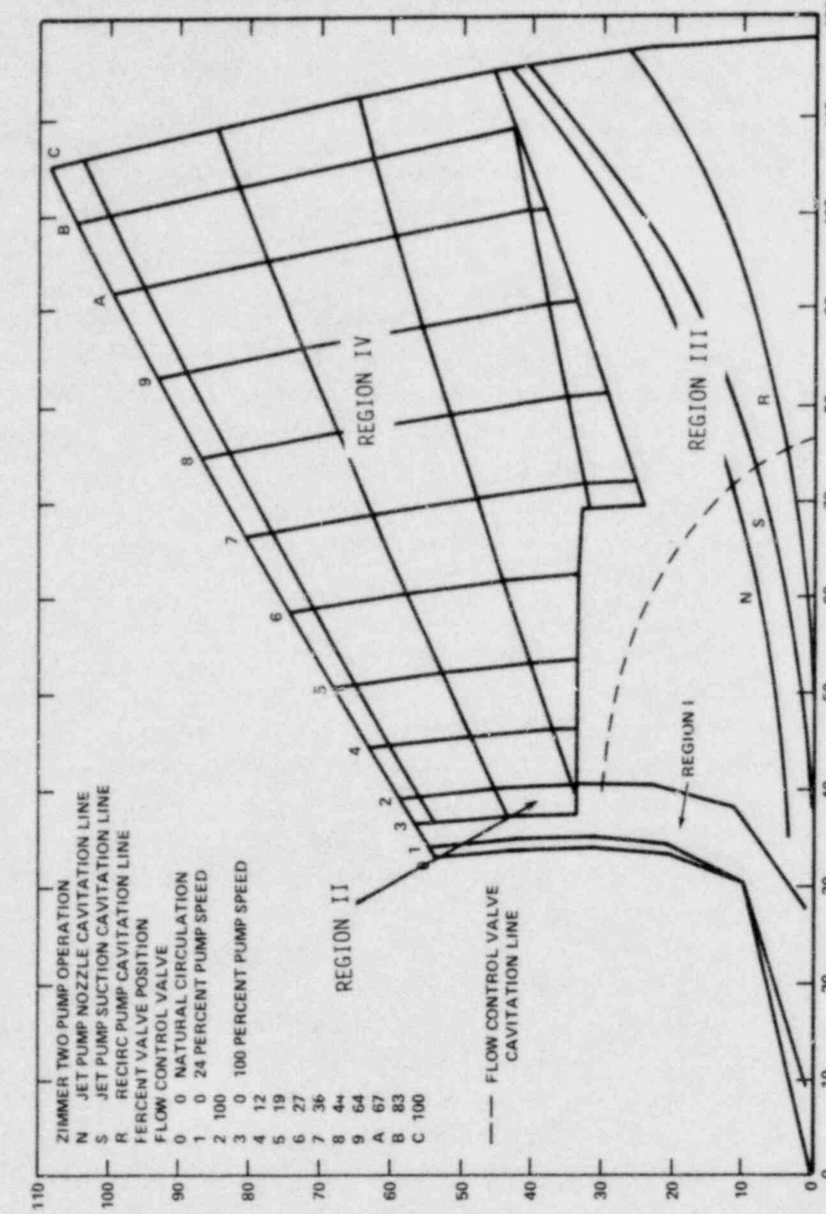

FIGURE 4-1 - REACTOR POWER FLOW OPERATING MAP

4.4.2 Summary

The thermal-hydraulic design of the core for Zimmer Unit 1 was reviewed. The scope of the review included the design criteria, implementation of the design criteria as represented by the final core design, and the steady-state analysis of the core thermal-hydraulic performance.

The applicant's thermal-hydraulic analyses were performed using analytical methods and correlations that had been previously reviewed by us and found acceptable. However, in some cases these methods may not be conservative. Our generic review and evaluation of the new ODYN code is expected to be completed prior to licensing of the Zimmer station. At that time we will determine an acceptable minimum critical power ratio limit.

We conclude that, with the exception noted above, the thermal-hydraulic design of the core conforms to the Commission's regulations and to applicable regulatory guides and our technical positions and is acceptable.

4.5 Reactor Materials

4.5.1 Control Rod System Structural Materials

The mechanical properties of structural materials selected for the control rod system components exposed to the reactor coolant satisfy Appendix I of Section III of the American Society of Mechanical Engineers Code, or Part A of Section II of the Code.

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

The compatibility of all materials used in the control rod system in contact with the reactor coolant satisfies the criteria for Articles NB-2160 and NB-3120 of Section II of the Code. Both martensitic and precipitation-hardening stainless steels have been given tempering or aging treatments in accordance with our positions. Cleaning and cleanliness control are in accordance with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

During routine maintenance inspections of a General Electric Company designed boiling water reactor in June 1975, dye penetrant inspections of control rod drive components revealed fine cracks in some of the control rod tubes. Subsequent inspections of other drives that had been in operation disclosed similar cracks. Conventional metallography and scanning electron microscopy identified the cracking as inter-granular in nature. The cracks were generally circumferential and appeared mainly where the wall thickness changes in the area between the ports. The cracks had developed from the outside of the tube, but none of the cracks were through the wall. They were generally shallow, less than half the wall thickness. Many of the cracks were "tight" and filled with oxide.

Operating experience obtained from 270 boiling water reactor years and results of the 779 control rod tubes (out of about 4000 drives in service) inspected by the dye penetrant technique at 22 sites disclosed that partial cracking had occurred in 78 tubes at 11 of the sites.

We have established a Category B Technical Activity (No. B-48) to address and resolve the issue of boiling water reactor control rod drive mechanical failures.

The control rod tube cracking that has occurred to date is generally shallow, intermittent, and very "tight", and has not impaired the control rod drive's ability to meet its functional requirements. We have requested information from the applicant as to the detailed resolution of this problem for the Zimmer plant. The subject will be reported in a supplement to this report.

Subject to resolution of the above generic concern, we conclude that conformance with the codes, standards and regulatory guides indicated above and the use of acceptable minimum tempering or aging temperatures for martensitic and precipitation-hardened stainless steels, constitutes an acceptable basis for meeting the requirements of General Design Criterion 26.

4.5.2 Reactor Internals Materials

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

The materials for reactor internals exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion on all materials is expected to be negligible based on guidance and criteria described in Standard Review Plan 4.5.2, Revision 0, "Reactor Internals Materials."

The controls imposed on reactor coolant chemistry provide reasonable assurance that the reactor internals will be adequately protected during operation from conditions which could lead to stress corrosion of the materials and loss of component structural integrity based on guidance and criteria described in Standard Review Plan 5.2.3, Revision 0, "Reactor Coolant Pressure Boundary Materials."

The controls imposed upon components constructed of austenitic stainless steel, as used in the reactor internals, satisfy the recommendations of our position on

Regulatory Guide 1.31, "Control of Stainless Steel Welding," Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Material selection, fabrication practices, examination procedures, and protection procedures performed in accordance with these recommendations provide reasonable assurance that the austenitic stainless steel used for reactor internals will be in a metallurgical condition which precludes susceptibility to stress corrosion cracking during service. The use of materials proven to be satisfactory by actual service experience and conformance with the recommendations of these regulatory guides constitutes an acceptable basis for meeting the applicable requirements of General Design Criteria 1 and 14.

4.6 Functional Design of Reactivity Control Systems

4.6.1 General

The control rod drive system and recirculation flow control system are designed for reactivity control during power operation. Reactivity is controlled in the event of fast transients by automatic rod insertion. In the event the reactor cannot be shut down with the control rods, the operator can actuate the standby liquid control system which pumps a solution of sodium pentaborate into the pressure vessel.

4.6.2 Control Rod System

Each control rod is moved by a separate hydraulic control unit. A supply pump provides each hydraulic control unit with water from the cycled condensate storage tank for cooling the rods and for moving them into and out of the core, with a spare pump on standby. The pump also provides water to a scram accumulator in each hydraulic control unit. The accumulator forces water into the control rod drive system to scram the control rod connected to that hydraulic control unit: at lower pressures the volume of water in the scram accumulator is sufficient to scram the rod--at higher pressures this volume of water is supplemented by water directly from the reactor vessel. A single failure in an hydraulic control unit would result in the failure of only one rod. In addition, any single component may be removed from the control rod drive system without disabling the protective system. The protection system has been designed to permit periodic functional testing during power operation with the capability to test individual scram channels and motion of individual control rods independently, thus complying with the requirements of General Design Criterion 21.

Preoperational tests of the control rod drive hydraulic system will be conducted. With all drives installed, flow rates, system pressures, and transient response of rods will be verified in both withdraw and insert directions; these parameters must be within the limits prescribed by design specifications. Periodic testing of the control rod drive system response times will be conducted to demonstrate continued acceptable performance.

The protection system is designed so that failure of all electrical power will cause the control rods to scram, thereby protecting the reactor. This complies with the requirements of General Design Criterion 23. The separation of the protection system which initiates a scram and the control system as required by General Design Criterion 24 is discussed in subsection 7.2 of this report.

A malfunction in the control rod system could result in a reactivity insertion. The applicant demonstrated in his safety analyses (Chapter 15 of the Final Safety Analysis Report) that the protection system limits these postulated transients within acceptable fuel response, as required by General Design Criterion 25.

The control rod drive system is designed to provide reactivity control under normal operation and anticipated operational occurrences with an appropriate allowance for a stuck rod. This capability is demonstrated by the safety analyses discussed in Chapter 15 of the Final Safety Analysis Report. This system is also capable of holding the core subcritical under cold shutdown conditions. The standby liquid control system is capable of accommodating reactivity changes during normal operation conditions (i.e., power changes, xenon burnout). Based on our evaluation, we conclude that these systems, taken together, satisfy the requirements of General Design Criteria 20, 26, and 28 as noted above.

The control rod drive system is capable of providing reactivity control following postulated accidents with an appropriate margin for a stuck rod. We have evaluated this capability which is demonstrated by the loss-of-coolant accident and rod dropout analyses presented by the applicant which, in turn, show that the consequences are acceptable and core cooling is maintained, as required by General Design Criteria 20, 27, and 28.

The control rod drive reactivity control systems satisfy the requirements of the General Design Criteria noted above and are acceptable to us.

4.6.3 Standby Liquid Control System

The standby liquid control system is intended to accomplish shutdown in the unlikely event the control rods remain fixed in the rated power pattern. It provides a manual, redundant, independent and diverse means to attain and maintain the reactor in a subcritical condition from any power operation to cold shutdown by injecting a solution of sodium pentaborate into the reactor vessel.

The system consists of a storage tank, a test tank, two positive-displacement pumps, two explosive valves, and associated local valves and controls located within the reactor building. A heater system maintains the solution temperature between 75 degrees Fahrenheit to 85 degrees Fahrenheit, to prevent precipitation of the sodium pentaborate from the solution during storage. High and low liquid level and temperatures are alarmed in the control room. The two parallel pumps draw the solution from the storage tank via a common suction line. The discharge from the two pump lines are provided with check valves, a cross over line and an explosive valve. They are arranged such that the failure of either pump or explosive valve will not prevent the sodium pentaborate solution from entering the reactor vessel. The explosive squib causes a plug to be sheared-off so as to permit entry of the solution.

System actuation is accomplished by two manually operated two position key-locked switches at the control console. Switching starts an injection pump, actuates the explosive valves and closes the reactor cleanup system to prevent loss or dilution or boron. Should the provided instrumentation indicate the solution is not entering the reactor vessel, the operator can immediately turn the second switch to actuate the alternate pump.

The standby liquid control system equipment for injection of the sodium pentaborate solution is designed to seismic Category I requirements. Based on our review we conclude that the design of the standby liquid control system conforms to the requirements of General Design Criterion 26 for a second reactivity control system and is, therefore, acceptable.

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

The principal components of the reactor coolant system are the reactor pressure vessel, reactor recirculation system, main steam lines including the outermost main steam isolation valve, and the pressure relief system. These components and systems comprise the major portion of the reactor coolant pressure boundary. The reactor coolant 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 out to the outermost isolation valve, are part of 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 American Society of Mechanical Engineers Code Class 1 reactor coolant pressure boundary components and their supports have been appropriately categorized with respect to the plant condition identified as normal, upset, emergency or faulted. The design limits used by the applicant for these plant conditions are consistent with the criteria recommended in Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components." Use of this criteria for the design of the reactor coolant pressure boundary components and supports provides reasonable assurance that (1) in the event an earthquake should occur at the site, or (2) other system upset, emergency or faulted conditions should develop, the resulting combined stresses imposed on the system components will not exceed the allowable design stresses and strain limits for the materials of construction. Limiting the stresses and strains under such loading combinations provides a basis for the design of the system components for the most adverse loadings postulated to occur during the service lifetime without loss of the system's structural integrity. Subject to satisfactory resoluton of the measures discussed in subsection 3.9.2 of this report the design load combinations and associated stress and deformation limits specified for American Society of Mechanical Engineers Code Class 1 components and their supports constitute an acceptable basis for design in satisfying the related requirements of General Design Criteria Nos. 1, 2 and 4.

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

Subject to satisfactory resolution of component testing procedures discussed in subsections 3.9.1 and 3.9.2 of this report the applicant's component operability assurance program constitutes an acceptable basis for implementing the requirements of General Design Criterion 1 as related to the operability of American Society of Mechanical Engineers Code Class 1 active valves.

The applicant was requested to provide stresses and corresponding margins of safety for critical reactor vessel support components due to transient pressure loads resulting from worst case blowdown (see subsection 3.9.1 of this report). The applicant provided the requested information in Amendment 76 to the Final Safety Analysis Report, August 28, 1978.

We conveyed our position to the applicant describing an acceptable procedure for assuring early detection of possible occurrence of cracks in the feedwater nozzles, control rod drive return line nozzles, and vessel blend radii. The applicant will provide a satisfactory response to our position regarding the augmented inservice inspection program. We initiated discussions with the applicant pertaining to this matter. Resolution of this matter will be provided in a supplement to this report. (See subsection 5.2.4 of this report).

Compliance with 10 CFR Part 50, Section 50.55a

Components of the reactor coolant pressure boundary as defined by the rules of 10 CFR Part 50, Section 50.55a, "Codes and Standards," have been properly classified as American Society of Mechanical Engineers Section III, Class A or Class 1, the Draft American Society of Mechanical Engineers Code for pumps and valves, Class 1, and American National Standards Institute B31.7, Class 1 components in Tables 3.2-1 and 3.2-2 of the Final Safety Analysis Report.

These Quality Group A reactor coolant pressure boundary components have been constructed to the maximum extent practical in accordance with the codes and addenda described in 10 CFR Part 50, Section 50.55a. The codes and addenda used for construction of the Quality Group A components are those that were required at the time of procurement of the components.

We reviewed the American Society of Mechanical Engineers codes and addenda used in the construction of the reactor coolant pressure boundary components and identified no major differences between these codes and addenda and those described in the codes and standards rule, 10 CFR 50.55a. We conclude that the American Society of Mechanical Engineers codes and addenda used in the construction of the Quality Group A reactor coolant pressure boundary components are designed in compliance with the codes and addenda described in 10 CFR 50.55a to the maximum extent practical; and that in accordance with considerations of 10 CFR 50.55a(a)(2), the Quality Group A components meet the requirements of 10 CFR 50.55a which provide adequate assurance that component quality is commensurate with the safety function of the reactor coolant pressure boundary.

Applicable Code Cases

The American Society of Mechanical Engineers Code Cases specified in Table 5.2-4 of the Final Safety Analysis Report whose requirements have been applied in the construction of pressure-retaining American Society of Mechanical Engineers Section III, Class A or Class 1, components with the reactor coolant pressure boundary (Quality Group Classification A), are in accordance with those code cases that are generally acceptable to us. We conclude that compliance with the requirements of these code cases, in conformance with the Commission's regulations, is expected to result in a component quality level that is commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

5.2.2 Overpressurization Protection

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

The Wm. H. Zimmer Nuclear Power Station has 13 safety/relief valves for overpressure protection during transients at operating conditions. These are located on the main steam lines with two valves on each of steam lines A and D, four on line B, and five on line C. The valves have a three-fold role:

(1) To limit the pressure within the reactor coolant pressure boundary to 110 percent of the design value (1.1 x 1250 = 1375 pounds per square inch gauge) by utilizing the valves in their safety mode.

(2) To relieve pressures at lower pressures in order to minimize safety action using the valve relief mode.

(3) To depressurize the reactor as part of the emergency core cooling system in the automatic depressurization system mode, for events involving small breaks in the reactor coolant system. This permits injection by the low pressure emergency core cooling system subsystems. There are six valves in the automatic depressurization system.

The valves are designed to meet seismic and quality standards consistent with requirements of Regulatory Guides 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants," and 1.29, "Seismic Design Classification," as discussed in subsection 3.9 of this report.

The nominal setpoints of the safety/relief valves are 1076 to 1116 pounds per square inch gauge in the relief mode and 1165 to 1205 pounds per square inch gauge in the safety mode.

The applicant analyzed a series of transients expected to require pressure relief actuation to prevent overpressurization. These are tabulated below:

Pressurization Events Resulting in Pressure Relief Actuation

(1) Generator Load Rejection with Bypass
(2) Turbine Trip with Bypass
(3) Turbine Trip w/o Bypass
(4) Closure of all Main Steam Isolation Valves
(5) Pressure Regulator Failure - Fail Open
(6) Loss of Auxiliary Power
(7) Feedwater Controller Failure - Maximum Flow

The applicant shows that the reactor coolant pressure at the vessel bottom remains below the American Society of Mechanical Engineers Code limit for the worst depressurization event, main steam isolation valve closure. The analysis used the high pressure scram in accordance with our requirements described in Standard Review Plan 5.2.2 "Overpressurization Protection." The analysis revealed that, even with as few as nine of the total number of thirteen valves operating to mitigate the transient, the maximum pressure was slightly less than the allowable limit of 1375 pounds per square inch gauge, corresponding to the American Society of Mechanical Engineers Code limit of 110 percent of the design pressure.

Trip of the recirculation pumps at high vessel pressure is used to provide partial mitigation of the consequences of anticipated transients without scram. It was found that the effects of this trip were not included in the analyses. We require that overpressurization calculations, including the effects of this trip, be submitted for evaluation. The results will be discussed in a supplement to this report.

The safety relief valve manufacturer tests the valves hydrostatically, for valve response, for set pressure, and for seat leakage prior to shipment to certify that design and performance requirements have been met. Specified manual and automatic actuation is verified during the preoperational test program. This complies with the preoperational testing of Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants." In addition, one-half of the valves will be tested to check set pressure at each refueling outage. We requested information concerning qualification tests and operating experience with the safety relief valves with respect to the safety mode of activation and performance of the automatic depressurization function. The results of our review of the response to this information request will be discussed in a supplement to this report.

The system designed to prevent overpressurization of the reactor coolant system was reviewed. We conclude that this system conforms to the requirements of General Design Criterion 15 and the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and is acceptable, subject to review of an analysis using the anticipated transient without scram pump trip, and analytical methods discussed in subsection 15.1 of this report. We will report on this matter in a supplement to this report.

5.2.3 Reactor Coolant Pressure Boundary Materials

Material Specifications and Compatibility with Reactor Coolant

The materials used for construction of components of the reactor coolant pressure boundary, including the reactor vessel and its appurtenances, have been identified by specification and found to be in conformance with the requirements of Section III of the American Society of Mechanical Engineers Code.

Based on meeting the guidance and acceptance criteria described in the Standard Review Plan 5.2.3, "Reactor Coolant Pressure Boundary Materials," we conclude that general corrosion of all materials except carbon and low alloy steel will be negligible. For these materials, conservative corrosion allowances have been provided for all exposed surfaces of carbon and low alloy steel in accordance with the requirements of the American Society Mechanical Engineers Code, Section III. The external nonmetallic insulation to be used on austenitic stainless steel components conforms with the requirements of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels."

Further protection against corrosion problems will be provided by control of the chemical environment. The composition of the reactor coolant will be controlled; and the proposed maximum contaminant levels have been shown by tests and service experience to be adequate to protect against corrosion and stress corrosion problems. The controls imposed on reactor coolant chemistry are in conformance with the recommendations of Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," and provide reasonable assurance that the reactor coolant pressure boundary components will be adequately protected during operation from conditions that could lead to stress corrosion of the materials and loss of structural integrity of a component.

The instrumentation and sampling provisions recommended in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," for monitoring reactor coolant water chemistry provide adequate capability to detect changes on a timely basis and to effect corrective actions to bring coolant chemistry within limits which will prevent stress corrosion. The use of materials of proven performance and the conformance with the recommendations of the regulatory guides constitutes an acceptable basis for satisfying the requirements of General Design Criteria 14 and 31.

Stainless Steel Pipe Cracking

In September 1974, cracking was experienced in the stainless steel piping at Dresden Nuclear Power Station, Unit No. 2. This was the first of a series of incidents of intergranular stress corrosion cracking that 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, we formed a special task group to investigate the causes of the cracking. The results and conclusions of the task group are given in our 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. It 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.

The applicant is implementing a number of the task group's recommendations for identified areas of high susceptibility. The two bypass lines have been entirely removed from the recirculation system. The core spray safe-ends are carbon steel. The control rod drive hydraulic return line and the existing reactor pressure vessel return nozzle will be valved off and the water returned to the reactor vessel by way of the drive seals. These measures are in accordance with task group recommendations specifically directed at plants under construction and are acceptable to us. By letter dated February 28, 1978, we requested that the applicant provide us with information regarding his implementation of the position stated in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." We will report on this matter in a supplement to this report.

Fabrication and Processing of Ferritic Materials

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

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

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

Compliance with these Code provisions and Commission Regulations constitutes an acceptable basis for satisfying the requirements of General Design Criterion 31.

The welding procedures used for ferritic steels in limited access areas comply with the intent of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessiblity." The fabrication practices and examination procedures performed in accordance with these recommendations provide reasonable assurance that ferritic stainless steels in the reactor coolant pressure boundary will be satisfactory in locations of restricted and visual accessibility.

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

Fabrication and Processing of Austenitic Stainless Steels

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

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

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection & Testing

We have completed our review of the applicant's preservice inspection and testing program but have not completed our review of requests for relief from the code requirements of 10 CFR, Part 50, Section 50.55a(g)(2). In addition, we are reviewing the augmented inservice inspection required for early detection of feedwater and control rod drive return nozzle and blend radii cracks (see subsection 5.2.1 of this report). The applicant has not submitted his revised inservice inspection program (10 CFR, Part 50, paragraph 50.55a(g)(4)). We will review this information when it becomes available to assure implementation of 50.55a(g)(4) requirements and will report the results of our review in a supplement to this report.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

A limited amount of leakage is to be expected from components forming the reactor coolant boundary. Components such as valve stem packing, circulating pump shaft seals, and flanges are not completely leak tight. This type of leakage (identified leakage) is monitored, limited, and separated from other leakage (unidentified).

Unidentified leakage may be an indication of a small through-wall flaw developed in the primary coolant boundary. Changes in the unidentified leakage may represent a change in flaw size which is a safety concern.

Identified leakage within containment is collected in the drywell floor drain sump and drywell equipment drain sump. The former normally collects leakage from control rod drive, valve flange leakage, floor drains, chilled cooling water system, and drywell cooling unit drains. The equipment drain sump collects condensate leakage from pump seals, reactor vessel head flange vent drain, and valve packing. Leakage in excess of background leakage would indicate initiation of unidentified leakage.

Drywell cooler condensate flow rate is indicated and alarmed in the control room; the alarm may be adjusted to annunciate when the flow rate approaches the technical specification limit. The condensate rate is related to drywell humidity content, indicating a possible reactor coolant pressure boundary leak from an unidentified source when an increase is noted. An unidentified leak can also be detected by an increase in drywell pressure. Drywell ambient temperature increase measured either directly, at various elevations, or by measuring an increase in water temperature leaving the air coolers will indicate an increase in leakage within the drywell. The air sampling system draws samples from the drywell atmosphere and monitors these samples for airborne radioactivity; an increase is annunciated in the control room.

Leakage through the first of the two seals forming the reactor vessel head closure is annunciated in the control room. Increased pressure between the seals is alarmed in the control room.

Recirculation pump seal leakage, indicated as a high flow rate in the drain line, is alarmed in the control room. Safety/relief valve leakage is detected by temperature sensors in the discharge line; leakage, indicated as temperature increase above ambient, is annunciated in the control room.

Valve stem packing leaks from power-operated valves in the nuclear boiler system, reactor water cleanup system, high and low pressure core spray systems, reactor core isolation system, residual heat removal system, and recirculation system are detected by monitoring packing leakoff for high temperature. Leaks are annunciated in the control room.

Temperature sensing systems are installed in rooms containing the reactor core isolation cooling, residual heat removal and reactor water cleanup systems equipment and in the main steam line tunnel. Annunciator and remote readouts from the temperature sensors are provided in the control room; an alarm is sounded and isolation initiated when a preset temperature is reached.

The reactor building sump normally collects leakage from the reactor water cleanup system and control rod drive systems and from other miscellaneous vents and drains. Instrumentation is present to monitor and indicate the rate of leakage into the sump which may be compared to the normal amount which is found during preoperational testing.

A differential flow measurement is used to measure reactor water cleanup systems leakage during operation; flow from the reactor vessel is compared with that returning to the reactor vessel. The signal is alarmed at a preset level.

Leak detection instrumentation is available after a safe shutdown earthquake as required by Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

The sumps in the drywell are capable of detecting an increase in unidentified leakage of one gallon within one hour which satisfies the sensitivity requirements of Regulatory Guide 1.45.

The leak detection systems may be tested for operability during plant operation. There are sufficient methods (at least three) to measure reactor coolant pressure boundary leakage within containment, in accordance with the requirements of Regulatory Guide 1.45.

The limiting conditions for operation in the technical specifications covering system operability specify that at least two of three containment leak detection systems must be operable (particulate radioactivity monitoring, sump flow integrating, and air cooler condensate flow rate monitoring) and provides for appropriate limitations in case of failure of one or two systems. In addition, leakage limits are specified as five gallons per minute unidentified leakage, averaged over a 24-hour period with a limit of an average of 25 gallons per minute of total leakage for 24 hours.

The leakage detection systems provide reasonable assurance for detecting small leaks across the reactor coolant pressure boundary as required by General Design Criterion 30 and Regulatory Guide 1.45 and are acceptable.

5.3 Reactor Vessel

5.3.1 Compliance with Code Requirements

We reviewed the materials selection, toughness requirements, and extent of materials testing performed by the applicant to provide assurance th * the ferritic materials used for pressure retaining components of the reactor coolant boundary will have adequate toughness under test, normal operation, and transient conditions. The reactor vessel is designed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, 1968 Edition, including Addenda through Summer 1970 (except N-335) based on the order date of November 1969.

The ferritic pressure boundary material of the reactor pressure vessel was qualified by impact testing in accordance with the American Society of Mechanical Engineers Code, Section III, 1968 Edition, including Addenda through Summer 1970.

We reviewed the method of compliance with 10 CFR, Part 50, Appendix G, proposed by the applicant and find the approach generally acceptable.

However, as with other plants similar to the Zimmer plant, we expect that there are areas where some of the fracture toughness tests and procedures are in noncompliance with certain requirements of 10 CFR, Part 50, Appendix G, "Fracture Toughness Requirements," and where some of the ferritic material surveillance program is in noncompliance with certain requirements of 10 CFR, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

The reason for noncompliance is that the reactor vessel was ordered and fabricated and its testing programs were developed well before the requirements of Appendices G and H to 10 CFR, Part 50, became effective.

We are reviewing information submitted by the applicant to assess if specific exemptions may be granted for items of noncompliance to 10 CFR, Part 50, Appendix G pursuant to 10 CFR, Part 50.12. The areas of noncompliance are similiar to those reviewed and approved by us for exemptions on other applications. However, the applicant has not provided all the information necessary to evaluate compliance with 10 CFR, Part 50, paragraph IV B. We will require that this information be submitted by the applicant to ensure that acceptable safety margins are obtained.

If we grant a similar exemption for the Zimmer Station, our safety evaluation supporting the matter will accompany the granting documents.

5.3.2 Operating Limitations

The reactor will be operated in a manner that will minimize the possibility of rapidly propagating failure. The pressure-temperature limit curves, for all phases of plant operation, were established using the available impact test data and conservative nil-ductility transition reference temperature estimates to perform a fracture toughness calculation by the methods of the American Society of Mechanical Engineers Code, Section III, Appendix G (Summer 1972 Addenda) for all areas of the vessel remote from discontinuities. These calculations were based on a postulated surface flaw equal to one quarter of the material thickness. All vessel shell and head areas remote from discontinuities were considered and the operating curves were developed based on the limiting area. The maximum through-wall temperature difference resulting in continuous heating or cooling at 100° degrees Fahrenheit per hour was considered. The safety factors applied were in accordance with American Society of Mechanical Engineers Code Section III, Appendix G, 10 CFR, Part 50, Appendix G, paragraph IV.A.2.c and General Electric Company Topical Report, NEDO-21778," Transient Pressure Rises Affecting Fracture Toughness Requirements for BWR's."

The applicant has not provided sufficient technical justification and data to demonstrate that the estimate of initial nil-ductility transition reference temperature is acceptable. We will require that this justification and data be submitted for our review. When the applicant submits this information, we will review it to ensure that acceptable estimates of initial nil-ductility transition reference temperature are used to construct the operating limits curves. When the review is complete, our evaluation will be included in a supplement to this report.

We reviewed the operating limit curves in the Final Safety Analysis Report and conclude they are acceptable. The use of Appendix G of the American Society of Mechanical Engineers Code as a guide in establishing safe operating limitations, and using results from the available fracture toughness tests and conservative estimates performed in accordance with the American Society of Mechanical Engineers Code and Commission Regulations, will ensure adequate safety margins during operation, testing, and mainterance. Compliance with these American Society of Mechanical Engineers Code provisions and Commission Regulations constitutes an acceptable basis for satisfying the requirements of General Design Criterion 31.

Reactor Vessel Material Surveillance Program

The toughness properties of the reactor vessel beltline material will be monitored throughout the service life with a surveillance program. The applicant requested that the we evaluate the method of compliance with 10 CFR, Part 50, Appendix H, based on page 19013 of the July 17, 1973 FEDERAL REGISTER.

The material surveillance program is similar to others approved by us for similar reactor vessels such as Hatch 2. The applicant has identified the areas of noncompliance with Appendix H to 10 CFR, Part 50.

Changes in the fracture toughness of material in the reactor vessel beltline caused by exposure to neutron radiation will be assessed properly, and adequate safety margins against the possibility of vessel failure can be provided if the material surveillance requirements of Appendix H, 10 CFR, Part 50 are met. Compliance with this document will ensure that 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. However, as stated in subsection 5.3.1 of this report, we expect that there will be areas where some of the ferritic material surveillance program is in noncompliance with certain requirements of 10 CFR, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

We will review the information when provided by the applicant, to determine whether, pursuant to 10 CFR 50.12, specific exemptions should be granted. The areas of noncompliance are similar to those reviewed and approved by us for examptions on other applications.

If we grant a similar exemption for the Zimmer Station, our safety evaluation supporting the matter will accompany the granting documents.

5.3.3 Reactor Vessel Integrity

We reviewed all factors contributing to the structural integrity of the reactor vessel and conclude there are no special considerations that make it necessary to consider potential vessel failure for the Wm. H. Zimmer Nuclear Power Station, Unit No. 1. The bases for our conclusion are that the design, material, fabrication, inspection, and quality assurance requirements conform to the rules of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, 1968 Edition, including Addenda and applicable Code Cases through Summer 1970.

Although we have identified areas of noncompliance with Appendices G and H of 10 CFR, Part 50, we will require that acceptable safety margins are obtained.

Operating limitations on temperature and pressure will be established for this plant using as a guide Appendix G, "Protection Against Non-Ductile Failure," of the 1972 Summer Addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III.

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

(1) Is designed, analyzed and fabricated to the high standards of quality required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, and pertinent Code Cases.

(2) Is made from materials of controlled and demonstrated high quality.

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

(4) Will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation or during most upsets in operation.

(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.

An acceptable procedure for assuring early detection of possible occurrence of cracks in the feedwater nozzle, control rod drive return line nozzle and vessel blend radius has not been provided by the applicant. Resolution of this issue will be discussed in a supplement to this report as stated in subsections 5.2 1 and 5.2.4 of this report.

5.4 Component and Subsystem Design

5.4.1 Reactor Core Isolation Cooling System

The reactor core isolation cooling system is designed to provide water to the reactor vessel in the event of low water level. The reactor core isolation cooling system is normally aligned to take water from the condensate storage tank and inject it into the reactor vessel through the head spray nozzle to maintain core cooling (the high pressure core spray system also starts on low water level). The reactor core isolation cooling system uses a centrifugal pump driven by a steam turbine. System valves are operated by the direct current power system. In this way the reactor core isolation cooling system has been made independent of alternating current power supplies for greater reliability. The reactor core isolation cooling system can operate for a minimum of eight hours with water from the condensate storage tank; the system can be lined up to inject water from the suppression pool as required.

The reactor core isolation cooling system is housed within the reactor and auxiliary buildings which are protected against the effects of earthquakes, floods, and other phenomena, in accordance with General Design Criterion 2, as discussed in subsection 3.2 of this report.

The reactor core isolation cooling system is utilized for safe shutdown and has proper seismic and quality group classifications for a safety-grade system in accordance with the specifications of Regulatory Guide 1.29, "Seismic Design Classification," as discussed in subsection 3.2 of this report.

The reactor core isolation cooling system is started automatically by reactor vessel low water level signal but may also be operated manually by the operator in the control room.

The high pressure core spray and reactor core isolation cooling systems are located in different corners of the reactor building and are thus protected against common mode failure. They use different energy sources for pump motivation (steam turbine for reactor core isolation cooling system, electric power for high pressure core spray system and different power systems for control power). This diversity conforms to the requirements of Section 5.4.6 of the Standard Review Plan.

The reactor core isolation cooling system design operating parameters are consistent with expected operational modes as noted in the process flow diagram.

The design and installation of the reactor core isolation cooling system is in accordance with applicable codes as discussed in subsection 3.9 of this report. The system is designed to operate under all operating and accident conditions, as noted in subsection 3.11 of this report. In addition, the system is protected against pipe whip inside and outside containment required by General Design Criterion 4, as discussed in subsection 3.6 of this report.

The reactor core isolation cooling system contains a miniflow line which discharges into the suppression pool when the line to the reactor vessel is isolated. In order to protect the reactor core isolation cooling system pump against the effects of water hammer when starting, a jockey pump keeps the system full. The system will be checked at least once every 31 days to assure that the lines are filled. Further, an alarm is sounded if the pump stops.

The reactor core isolation cooling system includes a full flow test line with water return to the condensate storage tank for periodic testing. Technical specifications include a flow test at least every 92 days and a system functional test at least every 18 months with simulated automatic actuation and verification of proper automatic valve position. In addition, the flow rate is measured.

Isolation between the reactor coolant system and the reactor core isolation cooling system is provided by: (a) two check valves and a closed motor-operated valve in the reactor core isolation cooling system discharge line, and (b) one closed motor-operated valve in the steam line to the reactor core isolation cooling steam turbine. We require that isolation valves be leak tested periodically in accordance with the provisions of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The applicant must provide leakage limits for each of the valves in the technical specifications.

The reactor core isolation cooling system and the high pressure core spray system maintain water level and core cooling within the reactor vessel in the event of a reactor coolant system leak. The reactor core isolation cooling system is acceptable based on conformance with applicable regulatory guides, our positions, and General Design Criteria.

5.4.2 Residual Heat Removal System

The residual heat removal system consists of three separate loops with a pump in each loop and a heat exchanger in each of two of the loops. The residual heat removal system operates in four different modes:

(1) Shutdown cooling
(2) Steam condensing
(3) Low pressure cooling injection
(4) Containment cooling.

The residual heat removal system is used to cool the reactor to the cold shutdown mode after the reactor coolant system temperature and pressure have been reduced to permit residual heat removal system operation. The primary system can be cooled to 125 degrees Fahrenheit within 20 hours after reactor scram. This temperature can be maintained with one reactor heat removal system loop during refueling and servicing. The residual heat removal system can be used to cool the vessel head by means of the head spray nozzle.

Steam from the main steam line is piped directly to the reactor heat removal system heat exchangers in the steam condensing mode. A pressure reducing valve is used to reduce the steam pressure to the desired operating pressure. The condensate may be returned to the vessel via the residual heat removal system or discharged to the suppression pool.

The reactor heat removal system operates as low pressure coolant injection system in the event of a loss-of-coolant accident as discussed in subsection 6.3 of this report. Containment cooling has two facets. suppression pool (wetwell) spray and containment (drywell) spray, which can be supplied by either of two loops.

The residual heat removal system has only a single suction line which is vulnerable to a single failure of either of the isolation valves. The applicant has an alternate cooling path using the relief valves and suppression pool cooling in the event of a failure in the residual heat removal system suction line which would preclude residual heat removal system operation. Both paths are operable from emergency power supplies. These alternate cooling provisions satisfy the single failure requirements of General Design Criterion 34.

Flow in the residual heat removal system is diverted to a miniflow line when residual heat removal system discharge valves are closed; the miniflow line is isolated when flow is sufficient to prevent damage to the pump by overheating. A jockey pump keeps the lines full in order to prevent water hammer on startup.

Isolation betwee, the reactor cooling system and the residual heat removal system is provided by: (a) a check valve and a closed motor-operated valve in each discharge line, and (b) three closed motor operated valves in the residual heat removal system pump suction line from the reactor coolant system. We require that isolation valves be leak tested periodically, in accordance with the provisions of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The applicant must provide leakage limits for these valves in the technical specifications.

The residual heat removal system will be tested during operation by taking water from the suppression pool. Valves required to actuate during safety operation will be tested periodically during power operation. Sequencing of low pressure coolant injection operation is tested during reactor shutdown periods. This periodic testing is in conformance with the requirements of General Design Criteria 34 and 61.

The residual heat removal system is designed to operate under both normal and accident conditions (discussed in subsection 3.11 of this report.) It is protected against missiles (discussed in subsection 3.5 of this report), pipe whip (discussed in subsection 3.6 of this report.) The residual heat removal system components are designed to function under normal operating and accident conditions as discussed in subsection 3.11 of this report. In this way, the residual heat removal system complies with the requirements of General Design Criterion 4.

The residual heat removal system is designed to seismic Category I requirements. It is protected against the effects of flooding, tornadoes, hurricanes, and other weather phenomena by the reactor building in which it is housed (as discussed in subsection 3.8 of this report), which conforms with the requirements of Regulatory Guide 1.29, "Seismic Design Classification," and General Design Criterion 2. The containment isolation requirements of General Design Criteria 55, 56, or 57 are discussed in subsection 6.2 of this report. Systems used for cooling the residual heat removal system conform to the requirements of General Design Criteria 44, 45, and 46, as discussed in subsection 9.2 of this report.

Instruments are provided to indicate flow rate through each loop. Temperature elements are placed upstream of each heat exchanger, designed to alarm if the inlet temperature exceeds a preset value. These measurements permit the operator to monitor and control the residual heat removal system in any of its operating modes.

We conclude that the residual heat removal system design meets all pertinent General Design Criteria and regulatory guides and is acceptable based on compliance with the requirements noted above.

5.4.3 Reactor Water Cleanup System

System Description

The reactor water cleanup system 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 containment isolation valve are part of the reactor coolant pressure boundary. The applicant's design objectives for the system were 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 to maintain serviceability during reactor operation.

The reactor water cleanup system flow rate is 100,000 pounds per hour. The reactor water cleanup system consists of two 50 percent capacity pumps, regenerative and nonregenerative heat exchangers, and two 50 percent capacity filter demineralizers. The demineralized water may be sent to the reactor through the shell side of the regenerative heat exchanger, to condensate storage, or to the liquid radwaste system.

Two isolation valves in the reactor water cleanup system close automatically on a signal from the reactor coolant pressure boundary leak detection system to prevent the loss of coolant and the release of radioactive material from the reactor vessel. These valves also operate if the standby liquid control system is activated or if the outlet temperature of the non-regenerative heat exchangers exceed a preset level. The design of the valves is such that they can be operated manually. Reverse flow isolation is provided by a check valve in the reactor water cleanup system or feedwater piping.

Flow will be maintained in the filter/demineralizers in the event of low flow or loss of flow by separate holding pumps provided for each filter/demineralizer unit. Resin loss to the reactor coolant is prevented by strainers on the outlet of each filter/demineralizer unit.

Those components of the reactor water cleanup system which are within the outermost isolation valve boundary are designed to Quality Group A and seismic Category I classification. Those components outboard of the outermost isolation valve boundary are designed to Quality Group C and non-seismic classification. The isolation valves are designed to Quality Group A and seismic Category I classification.

Evaluation Findings

The reactor water cleanup system is used to aid in maintaining the reactor water purity and to reduce the reactor water inventory as required by plant operations. The scope of the our review of the reactor water cleanup system included the system capability to meet the anticipated needs of the plant, the capability of the instrumentation and process controls to ensure operation within the recommended limits defined in Regulatory Guide 1.56, "Maintenance of Water Purity In Boiling Water Reactors," and the seismic design and quality group classifications relative to Regulatory Guides 1.26, "Quality Group Classificatic. . Standards for Water-, Steam- and Radioactive-Waste-Containing Components of Nuclear Power Plants," and 1.29, "Seismic Design Classification." Our review included piping and instrumentation diagrams and process diagrams along with descriptive information concerning the system design and operation.

The basis for acceptance in our review was conformance of the applicant's design and design criteria to the Commission's regulations and to applicable regulatory guides, as referenced above, as well as to our technical positions and industry standards. Based on the foregoing evaluation, we conclude that the proposed reactor water cleanup system is acceptable.

6.0 ENGINEERED SAFETY FEATURES

The purpose of the various engineered safety features in a nuclear power plant is to provide a complete and consistent means of assuring that the public will be protected from excessive exposure to the radioactive materials should a major accident occur in the plant. 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 15.0 of this report. They are designed to seismic Category I standards and they will function even with 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 and control system for each engineered safety feature is designed to the same seismic, redundancy, and quality requirements as the system it serves. Instrumentation and control systems are discussed in section 7.0 of this report.

6.1 Engineered Safety Feature Materials

6.1.1 Engineered Safety Features Metallic Materials

The mechanical properties of materials selected for the engineered safety features satisfy Appendix I of Section III of the American Society of Mechanical Engineers Code, or Parts A, B and C of Section II of the Code. The controls on the use and fabrication of the austenitic stainless steel of the systems satisfy the guidance and criteria of our position on Regulatory Guide 1.31, "Control of the Use of Sensitized Stainless Steel."

The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on austenitic stainless steel components of the engineered safety features are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

Conformance with the Codes and Regulatory Guides mentioned above, and with our positions on stainless steel constitute an acceptable basis for meeting applicable requirements of General Design Criteria 35, 38, and 41.

6.1.2 Organic Materials

The protective coating systems used inside the primary containment were evaluated as to their suitability under design basis accident conditions.

The applicant stated that nearly all the organic materials and coatings within the primary containment meet the requirements of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," and that those which do not meet the requirements will be present in insignificant amounts (total < 10 kilograms). Based on such information, we conclude that under accident conditions including those of a postulated design basis accident, these organic materials and paints will not adversely affect the performance of the engineered safety features equipment.

6.1.3 Post-Accident Chemistry

The Zimmer Station will not use containment sprays in the reactor building or drywell. The controls on the hydrogen ion concentration of the emergency core cooling system water following a loss-of-coolant accident will be adequate to ensure freedom from stress corrosion cracking of the austenitic stainless steel components and welds of the emergency core cooling system throughout the duration of the postulated accident to completion of cleanup.

6.2 Containment Systems

6.2.1 General

The containment functional design refers to the performance capability of the reactor containment structure following a postulated loss-of-coolant accident. For Zimmer, the Mark II type containment is the final fission product barrier in the unlikely event of a loss-of-coolant accident. It also serves as a heat sink for certain operational transients. Figure 6-1 shows the principal features of the Mark II containment concept. This design utilizes the effect of water pressure suppression and consists of the drywell, the pressure suppression chamber, a vent system connecting the drywell and suppression chamber, and a vacuum relief system.

The review of the containment included the temperature and pressure responses of the drywell and wetwell to a spectrum of loss-of-coolant accidents; suppression pool dynamic effects during a loss-of-coolant accident and following the actuation of one or more reactor coolant system pressure relief valves; the capability to withstand the effects of steam bypass from the drywell directly to the suppression pool; and the external pressure capability of the containment. The review considered the applicant's proposed design bases and design criteria for the containment and the analyses and test data in support of the criteria and bases.

Our containment review also included loads resulting from pool dynamic-related phenomena. Following a loss-of-coolant accident an air steam mixture will be forced from the drywell through the vent system into the suppression pool. The air component of the vent flow forms high pressure bubbles in the pool. Motion of the air bubbles results in an upward acceleration of the pool surface which can

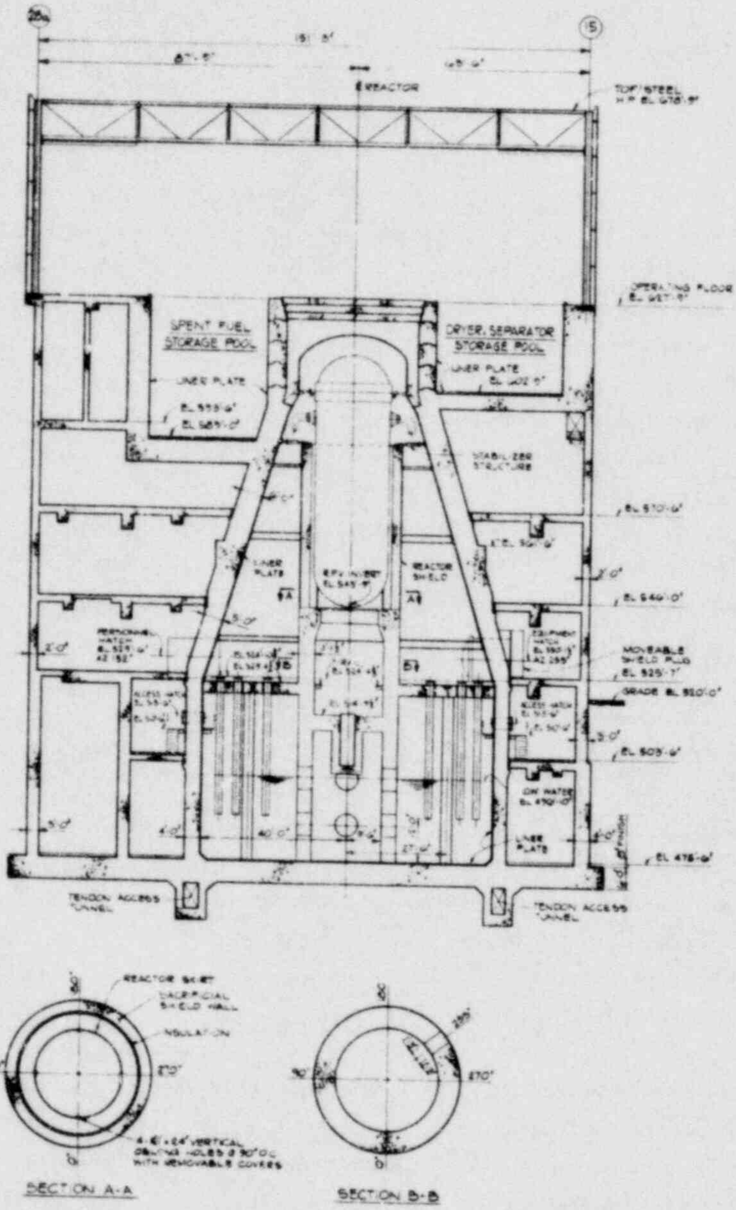

FIGURE 6-1 ZIMMER PRIMARY AND SECONDARY
CONCRETE CONTAINMENT STRUCTURES

impact internal containment structures. Additional containment loads result as the steam portion of the vent flow condenses in the pool. Actuation of relief valves also results in containment loads. Pressure waves are generated within the suppression pool when the relief valves discharge air and steam into the pool water.

The primary containment system is divided into two major subvolumes: a drywell system and the pressure suppression chamber. The drywell and the suppression chamber are connected by an array of vertical vents.

The drywell is a steel-lined prestressed concrete vessel in the shape of a truncated cone, closed by a steel dome. The drywell houses the reactor vessel, the reactor coolant recirculation loops and other branch connections of the reactor primary system. The net free volume of the drywell is 180,000 cubic feet and the design pressure is 45 pounds per square inch gauge.

The pressure suppression chamber is a cylindrical steel-lined prestressed concrete vessel located below the drywell. The suppression chamber consists of an air region and a water region (suppression pool) with net volumes of 95,350 cubic feet and 102,120 cubic feet, respectively; and the design pressure is 45 pounds per square inch gauge. 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 vent system would provide the energy transfer path.

The vent system consists of 88 straight down pipes (downcomers) each 24.0 inches inside diameter. These pipes extend from the drywell floor to the suppression pool and are submerged a minimum of ten feet in the pool water. The purpose of the downcomers is to channel the energy and mass released in the form of uncondensed steam from the primary system into the suppression pool where the steam is condensed.

To satisfy the design basis as a low leakage barrier, the primary containment system is designed for a leakage rate of 0.5 percent of the containment volume per day at a maximum calculated accident pressure of 40.4 pounds per square inch gauge. This leakage rate is within the leakage rate assumed for the accident analysis discussed in subsection 15.3.2 of this report.

An additional structure called the reactor building surrounds the primary containment. Its purpose 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 reactor building design is included in subsection "Secondary Containment," which follows.

Figure 6-2 illustrates the drywell and suppression chamber pressure response as a function of time following a recirculation line break, which has been identified as the design basis loss-of-coolant accident. The pressure response will be discussed as short-term and long-term transients. These transients are presented separately in following subsections of this report.

Figure 6-3 illustrates the drywell and suppression chamber pressure response as a function of time following a steam line break, which is the break yielding the limiting differential pressure across the drywell floor.

The drywell floor serves as the barrier separating the drywell from the suppression chamber. It is a conventional reinforced concrete structure and is supported on a cylindrical base at its center, on a series of concrete columns and from the containment wall and at the periphery of the slab. The drywell floor is rigidly connected to the primary containment wall. Thermal expansion of the floor is accounted for in the containment design. Our evaluation of the floor design is included in subsections "Short Term Pressure Response," and "Drywell Floor Reverse Pressure," which follow.

The drywell is divided into subcompartments by internal structures. Our evaluation of the subcompartment designs is discussed in subsection, "Subcompartment Pressure Analysis," which follows.

Review of Boiling Water Reactor Containment Technology

Pressure suppression designs of containments in the United States that are different from the Mark II containment and used in conjunction with boiling water reactor systems are the Humboldt Bay, the Mark I or "lightbulb-torus," and the Mark III. A comparison of design parameters for the Mark I, II, and III containment types is provided in Table 6-1.

The earliest widespread use of boiling water pressure suppression containment design is the Mark I. In the Mark I design, Figure 6-4, the drywell consists of an inverted lightbulb-shaped vessel, and the suppression chamber is a torus shaped steel vessel located below and encircling the drywell. The vent systems consist of ducts, vent header, and downcomers. The typical design pressure for both drywell and pressure suppression chamber is 56 pounds per square inch gauge, except for Oyster Creek and Nine Mile Point 1 where the pressure suppression chamber design pressure is 35 pounds per square inch gauge.

The Mark III design, Figure 6-4, is the latest boiling water reactor pressure suppression containment design. In this design, the containment (pressure suppression chamber) completely surrounds the drywell. 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. Located in the vertical section

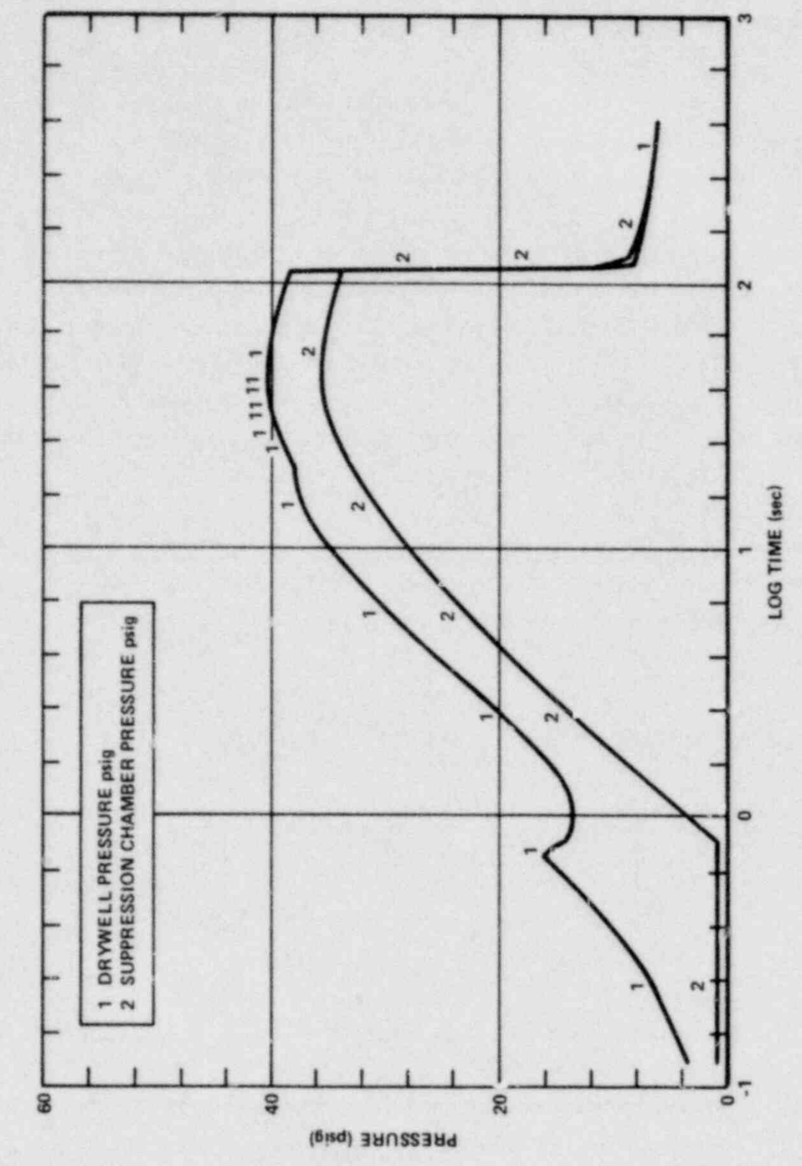

FIGURE 6-2 PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK

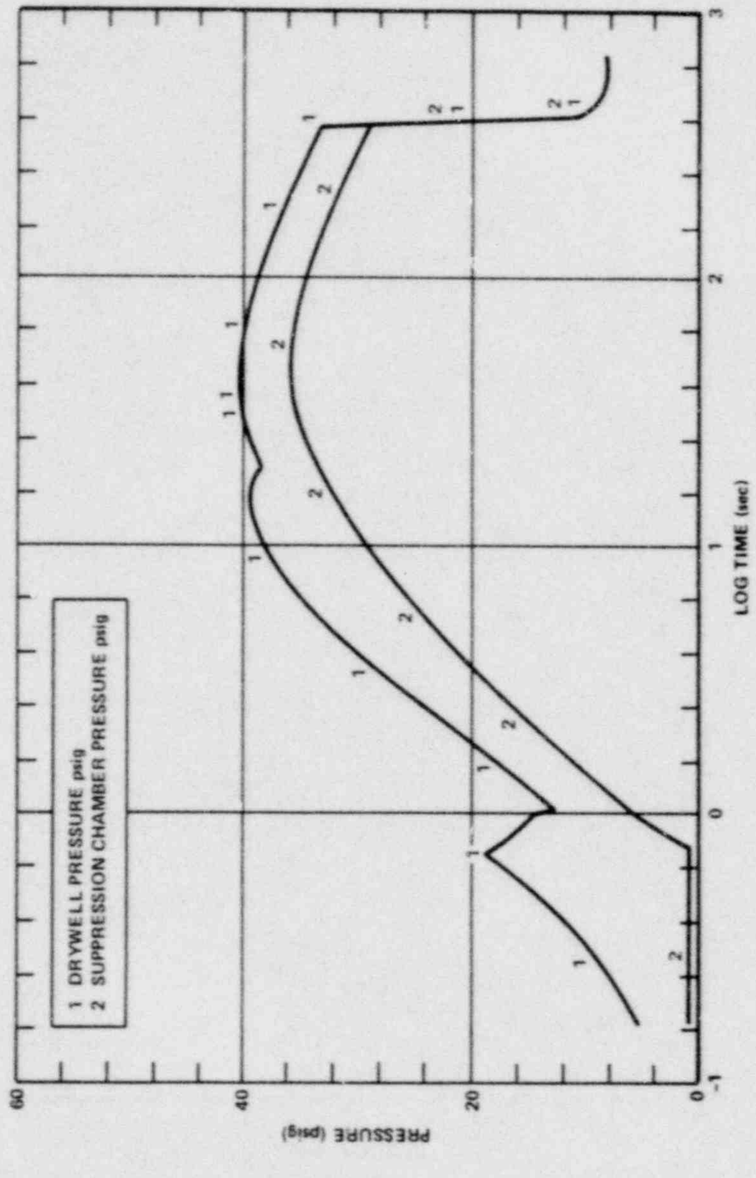

FIGURE 6-3 MAIN STEAMLINE BREAK RESPONSE
PARAMETERS PRESSURE RESPONSE

TABLE 6-1

COMPARISON OF BWR CONTAINMENT DESIGNS

DRYWELL	MARK I	MARK II	MARK III
	(HATCH 2)	(ZIMMER)	(GESSAR 238)
type of construction	steel shell	steel-lined prestressed	reinforced concrete
air volume (cubic feet)	146,266	180,000	274,500
design pressure (pounds per square inch gauge)	56	45	30
leak rate (percent per day)	1.2	0.5	NA
WETWELL			
type of construction	steel shell	steel-lined prestressed concrete	steel shell (containment)
air volume (cubic feet)	109,714	95,350	1.168×10^6
pool volume (cubic feet)	90,550	102,120	163,700
design pressure (pounds per square inch gauge)	56	45	15
leak rate (percent per day)	1.2	0.5	0.3
thermal power (megawatts thermal)	2537	2550	3758
loss-of-coolant accident break area (square feet)	4.378	2.238	3.94
vent area (square feet)	216	276	480
break area/vent area	.0202	.0081	.008

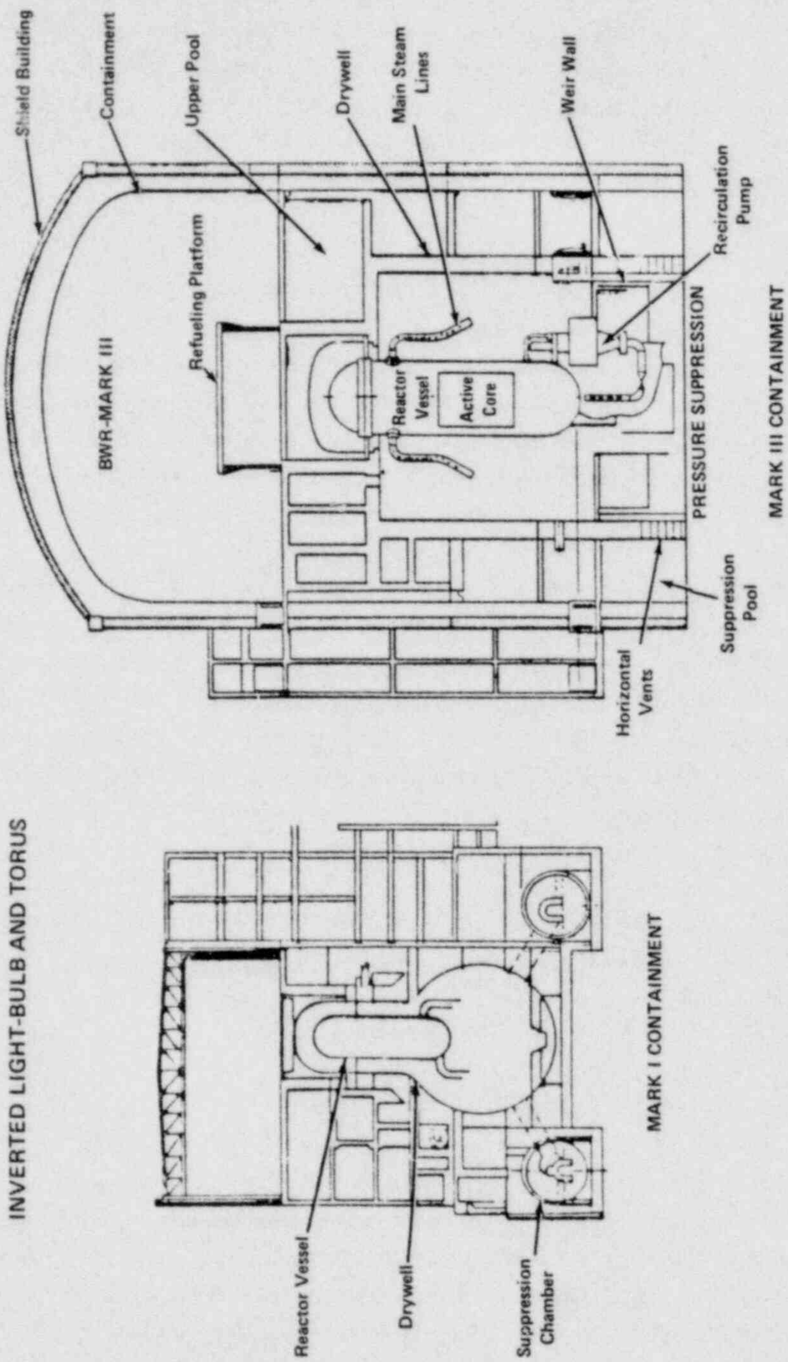

FIGURE 6-4 MARK I AND MARK III CONTAINMENTS

The wetwell and drywell of the Mark I and II designs are connected by a vent system which enters the suppression pool vertically at a constant submergence. The Mark III design uses a horizontal vent system at variable submergence. In both the Mark I and II containments, the peak drywell pressure occurs in the range of approximately ten to fifty seconds following the accident after the vent clearing process and during the vent flow part of the transient. Wetwell peak pressures occur in about the same time frame for the Mark I and II designs, due primarily to the compression of drywell air which is carried over to the wetwell.

The peak drywell pressure in the Mark III design occurs at about one second during the vent clearing process. The peak wetwell pressure for a Mark III design occurs in the long term. The peak pressure is primarily a function of the capacity of the containment heat removal system. It is not determined by the compression of drywell air carried over to the wetwell as in the case of the Mark I and II designs. This is due to the large volume of the wetwell in the Mark III design. The wetwell free volume for a Mark III design is about five times the drywell volume.

Mark II containments also experience a short-term drywell floor pressure differential which can occur either at the time of vent clearing or later during the vent flow transient. Generally, those Mark II plants with a relatively small break-area-to-vent-area ratio have vent clearing controlled peak floor differential pressures. The Zimmer containment falls in this category. In the long term, both the drywell and wetwell reach a secondary peak pressure due to continued decay heat generation; however, this transient is less severe than the short term and is therefore not controlling for establishing design pressures.

The analytical models, assumptions, and methods used by General Electric Company to evaluate the containment response during the reactor blowdown phase of a loss-of-coolant accident are described in NEDO-10320, "The General Electric Pressure Suppression Containment Analytical Model."

Short-Term Pressure Response

The limiting drywell, suppression chamber and drywell floor pressures occur during the blowdown phase of a loss-of-coolant accident transient. The duration of the blowdown period is about 45 seconds following a postulated break in the recirculation line. In the long-term, about three hours after a loss-of-coolant accident, both the drywell and wetwell reach a secondary peak pressure due to continued decay heat generation; however, this transient is less severe than the short-term and is therefore not controlling for establishing design pressures.

The applicant performed analyses of varied postulated primary system breaks including recirculation line, main steam line and a spectrum of liquid line and steam line breaks. Results of the analysis indicates that the recirculation line break yields the limiting drywell and suppression chamber pressure. The applicant, therefore, concludes that the recirculation line break is the design basis accident for the drywell and suppression chamber. We performed similar confirming analyses and agree with the applicant's conclusion.

Following the postulated double-ended rupture of a 20-inch recirculation pump suction line break, the mass and energy released from the primary system pressurizes the drywell. As the drywell pressure increases, water initially in the downcomers is accelerated downward. This downward motion continues until the entire water column is expelled. At this point, the air-steam water mixture begins to flow into the suppression pool. The steam is condensed in the pool and air is released into the suppression chamber air region.

The above process is called the vent clearing transient, which occurs less than one second following the postulated accident. As shown in Figure 6-2 the maximum pressure differential between the drywell and pressure suppression chamber occurs at the time of vent clearing.

The drywell pressure continually increases until it reaches the calculated peak pressure of 40.4 pounds per square inch gauge at 42 seconds after the accident. The drywell pressure then decreases slightly as the rate of energy dumped to the suppression pool via downcomers exceeds the rate of energy released into the drywell from the primary system.

Following vent clearing the pressure in the pressure suppression chamber increases at about the same rate as the drywell pressure due to the transfer of steam and noncondensibles from the drywell. The suppression chamber reaches a calculated peak pressure of 36 pounds per square inch gauge at 42 seconds after the accident.

At about 110 seconds the emergency core cooling system injection water floods the reactor vessel to the level of the break and the emergency core cooling system flow cascades into the drywell. This results in condensation of the steam in the drywell and a rapid reduction in the drywell pressure. As soon as the drywell pressure drops below the suppression chamber pressure, the drywell vacuum breakers will open and noncondensible gases from the suppression pool air volume will flow back into the drywell.

The limiting break for the drywell floor is the postulated double-ended, rupture of a main steam line. The applicant 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 swell in the reactor vessel.

The time at which the liquid level in the reactor vessel swells to the evaluation of the steam line nozzles following the break, determines the time at which the model changes from steam to two-phase blowdown assumptions. The peak drywell floor 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 a steam only blowdown and also introduces liquid water into the vent flow.

The applicant's calculated peak pressures for the design basis accident are 40.4 and 36 pounds per square inch gauge for the drywell and suppression chamber respectively. The design pressure of the drywell and suppression chamber is 45 pounds per square inch gauge, which provides a 10 percent design margin for the drywell and 20 percent for the suppression chamber. The applicant calculated a peak pressure differential across the drywell floor of 17.4 pounds per square inch at 0.75 seconds, the time of vent clearing. The design pressure is 25 pounds per square inch. The resulting margin for the drywell floor is more than 30 percent. We performed an analysis of the containment pressure response using the CONTEMPT-LT computer code. Our calculations of the peak pressures confirm those calculated by the applicant. Based on the design margin and our own independent verification of the applicant's assumptions and the analytical results, we conclude that the design pressures for the drywell, suppression chamber and the drywell floor are acceptable.

Long-Term Pressure Response

Following the short-term blowdown phase of the accident, the suppression pool temperature and suppression chamber pressure will continuously increase due to the input of decay heat and sensible heat into the suppression chamber. Referring to Figure 6-3 between about ten and 100 seconds after the accident, the drywell pressure has stabilized to approximately five pounds per square inch above the suppression chamber pressure. This differential pressure corresponds to the submergence of the downcomers. At a later time, the drywell and suppression chamber pressures will equalize due to the return of air from the suppression chamber.

During this time period, the emergency core coolig system pumps, taking suction from the suppression pool, have reflooded the reactor pressure vessel. Subsequently, emergency core cooling system water will flow out of the break and flow into the drywell. This relatively cold emergency core cooling system water will condense the steam in the drywell and bring the drywell pressure down rapidly as show in Figure 6-3 at about 110 seconds after the accident. At ten minutes following the accident, the containment cooling mode for the residual heat removal system is activated and suppression pool water is circulated through the residual heat removal system heat exchangers, establishing an energy transfer path to the service water system and ultimate heat sink.

In the long-term analysis, the applicant accounted for potential post-accident energy sources. These include decay heat, sensible heat and metal-water reaction energy. The applicant's long-term model also assumes that the suppression chamber atmosphere is saturated and equal to the suppression pool temperatures at any time. Therefore, the suppression chamber pressure is equal to the sum of the partial pressure of air and the saturation pressure of water corresponding to the pool temperature.

Based on the above assumption, the applicant calculated a peak suppression pool temperature of 206 degrees Fahrenheit for the most limiting residual heat removal system cooling mode; i.e., only one residual heat removal system cooling loop with one residual heat removal system pump available. The calculated long term, secondary peak suppression chamber pressure is about 23 pounds per square inch gauge. The suppression chamber is designed for 45 pounds per square inch gauge and 275 degrees Fahrenheit. On the basis of our review of the applicant's analysis, we conclude that the suppression chamber design pressure and temperature are acceptable.

Drywell External Pressure and Floor Reverse Pressure

Events which may result in drywell external pressure and drywell floor reverse pressures are:

(1) The containment spray is initiated following a loss-of-coolant accident.

(2) The containment spray is initiated following a small steam line break.

The most severe case for containment external pressure is case (1). Conservative calculations made by the applicant yield a maximum external pressure differential of -1.4 pounds per square inch. The containment external design differential pressure is -2 pounds per square inch.

We also investigated the potential for inadvertent operator actuation of the containment sprays during conditions of high containment temperature and humidity. The applicant provided information to show that this is an extremely unlikely event based on the hardware provided in the residual heat removal system and the manner in which it is operated.

This conclusion is based on the following:

(1) No operating procedures exist which specify drywell spray initiation.

(2) The drywell spray is isolated by two normally closed valves in series. Interlocks prevent opening both valves unless a high drywell pressure exists.

(3) The drywell spray valves are operated by key-operated switches in the main control room. These keys are controlled for distribution.

(4) The drywell spray mode is part of the residual heat removal system which is normally not operating.

Inadvertent initiation of the drywell sprays under normal operating conditions can only occur by assuming multiple failures or errors. It is therefore precluded.

We find the containment external design pressure acceptable based on:

(1) The unlikely potential for inadvertent initiation of drywell sprays under normal operating conditions; and

(2) A maximum calculated external pressure of -1.4 pounds per square inch differential.

For the drywell floor upward pressure, the applicant analyzed the cases described above. The applicant assumed that three out of a total of four sets of drywell floor vacuum breakers were functioning. The vacuum breakers are provided to limit the drywell vacuum conditions. Each set of vacuum breakers consists of two valves in series to reduce potential bypass leakage. For the limiting case of the conditions discussed above the applicant calculates a peak reverse pressure differential of 2.74 pounds per square inch. The design reverse pressure differential of the drywell floor is nine pounds per square inch. Our analysis using the CONTEMPT LT computer code as identified in Standard Review Plan 6.2.1.1C, "Pressure Suppression Type BWR Containments," confirms the applicant's results and we conclude that the floor design pressure is acceptable.

Subcompartment Pressure Analysis

Within the primary containment, internal structures form subcompartments or restricted volumes which are subject to differential pressures following postulated pipe ruptures. In the drywell there are two restricted volumes, the annulus formed by the reactor vessel and the sacrificial shield, and the drywell head region which is a cavity between the drywell head and the refueling seal. Since the suppression chamber is virtually an open space, no restricted volume exists.

The applicant performed a pressure response analysis for the drywell head region postulating a break in the residual heat removal system head spray line.

Based on the frictionless Moody blowdown flow at a reactor pressure of 1025 pounds per square inch gauge, the applicant calculated the pressure differential using the computer code WARLOC. The maximum calculated pressure difference is 3.2 pounds per square inch. This represents only a fraction of the design pressure (18 pounds per square inch) of the bulkhead plate. Our analysis confirms the applicant's result and we conclude that the design pressure for the head region is acceptable.

The applicant submitted the results of calculations of pressure differentials across the annulus formed by the sacrificial shield wall and the reactor pressure vessel.

Two situations were considered in the analysis of the annulus for a postulated line break. These are: (1) design pressure for the shield wall, and (2) forces on the reactor pressure vessel affecting the design of the support skirt.

Separate analyses were conducted by the applicant for two different breaks to determine the limiting case. The postulated breaks analyzed are the recirculation suction line and the feedwater line. For each of these cases a separate nodalization scheme was employed to maximize the loads obtained locally in the area of the break. In addition nodalization sensitivity studies were performed by the applicant to arrive at an acceptable nodalization scheme.

The applicant calculated a peak differential pressure within the sacrificial shield annulus of 113 pounds per square inch for the recirculation suction line break. The RELAP-4 computer code was used to perform the analyses. The actual pressure transients calculated by the applicant were utilized in dynamic structural analyses to determine the sacrificial wall design. We performed confirmatory analyses using the COMPARE computer code and the applicant's mass and energy release data. The computer codes cited above are referenced in the Standard Review Plan 6.2.1.2, "Subcompartments Analysis," as acceptable codes. Our calculations confirm the conservatism of the applicant's calculations. Based on our review of the applicant's analysis and our confirmatory calculations, we find the design of the shield wall acceptable.

The applicant provided us with additional information in Amendment 76 to the Final Safety Analysis Report to allow completion of our review of the forces on the reactor pressure vessel affecting the design of the support skirt. We find the applicant's methods of analysis acceptable.

Steam Bypass of the Suppression Pool

During a postulated primary system line break inside the drywell, possible bypass leakage paths between the drywell and suppression chamber air space could result in high containment pressures. The control of such bypass paths is important to ensure that the design pressure of the containment is not exceeded. There are several potential sources of steam bypass of the suppression pool associated with the Mark II containment used in the Zimmer Nuclear Power Station.

Since the drywell floor is of reinforced concrete construction, the potential exists for cracking of the floor under accident loading conditions. This can allow direct leakage of blowdown steam to the wetwell volume. In addition, the potential exists for direct leakage through the drywell floor vacuum breakers or around penetrations in the drywell floor. The containment hydrogen recombiner also introduces a potential path for steam bypass.

The applicant provided information regarding the operation of the recombiner to show that steam bypass through the recombiner is an extremely unlikely event based on the hardware provided and the manner in which the recombiner is operated. This conclusion is based on the following:

(1) The startup procedures consist of six individual sequential steps. A two-mode or greater failure must be postulated in this procedure to establish gas bypass.

(2) The recombiner system is provided with a condenser that is designed to condense water vapor from the recombiner inlet gas. The recombiner system includes a low-flow alarm on the condenser cooling water.

On the basis of our review of this information, we find that the applicant's basis for neglecting any steam bypass through the recombiner is acceptable.

The applicant analyzed the containment to determine the capability for pool bypass leakage, for a spectrum of break sizes. The limiting case for bypass capability corresponds to a small break. The applicant calculated an allowable bypass area for Zimmer of $A/\sqrt{K}$ = 0.04 square feet for a postulated small break. The allowable bypass area is considered to be that leakage area between the drywell and suppression chamber which could result in suppression chamber pressurization to design pressure following a small break.

The analysis consists of a transient analysis that takes credit for passive heat sinks in the wetwell chamber, actuation of the wetwell sprays following a ten minute delay to satisfy the system's emergency core cooling function, and a six hour plant shutdown time.

Based on the results of our study on Mark II steam bypass capability, we conclude that the Zimmer plant should meet the following design criteria.

(1) The containment should have a minimum steam bypass capability for small breaks of the order of 0.05 square feet ($A/\sqrt{K}$). (.04 square feet has been accepted by us based on the capability to control the overpressure of the containment to within design limits from this steam bypass, using wetwell sprays actuated in sufficient time (~15 minutes) following a postulated small break loss-of-coolant accident.)

(2) The wetwell spray should be automatically actuated ten minutes following a loss-of-coolant accident.

(3) A single preoperational high pressure leakage test should be performed prior to operation and post operational low pressure leakage tests should be performed at each refueling outage.

(4) The acceptance criterion for the leakage tests is measured leakage less than ten percent of that corresponding to $A/\sqrt{k}$ = .04 square feet steam bypass capability.

(5) Redundant position indicators should be placed on all vacuum breakers with indication and redundant alarms in the control room.

(6) Vacuum breakers should be operability tested at monthly intervals.

As a part of the pool bypass position, we requested that the applicant provide information to establish the availability of the wetwell spray system 10 minutes following a loss-of-coolant accident. The consequences of actuation of the wetwell spray system on the emergency core cooling system function also were to be evaluated. The applicant has not provided us with all of the information required. We will report the resolution of this matter in a supplement to this report (see subsection 6.3.4 of this report).

The applicant committed to abide by all of our requirements and we find the pool bypass capability of the containment acceptable.

Pool Dynamics

Mark II Pool Dynamics History

In the course of the General Electric Company testing program for the Mark II pressure suppression containment program, new containment loads associated with a postulated loss-of-coolant accident were identified which were not explicitly included in the original design of Mark II containment. These loads result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool during a postulated loss-of-coolant accident event. Other previously unaccounted for pool dynamic loads result from the actuation of safety relief valves in the Mark II containment.

In view of the potential significance of these loads, it was determined that a reassessment of the Mark II containment system design would be required. A letter was sent by us to each domestic Mark II owner on April 11, 1975 notifying them of the need for this reassessment.

As a result of our letter, an "ad hoc" Mark II Owners Group was formed which is an organization of all domestic utilities owning Mark II facilities. They have engaged General Electric Company as their program manager for resolution of the Mark II containment pool dynamic concerns. A program was developed to establish generic pool dynamic loads, load combinations and design criteria. The program consists of a number of experimental and analytical tasks. These tasks form the basis for establishing the generic Mark II pool dynamic loads.

Description of Phenomena

Loss-of-Coolant Accident Pool Dynamics: Figure 6-5 shows the sequence of events following a postulated loss-of-coolant accident and the potential loading conditions associated with those events. Following a postulated loss-of-coolant accident the drywell pressure increases due to blowdown of the reactor system. Pressurization of the drywell causes the water initially in the vent system to be accelerated out through the vents. During this water expulsion process the resulting water jets cause impingement loads on local containment structures.

Following vent clearing, an air/steam bubble forms at the vent exit which causes a hydrostatic pressure increase in the pool water resulting in a loading condition on the pool boundaries. The steam condenses in the pool. However, the continued addition and expansion of the drywell air causes the pool volume to swell, resulting in a rise of the pool surface. Upward motion of this slug of water creates a drag load on structures submerged in the pool and impact loads on unsubmerged structures located just above the initial pool surface.

After the pool has risen approximately 1.5 times the initial submergence of the main vents, the rising slug of water breaks apart. Subsequent pool swell involves a two-phase air water froth which produces further structural-impingement loads. A gravity induced fallback of the pool returns the pool surface to the post loss-of-coolant accident elevation.

At about the time of slug breakup, the drywell floor can be subjected to an upward load due to an imbalance in pressure between the compressed air in the wetwell free air space and the air purged drywell volume.

Following the pool swell transient, there will be a period of high steam flow through the main vent system. At these high steam flow conditions, the water/steam condensation interface oscillates due to bubble growth and collapse. These condensation oscillations result in an oscillatory load on the pool boundary. At low vent flow rates, the water/steam condensation interface can oscillate back and forth in the vents causing "chugging". The chugging action results in loads on both the downcomer vents and the containment boundaries.

Relief Valve Dynamics: Actuation of safety/relief valves produces transient loadings on components and structures in the suppression chamber region. Prior to actuation, the discharge piping of an safety/relief valves line contains atmospheric air and a column of water corresponding to the line submergence. Following safety/relief valves actuation, pressure builds up inside the piping as

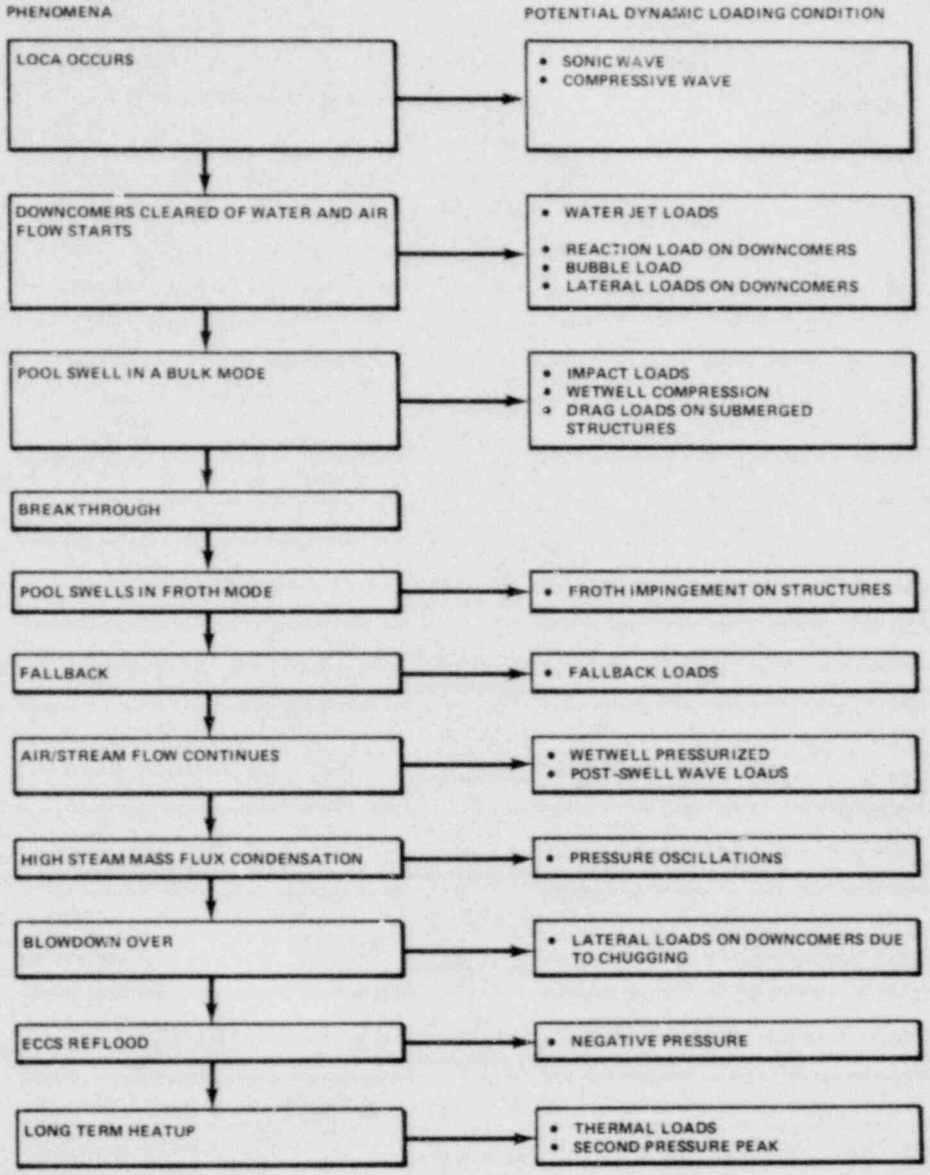

FIGURE 6-5 SEQUENCE OF EVENTS AND POTENTIAL LOADING CONDITIONS FOLLOWING A POSTULATED LOCA

steam compresses the air in the line. The resulting high pressure air bubble that enters the pool oscillates in the pool as its goes through cycles of overexpansion and recompression. The bubble oscillations resulting from safety/relief valves actuation and discharge cause oscillating pressures throughout the pool, resulting in dynamic loads on pool boundaries and submerged structures.

Severe steam condensation vibration phenomena can occur when high-pressure, high temperature steam is continuously discharged at high-mass velocity into the pool, if the pool is at elevated temperatures. These steam quenching vibrations also result in loads on pool boundaries and submerged structures.

The characteristics of the safety/relief valves load varies depending on the discharge device (ramshead or quencher) located at the exit of the safety/relief valves line. The applicant initially was using a ramshead device for the Zimmer Nuclear Power Station but made the decision to change to a quencher device in order to help mitigate pool temperature effects and dynamic forces.

Mark II Owners Group - Generic Program

The Mark II Owners Group developed a generic program to establish pool dynamic loads, load combinations and design criteria. The program includes a number of analytical and experimental programs to provide the data base to support the proposed loads and load prediction models. Based on the results of some of the early tasks in the program, the Mark II Dynamic Forcing Function Information Report was prepared and submitted to us in November 1975. This report specifies the generic Mark II pool dynamic loads, load combinations and design criteria to be utilized by each domestic power plant with a Mark II containment.

The Mark II Owners Group program consists of a "short-term program" or lead plant effort and a "long-term program." The purpose of the short-term program is to demonstrate that a sufficient technical understanding of the pool dynamics phenomena and principles of interest exists to allow the utilization of the loads and methods prescribed in the Dynamic Forcing Function Information Report for the licensing of the lead Mark II plants. As a result, in some cases, a bounding interpretation of test data was utilized to assure conservatism in the Dynamic Forcing Function Information Report loads. The primary purpose of the long-term program is to confirm the loads utilized in the short-term program. In addition, the long-term program tests and analyses may be used to justify the application of less conservative loads in the future. A description of the loss-of-coolant accident and safety/relieve valve-related tasks and documentation comprising the Mark II Owners Groups program is provided in NEDO 21297, "Mark II Containment Supporting Program Report."

Data from a number of experimental programs have been used in conjunction with analytical models to support the short-term program loads specified in the Dynamic

Forcing Function Information Report. The major Mark II-related experimental programs include the temporary tall test tank full scale, Electric Power Research Institute 1/13 scale, pressure suppression test facility 1/3 scale, and the Monticello in-plant tests.

The temporary tall test tank facility consists of a single cell representative of a typical Mark II containment. The test facility utilized a single full-size vertical downcomer in a tank. A total of 46 steam blowdown tests were conducted by the General Electric Company in this full scale facility during a test program consisting of three phases. These tests provided the primary data source for the Dynamic Forcing Function Information Report loss-of-coolant accident pool dynamic loads including those related to pool swell, wetwell pressurization, vent flow, pool thermal response, condensation and chugging. The test matrix included a range of vent submergence, break size, vent size, blowdown fluid, vent bracing and initial conditions to reflect Mark II plant-to-plant differences.

Test results from the General Electric Company pressure suppression test facility supplied the data base for pool impact loads on representative small containment structures including pipes, I-beams, and grating. Impact pool velocity correlations were developed from these data which are used in combination with calculated pool swell velocities and temporary tall test tank data to establish the Mark II impact loads. The pressure suppression test facility used for these tests represents a segment of a Mark III containment with a one-third scale vent test section (vent area scaled) with a one-third scale suppression pool (pool area scaled). The impact load data were obtained from the pressure suppression test facility test series 5805.

The Electric Power Research Institute Mark II facility consists of a 1/13 scale model of a typical Mark II containment system. The facility containing 21 vents represents a 90 degree sector of the suppression chamber including the pedestal region. The test consisted of air charged tests in contrast to the temporary tall test tank steam blowdown tests. About 90 tests were performed by Stanford Research Institute for the Electric Power Research Institute to provide data related to Mark II pool swell phenomena. Specifically, data from these tests were used to verify the adequacy of the temporary tall test tank unit cell approach to study pool swell phenomena, validate the temporary tall test tank air/steam tests, and validate the Dynamic Forcing Function Information Report pool swell analytical model.

The Monticello plant of Northern States Power Company was made available for in-plant safety/relief valve tests. The containment for this plant is a Mark I or lightbulb torus containment. The steel torus comprising the suppression chamber was provided with instrumentation to permit determination of safety/relief valve loads. The primary objective of this test program, as it relates to the Mark II pool dynamic program, was to measure pressures and temperatures in the torus and safety/relief valve piping associated with relief valve actuations. The data base established by these tests is to be used for verifying and improving analytical models that are used to predict the loads produced by safety/relief valve discharges through a ramshead device. A total of 37 tests involving single, multivalve and consecutive valve actuation were conducted at a pool temperature range of 75 degrees Fahrenheit to 95 degrees Fahrenheit.

In addition to the above test programs, the Mark II Owners Group has provided information relating to tests conducted outside the scope of the Mark II program to support some of the loads specified in the Dynamic Forcing Function Report. This includes data for steam blowdown tests from the Marviken test facility and data resulting from tests in General Electric Company foreign licensee single and multivent, large scale facilities. Data from these tests were used in the Mark II program to support the conservatism of single vent tests for vent lateral loads and pool boundary chugging loads.

All of the above programs associated with the short-term program have been completed. In addition, the interpretation of test results, verification of analytical models and test documentation is also complete for those tests associated with the Mark II short-term program.

Several of the lead Mark II plants (Zimmer and Shoreham) have advised us that they will utilize a T-Quencher device designed by the Kraftwerk Union of Germany. Documentation of the design and supporting test data is expected to be filed with us in early 1979. In addition the applicant committed to perform in-plant tests to measure safety/relief valve loads that are scheduled for completion prior to operation of the Zimmer plant.

Additional confirmatory tests and analytical programs will be conducted as a part of the Mark II Owners Group long-term program, as discussed in NEDO-21297. Several of the more important long-term program test programs are the safety/relief valve quencher in-plant CAORSO tests and multivent subscale tests. The CAORSO tests will be conducted in the CAORSO plant, a boiling water reactor plant in Italy. These tests were originally included as part of the long-term program since this information was not needed for the lead Mark II plants using ramshead devices. The data from the CAORSO tests will provide the data base for the verification of safety/relief valve loads associated with a four arm quencher device.

The multivent subscale tests will consist of multivent steam tests conducted at several scales and with several different vent configurations to provide confirmatory data related to multivent chugging effects. In addition, the results of these tests may be used to establish the margin inherent in the use of the temporary tall test tank single vent steam chugging loads.

The Mark II long-term program experimental and analytical programs will be conducted over a period of approximately two years. Documentation for the last task is scheduled for the last quarter of 1979.

Zimmer Design Assessment Report

The applicant submitted a Design Assessment Report of the pool dynamic loads for the Zimmer Mark II containment. This report provides a description of the specific application of the generic Mark II pool dynamic loads and methods for the Zimmer plant. The plant unique loads were used in assessing the capability of the Zimmer containment and components to pool dynamic phenomena.

We are conducting an evaluation of the Zimmer design assessment report in parallel with our review of the generic Mark II owner's pool dynamic load program. We requested additional information dealing with this report. As a result of our requests the applicant provided a structural capability study and a closure report. The structural study investigated the capability of the Zimmer containment structure to accommodate safety/relief valve and chugging loads higher than those specified in the Dynamic Forcing Function Information Report. We are currently reviewing this information and the applicant's response to our other Zimmer Design Assessment Report requests.

The closure report was recently submitted by the applicant to reflect changes in pool dynamic loads specifications that have arisen since the Design Assessment Report was issued. We are currently reviewing the closure report. We will review this information and report our evaluation in a supplement to this report.

Status of Our Evaluation

We have been actively reviewing the generic Mark II pool dynamic load program since November 1975, when it received some of the initial documentation for tests and analyses associated with the short-term program. This information provided an early indication of the magnitude of the Mark II pool dynamic loads.

In the course of our review of this early documentation, we issued several rounds of requests for information which resulted in an expansion of the generic Mark II Owners program. A current list of the program tasks included in the Mark II Owners short-term program and long-term program is provided in NEDO-21297. Several tasks that were added to the program as a result of concerns raised by us are the Electric Power Research Institute 1/13 scale tests (A.4), multivent subscale testing and analysis (A.11) and justification of temporary tall test tank bounding loads. These tests will provide data to establish: three-dimensional pool swell effects, pool swell phenomena associated with air vent flow and multivent steam condensation and chugging loads.

In addition, information should be provided to justify the bulk pool temperature limit specified for safety/relief valve operation. As previously discussed severe vibrations may occur if the suppression pool is operating at an elevated temperature. To avoid this steam quenching vibration, a pool temperature limit is specified for safety relief valves operating in response to a normal plant transient. We requested that the applicant provide us with additional information to justify a temperature limit and to analyze the suppression pool temperature response to normal plant transients. We will report the resolution of this issue in a supplement to this report.

We are currently conducting our review of the Mark II pool dynamic loads under two of our generic technical activities (Task A-8, "Mark II Containment Pool Dynamic Loads"; and Task A-39, "Determination of Safety Relief Valves (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containments"). These two activities are (1) resolution of the Mark II containment pool dynamic loads (i.e., loss-of-coolant accident-induced), and (2) resolution of the safety/relief valve pool dynamic loads and temperature limits for boiling water reactor containment. Our review schedule for the Mark II short-term program called for development of our acceptance criteria in September 1978 and a short-term safety evaluation report in October 1978. Both of these tasks have been completed and the acceptance criteria was sent to the lead plant Mark II owners (including Zimmer) by letter dated September 14, 1978. The applicant has committed to accept most of our acceptance criteria with a limited number of exceptions. The seven generic exceptions taken by the lead plant applicants are described in a November 24, 1978 letter "Mark II Generic Acceptance Criteria for Lead Plants," from Roger S. Boyd to each of the Mark II lead plant applicants. We will report our evaluation of alternate criteria proposed by the lead plant owners in a supplement to this report.

We believe that the short-term test with the additional task described above is adequate to establish the pool dynamics loads for the lead Mark II plants. We anticipate completion of our review of the Zimmer pool dynamic loads, as described in the Design Assessment Report and the closure report, following completion of our review of the generic short-term program Mark II pool dynamic program. We will provide our Zimmer evaluation for pool dynamic loads based on the above acceptance criteria in a supplement to this report.

The long-term program serves as a confirmatory program for the loads utilized in the lead Mark II plants. This program is scheduled for completion the last quarter of 1979. Our review of this program is scheduled for completion in mid-1980.

Secondary Containment

The secondary containment system includes the structures and systems to be used to control and treat radioactive leakage from the primary containment in the event of a loss-of-coolant accident. For the Zimmer plant the secondary containment structures consist of the reactor building, a small reactor building recirculation fan room, the equipment access structure and a portion of the main steam tunnel.

The reactor building is a structure surrounding the primary containment with a free volume of 2.65×10^6 cubic feet. The reactor building ventilation system controls the pressure in the secondary containment during normal operation to -0.25 inches of water gauge. Following a postulated loss-of-coolant accident, the standby gas treatment system will maintain the secondary containment pressure at -0.25 inches of water gauge.

The standby gas treatment system is designed to seismic Category I criteria and is located within seismic Category I structures. The standby gas treatment system consists of redundant exhaust fans and filtration trains each consisting of a demister, heating coil, pre-filter, high-efficiency particulate air filters and carbon adsorbers.

Following a postulated loss-of-coolant accident, the pressure in the secondary containment could increase due to inleakage, air expansion due to equipment heat and the starting time required for the standby gas treatment system. The applicant performed an analysis of the secondary containment pressure transient which considers the above phenomena. The results of these analyses show that approximately three minutes after the loss-of-coolant accident, the secondary containment pressure is reduced back to -0.25 inches of water gauge. During the transient the calculated pressure does not rise above -0.16 inches of water gauge. Operation of only one of the two redundant standby gas treatment system trains was assumed for the analysis. In calculating the offsite radiological consequences of this pressure transient the applicant assumed that bypass of the secondary containment occurs for five minutes after a loss-of-coolant accident. The applicant did calculations to determine the time needed to achieve a -0.25 inch of water gauge in the secondary building using the standby gas treatment system. These calculations indicate a draw down time of less than five minutes. We conclude from our review that the analysis is reasonable and acceptable.

To confirm this analysis, the applicant committed to leakage testing of the secondary containment volumes to verify the inleakage assumption (reactor building inleakage of 1400 cubic feet per minute at -0.25 inches of water gauge) and the five minute drawdown time to reestablish a -0.25 inches of water gauge pressure.

Although the primary containment is 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 associated with 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 controlled regions. The criteria by which potential bypass leakage paths are determined has been set forth in Branch Technical Position, Containment Systems Branch 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants."

We reviewed the design of the secondary containment systems for the Zimmer Nuclear Power Station. Our review included the applicant's design bases, analysis of the functional capability of the secondary containment system and an evaluation of the systems against the criteria specified in Containment Systems Branch 6-3. Based on this review we find the design of the secondary containment systems acceptable.

Summary and Conclusions of Containment Functional Design

The applicant calculated the short and long-term drywell and suppression chamber pressures and temperatures and the drywell floor differential pressure as previously described. Based on our review of the applicant's analytical methods and our confirmatory analyses, we conclude that the drywell and suppression chamber design pressures and temperatures and the drywell floor design pressure are acceptable. We also conclude that the drywell floor design reverse pressure is acceptable.

We have not completed our evaluation of the tests and analytical programs comprising the generic Mark II pool dynamic load program. This program forms the basis for the pool dynamic, loss-of-coolant accident and safety/relief valve loads utilized in the Zimmer Design Assessment Report. Nor have we completed our evaluation of the Zimmer plant unique application of the generic pool dynamic load as described by the Zimmer Design Assessment Report and closure report.

6.2.2 Containment Heat Removal System

The containment residual heat removal system includes the piping, valves and mechanical components which will be used to remove energy from the containment to limit temperature and pressure in the drywell and suppression chamber following a postulated loss-of-coolant accident.

The residual heat removal system consists of two complete loops including two heat exchangers and three main system pumps. Each loop is designed such that a failure in one loop cannot cause a failure of another. In addition each of the loops and associated equipment is located in a separate protected area of the reactor building to minimize the potential for single failures including loss of onsite or offsite power causing the loss of function of the entire system. The system equipment piping and support structures are designed to seismic Category I criteria. Provisions have been made in the residual heat removal system to permit inservice inspection of system components and functional testing of active components.

Operating in the containment cooling mode, the residual heat removal pumps take suction from the suppression pool. Flow is then directed through the residual heat removal heat exchangers to the suppression pool, the reactor vessel, or the containment spray headers. The location of system and return lines in the suppression pool facilitates 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. The applicant provided analyses of the long-term post-accident containment pressure and temperature response assuming various combinations of containment cooling availability. Our evaluation of this analysis is discussed in the subsection, "Long-Term Pressure Response," above.

The applicant analyzed the net positive suction head that is available at the residual heat pump inlets assuming the containment will be at atmospheric pressure and the pool at saturation temperature. In addition, the applicant designed the suction piping from the suppression pool so that if any one suction strainer is 50 percent plugged, the maximum required net positive suction head to the residual heat removal system pumps during containment cooling will be provided. The above assumptions are in agreement with the provisions of Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Sumps," and the Standard Review Plan 6.2.2 (Rev. 2), "Containment Heat Removal Systems" and, therefore, are acceptable.

The potential for debris to clog the residual heat removal system suction lines was evaluated. Each residual heat removal system pump is provided its own suction line and strainer assembly. The pipe insulation used in the drywell, metal reflective insulation, is of a type to minimize the potential for its breaking away from piping and being carried through the vent system into the suppression pool. This design minimizes the potential of clogging the suction line. Therefore, we find the design of the pump suction strainers acceptable.

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 will conform to General Design Criteria 38, 39, and 40 and is acceptable.

6.2.3 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.

The design objective of the containment isolation system is to allow the normal or emergency passage of fluids through the containment 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. The applicant specified design bases and design criteria as well as the isolation valve arrangements to be used for isolation of primary containment penetrations.

The containment isolation system is designed to automatically isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of two isolation valves in series or a closed system and isolation valves, are provided to assure that no single active failure will result in the loss of containment integrity. The containment isolation system components, including valves, controls, piping and penetrations, are protected from internally or externally generated missiles, water jets and pipe whip (see subsection 3.6 of this report).

The basis for our acceptance has been the conformance of the containment isolation provisions to the Commission's regulations as set forth in the General Design Criteria, and to the guidance provided in the Standard Review Plan noted below.

The containment isolation systems are designed to the American Society of Mechanical Engineers Code, Section III, Class 1 or 2 and are classified as seismic Category I design systems.

The containment isolation provisions for the lines penetrating containment conform to the requirements of the General Design Criteria 55, 56 or 57, as appropriate. As provided by General Design Criteria 55 and 56, there are containment penetrations whose isolation provisions do not have to satisfy the explicit requirements of the General Design Criteria but can be acceptable on some other defined basis.

Most of those penetrations not satisfying the explicit requirements of the General Design Criteria were found acceptable based on their meeting the alternative criteria specified in Section 6.2.4.II of the Standard Review Plan, "Containment Isolation System". These alternative acceptance criteria are summarized below:

(1) Lines that must remain in service following an accident and lines which should remain in service during normal operation for safety reasons are provided with at least one isolation valve. A second isolation boundary is formed by a closed system outside the containment.

(2) Where a closed system outside the containment forms the second isolation boundary, each of the systems and all components which form its boundary are designed to Quality Group B and seismic Category I standards. Valves which isolate the branch lines of these closed systems outside containment are normally closed and under strict administrative control.

(3) On some engineered safety features or related system remote manual valves are used in lieu of automatic valves since these lines must remain in service following an accident. Where remote manual valves are used leakage detection capabilities are provided.

(4) On some penetrations the containment isolation provisions consist of two valves in series both of which are outside the containment. The location of a valve inside containment would subject it to more severe environmental conditions (including suppression pool dynamic loads) and it would not be easily accessible for inspection.

Those lines which we found acceptable based on the criteria specified in the Standard Review Plan include: The reactor core isolation cooling system and residual heat removal system head spray, residual heat removal system-low pressure core injection system to reactor pressure vessel, low pressure core spray system to reactor pressure vessel, high pressure core spray system to reactor pressure vessel, residual heat removal system-drywell spray, residual heat removal system pump suction, low pressure spray system suction, high pressure core spray system suction, residual heat removal system-suppression pool spray, low pressure core spray system minimum flow line, residual heat removal system minimum flow line, high pressure core spray system test line, drywell equipment sump drain line, drywell floor sump drain, standby liquid control system, residual heat removal system test line A, drywell purge exhaust, drywell purge inlet, suppression chamber purge exhaust, combustible gas control system, traversing incore probe system, instrument air supply, control rod drive system hydraulic, and instrument lines.

Other lines penetrating the containment described below do not meet either the explicit requirements of the General Design Criteria or the alternative Standard Review Plan acceptance bases but meet acceptable isolation criteria on other defined acceptance bases.

Feedwater Lines

The feedwater line penetrates the drywell to connect with the reactor pressure vessel. It has three isolation valves. The isolation valve inside the drywell is a check valve. Outside the primary containment is another check valve. Farther away from the primary containment is a motor-operated gate valve. Should a break occur in the feedwater line, the check valves prevent significant loss of reactor coolant inventory and offer prompt primary containment isolation. During the postulated loss-of-coolant accident, it is desirable to maintain reactor coolant makeup from all sources of supply. For this reason, the outermost valve does not automatically isolate upon a signal from the protection system. Instead of using a leakage detection system in conjunction with use of a remote manual valve, this valve will be procedurally controlled and remotely closed from the control room to provide long-term leakage protection 20 minutes after a postulated loss-of-coolant accident.

We find this acceptable, since the time to close this valve is approximately the same as if a leakage detection system were employed.

Control Rod Drive Return to Reactor Pressure Vessel

This line which injects into the reactor vessel from the control rod drive system has three isolation valves. In addition to a simple check valve inside the drywell and a check valve outside the drywell, a motor-operated and remote manually actuated gate valve functions as a third isolation valve. During the postulated loss-of-coolant accident, it is desirable to maintain reactor coolant makeup from all sources of supply. For this reason, valves which automatically isolate upon a signal are not included in the design of this system. Consequently, a third valve provides long-term leakage control. Should a break occur in the control rod drive return line, the check valves would prevent significant loss of inventory and offer immediate isolation. The outermost isolation valve would provide long-term leakage control. The remote manual gate valve would be procedurally closed 20 minutes after a postulated loss-of-coolant accident. We find this acceptable since the closure time associated with a detection system is approximately the same.

Emergency Core Cooling System Heat Exchanger Relief Valves

These system penetrations meet the two barrier criteria described in the Standard Review Plan with a closed system outside containment and a containment isolation valve. However, the isolation valve consists of a system relief valve, which discharges into the containment. We find use of this relief valve acceptable as an isolation valve since accident pressure seats rather than unseats the valve.

Reactor Building Closed Cooling Water System

Containment isolation barriers are normally designed to Quality Group B design criteria. However, the reactor building closed cooling water systems, which the applicant has identified as containment isolation barriers, are designed to Quality Group C requirements. The applicant has committed to perform inservice inspections of the system in accordance with Section XI of the American Society of Mechnical Engineers Code. These periodic inspections will assure the reliability of these systems. Accordingly, we find the design of these isolation barriers, when supplemented by the above cited inservice inspections, acceptable.

Reactor Core Isolation Cooling System Turbine Exhaust and Vacuum Pump Discharge Line

The applicant proposed for the reactor core isolation cooling system turbine exhaust and vacuum pump discharge lines, the use of simple check valves outside containment. For these penetrations, the check valves serve as antisiphon devices to prevent the suppression pool water from being siphoned from the containment. Replacement of the check valve by any other type of valve would eliminate this desirable feature. Therefore, we find the isolation provisions for these lines acceptable.

Conclusion

Based on the above evaluation, we conclude that the applicant's proposed design of the containment isolation system satisfies the requirements of General Design Criteria 54, 55, 56 and 57 and is acceptable.

6.2.4 Containment Purge System

The applicant proposed use of the drywell and suppression pool purge system from four to six times per year to cleanup the air purged from the primary containment during hot shutdown, cold shutdown and refueling modes. The applicant does not anticipate that it will be necessary to operate these systems during plant operation but he stated that operation flexibility in the use of this system is desirable.

The containment purge system consists of 18-inch inlet and outlet lines, complete with redundant 18-inch butterfly isolation valves provided for each of the drywell and suppression pool purge lines. In addition, a two-inch bypass valve is provided so that the station operator may relieve a small quantity of air from the drywell to reduce pressure.

We consider purging of the primary containment under plant operating conditions undesirable. However, we find it acceptable if the design criteria set forth in Technical Position Containment Systems Branch 6-4, "Containment Purging During Normal Plant Operation," (Standard Review Plan 6.2.4, "Containment Isolation System") are met by the design.

We reviewed the design of the containment purge system based upon the criteria specified in the technical position and find that the purge system will meet these criteria, satisfy the requirements of General Design Criteria 54 and 56 and therefore is acceptable.

6.2.5 Combustible Gas Control

The combustible gas control systems include the piping valves, components and instrumentation necessary to detect the presence of combustible gases within the primary containment and to control the concentration of these gases.

The scope of review of the design and functional capability of the combustible gas control system for the Zimmer Nuclear Power Station included drawings and descriptive information of the equipment to mix the containment atmosphere, monitor combustible gas concentration, and reduce combustible gas concentrations within the containment following the design basis accident. The review also included the applicant's proposed design bases for the combustible gas control systems, and the analyses of the functional capability of the system provided to support the adequacy of the design bases.

The bases for our acceptance are the conformance of system design and design bases to the Commission's regulations as set forth in the General Design Criteria, and to applicable regulatory guides, branch technical positions, and industry codes and standards.

Following a loss-of-coolant accident, hydrogen may accumulate within the containment as a result of metal-water reaction between the fuel cladding, and as a result of radiolytic decomposition of the post-accident emergency cooling water. The applicant analyzed the production and accumulation of hydrogen from the above sources in accordance with the guidelines of Branch Technical Position Containment Systems Branch 6-2, "Control of Combustible Gas Concentrations in Containment." The guideline regarding the metal-water reaction states that hydrogen production is five times the maximum calculated reaction under 10 CFR 50.46, or that amount that would be evolved from a core-wide average depth of reaction into the original cladding of 0.23 mils, whichever is greater, in two minutes. Results of the applicant's emergency core cooling system evaluation shows that the metal-water reaction is less than 0.1 percent of the fuel cladding. However, based upon Containment System Branch 6-2 the zirconium metal water reaction used for the Zimmer analysis is 0.7 percent of the cladding. We conclude that the applicant calculated the hydrogen source in accordance with the guidelines of Branch Technical Position Containment System Branch 6-2 and Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." The hydrogen source is therefore acceptable.

We find that the hydrogen concentration inside the containment would not reach the flammable limit for several hours; therefore, in accordance with the criteria described in 10 CFR 50.44, inerting the containment is not required.

The applicant proposed a hydrogen recombiner subsystem described in NEDE-21071-P, "Flammability Control System," May 1976, to limit the hydrogen concentrations within the containment to below four volume percent. The recombiner is manufactured by General Electric Company. It is designed to seismic Category I quality group B requirements. A single 100 cubic feet per minute recombiner subsystem was proposed. However, all active components of the recombiners are provided with redundancy such that no single failure will impair the normal function of the recombiners. The recombiner is located in the reactor building where it is protected from missile and jet impingement. Suction is taken from the drywell area and the discharge is returned to the suppression pool above water level. Our evaluation of this General Electric Company recombiner design is provided below.

In addition to the recombiner a backup controlled purge system has also been proposed. The capacity of this purge system is 100 cubic feet per minute.

Since the primary containment atmosphere is not inerted, the drywell hydrogen concentration is limited to four volume percent. The applicant calculates that the drywell hydrogen concentration will not reach this limit until about 16 hours after a postulated loss-of-coolant accident without the recombiner system in operation.

Following a postulated loss-of-coolant accident the operator is instructed by procedures to start the recombiner at 30 minutes after the assumed loss-of-coolant accident. After a three hour warm-up period the operator can initiate recombiner process flow.

The applicant performed calculations of the containment hydrogen concentration. Based on 100 cubic feet per minute flow at the recombiner at 3.5 hours after a loss-of-coolant accident the applicant calculates a maximum drywell hydrogen concentration of 3.5 volume percent occurring approximately 20 hours after the postulated accident. The maximum calculated wetwell hydrogen concentration is 2.5 volume percent. We performed similar analyses that confirm the results of the applicant's calculations.

Hydrogen sample points are located in the primary containment compartments so as to allow detection of nonuniform hydrogen concentrations. Two sample points are located in each compartment to ensure sampling capability following a single failure. The containment subcompartments with sample points include: wetwell, drywell, vessel skirt area, and the drywell head area.

We conclude that the design of the combustible gas control system conforms to all applicable regulations, guides, our positions, and industry standards and is acceptable.

Containment Hydrogen Recombiner Evaluation

A General Electric Company thermal hydrogen recombiner is used to control the hydrogen concentration in the Zimmer containment following a loss-of-coolant accident. This is the first applicant to propose such a unit. The recombiner system consists of the flammability control unit, and control panels. The General Electric Company qualification program for this recombiner is described in NEDE-23597, "Qualification Report - Flammability Control System," June 1977.

The flammability control unit is skid-mounted and located outside the containment. However, the control panels are located in the main control room. The atmosphere from the containment drywell is circulated through the flammability control unit and returned to the pressure suppression chamber. The major components of the flammability control unit are the blower, electric heater, heat exchangers and associated piping.

The design pressure of the recombiner system is 62 pounds per square inch gauge. The recombiner is capable of circulating 100 standard cubic feet per minute of containment atmosphere. However, the system is designed to recycle a portion of the 100 standard cubic feet per minute effluent gas back to the blower inlet for inerted containments. The recycled gas flow rate is regulated between 0 standard cubic feet per minute and 60 standard cubic feet per minute by a motor-operated flow control valve. Since the Zimmer containment is not inerted, the recycle system will not be used.

The qualification test program included (1) a thermal cycle test to verify that the recombiner heater and reaction tube will withstand the thermal cycling that will occur during the surveillance testing during their design life, and (2) functional tests to determine flow capacity to demonstrate that the recombiner will perform as intended in a post-loss-of-coolant accident containment environment.

The thermal cycle test consisted of heating the recombiner to its operating temperature and then cooling the recombiner. This procedure was repeated 80 times. The functional tests consisted of various tests which determined system flow rates and effluent hydrogen concentrations for different hydrogen, oxygen and steam concentrations. In addition the environmental and the 30 day test were performed with the test enclosure structure maintaining a hot, humid, condition by circulating steam through the enclosures.

The results of the tests conducted on the General Electric Company recombiner system demonstrate that the recombiner (1) is capable of withstanding the thermal induced cycling as a result of the surveillance testing during the 40-year design life, and (2) will perform as intended in a post-loss-of-coolant accident environment with a flow rate of 100 standard cubic feet per minute. For all cases in which there was not any recycled flow, the hydrogen concentration in the effluent was less than 1.0 percent.

Based on our review we conclude that the General Electric Company thermal hydrogen recombiner is capable of processing 100 standard cubic feet per minute of containment atmosphere for the Zimmer nuclear power plant with a hydrogen concentration up to four volume percent in the drywell. The test results provide adequate assurance that effective control can be maintained of the hydrogen concentration within the Zimmer containment following a loss-of-coolant accident. We find the General Electric Company recombiner acceptable for use in the Zimmer containment.

6.2.6 Containment Leakage Testing

We reviewed the applicant's containment leak testing program for compliance with the containment leakage testing requirements specified in Appendix J to 10 CFR, Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Such compliance provides adequate assurance that the containment leak-tight integrity can be verified throughout service lifetime and that the leakage rates will be periodically checked during service on a timely basis to maintain such leakage within the specified limits. Maintaining containment leakage with such limits provides reasonable assurance that, in the event of any radioactivity release within the containment, the loss of the containment atmosphere through potential leak paths will not be in excess of the limits specified for the site.

Specifically, we reviewed the containment leak testing program to assure that the containment penetrations and system isolation valve arrangements are designed to satisfy the containment integrated leak rate testing requirements and the local leak testing requirements of Appendix J. The proposed leak testing practices for the containment personnel airlocks and main steam isolation valves, however, differ from the explicit requirements of Appendix J. The acceptability of this is discussed below.

We have determined that an exemption from certain requirements of Appendix J to 10 CFR, Part 50 is required regarding the leak testing practices proposed below. We are currently reviewing the information provided by the applicant to determine whether or not a specific exemption should be granted. The areas of noncompliance are similar to those reviewed and approved by us for exemption on other applications. If we grant a similar exemption for Zimmer, our evaluation supporting the matter will accompany the granting documents.

Containment Personnel Airlocks

Appendix J to 10 CFR, Part 50 requires the containment personnel airlocks to be leak tested at 6-month intervals and after each opening during such intervals (III.D.2). Appendix J further requires that the test be conducted at the peak calculated containment pressure related to the design basis accident; i.e., Pa (III.B.2).

Based on plant operating experience, the requirement that airlocks be leak tested after each opening between six-month tests is an impractical requirement when frequent airlock usage is necessary over a short period of time. Since the airlock design permits leak testing of the door seals at a reduced pressure, the applicant plans to leak test the door seals within three days after opening an airlock by pressurizing the volume between the door seals to 10 pounds per square inch gauge. The acceptance criterion for the test is that the leak rate shall not exceed five standard cubic feet per hour. Testing an airlock in this manner is more practical, and still provides the desired confidence that the door seals have not been damaged. Furthermore, the six-month test of an airlock, which involves pressurization of the entire airlock to Pa, will be retained.

Main Steam Isolation Valves

Appendix J to 10 CFR, Part 50 requires local leak testing of boiling water reactor main steam isolation valves (II.H.4) at the peak calculated containment pressure related to the design basis accident (III.C.2). Furthermore, Appendix J requires that the measured leak rates be included in the summation for the local leak rate tests (III.C.3).

The applicant, however, proposes to leak test the main steam isolation valves at a reduced pressure and exclude the measured leakage from the combined leak rate for the local leak rate tests.

Each main steam line is provided with two main steam isolation valves which are positioned to provide better sealing in the direction of post-accident containment atmosphere leakage. In the event of a loss-of-coolant accident the main steam leakage control system will maintain a negative pressure between the main steam isolation valves. The effluent will be discharged into a volume where it will be processed by the standby gas treatment system prior to being released to the environs. A radiological analysis for this potential source of containment atmosphere leakage was performed, based on an assumed leak rate past the inboard main steam isolation valve of 11.5 standard cubic feet per hour.

The applicant plans to periodically leak test the main steam isolation valves to assure the validity of the radiological analysis.

The design of the main steam isolation valves is such that testing in the reverse direction tends to unseat the valve. Testing of the two valves simultaneously, between the valves, at design pressure would lift the disc at the inboard valve. This would result in a meaningless test. The proposed test calls for a test pressure of 20.2 pounds per square inch gauge to avoid lifting the disc at the inboard valve. The total observed leakage through both valves (inboard and outboard) is then conservatively assigned to the penetration. We conclude that this procedure is acceptable. Furthermore, excluding the leakage from the summation for the local leak rate tests is acceptable since the leakage has been accounted for separately in the radiological analysis of the site.

Conclusion

With the exception of the leak testing practices for the containment's personnel airlocks and main steam isolation valves, described above, the proposed reactor containment leak testing program complies with Appendix J to 10 CFR, Part 50 and is acceptable.

6.3 Emergency Core Cooling System

The emergency core cooling system is designed to provide water to the reactor coolant system in the event of a break in the pressure boundary. The emergency core cooling system capability extends to failures as large as a double ended rupture of the largest pipe carrying water or steam, and spurious safety/relief valve operation.

The basis for the design of the emergency core cooling system system is to limit damage to the fuel cladding in accordance with 10 CFR 50.46. The system must be capable of performing its design function even without offsite power and with a single failure, including loss of an emergency diesel.

6.3.1 System Design

The emergency core cooling system consists of the following systems:

(1) High Pressure Core Spray System;
(2) Automatic Depressurization System;
(3) Low Pressure Core Spray System; and
(4) Low Pressure Coolant Injection System

The high pressure core spray system maintains coolant level for small breaks and provides spray to cool the core for larger breaks; the reactor core isolation cooling system, also serves to cool the core in the event of low water level. The high pressure core spray system takes water from the condensate storage tank and pumps it into the spray sparger within the reactor vessel. The high pressure core spray suction is diverted to the suppression pool if the water supply in the condensate storage tank is exhausted, or if the water level in the suppression pool exceeds a prescribed value.

The automatic depressurization system is used to depressurize the reactor coolant system in the event of a small break in the reactor coolant system. The automatic depressurization system uses six of the safety/relief valves to reduce the system pressure so that the low pressure systems may inject water to cool the core. Operation of the automatic depressurization system is delayed for two minutes as a compromise between the need to depressurize the reactor coolant system during a loss-of-coolant accident and the need to minimize potential pressure transients in the reactor coolant system.

The low pressure coolant spray system pumps water from the suppression pool to a core spray sparger within the reactor vessel. The low pressure coolant injection system has three loops with a pump in each. Two of the loops can be used to cool the containment and the suppression pool (wetwell).

6.3.2 Evaluation

Single Failures

We reviewed the system description and piping and instrumentation drawings to assure that abundant core cooling will be provided during the injection phase with and without offsite power and assuming a single failure. A low reactor vessel water level and/or high containment pressure signal is required to start pumps and open discharge valves. A single failure of a pump, suction, or discharge valve, could result in the loss of a single emergency core cooling system loop.

A diesel generator failure could result in failure of either the high pressure core spray train or a low pressure core injection and low pressure core spray train or two low pressure core injection trains. Failure of individual components would cause failures of only single trains. For this reason, loss-of-coolant accident calculations were predicated upon failures of diesel generators in order to provide conservative results.

The applicant states that after 10 minutes, the core is reflooded and either one of the core spray pumps or a low pressure core injection system pump is sufficient to maintain core cooling. Only one low pressure core injection system loop (containing a heat exchanger) is required for long term removal of decay heat from the containment.

Qualification of the Emergency Core Cooling System

The emergency core cooling system is designed to meet the requirements of Seismic Category I requirements in compliance with Regulatory Guide 1.29, "Seismic Design Classification," as discussed in subsection 3.2 of this report; it is housed in structures designed for seismic events, tornadoes, floods, and other phenomena in accordance with the requirements of General Design Criterion 2 as discussed in subsection 3.8 of this report. All emergency core cooling system equipment is designed in accordance with Quality Group A and B standards in compliance with Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," as discussed in subsection 3.2 of this report.

The emergency core cooling system is protected against pipe whip inside and outside of containment and against discharging fluids resulting from failure of equipment in compliance with the requirements of General Design Criterion-4, Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," and our applicable criteria as discussed in subsection 3.6 of this report.

The emergency core cooling system is designed to permit inservice inspection in accordance with the requirements of General Design Criterion 36, as noted in subsection 5.4 of this report.

The high pressure core spray system serves to complement the reactor core isolation cooling system in cases of low water level in the reactor vessel. The alternate functions of the residual heat removal system include shutdown cooling, steam condensing mode, containment cooling, and low pressure coolant injection. Use in one mode does not impair the operational capability in another mode. The residual heat removal system valve logic requires low pressure core injection system alignment in the event of a loss-of-coolant accident, taking precedence over other residual heat removal system functional modes.

Emergency core cooling system equipment is designed to operate during normal and accident conditions in accordance with the requirements of General Design Criterion 4 as discussed in subsection 3.11 of this report.

Functional Design

The available net positive suction head for the pumps in the emergency core cooling system subsystems, has adequate margin to prevent cavitation and assure pump safety in accordance with Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." Three jockey pumps are used to keep the emergency core cooling system injection lines filled with water in order to prevent water hammer on startup. Each pump is powered by a different emergency bus so that loss of offsite power coupled with single failure can only disable one filling system. Failure of a jockey pump is alarmed in the control room.

Containment isolation of the emergency core cooling system lines complies with the requirements of General Design Criteria 55, 56, and 57, as discussed in subsection 6.2 of this report.

For pump protection, all of the emergency core cooling subsystems (high pressure core spray, low pressure core spray and low pressure core injection systems) have miniflow lines to permit a limited amount of flow in the event injection to the reactor coolant system is prevented for any reason. When a quantity of flow, sufficient to protect the pump, is detected in the injection line, the valves in the miniflow bypass lines close.

We also reviewed the potential for passive failures in the emergency core cooling system following an assumed loss-of-coolant accident. The passive failures considered include failures such as pump seals, instrument penetrations, or valve stem leakage.

Leakage can be detected by changes in suppression pool level and the reactor building sumps which have level alarm detection systems. Collected leakage is normally pumped to radwaste but could be diverted to the tendon tunnel which has a volume of 66,000 gallons. It is concluded that leakage due to passive failures can be detected in sufficient time to isolate the affected system and permit transfer to redundant equipment.

Isolation between the reactor cooling system and each emergency core cooling system is provided by a check valve and a closed motor operated valve. We require that isolation valves be leak tested periodically in accordance with the provisions of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. Leakage limits for these valves will be provided in the Technical Specifications.

6.3.3 Testing

The applicant will demonstrate the operability of the emergency core cooling system by preoperational and periodic testing, as required by Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants," and General Design Criterion 37.

Preoperational Tests

Preoperational tests will assure proper functioning of controls, instrumentation, pumps, piping, and valves. Pressure differentials and flow rates will be measured for later use in determining acceptable performance in periodic tests. The applicant will meet the guidelines of Regulatory Guide 1.68 mentioned above for preoperational and initial startup testing of the emergency core cooling system.

Periodic Component Tests

The applicant will test the subsystems comprising the emergency core cooling system periodically to show that specified flow rates are attained. Every 18 months a test is performed in which each subsystem is actuated through its emergency operating sequence. These tests comply with General Design Criterion 37.

6.3.4 Performance Evaluation

We reviewed the loss-of-coolant-accident analyses presented by the applicant in Section 6.3.3 of the Final Safety analysis Report. Calculations were conducted in accordance with the methods described in General Electric Topical Report NEDO-20566, "General Electric Company Analytical Model for Loss-of-Coolant Analyses in Accordance with 10 CFR Part 50, Appendix K," dated August 1974 and "General Electric Refill Reflood Calculation" transmitted December 20, 1974. During 1977, the General

Electric Company proposed several changes to its emergency core cooling system evaluation model. These changes have been approved by us and are described in our "Safety Evaluation for General Electric emergency core cooling system Evaluation Model Modification."

As the lead BWR-5 plant, a full break spectrum was presented for Zimmer. As a minimum the water level inside the shroud, the reactor vessel pressure, the convective heat transfer coefficient, and the peak cladding temperature were presented for each break analyzed. For the limiting break (double-ended offset shear with a discharge coefficient = 1.0), the applicant also calculated the maximum cladding oxidation and maximum hydrogen generation. The most limiting break was found to be a double-ended break of the recirculation line on the suction side of the pump.

For the loss-of-coolant accidents which include the limiting break area and breaks with areas 0.8 and 0.6 times the design basis accident break area, the applicant reported that the worst single failure was the failure of a diesel generator which supplied power to low pressure coolant injections system trains B and C. The worst single failure for the transition break (1.0 square foot) in the recirculation line on the suction side of the pump was reported to be failure of the high pressure core spray system train. The applicant reported that the worst failure for the other small breaks (0.07 and 0.2 square feet), both in the recirculation line, was also the high pressure core spray system train failure. Other breaks were also considered, as noted below.

Break Location	Size	Worst Single Failure
Core Spray Line	0.278 square feet	Diesel generator supplying power to low pressure core spray system and low pressure coolant injection system train A
Feedwater Line	0.635 square feet	High pressure core spray system train failure
Main Steam Line		
Inside Containment	2.598 square feet	Diesel generator supply power to low pressure coolant injection system train B&C.
Outside Containment	2.723 square feet	High pressure core spray system train failure

The applicant assumed the failure of one automatic depressurization system valve in all calculations, in addition to the other single failures con·idered. The applicant also assumed that power to the recirculating pumps was lost at the beginning of the loss-of-coolant accident. It was shown that the consequences were less severe if pumps were assumed to continue running.

For the design basis accident, results were as follows:

	Design Basis Accident Calculated	Allowable
Peak Clad Temperature	1821 degrees Fahrenheit	2200 degrees Fahrenheit
Maximum Cladding Oxidation	0.6 percent	17 percent
Maximum Total Hydrogen Generation	0.05 percent	1 percent

The applicant cited NEDO-20566 which shows that core damage as a result of the worst loss-of-coolant accident will not be sufficient to prevent effective cooling by the emergency core cooling (i.e., a coolable geometry is maintained). Additionally, the long term cooling will be maintained by at least one core spray system (either high or low pressure) in the event of a recirculation line break or by reflooding the reactor vessel by means of at least one low pressure coolant injection system train.

We are reserving judgment on the conformance of these analyses to 10 CFR 50.46 pending our review and assessment of the impact of the recent Two Loop Test Apparatus Tests performed by the General Electric Company on the loss-of-coolant evaluation model.

The Two Loop Test Apparatus is part of the Blowdown/Emergency Core Cooling Program being conducted by General Electric at San Jose, California, under sponsorship by the Electric Power Research Institute, General Electric Company and the Nuclear Regulatory Commission. The purposes of the program are:

(1) Obtain and evaluate basic blowdown and emergency core cooling data from test configurations which have calculated performance characteristics similar to a boiling water reactor with 8x8 fuel bundles during a hypothetical loss-of-coolant accident.

(2) Determine the degree to which models for the boiling water reactor system and fuel bundles describe the observed phenomena, and as necessary, develop improved models which are generally useful in improved loss-of-coolant accident analysis methods.

The Two Loop Test Apparatus configuration used for blowdown and emergency core cooling is scaled to a BWR/6 design (624 bundles) and includes the following major components: (1) pressure vessel and internals, (2) 8x8 heated bundle, (3) two recirculation loops, (4) emergency core cooling systems (high pressure core spray, low pressure core spray, low pressure coolant injection and automatic depressurization systems) and (5) auxiliary systems.

The applicant investigated the consequences of diverting low pressure coolant injection pumps to the wet well spray system 10 minutes after the hypothetical loss-of-coolant accident. The worst single failure/break type combination with low pressure coolant diversion was stated to be a break in the high pressure core spray line of about 0.02 square foot combined with the failure of the diesel generator which supplies power to the low pressure core spray system pump and one low pressure coolant injection system pump. The resulting peak clad temperature was approximately 1725 degrees Fahrenheit. While the calculated peak clad temperature with low pressure coolant injection system diversion to the wet well spray system satisfies emergency core cooling system acceptance criteria, we have not completed our review of the proposed automatic diversion of low pressure coolant injection system pumps and operator procedures relative to diversion. We will report this matter in a supplement to this report (See subsection 6.2.1 of this report.)

The core spray sparger for both the high and low pressure core spray systems each consists of two semicircular segments which form an essentially complete circular sparger. Water is sprayed radially onto the tops of the fuel assemblies by short elbow nozzles spaced around the sparger. Tests of this type of spray system to ensure that adequate coolant is delivered to each fuel assembly were made in a full-scale test in which air at atmospheric pressure simulated the post loss-of-coolant accident steam environment. However, recent tests conducted on a single nozzle indicate that the actual steam environment may adversely affect the distribution of flow from certain types of core spray nozzles. As discussed in NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," this problem is being studied by us under Task A-16 entitled, "Steam Effects on BWR Core Spray Distribution." Preliminary analyses and measurements have been made which support the existence of a significant safety margin between that amount of spray flow provided to each fuel assembly in the post-loss-of-coolant accident steam environment and that used to calculate the spray cooling coefficients assumed in the loss-of-coolant accident analyses. Tests will be conducted by General Electric Company to confirm spray flow margins used in the emergency core cooling system loss-of-coolant accident analyses. In the interim period, we believe there is a sufficient technical basis to permit licensing of Zimmer, Unit 1. This conclusion is based on:

(1) The existence of the safety margins between available and required spray flow indicated by preliminary analyses and measurements,

(2) The existence of counter-current-flow-limiting phenomena should provide a steam/water layer on top of the core and force a more even distribution of the core spray,

(3) The timely confirmation of the spray flow margin presently believed to exist which should be provided by the aforementioned tests and information.

6.3.5 Conclusions

We reviewed piping and instrumentation drawings and description of the emergency core cooling system presented in the Final Safety Analysis Report. We, subject to the reservation stated in subs ction 6.3.4 of this report, find that the emergency core cooling system conforms t pertinent General Design Criteria and Regulatory Guides and is acceptable.

6.4 Habitability Systems

The following discussion is related to the emergency protection provisions for the control room with respect to radiological and toxic gas hazards.

6.4.1 Radiological Dose Protection

The applicant's design meets General Design Criterion 19 by use of adequate shielding and by installing a dual fresh-air inlet system containing redundant, once-through, deep bed charcoal filter trains for control room pressurization. In addition, the system provides for 50 percent full flow filtered recirculation through smoke and odor removal charcoal filters which are installed in each of the redundant normal air handling equipment trains.

Each of the pressurization filter trains contains a 2000 cubic foot per minute once-through charcoal filter for removing radioactive iodine from the incoming outside air. The recirculation charcoal filters are sized for 100 percent full flow (39,090 cubic feet per minute) capacity for smoke and odor removal. Under radiological emergency conditions, they will be used in a 50 percent (19,545 cubic feet per minute) recirculation mode for radioactive iodine removal from the control room air. All emergency ventilation filtration will be initiated automatically upon a high radiation signal from the detectors located on the outside air intakes.

The control room ventilation system is equipped with a separate air intake which is to be used for purging the control room of smoke or noxious gases. The intake is normally isolated by a single damper. Since this damper does not meet the single failure criterion, we required the applicant to provide suitable modifications or administrative procedures, such as:

(1) Provision of a second damper in series with the single damper, or

(2) Appropriate administrative procedures and/or design modifications for the single damper. Specifically, a technical specification limiting the "open" status of the damper to purging operations, and adequate differentiation (e.g., separation distance on control board) between the manu?' control of this damper and the other isolation dampers, so as to minimize the probability of the single damper being inadvertently opened.

The applicant committed to the second alternative above and we find the commitment acceptable.

We determined that the potential radiation doses to control room personnel following a loss-of-coolant accident are within the guidelines of General Design Criterion 19. We conclude that the design of the control room emergency ventilation system is acceptable for the purpose of preventing significant radiological exposure of operating personnel.

6.4.2 Toxic Gas Protection

The toxic gas hazards with respect to the control room were evaluated using 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," and 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." The applicant made provisions for storing ten one-ton cylinders of chlorine within the circulating water pump structure which is about 500 feet away from the nearest control room outside air intake. The control room outside air intakes is equipped with quick acting redundant chlorine detectors which will, upon chlorine detection, isolate the control room ventilation system from the outside atmosphere. Our evaluation of the chlorine hazard from the onsite chlorine tanks finds the control room to be protected adequately. With respect to offsite toxic gas hazards, a preliminary review of the river barge traffic indicates that the frequency of shipment of toxic chemicals is acceptably low in accordance with the Regulatory Guide 1.78 and does not pose an undue hazard to the control room operators.

At our request the applicant determined the shipping data for toxic chemicals known to be transported regularly on the Chesapeake and Ohio Railway near the site (about 2300 feet from the control room). In accordance with the guidelines of Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," the data obtained by the applicant from the C&O Railroad indicate that only chlorine and anhydrous ammonia are shipped in sufficient frequency and quantity to warrant design basis toxic gas protection.

The applicant committed to providing redundant chlorine and ammonia detectors in the control room outside air intakes. The detectors will be designed to close automatically the outside air dampers, thus isolating the control room prior to the introduction of significant amounts of the above chemicals into the control room. In addition, the applicant will provide breathing apparatus and a 30-hour bottled air supply for at least five persons within the control room area.

On the basis of the above we find that the control room is protected adequately against the hazards associated with the shipment of toxic gases along the C&O railway and Ohio River near the site.

With respect to U.S. Route 52 near the Zimmer site, our review indicates that the daily commercial traffic is in the range of 400 to 600 trucks per day. Since some of this traffic may involve shipment of toxic material, there is a potential hazard to the plant with respect to this route. According to Regulatory Guide 1.78, truck shipments of toxic materials within 5 miles of a plant should be analyzed if the frequency is 10 shipments per year or more. Since information on the type of cargo being carried by the commercial vehicles have not been presented in the Final Safety Analysis Report and since our initial efforts have indicated that this information is not readily available at the present time, we are not able to determine whether or not a hazard exists to the control room operators due to potential toxic gas releases on U.S. Route 52 near the Zimmer site. It should be noted, however, that the Zimmer control room ventilation system is equipped with isolation capability due to the protection requirements against chlorine and ammonia.

In view of the above, we consider this to be a confirmatory issue and we will pursue its resolution with the applicant and report on the matter in a supplement to this report.

6.5 Fission Product Removal and Control Systems

6.5.1 Engineered Safety Features Filter Systems

The engineered safety feature atmosphere cleanup systems for the Zimmer Nuclear Power Station consist of process equipment and instrumentation to control the releases of radioactive material in gaseous effluents (radioiodine and particulate matter) following a design basis accident. In the Zimmer Station design, there are two filtration systems designed for this purpose, the control room habitability system, and the standby gas treatment system.

6.5.2 System Description and Evaluation

Control Room Habitability System

The function of the control room habitability system is to supply non-radioactive air to the control room after a design basis accident and to pressurize the control room to a slight positive pressure. This system will permit operating personnel to remain in the control room following a design basis accident. The standby makeup air filter train of the control room has redundant active components with an intake design capacity of 2,000 cubic feet per minute and contains the following components: two fans, two heating coils, a demister, a prefilter, two high effiency particulate air filters, and a charcoal adsorber. The normal air supply and return portion of the habitability system consists of redundant trains with an intake and recirculating design capacity of 39,000 cubic feet per minute. Each train contains the following components: a high efficiency particulate air filter, a charcoal adsorber, a humidifier, and a fan.

The equipment and components are seismic Category I design and are located in a seismic Category I structure. Following a design basis accident , the pressurization and recirculation system will be automatically activated by a signal from radiation monitors located in the inlet ducts or be activated manually from the control room.

We determined that the control room habitability system is designed in accordance with the guidelines of Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineering Safety Features Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and is capable of maintaining an acceptable control room environment following a design basis accident. We, therefore, find the design of the system to be acceptable.

Standby Gas Treatment System

The function of the standby gas treatment system is to produce and maintain a slightly negative pressure (0.25 inch of water gage) in the secondary containment and to provide control of the releases of radioactive materials in gaseous effluents following a design basis accident. The system is automatically activated by a high drywell pressure signal, low reactor water level, and airborne radiation monitors in the reactor building ventilation system, or can be manually controlled from the control room. The standby gas treatment system consists of redundant systems, with each system having a treated exhaust capacity of 2,300 cubic feet per minute of air and a recirculation design capacity of 80,000 cubic feet per minute. Each engineered safety feature filter train contains the following components: fan, demister, electric heating coil, prefilter, upstream high efficiency particulate air filter, two carbon adsorbers in series, and downstream high efficiency particulate air filter. The equipment and components are seismic Category I design and are located in a seismic Category I structure.

We determined that the standby gas treatment system is designed in accordance with the guidelines of Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Engineering Safety Features Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and is capable of maintaining suitable control of gaseous effluents following a design basis accident. We, therefore, find the design of the system acceptable.

7.0 INSTRUMENTATION AND CONTROLS

7.1 General Information

7.1.1 Introduction

Section 7.1 of the Final Safety Analysis Report contains information pertaining to safety-related instrumentation and control systems, their design bases, and the applicable acceptance criteria. We review this information as detailed in the Standard Review Plan, and also determine the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format for Safety Analysis Reports.

7.1.2 General Findings

We audited the applicable design criteria listed in the Final Safety Analysis Report Tables 7.1-3 thru 7.1-9, the design bases and the descriptions of the safety-related systems, and conclude that they comply with those criteria which we reviewed and approved for the issuance of the construction permit. In addition, the criteria used for the Zimmer design are similar to the acceptance criteria of Section 3.10 and Table 7-1 of the Standard Review Plan except for the following major differences:

(1) Environmental qualification of Class IE equipment was conducted in accordance with Institute of Electrical and Electronic Engineers Standard 323-71 standard instead of completely in accordance with the Institute of Electrical and Electronic Engineers Standard 323-74 standard. However the criteria used for the Zimmer plant are acceptable based on the implementation guidance described in Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants."

(2) Institute of Electrical and Electronic Engineers Standard 384 and Regulatory Guide 1.75, "Physical Independence of Electrical Systems," did not exist when the construction permit was issued. However, with the exceptions as indicated in sections 7 and 8 of this report, the Zimmer Station meets the requirements of these latter criteria.

We have concluded, on the basis of this information, that the Zimmer criteria are acceptable except as noted for specific systems which were found unacceptable and are so reported below.

Our review of the Final Safety Analysis Report, Chapter 15, revealed that certain systems, classified as not being required for safety, are assumed to function to mitigate the consequences of design bases events. These systems are identified below under the appropriate subheadings of this report.

7.1.3 Specific Findings

During the course of our review, we found a pattern of "repetitive" problems in the implementation of the design of Class IE equipment. Because these problems involved more than one category of safety-related systems and/or more than one system in a particular category, they are presented in this section of this report.

Protection of Motor-Operated Valves

Torque switches and limit switches are used to prevent mechanical damage to the valve operators. Traditionally, drift in the adjustment of torque switches has been a major cause for valve failures in operating nuclear power plants. In order to improve the reliability of motor-operated valves, torque switches were removed from the safety function position circuits of many Class IE valves. Torque switches remained installed in the "normal" position circuits (circuits provided for manual test operation) of the valves. We requested the procedures and frequency which will be used to verify proper torque switch operation for these remaining torque switches. We reviewed this information and found it acceptable so that there is a low likelihood of undetected valve damage during routine valve testing.

Our review of the method for implementing thermal overload protection indicated that the design is in compliance with the requirements of Branch Technical Position, Electrical Instrumentation and Control Systems Branch 27 and is acceptable.

Testing

The applicant proposed to use a General Electric Company supplied computer system known as "Startrek" to monitor plant startup testing and that both safety and non-safety inputs will be connected to the computer. We were concerned that such equipment may compromise the electrical independence and physical separation of Class IE redundant divisions, and requested more information. Our review of the responses indicates that this system will be physically separated and electrically isolated from Class IE equipment using isolation devices which satisfy the requirements of Institute of Electrical and Electronic Engineers Standard 279-1971 (10 CFR Section 50.55a) and wiring which satisfies the Zimmer separation criteria. We will review the qualification testing of the isolation devices when the tests are completed and provide our evaluation in a supplement to this report.

Safety System Setpoints

The compatibility of the setpoints used to automatically control Class IE equipment with the range and accuracy of the sensors used was reviewed because of instrument setpoint drift in similar plants. We found that the setpoints for some

Class IE systems were not listed in the Final Safety Analysis Report and that, for some of the sensors listed, the worst-case combination of setpoint and accuracy exceeded the sensor range. The resolution of this issue may require the replacement of some sensors and will be provided in our approval of the technical specifications to be part of the operating license.

Response Time Testing

We requested a description of the provisions in the design for conducting response time testing and a description of the methods to be used to assure that the Class IE equipment can meet its design bases. Further discussion was postponed until the revised technical specifications are submitted. We will pursue this matter and provide the resolution with our issuance of the technical specifications to be a part of the operating license.

7.1.4 Evaluation

With the exception of the specific issues which are presented above and in the following subsections of this section, our review indicates that there is reasonable assurance that the safety-related instrumentation and control systems have been designed and implemented to the applicable safety criteria identifed in subsection 7.1.2 of this report. We conclude that implementation of all safety-related systems in accordance with the criteria identified in subsection 7.1.2 of this report will provide assurance that the plant will perform as designed in normal operation, anticipated operational transients and postulated accident conditions, and will meet the applicable requirements of General Design Criterion 1 with regard to appropriate records of design, fabrication, erection and testing of systems and components important to safety.

7.2 Reactor Trip System

7.2.1 General Discussion

Section 7.2 of the Final Safety Analysis Report describes the reactor trip system. The reactor trip system, which is part of the reactor protection system, includes those power sources, sensors, initiation circuits, logic matrices, bypasses, interlocks, racks, panels and control boards, and actuation and actuated devices that are required to initiate reactor trip. This trip system has been designated as the reactor protection system. The reactor protection system is designed to initiate automatically the release of the control rods in the reactivity control system to assure that specified acceptable fuel design limits are not exceeded. It also includes those safety-related portions of control systems, the actions of which inhibit or limit the response of the reactivity control system, to ensure that fuel design limits and safety limits are not exceeded.

7.2.2 General Findings

The design of the reactor protection system is similar to the designs for Hatch 1 and 2. The basic design, which utilizes one-out-of-two-taken-twice logic, has been reviewed extensively in the past. There are design changes incorporated in Zimmer to satisfy the requirements of Institute of Electrical and Electronic Engineers Standard 279-1971. These changes are discussed in the following paragraphs. With the exception of the items noted below, we evaluated the design and implementation of the reactor trip system and conclude that this design meets the Commission's requirements identified in subsection 7.1.2 of this report and (except as noted in subsection 7.1.3 of this report and in the following paragraphs) is therefore acceptable.

7.2.3 Specific Findings

Detector Location and Manual Trip Capability

The smaller core size, than LaSalle for example, has resulted in a reduction in the number of local power range monitor assemblies with the number of average power range monitors remaining unchanged. The assignment pattern remained the same; thus, the quality of averaging is maintained.

The Zimmer design (BWR-5) includes the BWR-6 physical separation of the reactor trip system detectors into four separate areas within the reactor building, which provides for improved physical independence between redundant equipment. The reactor manual trip circuit was modified to meet the single failure criterion of Section 4.17 of Institute of Electrical and Electronic Engineers Standard 279-1971. We find the criteria used for detector location and manual trip capability acceptable.

Alternate Reactor Protection System Power Sources

The reactor protection system receives normal electrical power from two motor generator sets. These sets provide isolation from the power sources for the two logic systems and enable the sensor wiring to be electrically isolated from the engineered safety features. Backup power to the logic is provided from the division 1 instrument bus 1A (Class IE source). This action results in redundant division cables (120 volt alternating current division 1 and [illegible] volt direct current division 3) running in the same conduit on instrument racks A and C when the alternate source is used to replace motor generator sets. The applicant agrees that this situation is unacceptable and is considering several design modifications. However, the resolution of the separation between the reactor protection system alternate power supply and division 3 direct current has been partially resolved by the General Electric Company which has advised the applicant that a non-Class IE supply should be used, but the applicant has identified two similar situations which involve the use of a common instrument and a common conduit for some division 2 direct current circuits and reactor protection system wiring at the instrument racks.

A visual inspection of these instruments and racks indicates that it is not possible to modify the instrument to provide two separate conduit entrances and that there is not sufficient space on the racks to properly mount an additional instrument and its associated calibration manifold. The physical location of these racks does not permit the installation of additional racks. Therefore, we are of the opinion that the present design of 1H22P004 (a typical rack) presents the best practical solution in which the opposing circuits are pulled from an instrument in common flexible conduit for a distance of two feet and terminated in a junction box.

The circuits are separated at this junction box and pulled through separate conduits to the termination cabinets at the top of the rack. The applicant has tentatively agreed to assuring that this technique is used for all of the affected circuits and racks but will not commit to this resolution until a more detailed study of the present designs is completed. (The applicant states that he is still considering installing an additional instrument on each rack and justifying the use of a Class IE alterna'e source.)

The circuits which are .nvolved in a lack of ideal separation are:

a. Reactor protection system channel A1 low reactor water level and high pressure core spray system high level stop cut off (division 3 direct current).

b. Reactor protection system channel A2 low reactor water level and high pressure core spray system high level cut off (division 3 direct current).

c. Reactor protection system channels B1 and B2 and reactor core isolation cooling system high level cut off.

d. Reactor protection system channel A2 and Division 2 PAM reactor vessel level indication.

This lack of ' ... .ion for two feet in each of four racks is deemed acceptable because:

a. The high level cut off of the high pressure core spray system and the reactor core isolation cooling system is not required for safety.

b. The low independence of the direct current circuitry would most likely lead to a loss of reactor protection system voltag^ and a half channel trip if faulting should occur.

c. Multiple insulation faults are required to cause a circuit failure.

d. The reactor protection system power sources are isolated from the Class IE buss by either a motor-generator set or a transformer and the batteries are isolated from the alternating current sources by their chargers.

e. The direct current circuits include ground fault alarms and one leg of each reactor protection system supply is grounded.

f. All affected circuits are in continuous operation and are routinely monitored in the control room.

We will continue to pursue this issue with the applicant and will provide the resolution in a supplement to this report.

Backup Scram Capability

Like earlier General Electric Company product lines, this design has a backup scram system which utilizes energy to open direct current solenoid valves. Our review of the wiring for this function indicated that division 1 and division 2 direct current (redundant divisions) are routed in the same conduits within the logic cabinets with the inputs to the non-Class IE process computer. In addition, the separation between the redundant direct current divisions, which appear on the same terminal strip, is insufficent. The applicant agrees that these are unacceptable designs and stated that the direct current wiring will be physically separated by distance from all other wiring or placed in separate metallic conduits.

Cabinet Lighting

The wiring associated with the reactor protection system cabinet penetrations 187 and 194, which were identified during the site visit, was found to be in error. Penetrations 187 and 194 are in the opposite holes. This results in the cabinet lighting circuit in 194 crossing and becoming associated with the isolation system wiring in 187. The applicant agreed that this is unacceptable, because the lighting circuits are not treated as associated circuits, and stated that this will be corrected. This matter must be resolved prior to permitting the plant to reach criticality.

Anticipated Transients Without Scram

We have not completed our review of the anticipated transients without scram requirements for boiling water reactors. The current status of our review of anticipated transients without scram is described in subsection 15.2 of this report.

Field Modifications to Protection System

During discussions on the resolution of the separation problems which are identified above, the applicant stated that the reactor protection cabinets will be extensively rewired in the field. In addition to correcting the problems which have been identified by us, the applicant will remove circuitry which is not used in the Zimmer design (prompt relief trip) and will install the logic for the recirculation pump trip.

Motor Generator Set Protection

The applicant has been informed of the concern which has been identified by us as a result of our review of the Hatch 2 application for an operating license.

This concern deals with the postulated single undetected failure of an output voltage sensor for either motor-generator set which could result in damage to the reactor protection system components and consequently potential loss of capability to scram.

We have also informed the applicant that, as a minimum, we will require all BWR 5 plants to implement those features and/or procedures which are required to resolve our concern for the E. I. Hatch Unit 2.

We are waiting for detailed information on the applicant's proposed resolution of this safety matter.

7.2.4 Evaluation

In addition to a site review, the scope of the review included the descriptive information, functional logic diagrams, functional instrumentation and electrical diagrams, final physical arrangement drawings and schematics. The review included the applicant's design bases and the relation to the design of the reactor trip system. The review also included the proposed means for identification of cables and equipment, periodic testing capability, and the qualification test program results for demonstrating the suitability of the reactor trip system.

With the exception of adequate physical separation and electrical isolation within the protection panels and cabinets, and the concern discussed in subsection 7.2.3 above, our review indicates that the applicant's designs for the reactor trip system and supporting systems satisfy the Commission's requirements as set forth in the General Design Criteria and to applicable regulatory guides, branch technical positions, and industry standards and are therefore acceptable. The Commission's requirements are listed in Table 7-1 of the Standard Review Plan. We will verify proper implementation of the resolutions outlined above and report our findings in a supplement to this report.

7.3 Engineered Safety Features Systems

7.3.1 General Discussion

Section 7.3 of the Final Safety Analysis Report describes the portion of the protection system used to initiate and control operation of the engineered safety features systems and their auxiliary supporting systems. The descriptive information, functional control diagrams, piping and instrument diagrams, electrical schematics and physical arrangement drawings, as presented in the Final Safety Analysis Report, were reviewed. The objective of our review was to determine that the engineered safety feature systems satisfy applicable design criteria and will perform as intended during all plant operating conditions and accident conditions for which its function is required.

7.3.2 General Findings

We evaluated the design and implementation of the engineered safety features systems and conclude that the designs meet the Commission's requirements identified in subsection 7.1.2 of this report and (except as noted in subsections 7.1.3, 7.2.3 and in the following paragraphs of this report) are therefore acceptable.

7.3.3 Specific Findings

Surveillance Testing

The validity of the models used as bases for the surveillance frequency in the Final Safety Analysis Report Appendix 6A was reviewed during our audit of the associated emergency core cooling system schematics. We find that the methods used and the results form an acceptable basis for establishing the surveillance frequency. The basis for this finding is that the calculations result in test intervals which are equal to or less than those which have been established in the Standard Technical Specifications which have been previously reviewed and approved for similar plants such as Hatch 2.

Initiation Logic for Engineered Safety Features Systems

The basic design of the initation and control systems for the engineered safety features systems is identical to the designs used for Hatch 2 with the exception of changes to satisfy the requirements of Institute of Electrical and Electronic Engineers Standard 279-1971. These changes are included in the Zimmer designs. The designs provide redundant equipment for manual initiation of protective actions, at the system level, for the emergency core cooling system and the containment and reactor vessel isolation control systems. Our review found and we conclude that the designs satisfy the acceptance criteria identified in Section 7.1.2 of this report and are therefore acceptable with the exception of discrepancies noted in subsections 7.1.3 of this report.

Emergency Core Cooling System

The applicant's Final Safety Analysis Report states that the emergency core cooling system includes the following subsystems: (1) high pressure core spray system, (2) automatic depressurization system, (3) low pressure core spray system, and (4) low pressure coolant injection mode of the residual heat removal system.

The purpose of emergency core cooling system instrumentation and controls is to initiate appropriate responses from the system to ensure that the fuel is adequately cooled in the event of a design basis accident. The cooling provided by the system restricts the release of radioactive materials from the fuel by preventing or limiting the extent of fuel damage following situations in which reactor coolant is lost from the nuclear system. The emergency core cooling system instrumentation is designed to detect a need for core cooling systems operation, and to initiate the appropriate actions. Successful core cooling for a specified line break accident is depicted in Figure 7.3-1 of the Final Safety Analysis Report for small line breaks: (1) the depressurization phase is accomplished by the high pressure core spray system, automatic depressurization system A or automatic depressurization system B, and (2) the low pressure core cooling phase is accomplished by low pressure core spray system, any two residual heat removal system pumps or the high pressure core spray system.

Similarly, the large break model uses the low pressure core spray system, the high pressure core spray system or the three residual heat removal system pumps for successful core cooling.

As a result of the response to acceptance review Request 222.14 and in accordance with the assumptions and results of the Final Safety Analysis Report, Appendix B and Chapter 15, it is our position that the reactor core isolation cooling system is also an engineered safety feature because it is required to be redundant to the high pressure core spray system to provide early core cooling in the event of a control rod drop accident at power. The reactor core isolation cooling system did not have a seismic Category I water source. The applicant did not agree with our position; however, the design was modified to provide automatic transfer to the suppression pool. This design change results in a reactor core isolation cooling system which satisfies our acceptance criteria.

Fill Pumps

Each subsystem of the emergency core cooling system, with the exception of the automatic depressurization system, is provided with a fill pump to keep the pump discharge piping full. The design basis for the fill system is to prevent water hammer when an emergency core cooling system pump is started. The fill pumps are powered from the same division as their respective emergency core cooling system pumps. System pressure monitoring is provided to alert the operator as to fill pump status. These fill pumps are under manual control and are listed in the Final Safety Analysis Report, Chapter 8, as normal and emergency electrical loads. We find that, based on past experience with equipment of this quality, that the electrical part of the system will perform its design function and is therefore acceptable.

High Pressure Core Spray System

The high pressure core spray system is a division 3 emergency core cooling system. The high pressure core spray system is designed to provide and maintain coolant inventory inside the reactor vessel to maintain cladding temperatures below fragmentation temperature in the event of breaks in the reactor coolant pressure boundary. The system is initiated by either high pressure in the drywell or low water level in the vessel. It is designed to operate independently of all other systems over the entire range of reactor coolant pressure. The high pressure core spray system pump motor is powered by a diesel generator which is independent of other emergency core cooling systems if offsite power is not available, and the system may also be used as a backup for the reactor core isolation cooling system for isolation cooling and the rod drop accident. Our review found that the design satisfies the applicable regulatory requirements given in Table 7.1 of the Standard Review Plan and, therefore, the design is acceptable except for some of the previously listed "repetitive" problems which are listed in subsections 7.1.3 and 7.2.3 of this report. We will pursue these concerns and will provide the resolution in a supplement to this report.

Low Pressure Core Spray System

The low pressure core spray system is a division 1 emergency core cooling system. It consists of one independent pump and the valves and piping to deliver cooling water to a spray sparger over the core. The system is actuated by conditions indicating that a breach exists in the reactor coolant pressure boundary, but water is delivered to the core only after reactor vessel pressure is reduced. This system provides the capability to cool the fuel by spraying water into the fuel channels. The low pressure core spray system functioning in conjunction with the automatic depressurization system is designed to maintain the fuel cladding below fragmentation temperature. Our review found that the design satisfies the applicable regulatory requirements given in Table 7-1 of the Standard Review Plan and, therefore, the design is acceptable with the exception of some of the previously listed "repetitive" problems which are listed in subsections 7.1.3 and 7.2.3 of this report. We are pursuing these issues and will provide the resolution in a supplement to this report.

Low Pressure Coolant Injection

The low pressure coolant injection system serves the following functions: (1) emergency core cooling by coolant injection to the core, (2) emergency core cooling by containment spray, and (3) residual heat removal.

Low pressure coolant injection is an operating mode of the residual heat removal system described in subsection 7.4.2 of this report, but it also acts as an engineered safety feature in conjunction with the other emergency core cooling systems. The low pressure coolant injection system uses the pump loops of the residual heat removal system to inject cooling water directly into the pressure vessel. The low pressure coolant injection system is actuated by conditions indicating a breach in the reactor coolant pressure boundary, but water is delivered to the core only after reactor vessel pressure is reduced. The low pressure coolant injection system operation is designed to provide the capability of core reflooding following a loss-of-coolant accident in time to maintain the fuel claol ing below final acceptance criteria limits.

Low pressure coolant injection system A is powered from division 1 and receives its initiation signals from the low pressure core spray system. Low pressure coolant injection system B and low pressure coolant injection system C are powered from division 2 and have their own shared initiation logic. Our review found that the design satisfies the applicable regulatory requirements given in Table 7-1 of the Standard Review Plan and, therefore, is acceptable with the exception of some of the previously listed "repetitive" problems which are listed in subsections 7.1.3 and 7.2.3 of this report, and the prohlem related to "Containment Spray" noted below. We will report on the resolution of these matters in a supplement to this report.

Containment Spray

The suppression pool spray was originally manually controlled with an interlock requiring that the injection valve be closed. We completed our review of the containment bypass leakage capability and as a result require automatic actuation of the spray system. We will review the electrical system which is provided for automatic actuation and report on this matter in a supplement to this report.

Heat Exchangers

Two residual heat removal system heat exchangers are provided. Heat exchanger 1A normaly receives cooling from service water pump 1WS01PC (division 2) or 1WSOPD (division 3) and is isolated by normally closed inlet and outlet valves powered from division 1. Heat exchanger 1B normally receives cooling from service water pump 1WS01PA (division 1) or 1WSP1PB (division 2) and is isolated by normally closed valves which are powered from division 2. The two service -r loops are cross-tied by a normally closed and locked-out valve which is powered from division 1. Because of this design and the fact that only two pumps 1WS01PB and

1WSO1PD are automatically sequenced onto the diesel generators, we informed the applicant that the limiting conditions for operation in several of the proposed technical specifications are unacceptable. We are pursuing the matter of technical specifications and will provide the resolution in the issuance of the technical specifications to be a part of the operating license.

Automatic Depressurization System

The automatic depressurization system is designed to provide a means for opening the safety/relief valves in sufficent time so that the low pressure coolant injection system or low pressure core spray system can serve to mitigate against a small break. The concept is similar to that provided on earlier plants; however, the implementation is greatly improved in the areas of separation and testing. Each automatic depressurization system valve is controlled by three solenoid operated air valves and an associated air accumulator system. Two solenoids and an accumulator are associated with division 1 and the third with a separate accumulator are associated with division 2. In each division, one set of logic and one solenoid is used to initiate the automatic depressurization system function. Either division can open an automatic depressurization system valve on a combination of high drywell pressure, low vessel water level, confirmation that a low pressure coolant injection system or low pressure core spray system pump is running, and a 120-second time delay. The third solenoid (in division 1) is used in the relief function and for manual depressurization. Each sensor and the timer can be tested during normal operation because the final logic is an "and" function (e.g., low level in channel A after 120 seconds and low level in channel D to initiate). Continuity lights are provided in parallel with each pair of logic relay contacts. If a solenoid burns out or logic power is lost, both lights extinguish and an alarm sounds. If a single relay closes its light goes out and the other light brightens. If both relays close, both lights extinguish and a different alarm sounds. Additional red and green lights and a plant process computer printout indicate the state to which each automatic depressurization system valve has been ordered by its logic. Each electrical penetration is protected by a 10-ampere fuse and fully coordinated with the 125 volt direct current bus protection.

Our review found that the automatic depressurization system satisfies the requirements of the applicable criteria in Table 7-1 of the Standard Review Plan and therefore the design is acceptable.

Primary Containment and Reactor Vessel Isolation Control System

The primary containment and reactor vessel isolation control system is designed to automatically initiate closure of isolation valves to close off all potential leakage paths for radioactive material to the environs. This action is taken upon indication of a potential breach in the nuclear system process barrier. The basic design is similar to that employed on previous plants such as Hatch 2; however, some of the initiation sensors and their signal processing equipment are different. The following subsections discuss specific concerns which have been encountered during our review.

Power Sources

The isolation system logic is powered from the reactor protection system motor generator sets and the valves are powered from division 1 and division 2 emergency buses or the reactor protection system motor generator sets. Because of the problems which have been discussed in subsection 7.2.3 of this report. We were concerned that the inboard and outboard isolation valves may not have adequate physical separation and electrical independence. The proposed resolution of this issue is presented in subsection 7.2.4 of this report.

Temperature Monitoring
Power Sources

Two redundant leak detection temperature monitoring systems are provided. One is for the inboard and one is for the outboard isolation valves. Each system has two channels which share calibration equipment. Each system is powered from a separate Class IE instrument bus. We were concerned that the power supply from both bus 1A and 1B in cabinet 1H13-P642 may compromise the electrical independence and physical separation unless power supply 1E31-K600 is qualified as an isolation device. We will pursue this issue and will provide the resolution in a supplement to this report.

Calibration

The design of this system presents a potential improvement over previous designs in that thermocouples are used instead of temperature switches (which were difficult to calibrate). The new equipment is designed to permit each thermocouple to be compared with every other thermocouple measuring the same parameter, to have its converter standardized, and to inject test signals to the trip bistable for setpoint calibration. We find this aspect of the system design to be acceptable.

Main Steam Line Isolation Valve Testing

The design of the isolation system and its integration into the overall plant design requires that the main steam line isolation valve close in not less than three seconds nor more than five seconds. The testing scheme which has been proposed is designed to demonstrate valve operability. This scheme is adequate for demonstrating acceptable (i.e., $3<t<5$ seconds) closing time limits in the technical specifications.

Feedwater Isolation Valve Control

The feedwater pumps are driven by steam turbines. The feedwater line is isolated on both sides of the drywell penetration by check valves and a motor-operated isolation valve upstream of the outboard check valve. This motor-operated valve has a 40-second stroke time and is manually controlled.

Other boiling water reactor designs such as Hatch 2 with this arrangement have been found to be acceptable because: (1) the design satisfies the single failure criterion, and (2) the design permits the use of the steam-driven feedwater pumps to supplement the reactor core isolation cooling system when the reactor becomes isolated. On this basis we also find the Zimmer feedwater isolation valve control to be acceptable.

Standby Gas Treatment System

The standby gas treatment system consists of two identical filter trains, two identical sets of recirculation fans, and interconnecting ductwork for the station. Either train alone is designed to be capable of exchanging the total reactor building volume once in a 24-hour period. Each filter train contains electric heaters, a prefilter, high efficiency particulate filters (water-resistant and fire-resistant), and iodine filter (high ignition temperature), instrumentation to measure flow, and a pair of recirculation fans connected to the reactor building ventilation supply and exhaust duct system. The system is designed to maintain a slightly negative internal building pressure and process all gaseous effluent prior to its discharge from the standby gas treatment vent stack. All equipment is connected to the essential buses and is designed to start either automatically or manually from the control room.

Subject only to the completion of the seismic qualification program as stated in subsection 3.10 of this report, we find the design to be acceptable.

Control Room Heating, Ventilation and Air Conditioning System

The control room heating, ventilation and air conditioning system is designed to maintain proper air quality for personnel comfort and safety under all station conditions. All air distribution systems are designed so that airflow is directed from areas of lesser potential contamination to areas of progressively greater potential contamination. The system supplies filtered and conditioned air to the control room, auxiliary electrical equipment room, computer room and miscellaneous offices. The system consists of two 100 percent air conditioning systems, one of which is normally operating and one of which is normally on standby.

The control room heating, ventilation and air conditioning system has three air intakes. Two of these intakes are designed to provide sufficient air for control room pressurization and are isolated by two dampers in series. The third intake is designed to provide sufficient air to both trains for purging smoke and appears to be isolated by a single damper. We determined during the review that the outside air intakes are also isolated by a third set of dampers at the inlet to the filter trains. Therefore, the design of the three air intakes damper arrangement does satisfy the single failure criterion and is, therefore, acceptable.

The design of the initiation logic was modified in the Final Safety Analysis Report Revision 29, to include redundant ammonia detectors. We requested that this detection system be described in the Final Safety Analysis Report as required by Section 7.6 of the Standard Format for Safety Analysis Reports. This information was supplied in Revision 33, except for the information on seismic and environmental qualification of the detection system. The environmental information was provided in Amendment 34. Therefore, subject only to the completion of the seismic qualification program as stated in subsection 3.10 of this report, we find the design of this system to be acceptable.

Station Service Water System

Service water is supplied from service water pump as shown in the Final Safety Analysis Report, Figure 9.2-1. Four service water pumps are provided for the plant. The system is designed to remove heat from various equipment located within the plant. The system consists of two subsystems, one of which is essential and the other nonessential. The essential subsystem is an engineered safety feature providing service water to the residual heat removal system, reactor building closed cooling water system and diesel generator heat exchangers. The nonessential subsystem furnishes service water to the turbine building, traveling screens and other plant buildings and is automatically isolated from the essential system upon low pressure in the essential system. This isolation system satisfies the requirements for Class IE service.

Our review of the service water system logic indicated that the diesel generator load sequences have a significant impact on this system. Our review include schematics and logic drawings for the sequencers and indicates that the design of the sequencers is consistent with the information for the controlled systems and that the sequencers are similar to those provided and approved by us for Calvert Cliffs Units 1 and 2. Therefore, we find the design of this system to be acceptable.

Reactor Building Closed Cooling Water System

The reactor building closed cooling water system consists of four pumps, three heat exchangers, and control and instrumentation to provide adequate cooling for the essential reactor auxiliary systems. This equipment is provided to ensure adequate cooling capacity during both normal and accident conditions including the design basis loss-of-coolant accident. Two of the four pumps and two of the three heat exchangers are provided for the combined accident situation of a loss-of-coolant accident and a loss of offsite power. It is similar to previously reviewed plants that we have approved such as Clinton. As a result of our review we find the design to be acceptable subject only to the completion of the seismic qualification program as stated in subsection 3.10 of this report.

Combustible Gas Control System

The combustible gas control system consists of a hydrogen recombiner and the equipment required during the accident mode of the containment purge system. The recombiner is designed to be manually started to prevent the hydrogen-oxygen level within the primary containment from exceeding the flammability limit in the event of a loss-of-coolant accident. To provide a high degree of reliability, redundant valves, fans and electric heaters are provided. Redundant components, controls and instrumentation are powered from separate Class IE buses.

The process gas stream is cooled by the after-cooler prior to injection into the containment. The instrumentation which monitors the temperature of this cooler is not redundant and we expressed the concern that an undetected failure could result in returning heated gas which could increase containment pressure beyond design conditions. The response stated that sufficient Class IE instrumentation is available to monitor the secondary (cooling) side flows to demonstrate proper operation. Our review verified this response and we find it acceptable.

The applicant described the environmental and seismic qualification program test procedures and results by incorporation of General Electric Company Topical Report NEDE-23597, "Qualification Report - Flammability Control System," into the material submitted for review. Our review of the subject report reveals that it is possible for contaminated steam to condense in the blower casing. The manufacturer recommends that the blower be periodically shutdown and drained. We questioned the safety of such operation and requested additional information on shielding of personnel access to the drain valves and the disposal of the contaminated effluent water and gas. General Electric Company responded by stating that the procedure could be accomplished from the control panel by shutting down the blower and opening the recirculation line. This action permits the blower to drain by gravity to the suppression pool via the return line. This design is, therefore, acceptable.

7.3.4 Evaluation

The engineered safety feature systems include the instrumentation and controls used to detect a plant condition requiring operation of an engineered safety feature system, to initiate action of the engineered safety feature, and to control its operation. The scope of review included single line diagrams and schematic diagrams and descriptive information for the engineered safety feature and for those auxiliary supporting systems that are essential to the operation of the engineered safety features systems. The review included the Final Safety

Analysis Report design criteria and design bases for the engineered safety feature and the instrumentation and controls of auxiliary supporting systems, and the analysis of the adequacy of those criteria and bases. The review also included the Final Safety Analysis Report analyses of the manner in which the design of the engineered safety feature and auxiliary supporting systems conforms to the design criteria.

The basis for acceptance in our review is conformance of the designs, design criteria and design bases for the engineered safety features actuation systems and necessary auxiliary supporting systems to the Commission's regulations as set forth in the General Design Criteria, applicable regulatory guides, branch technical positions and industry standards. These are listed in Table 7-1 of the Standard Review Plan.

With the exception of the specific issues presented in subsections 7.1.3, 7.2.3 and 7.3.3 we conclude that the implementation of the engineered safety features systems conforms to all applicable regulations, guides, branch technical positions and industry standards and are, therefore, acceptable.

7.4 Systems Required for Safe Shutdown

7.4.1 General Discussion

The systems in the Final Safety Analysis Report, Section 7.4, are those instrumentation and control systems associated with parts of the nuclear steam supply system used to achieve and maintain a safe shutdown condition of the plant. We review this information as detailed in the Standard Review Plan, and also determines the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format for Safety Analysis Reports.

7.4.2 General Findings

The following systems have been identified as required for safe shutdown: (1) reactor core isolation cooling system, (2) standby liquid control system , (3) reactor shutdown cooling system, and (4) remote shutdown system (see subsection 7.5 of this report).

Our review of the systems identified as required for safe shutdown included the applicable design criteria design, bases, and descriptions which are identified in subsection 7.1.2 of this report. With the exception of some of the "repetitive" issues which are identified in subsections 7.1.3, 7.2.3 and 7.3.3 of this report, we conclude that these systems are acceptable.

7.4.3 Specific Findings

Reactor Core Isolation Cooling System

The reactor core isolation cooling system was classified as a system required for safe shutdown. We did not agree with this classification because the Operational Analysis Final Safety Analysis Report, Appendix B also states that it is required to mitigate the consequences of a control rod drop with the failure of the high pressure core spray system. The resolution of this matter is given in subsection 7.3 of this report.

Residual Heat Removal System

The equipment which is used for residual heat removal is composed of the same pumping equipment discussed in subsections 7.3 of this report. This equipment is augmented by a single suction line to one of the two reactor recirculation loops, steam pressure reducing valves for use of the emergency core cooling system heat exchangers to condense steam when the main condenser is not in use, a pressure vessel head spray to help reduce vessel pressure, and the associated instrumentation, valve interlocks and controls. The suction line is isolated by two valves in series. The inside drywell valve is powered from division 2 alternating current and the outside valve is powered from division 1 direct current. These valves satisfy Branch Technical Position Electrical Instrumentation and Control Systems Branch-3 (Standard Review Plan Appendix 7A) except that diverse pressure interlocks are not provided. However, inputs for interlocks from the leakage detection system, excess flow trips, and containment isolation signals which are diverse are provided. The entire system of pumps is backed up by the reactor core isolation cooling system for the early stages of reactor cooldown. In concept the system is similar to previous operating boiling water reactors and satisfies the applicable criteria of Table 7-1 of the Standard Review Plan and, when the outstanding issues which are identified in subsections 7.1.3 and 7.3.3 of this report are resolved, will be acceptable to us subject only to the completion of the seismic qualification program as stated in subsection 3.10 of this report.

Standby Liquid Control System

The standby liquid control system is the same as that presented in past applications; except that the design has been improved by powering the heaters from two independent onsite sources. We find this design acceptable.

7.4.4 Evaluation

The review of systems required for safe shutdown included the sensors, initiating circuitry, logic elements, interlocks, redundancy features, actuated devices and auxiliaries that provide the instrumentation and control functions that prevent the reactor returning to criticality and provide means for adequate residual heat removal from the core, containment and other vital components and systems. The scope of review of systems required for safe shutdown for the plant included single line diagrams and schematic diagrams and descriptive information for these systems and for auxiliary systems essential for their operation. The review included the design criteria, design bases and analyses. The review also included the Final Safety Analysis Report analyses of the manner in which the design of these systems and their auxiliary supporting systems conforms to the proposed design criteria.

The basis for acceptance in our review is conformance of the design, design criteria and design bases for systems required for safe shutdown and essential supporting auxiliaries to the Commission's regulations as set forth in the General Design Criteria, applicable regulatory guides, branch technical positions, and industry standards. These are listed in Table 7-1 of the Standard Review Plan. With regard to Electrical Instrumentation and Control System Branch-3, we noted that this position was not applied to the construction permit review and that diverse interlocks other than pressure are provided.

We conclude that the design of systems required for safe shutdown conform to the applicable regulations, guides, technical positions and industry standards and are acceptable subject to resolution of the outstanding issue regarding the emergency core cooling system, as defined in subsections 7.1.3 and 7.3.3 of this report.

7.5 Safety-Related Display Instrumentation

7.5.1 General Discussion

Information presented in Final Safety Analysis Report, Section 7.5, describes the design of safety-related display instrumentation required for safe functioning of the plant during operating and accident conditions. We review this information as detailed in the Standard Review Plan, and also determines the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format for Safety Analysis Reports.

7.5.2 General Findings

The systems which provide input to the safety-related displays satisfy the applicable requirements of Table 7-1 in the Standard Review Plan. However, many of these systems suffer from some of the "repetitive" problems which are dicussed in subsections 7.1.3, 7.2.3 and 7.3.3 of this report.

7.5.3 Specific Findings

Tables of System Variables and Components to be Indicated and Recorded

The review of the instrumentation identified that the Final Safety Analysis Report does not provide a table of variables and components to be indicated and recorded. The accuracies and ranges of most instruments have not been indicated within the text of the Final Safety Analysis Report; however, some of this information is available in the indiv..ual system drawings. The following systems are identified in Section 7.5 of the Final Safety Analysis Report.

Reactor Water Level

(1) Two wide-range instruments monitor reactor level from the top of the active fuel to the top of the feedwater control range (366.75 inches to 576.75 inches above cold vessel zero). The water level transmitters are most accurate at operating temperature and pressure and are only required for upset operations under these conditions. Each channel is powered from a separate Class IE source and both signals are recorded on separate recorders. The review found that this instrumentation is typical of previous boiling water reactors approved by us (e.g., Hatch 1) and is, therefore, acceptable to us.

(2) Two independent shroud water level instruments are provided. One is indicated and one is recorded. The signals represent vessel level from near the bottom of the active fuel to over the top of the active fuel (202.5 inches to 402.5 inches above cold vessel zero). The transmitters are uncompensated for variation in density and are calibrated to be most accurate at atmospheric conditions. These conditions are those expected during the post loss-of-coolant accident injection recovery. The review found that this instrumentation is typical of previous boiling water reactors and is, therefore, acceptable to us.

Reactor Pressure

Two reactor pressure signals are provided by two independent pressure transmitters and are recorded on the second pen of the separate wide range water level recorders. The range is 0 pounds per square inch gauge to 1500 pounds per square inch gauge. The accuracy is unknown to us but the Final Safety Analysis Report indicates it is the same as Hatch 1. We are pursuing the matters of the range and accuracy of all sensors which are required for safety (see subsection 7.1.3 of this report regarding safety system set points) and will provide the resolution in the final technical specifications to be a part of the operating license.

Drywell Pressure Monitoring

The Final Safety Analysis Report Section 7.5.1.4.2.4(a), indicates two drywell pressure signals are provided by two independent pressure transmitters and indicated in the control room. This indication is also available at the remote shutdown panels but neither channel is recorded. This design satisfies the construction permit commitments and is acceptable.

One area of concern was identified by us during discussions with the applicant for the resolution of the suppression pool spray design problem (see subsection 7.3 of this report). The resolution which has been proposed by the applicant requires monitoring of drywell pressures up to at least 35 pounds per square inch gauge. However, the present instrumentation for pressure monitoring is limited to a range of 0 pounds per square inch gauge to 25 pounds per square inch gauge (Final Safety Analysis Report Table 7.3-5). The applicant stated that two new sensors of suitable range and accuracy will be added to the present design. We are pursuing this matter and will provide the resolution in the final technical specifications to be a part of the operating license.

Drywell and Suppression Pool Temperature

The drywell atmosphere and suppression pool temperature are each monitored by four sensors. Two sensors monitor the temperature above the suppression pool surface and two sensors monitor the pool water temperature. One sensor in each pair is indicated and the other is recorded.

Emergency Core Cooling

The Final Safety Analysis Report pages 7.5-3 and 7.5-4, list the information which is furnished to the control room operator to permit assessment of reactor shutdown, isolation and availability of emergency core cooling following the postulated accidents.

We evaluated this information. The results of our review indicate that:

(1) The information contained in a.1 and a.2 on the Final Safety Analysis Report page 7.5-3 contradicts the information contained in the applicant's description of the reactor manual control system, which indicates that the power source for item a.1 is instrument bus 1B and item a.2 is powered from the non-Class IE reactor protection system motor generator sets. This design may not satisfy the single failure criterion because a failure in division 2 coincident with a loss of offsite power would result in a loss of all rod insertion indication.

(2) The annunciators and process computer are not Class IE.

As a result of our findings, we concluded that provisions for monitoring reactor shutdown were not acceptable. The applicant was informed that equipment which has not been designed and qualified to the requirements for Class IE service cannot be relied upon as a basis for the protection of public health and safety.

The applicant has proposed that the rod position information system power source be transferred to the computer uninterruptable power supply in order to resolve our concern. We find that this supply and the rod position information system can form a suitable substitute for the rod bottom information which is processed via the reactor manual control system and the division 2 alternating current instrument bus (bus 1B) if it is properly implemented. Furthermore, the use of the plant process computer for post accident monitoring was found acceptable in the construction permit Safety Evaluation Report.

Therefore, we have requested that the applicant provide revised drawings, which show these changes, for our review. We will pursue these issues with the applicant and provide the resolution in a supplement to this report.

Suppression Pool Level

In our review of the resolution of the question of passive failures following a loss-of-coolant accident, we expressed concern about the adequacy of the leakage detection systems for the emergency core cooling system pumps and valves.

Because the Class IE area temperature monitors which initiate containment isolation have setpoints which are higher than the temperature that the containment effluent reaches after the first 24 hours (approximately), and because the reactor building and emergency core cooling system sump pump leakage instrumentation systems are not Class IE, we concluded that the suppression pool level indication system is the only reliable method for detecting leakage out of the emergency core cooling system after a loss-of-coolant accident. Therefore, we requested additional information be provided on the suppression pool level indication system. The response indicates that a 50 gallons per minute leak can be detected in one hour and would be indicated as a one-inch drop in suppression pool level. Evaluation of this system is provided in subsection 6.3.2 of this report.

Seismic Qualification of Indicators and Recorders

Some of the indicators and recorders which are used in the design for post-accident monitoring are qualified for proper operation after a seismic event. This design feature is in accordance with the applicant's Preliminary Safety Analysis Report commitment to satisfy Institute of Electrical and Electronic Engineers Standard 279-1971. In view of this and other improvements (in terms of redundancy and physical separation) which the design provides in post-accident monitoring, our review found that this aspect of the design is acceptable.

Display of System Level Automatic Bypass Indication

The construction permit issuance preceded the issuance of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indications For Nuclear Power Plant Safety Systems." Our review of the information presented in Final Safety Analysis Report, Section 7.8, and some of the schematic diagrams of Final Safety Analysis Report Sections 7.2 and 7.3 indicate that the design satisfies the objectives of the Regulatory Guide 1.47.

Provisions for Remote Shutdown

The major provisions of the design which are included to meet the requirements of General Design Criterion 19 for shutdown from outside of the control room are: (1) the provision of remote shutdown panels in Class IE switchgear rooms A and B, and (2) the provision of the capability to initiate reactor scram via panel 1C71-P001 in the auxiliary equipment room.

The provision of two separate and independent remote shutdown panels presents a significant improvement over previously licensed designs and, conceptually, is a desirable design feature. We find that this aspect of the design is acceptable.

7.5.4 Evaluation

The safety-related display instrumentation provides the operator with information on the status of the plant to allow manual safety actions to be performed whenever necessary and to assess the condition of the plant during post-accident conditions. The scope of review of safety-related display instrumentation included description of system variables and component states to be indicated, functional control diagrams, electrical and physical layout drawings, and descriptive informaton. The review included the design criteria and design bases, including that for indication of bypassed or inoperable safety-related systems. The review also included the Final Safety Analysis Report analyses of the manner in which the design of safety-related display instrumentation conforms to the design criteria.

The basis for acceptance in our review has been conformance of the designs for safety-related display instrumentation to the Commission's regulations as set forth in the General Design Criteria, applicable regulatory guides, branch technical positions and industry standards. These are listed in Table 7-1 of the Standard Review Plan. Resolution of the matters described above will be reported in a supplement to this report.

7.6 All Other Instrumentation Systems Required for Safety

7.6.1 General Discussion

The group of instrumentation systems described in Final Safety Analysis Report, Section 7.6, are those required for safety that are not identified as part of the reactor protection system, engineered safety features systems, safety-related display instrumentation systems, or systems required for safe shutdown. We review this information as detailed in the Standard Review Plan, and also determines the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format for Safety Analysis Reports.

7.6.2 General Findings

The instrumentation systems required for safety in the design satisfy the applicable requirements of Table 7-1 in the Standard Review Plan. However, many of these systems are subject some of the "repetitive" problems which are discussed in subsections 7.1.3, 7.2.3 and 7.3.3 of this report. As a result of these "repetitive" problems, we have not been able to complete our review. However, we will pursue these issues and will provide the resolution of these issues in a supplement to this report.

7.6.3 Specific Findings

Refueling Interlocks

The refueling interlock system is similar to that of the other boiling water plants such as Hatch 2 except that the reactor manual control system is different from all previously licensed facilities. Because the reactor manual control system has been found acceptable (see Section 7.7.3 of this report), we find the design of these interlocks to be acceptable too.

Reactor Vessel Instrumentation

Our review determined that the reactor vessel instrumentation is the same as Hatch 1 with the exception of changes in range for this design which increase instrument sensitivity, improvements in separation of instrument lines, improved routing of instrument lines to give better drainage, improved recording techniques, and reduction of unnecessary temperature compensation. Based on this review, we find this aspect of the design acceptable.

Leak Detection System

Our review of the leak detection system indicates that this system has the same design basis as previously licensed boiling water reactors (e.g., Hatch 2). As a result, the subsystems which have been clearly identified as required for safety (those which initiate isolation of pressure vessel or drywell or initiate ventilation system isolation or the standby gas treatment system) were reviewed and evaluated with other engineered safety features systems. The result of this aspect of our review is presented in subsection 7.3 of this report. The basis for accepting the non-Class IE parts of this system is discussed in subsection 5.2.5 of this report.

Reactor Water Cleanup System

The reactor water cleanup system recirculates a portion of reactor coolant through a filter demineralizer to remove particulate and dissolved impurities from the reactor system under controlled conditions.

We reviewed the instrumentation and control to assure that the inlet line would be automatically isolated in the event of a loss of reactor coolant and upon initiation of the standby liquid control system. As a result of our review we determined that the isolation of this system is initiated by the equipment described in subsections 7.3 and 7.4 of this report and is, therefore, acceptable.

Neutron Monitoring System

Our review found that the neutron monitoring system for the Zimmer station is similar to all previous designs, which have been approved by us since Dresden Unit 2, except in the following areas.

Source Range Equipment

The source range equipment is similar to previous designs as noted above. However, the equipment is housed in an interconnected series of instrumentation racks which houses all of the other nuclear instrumentation, the rod block monitor, and the process radiation monitoring equipment which initiates reactor building and fuel pool isolation. Furthermore, the relays which provide trip and alarm isolation for the source range, intermediate range and process radiation monitoring equipment are housed in these racks. Our physical examination of these racks disclosed that the safety inputs and isolated alarm relay outputs are bundled together in each of the four auxiliary trip units. The applicant was informed that this appears to present an unacceptable violation of their separation criteria, and we requested the wiring diagrams for these units so that we could further evaluate the impact on public health and safety. We will pursue this matter and will provide the resolution in a supplement to this report.

Intermediate Range Monitor

As mentioned in the preceding subsection, we have not completed the review of the nuclear instrumentation system. We will provide the results of our review in a supplement to this report.

Rod Block Monitor

Our review of the rod block monitor identified the following concerns: (1) the four flow monitors are interconnected by armored cable and shielded cables. In addition, there are open spaces around the cables which penetrate fire barriers between redundant channels; (2) both rod block monitor channels are connected by data buses which are enclosed in a metal shield and run along the top of the cabinet; (3) the rod block monitor is a modified design and contains multiplexing circuitry which interfaces with the new reactor manual control system; and (4) the rod block monitor is assumed to function in the safety analyses of the Final Safety Analysis Report, Chapter 15. The Final Safety Analysis Report states that, "The RBM for Zimmer is designed to the same criteria as for Duane Arnold, which does not include criteria for a reactor protection system."

The applicant was informed that item (1) noted above is unacceptable and the applicant offered to provide suitable cable penetrations of these barriers. We advised that no action should be taken until the other concerns were resolved. We will provide the resolution of this item in a supplement to this report.

We informed the applicant that item (2) may not be acceptable from a fire protection standpoint and further evaluation will be performed as a part of the fire review. The resolution of this concern will be provided in the fire protection safety evaluation report (see subsection 9.5 of this report).

We required that, because of the new circuitry which is involved in interfacing the rod block monitor with the new reactor manual control system, the Final Safety Analysis Report Volume 11a be augmented with the schematic, logic and wiring diagrams of all of the nuclear instrumentation and radiation monitoring panels. These diagrams should be of sufficient detail to show the interfaces between the radiation monitoring, source range monitor, intermediate range monitor, average power range monitor and rod block monitor, and all inputs and outputs (including bypass switches). We will review these drawings and will provide the resolution of item (3) in a supplement to this report.

The applicant was informed that, "It is the staff's position that the RBM is a protection system and must be designed, fabricated, installed, tested and subjected to all of the design criteria which are applicable to a reactor trip system. Revise the FSAR to reflect the importance of the RBM in accordance with the requirements of Section 7.2 of the Standard Format." The applicant also was informed that our position is supported by the fact that after its 109th meeting, May 8-10, 1969, the Advisory Committee on Reactor Safeguards indicated its concern "that the rod block monitor system can perform an important safety as well as operational function and that incorporation of such a system, or its equivalent, is necessary and the system which performs this function should be built to meet appropriate protection system criteria. The criteria to be used for each system should be established on a basis acceptable to the regulatory staff."

In addition, the applicant was informed that the wiring of the rod block monitor bypass switch does not satisfy the separation criteria. Therefore, we requested the applicant to identify and justify any and all exceptions to our position and the separation criteria. The response was incomplete. We will pursue this issue and provide the resolution in a supplement to this report.

Local Power Range Monitor

For the reasons given in the preceding subsections, we have, concerns regarding the design and implementation of the nuclear instrumentation in general.

Because the local power range monitors provide inputs to the average power range monitors and the rod block monitor along with nonsafety inputs to annunciators and the plant process computer, our primary concern is physical separation, electrical independence and seismic qualification.

When we have completed the review of the requested drawings, the seismic qualification and the qualification of the isolation devices (see subsection 7.1.3 of this report), we will provide the results of our review in a supplement to this report.

Class IE Heat Removal Systems

The Class IE heat removal systems which have been identified by the applicant are: (1) diesel generator ventilation system, (2) emergency core cooling system equipment area cooling system, (3) service water pump cooling system, and (4) switchgear heat removal system.

Our audit of these system indicates that the instrumentation and controls are designed and implemented to Class IE requirements and the systems themselves are redundant and physically separated and electrically independent of their redundant counterparts. Therefore, we find these systems to be acceptable.

Recirculation Pump Trip System

The Final Safety Analysis Report states, "The recirculation pump trip system includes the sensors, logic circuitry, load drivers, switches and circuit breakers that cause main power to be disconnected from both recirculation pumps upon closure signals from the turbine stop valves or turbine control valve in the event of a turbine trip or generator load rejection." Final Safety Analysis Report Figure H.A-2 also indicates that reactor vessel low level or high reactor pressure will also initiate a reactor pump trip. The latter trips are part of the anticipated transients without scram protection.

The applicant's Final Safety Analysis Report states, "The recirculation trip system is designed to aid the reactor protection system in protecting the integrity of the fuel barrier. Turbine stop valve closure or turbine control valve fast closure will initiate a scram and concurrent recirculation trip in order to keep the core within the thermal-hydraulic safety limits during operational transients."

As a result of this statement, we find that this system is required for safety. However, this system receives nonsafety signals from the turbine building and assumes functioning of the non-Class IE relief systems. The system logic is powered from the reactor protection system power supplies, the breaker trip coils are powered from two divisions of Class IE batteries, and the motor power is received from non-Class IE (6.9 kilovolt) buses. As a result of this mix of safety and nonsafety inputs, we were concerned about the possibility of violation of the criteria for physical separation and electrical independence. However, we did an independent analysis of the consequences of a turbine trip without dependence on the non-Class IE relief system and find that the radiological consequences in the absence of a seismic event are a small fraction of the loss-of-coolant accident consequences and therefore are acceptable. This system is not required to function during a seismic event because fuel damage during a seismic event is acceptable.

Rod Sequence Control System

The rod sequence control system is a subsystem of the reactor manual control system discussed in Section 7.7 of this report.

This system performs required safety functions since the purpose of the rod sequence control system is to reduce the consequences of the postulated rod drop accident to an acceptable level by restricting the patterns of control rods that can be established to predetermined sets and to prevent a control rod withdrawal accident at low power.

The rod sequence control system will operate from the same instrument bus as the rod position information system, the subsystem of the reactor manual control system that is the primary data source for the rod sequence control system. The rod sequence control system is designed so that it will apply rod movement inhibits to the rod drive control system in the event of loss of input power.

In addition, the applicant was informed that this system cannot be relied upon to prevent a control rod withdrawal accident. The applicant has accepted this position and demonstrated to our satisfaction that the average power range monitors and intermediate range monitors provide adequate protection.

We reviewed the functional requirements for this system and found the design acceptable.

Main Steam Line Isolation Valve Leakage Control System

The main steam line isolation valve leakage control system is designed to minimize the release of fission products which could bypass the standby gas treatment system after a loss-of-coolant accident. This is accomplished by directing the leakage through the closed main steam line isolation valves to a bleed line into the reactor building.

The flow is effected by a blower which directs the leakage into the reactor building and eventually through the standby gas treatment system. Thus, leakages through the main steam isolation valves will be processed by the standby gas treatment system prior to release to the atmosphere.

The system is described in Section 9.3 of the applicant's Final Safety Analysis Report. The instrumentation and controls are described in Final Safety Analysis Report Section 7.3.1.1.10. As a result of our review of this system, we find that the design satisfies the recommendations of Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," but, that in addition to inconsistencies in the drawings, we have identified a single failure which could lead to possible failure of the system during testing or operation. The proposed annual test frequency also appeared to be insufficient. The applicant was informed of our concerns and we will provide the resolution of these issues and the results of the seismic qualification program in a supplement to this report.

7.6.4 Evaluation

The other instrumentation systems required for safety consist of safety-related instrumentation systems not identified as part of the reactor protection system, engineered safety features systems, safety-related display instrumentation systems, or systems required for safe shutdown. They are, to a large extent, groups of interlocks intended to protect other vital systems from potentially damaging transients during normal operating and accident conditions or provide inputs to the safety systems.

Our review encompassed the sensors, initiating units, logic, bypasses, interlocks, redundancy and diversity of features, actuated devices, testing provisions and equipment qualifications. The review included single line diagrams, schematic diagrams and descriptive information on this group of systems and supporting auxiliaries that are essential for their operation. The review included the design criteria and design bases and analyses of the manner in which the design of these systems conforms to the design criteria.

The basis for acceptance in our review of these systems has been conformance of the design criteria and design bases to the Commission's regulations as set forth in the General Design Criteria, applicable regulatory guides, branch technical positions and industry standards listed in Table 7-1 of the Standard Review Plan. Resolution of the matters discussed above will be reported in a supplement to this report.

7.7 Control Systems Not Required For Safety

7.7.1 General Discussion

The areas reviewed in the Final Safety Analysis Report Section 7.7 include such control systems as the primary system pressure, water level controls, and main turbine controls. We review this information as detailed in the Standard Review Plan, and also determine the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format for Safety Analysis Reports.

7.7.2 General Findings

The applicant identified the following systems as control systems which are not required for safety: (1) reactor manual control system, (2) recirculation flow control system, (3) feedwater control system, (4) pressure regulator and turbine controls, and (5) radwaste system.

7.7.3 Specific Findings

Reactor Manual Control System

Although the reactor manual control system is not a Class IE system, we classified this as a system which is required for safety to the extent that the rod sequence control system operates from the same instrument bus as the rod position information system (a reactor manual control subsystem) (subsection 7.6.3 of this report) and requires proper operation of the reactor manual control system to perform its safety function.

Reactor Manual Control System

General Electric Company introduced a new design for its reactor manual control system. The reactor manual control system design submitted with recent BWR-4 and BWR-5 plants has the new self-monitoring, solid-state circuitry.

The motivations for the new design are improved operability and reliability through the self-monitoring feature, reduced field wiring and reduced installation problems. Since the new design uses multiplexed information transfer which is new to nuclear power plant control and indication system designs, we implemented, on a generic basis, a program to perform an in-depth evaluation.

The reactor manual control system consists primarily of two major subsystems: the rod drive control system and the rod position information system. Both of these systems use solid-state components throughout with multiplexed information transfer (Multiplexed information transfer is the sequential transfer of binary data words from one point to another using a single transmission line).

The rod drive control system moves the control rods (137) individually in six-inch increments called "notches" in or out of the reactor, under the direction of the plant operator, for the purpose of changing reactivity (and thus power level).

The rod position information system is a companion of the rod drive control system and senses the vertical position of control rods. The information representing the vertical rod position is distributed to the rod and detector display and the four-rod display for use by the plant operator. The four-rod display presents to the plant operator the vertical position of each of the four rods in the four-rod group which contains the currently selected rod. The rod and detector display displays a total of six parameters concerning each control rod. They are:

1. Full-in — lighted if the rod is fully inserted into the core.

2. Full-out — lighted if the rod is fully withdrawn from the core.

3. Drifting — lighted if a rod has moved more than three inches when it should not have moved.

4. Selected — lighted if the rod has been selected by the operator for movement.

5. Accumulator — lighted if the nitrogen pressure or water level is abnormal in the hydraulic control unit associated with the rod.

6. Scram Valves — lighted if both scram valves in the hydraulic control unit have moved to the open position.

Also dispersed throughout the rod and detector display are local power range monitor display lights:

1. Up-Scale — lighted if neutron level at the chamber is above the set point of the up-scale trip on the local power range monitor card associated with the chamber.

2. Down-Scale — lighted if neutron level as seen by local power range monitor is abnormally low.

As noted above, the rod position information system also provides input to the rod sequence control system.

We conclude that this system is acceptable on the basis discussed in subsection 7.7.4 below.

Recirculation Flow Control System

In our Safety Evaluation Report for the Zimmer nuclear station dated February 18, 1972, we reported the results of our evaluation of the control systems which were not required for safety, and identified several items as appropriate subjects of indepth evaluations during the operating license stage review. The recirculation flow control system is different from that provided in previously licensed boiling water reactors (BWR/4) and is described in subsection 4.4 of this report. The objective of the recirculation flow control system is to control reactor power level, over a limited range, by controlling the flow rate of the reactor recirculation water. The recirculation flow control system consists of the electrical circuitry, switches, indicators, motors, and alarm devices provided for operation manipulation of the recirculation flow control valves and the low frequency motor-generator set and the surveillance of associated equipment. Recirculation flow control is either by manual operation or automatic if the power level is above 65 percent of rated, when the plant is operating on a rod pattern where rated power is produced with rated recirculation flow. During periods of low power level such as plant startup and shutdown, the recirculation pump and motor will be powered by the low frequency motor generator set and will operate at approximately 25 percent rated full load speed.

In addition to the fact that the recirculation flow control system is an intrinsic part of the recirculation pump trip system (see Subsection 7.6 of this report), we expressed concern that the valve interlocks which are a part of the reactor flow control system may be required to prevent erosion damage to the pressure vessel boundary (e.g., recirculation flow control body). We requested verification that these interlocks are not required for safety. Additional design information was provided in the Final Safety Analysis Report, Appendix H. Our review of this material has been completed and we find that the interlocks are not required for safety and are, therefore, acceptable.

Feedwater Control System

The feedwater control system is typical of previous boiling water reactor (e.g., Hatch 2). For the reasons presented in section 6 of this report, however, we determined that the feedwater high level trip ("level 8 trip") is one of two non-Class IE control systems which are required to provide adequate margin during certain transients. In one transient, the excess feedwater transient, this level trip equipment may also cause the transient. Accordingly we requested that the applicant demonstrate the adequacy of the electrical isolation between the three level sensors and their trip criteria such that a single failure will not cause an excess feedwater transient and prevent the level 8 trip from functioning. We will continue to pursue this issue with the applicant and will provide the resolution in a supplement to this report.

Pressure Regulator and Turbine Controls

As with the feedwater control system this instrumentation is typical of all previous boiling water reactors. One of the subsystems in this class of controls is the turbine bypass valve control. This subsystem is the other non-Class IE system which we determined is important to safety. Accordingly, both the feedwater control and the turbine bypass control systems will be subjected to a technical specification requirement for periodic surveillance. The applicant has been informed that he must establish a suitable surveillance frequency and provide the basis for our review. He has also been informed that he must describe how such tests will be conducted. This information will be evaluated by us and the resolution will be provided in a supplement to this report.

7.7.4 Evaluation

We reviewed the controls for systems not required for safety, as noted above, to determine the effects of failures or malfunctions of these controls on the reactor protection system and other plant safety-related systems. We conclude that, with the exception of the items noted in subsection 7.7.3 above, failures or malfunctions of these controls would not be expected to degrade the capabilities of plant safety systems or lead to plant conditions more severe than those for which safety systems are designed, and are, therefore, acceptable.

8.0 ELECTRIC POWER

8.1 Introduction

8.1.1 General Discussion

Section 8.1 of the Final Safety Analysis Report provides a brief description of the utility grid, its interconnection to other grids, and the onsite electric system in general terms. We review this information as detailed in the Standard Review Plan, and also determines the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format for Safety Analysis Reports.

8.1.2 General Findings

We reviewed the applicable design criteria listed in the Final Safety Analysis Report, Tables 7.1-3 thru 7.1-9, and the design bases and descriptions of the electrical systems and conclude that they are in agreement with Table 8-1 of the Standard Review Plan except for the deviation from Regulatory Guide 1.75, "Physical Independence of Electrical System."

The deviation from Regulatory Guide 1.75 involves the use of fault current operated devices in the Zimmer plant as isolation devices for the purpose of satisfying Institute of Electrical and Electronic Engineers Standard 279-1971.

However, we have conducted some independent short circuit calculations which verified the findings made by the applicant. Therefore, we conclude the criteria used for the Zimmer plant is acceptable based on our evaluation and the implementation guidance described in Regulatory Guide 1.75.

8.1.3 Specific Findings

Identification of Safety Loads

Our review of the load lists in the Final Safety Analysis Report, Chapter 8.3, identified the following types of inconsistencies: (1) some of the motor loads listed in the tables were different from those shown in the electrical schematics, and (2) the turbine generator turning gear was listed as an electrical load at full power and also as a loss-of-coolant accident load and the feed pump turning gear was listed as a loss-of-coolant accident load (see subsection 7.3 of this report for further discussion of the use of the feedwater system during plant emergencies).

Revised tables were provided and clearly indicated that the loads listed in these tables represent the latest information. Our review of these tables indicates that our concerns have been resolved and the tables are now acceptable.

8.1.4 Evaluation

The applicant identified safety-related electric power systems, applicable power system criteria, and documented his intent to construct these systems in accordance with the criteria of Standard Review Plan, Table 8-1, which were in effect when the construction permit was issued, and to the extent described in subsection 8.1.2 of this report. We conclude that construction of safety-related electric power systems in accordance with these criteria provides assurance that these systems will perform as designed.

8.2 Offsite Power Systems

8.2.1 General Discussion

Section 8.2 of the Final Safety Analysis Report presents descriptive information, analyses and referenced documents, including electrical single line diagrams, electrical schematics, logic diagrams, tables, and physical final arrangement drawings for the offsite power systems. We audited the application against the requirements of the Standard Review Plan, and also against the information requirements of the corresponding section of the Standard Format for Safety Analysis Reports.

The offsite power system is referred to in industry standards and regulatory guides as the "preferred power system."

Final Safety Analysis Report, Section 8.2.1, describes the offsite power system which includes multiple offsite sources, transmission paths and transformers. Each transformer is capable of supplying sufficient electrical power at a sufficient voltage to operate Unit 1 at full power. Full power operation presents the largest electrical load demand on these transformers.

8.2.2 General Findings

The offsite alternating current power sources include three 345 kilovolt transmission lines and one 69 kilovolt transmission line. Three-out-of-four of these transmission lines approach the switchyard on separate rights-of-way. Control power for the switchyard substation is independent of the engineered safety features buses. Load flow and stability studies were performed which indicate that a full load trip of the plant or largest external generating station would not impair the ability of the transmission system to supply power to the engineered safety features buses.

The Final Safety Analysis Report states that, "Control of transmission line circuit breakers and associated instrumentation will be by supervisory control from a system load dispatcher officer located remote from the plant." Because the dispatcher has more complete information as to the status of the incoming power sources and has direct communications with the power plant operator, this is an acceptable design. However, the applicant also stated that he will modify this design to provide backup control in the Zimmer control room for the Zimmer switchyard.

Our review found that the 345 kilovolt and 69 kilovolt sources do comply with the requirement of General Design Criteria 17 for two physically-independent offsite power sources and that these are both immediate access lines, and we conclude that the design meets the provision of Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," that is, two immediate access lines are preferable to only one immediate access line.

8.2.3 Specific Findings

Turbine Control

We noted that the turbine control system in conjunction with the recirculation flow control system could provide a method by which the remote dispatcher could change the reactor power level. However, the applicant indicated that the dispatcher will not have control of the turbine load limiter and, therefore, control of reactor power will remain within the Zimmer control systems.

Accordingly, we find that there is no event related to the turbine control system which is more severe than those events which are presented in the Final Safety Analysis Report, Chapter 15 and evaluated by us in section 15.0 of this report. Therefore, we conclude that the turbine control system and the interface between the main generator and the grid are acceptable.

Grid Stability

The applicant described the results of the grid stability studies in response to Acceptance Review Question 222.12 (b). We find that these studies satisfy the requirements of Section 8.2.2 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" and the system is designed and operated in accordance with the criteria established by the East Central Area Coordination Council.

We believe that the combination of the coal miner's strike and severe winter weather of 1976-1977 with their resultant heavy demands on the grid demonstrated both the adequacy of the council's criteria and the design of the existing grid to support the Zimmer Station. The basis for this conclusion is the fact that there were no major or prolonged interruptions in service to the applicant's customers yet the system was involved more than usual in the exchange of power across its interties with neighboring utilities. (For further information see the Government Accounting Office report EMD-78/06 dated October 10, 1978 page 30 paragraph 1).

The stability studies satisfy our criteria described in the Branch Technical Position, Instrumentation and Control Systems Branch 11, "Stability of Offsite Power Systems," contained in Table 8-1. The grid has demonstrated that it has sufficient margins to satisfy its load demands under severe conditions, and the addition of the Zimmer transmission lines will provide additional grid security. Accordingly, we find this aspect of the plant design and implementation acceptable.

Switchyard Location

Our review indicates that the switchyard and the cables from the auxiliary transformers are below the level of the maximum probable flood (545.4 feet). As a result of the location of the nuclear power plant in the flood plain of the Ohio River, there exists a possibility that all offsite sources will fail simultaneously even if a perfect switchyard control scheme were provided (see subsection 2.4 of this report for additional information on the probability of floods). This represents a change from the construction permit (when the switchyard was located at an elevation above 640 feet). Although the present location of the transformers is consistent with the construction permit, we noted that this arrangement might not satisfy the requirement of General Design Criterion 17 for "two physically independent circuits," and might not satisfy the construction permit commitment to demonstrate that a second independent 69 kilovolt source could be reinstated within 30 minutes.

We pursued the question of this aspect of the design satisfying the General Design Criteria 17 requirement to "minimize to the extent practical the likelihood of the simultaneous failure under postulated environmental conditions" with the applicant.

As a result of the information which is contained in the Final Safety Analysis Report, Section 2.4 and Chapter 8, we conclude that the plant can be in cold shutdown before a flood can damage the offsite sources and that either of the diesel generators for division 1 or 2 can continuously supply all necessary cold shutdown loads for 200 days. We also conclude that the design basis flood will recede in 14 days and that there is sufficient diesel oil onsite to operate one diesel for 21 days. Furthermore, the equipment of either division 1 or 2 is sufficient for long term shutdown cooling.

On this basis, we conclude that the present design satisfies General Design Criterion 17 and is, therefore, acceptable.

Independence of Preferred and Standby Power Systems

The 4160 volt switchgear consists of three indoor metal-c'.. assemblies located at elevations 510 feet, 525 feet and 546 feet of the auxiliary building. Each switchgear assembly supplies one division of safety-related Class IE loads as well as non-Class IE loads. The three 4160 volt switchgear assemblies are physically separate from each other. In addition, each switchgear assembly has assigned to it a diesel engine generator set for onsite standby power and one Class IE battery which is housed in a masonry compartment built into each switchgear room. Each switchgear assembly receives offsite power from the turbine generator (via the unit auxiliary transformer) and the two reserve auxiliary transformers through three sets of cables. Main generator electrical controls are located in the main control room. These include main generator circuit breaker controls, synchronizing equipment, generator excitation and voltage control equipment, and circuit breaker controls for all main supply circuits for the auxiliary power system. High speed protective relaying equipment is provided for the main generator, main and auxiliary transformers, main buses, transmission lines, and interconnecting cables and bus ducts to provide proper clearing of this equipment in the event of electrical faults. The protective relay system includes breaker failure protection and backup relaying which is designed to ensure proper clearing of electrical faults in event of a failure of the primary protective relaying. The instrumentation is provided in the main control room for the main generator connections and equipment. This includes indicating instruments for voltage, current, power, reactive power and frequency. Recording instruments are provided for generator power output and main bus voltage. Kilowatt-hour meters are provided for main generator outputs and for auxiliary power system loads. Instrumentation also is provided for monitoring generator and transformer temperatures. From the information which has been supplied, we conclude that there is adequate physical separation between the offsite and onsite sources.

Because of our concerns with regard to the accuracy of the load lists (see subsection 8.1.3 of this report), we were not certain that the breakers had sufficient short circuit or interrupt capacity nor were we certain that the protective relaying would be properly set so as to assure the electrical independence of these sources. We acknowledged the fact that the individual errors in the load lists were small (50 kilowatts or less in most cases) but the cumulative effect of these errors plus the additional loads which may be placed on switchgear "B" by nonjudicious operation of installed backup pumps was a concern which had to be resolved. We have reviewed the subsequent changes to the Final Safety Analysis Report and conclude that the present design is now acceptable regarding the concerns noted above.

Equipment Qualification

The offsite equipment is not required to meet Class IE requirements. However, General Design Criterion 17 does require that an offsite event not compromise the onsite power supply. One such offsite event which could potentially damage onsite equipment is prolonged exposure of the Class IE equipment to undervoltage and/or underfrequency conditions. For this reason, we requested additional information on the Class IE equipment with regard to its qualification for the extremes of the energy supply. It was our objective to assure that the offsite sources will be automatically separated prior to violation of the limits of qualification so that the safety related plant equipment will continue to satisfy the requirements of Institute of Electrical and Electronic Engineers Standard 279-1971 Section 3(7). Based on our review, we find that the undervoltage protective relaying will cause separation prior to violating the equipment design limits (as is required by Institute of Electrical and Electronic Engineers Standard 279-1971) and conclude that the present design is acceptable.

8.2.4 Evaluation

Our review found that the offsite power system includes three independent power sources from the grid, transmission lines, transmission line towers, transformers, switchyards and switchyard component control systems, switchyard battery systems, the main generator, and disconnect switches used to supply electric power to safety-related and other equipment. The review of the offsite power system included single line diagrams, station layout drawings, and schematic diagrams and descriptive information. The review included design criteria and design bases for the offsite power system and analyses of the adequacy of those criteria and bases The review also included the analyses of the manner in which the design of the offsite power system conforms to the design criteria.

The basis for acceptance in our review is an acceptable demonstration of conformance of the designs, design criteria and design bases for the offsite power system to the Commission's regulations as set forth in the General Design Criteria and applicable regulatory guides and standards.

8.3 Onsite Power Systems

Alternating Current Power Systems

8.3.1 General Discussion

The Final Safety Anlysis Report, Section 8.3, provides the descriptive information, including functional logic diagrams, functional piping and instrument diagrams, electrical single line diagrams, physical arrangement drawings, and electrical schematics for the alternating current onsite power system. We review this information as detailed in the Standard Review Plan, and also determine the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format for Safety Analysis Reports.

Areas of review associated with this system that are covered elsewhere in this report are as follows: (1) environmental design and qualification testing of electrical equipment are addressed in subsection 3.11, (2) onsite direct current control power feeds to the standby power system are addressed in subsection 8.4.3, (3) technical specification requirements imposed upon the operation of the standby power system are discussed in Sections 7.1.3. and (4) the criteria for seismic qualification and the test and analysis procedures and methods to assure the operability of seismic Category I instrumentation and electrical equipment, including cable trays, switchgear, control room boards and instrument racks and panels, in the event of a seismic occurrence, are addressed in subsection 3.10 of this report.

8.3.2 General Findings

As a result of our review, and with the exception of those items presented in subsections 3.10 and 7.1.3, we find that the designs of the onsite alternating current systems satisfy the applicable criteria presented in Table 8-1 of the Standard Review Plan, except as noted below, and are, therefore, acceptable.

8.3.3 Specific Findings

Separation

The design criteria for the physical separation evolved during the same period when Regulatory Guide 1.75, "Physical Independence of Electrical Systems," and Institute of Electrical and Electronic Engineers Standard 384-1974 were first drafted. The criteria for Zimmer are close to the requirements of the standard. However, our review identified several examples of non-Class IE equipment which are powered by an onsite bus and are not removed by an accident signal. Typical examples are: (1) turbine generator turning gear, (2) reactor feed pump turning gear, (3) hydrogen seal oil pumps, (4) emergency lighting, and (5) plant annunciators. All such equipment are listed in the tables of the Final Safety Analysis Report, Chapter 8.

In order to provide reasonable assurance that such usage would not present an unacceptable risk to the Class IE sources, we requested detailed information on the short circuit analyses and protective relay coordination. This information was provided and, as a result of our review, we are satisfied that the systems are adequately protected.

Diesel Generators

Three tandem diesel generators (one pair per division), located in separate seismic Category I rooms, are provided. The direct current control power is provided by the associated divisional battery. Each generator is rated at 3,500 kilowatts continuous, 3,731 kilowatts for 2,000 hours, and 3,927 kilowatts for 30 minutes. The maximum load is approximately 3,000 kilowatts (division 3) under loss-of-coolant conditions and 3,416 kilowatts (division 2) under safe shutdown conditions. The diesels satisfy the recommendations of Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between their Distribution Systems," and Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies." The load requirements will be verified by preoperational testing. The preoperational testing program is described in the Final Safety Analysis Report, Section 14.1, and the evaluation of this program is provided in section 14 of this report.

Testing

Because of our concerns such as those discussed in the previous sections of this report, we have not completed the review of the preoperational and periodic testing programs and provisions. (See Section 14.0 for a further discussion of the status of our review of the provisions and procedures for preoperational and periodic testing.)

In the course of a site visit we noted that non-Class IE convenience outlets are mounted on many Class IE instrument panels. We requested that the applicant describe the provisions and procedures which are to be provided to assure that a destructive ground current will not develop between the Class IE equipment and test equipment which may be powered from these outlets. The applicant stated that the outlet ground will be common to the instrument ground system and that this design modification will prevent destructive ground loops; we agree. Furthermore, we note that such practice is consistent with the General Electric Company grounding criteria presented in the General Electric Company Document 22A2718BC, Figure 4, which we conclude provides acceptable industry grounding practice. Therefore, we conclude that this concern has been suitably resolved.

8.3.4 Evaluation of Onsite Alternating Current Systems

Our review found that the standby power system includes the onsite power sources, distribution systems, vital auxiliary supporting systems, instrumentation, and controls utilized to supply power to safety-related components and systems. The scope of review included the descriptive information, functional logic diagrams, functional piping and instrument diagrams, electrical single line diagrams and final physical arrangement drawings, and electrical schematics for the standby power system and for those auxiliary systems that are vital to the proper operation of the Class IE standby power system and its connected Class IE loads. The review included the design bases and their relation to the design criteria for the standby power system and for the vital supporting systems and the analyses of the adequacy of those criteria and bases. The review also included the means for identifying safety-related cables, cable trays and terminal equipment in the plant; the qualification test programs and results demonstrating the suitability of the diesel generators as standby power supplies; and the seismic qualification test program and results and analyses.

The basis for acceptance in our review is a timely and complete documentation of the implementation of the designs to the applicable design criteria and the design bases for the standby power system and vital supporting systems, and to the Commission's regulations as set forth in the General Design Criteria, applicable regulatory guides, branch technical positions, and industry standards listed in Table 8-1 of the Standard Review Plan to the extent described in subsection 8.1.2 of this report.

8.4 Onsite Power Systems

Direct Current Power Systems

8.4.1 General Discussion

The Final Safety Analysis Report Section 8.3, also describes the direct current power systems including those direct current power sources and their distribution systems and vital supporting systems provided to supply motive or control power to safety related equipment. We review the information as detailed in the Standard Review Plan, and also determines the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format for Safety Analysis Reports.

8.4.2 General Findings

Areas of review associated with these systems which are covered elsewhere in this report are as follows: (1) environmental design and qualification testing of electrical equipment are addressed in subsection 3.11, (2) technical specification requirements imposed upon the operation of the direct current power system are discussed in subsection 7.1.3, and (3) separation is discussed in subsection 8.3.

We find that the direct current systems comply with the applicable requirements of Table 8-1 of the Standard Review Plan and are, therefore, acceptable.

8.4.3 Specific Findings

System Capacity

We reviewed the general methods which were used to determine battery and battery charger capacity and found that they satisfy General Design Criterion 1 and that each battery will supply its essential loads for a minimum of two hours and are, therefore, acceptable.

We also found that the batteries are physically separated into three electrically isolated divisions similar to the BWR-4 designs such as Hatch 1 and the design is acceptable.

8.4.4 Evaluation

Our review found that the direct current power system includes the batteries, battery chargers and distribution centers used to supply power to direct current operated safety-related equipment. The scope of review of the direct current power system included single line diagrams, schematic diagrams, and descriptive information for the direct current power system and for those auxiliary supporting systems that are essential to the operation of the direct current power system. The review included the design criteria and analyses of the adequacy of those criteria and bases. The review also included the analyses of the manner in which the design of the direct current power system conforms to the design criteria identified in subsection 8.1.1 of this report. The basis for acceptance in our review is conformance of the design, design criteria and design bases for the direct current power system to the Commission's regulations as set forth in the General Design Criteria, applicable regulatory guides, branch technical positions, and industry standards as listed in Table 8-1 of the Standard Review Plan.

We conclude that the design of the direct current power system conforms to applicable regulations, guides, technical positions and industry standards, noted above, and is acceptable.

9.0 AUXILIARY SYSTEMS

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

The auxiliary systems necessary for safe reactor operation or shutdown include: reactor building closed cooling water system; service water system; ultimate heat sink; the heating, ventilation and air conditioning systems for the control room and all other safety related areas; diesel generator fuel oil storage and transfer system; the diesel generator auxiliary systems; and essential portions of the compressed air system. The systems necessary to assure safe handling of fuel and adequate cooling of the spent fuel include: new and spent fuel storage facilities, the spent fuel pit cooling and cleanup system, the fuel handling system and portions of the fuel area ventilation system.

We reviewed those auxiliary systems or portions of the systems whose failure would not prevent safe shutdown but could, either directly or indirectly, be a potential source of radiological release to the environment. These systems include the plant equipment and floor drainage system and the main steam isolation valve leakage control system.

Other systems that are non-safety related include the turbine building cooling water system, compressed air system, condensate storage facilities, demineralized water makeup system, the potable water system, portions of the plant service water system, the communications and lighting systems and the nonessential heating and ventilation systems such as the turbine building area system. The acceptability of these systems was based on determining that: (a) where the system interfaces are connected to seismic Category I systems or components, seismic Category I isolation valves will be provided to physically separate the nonessential portions from the essential system or component, and (b) the failure of non-seismic systems or portions of the systems will not preclude the operation of safety-related systems or components located in close proximity. We find the above listed systems meet our criteria and, therefore, find them acceptable.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

The new fuel storage racks are designed for dry storage of approximately 30 percent of a full core load. The racks provide for a center-to-center spacing of 6.625 inches between adjacent fuel assemblies which is sufficient to maintain an effective multiplication factor of 0.95 or less in the event that the new fuel area were flooded with water. The outer structure of the rack design precludes the inadvertent placement of a fuel assembly in the rack closer than the prescribed spacing.

The new fuel storage racks are bolted together and fixed to the new fuel storage vault. The new fuel racks and storage vault are designed to seismic Category I requirements.

We reviewed the adequacy of the applicant's design for the new fuel storage facility necessary to maintain a subcritical array during normal, abnormal, and accident conditions. We conclude that the design is in conformance with General Design Criterion 62 and the positions of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," including the positions on seismic design and missile protection, and is, therefore acceptable.

9.1.2 Spent Fuel Storage

Spent fuel storage space is provided in the spent fuel storage pool. The fuel storage pool contains storage space sufficient for two full core fuel loads. Safety curtains are installed which serve as a structural member to prevent incorrect insertion of a fuel assembly. With the rack spacing provided, a maximum effective multiplication factor of 0.95 will not be exceeded. The spent fuel storage racks are designed to withstand the impact of a dropped fuel assembly. The spent fuel racks and the storage pool are designed to seismic Category I requirements.

The design of the spent fuel pool walls and the structure housing the spent fuel pool will prevent tornado missiles from penetrating the pool. The cask handling crane is fully redundant and can accept any single active failure, thereby precluding the possibility of dropping a cask while handling. The applicant, provided electrical interlocks to prevent crane travel over the spent fuel pool even though the crane can withstand any single active failure.

We reviewed the adequacy of the applicant's design for the spent fuel storage facility necessary to maintain a subcritical array during all normal, abnormal, and accident conditions. We conclude that the design for the spent fuel storage facilities is in conformance with the requirements of General Design Criterion 61 and 62 and the applicable positions of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," including the positions on seismic design, missile protection, design compatibility with the maximum crane loads that can travel over the pool, and are, therefore, acceptable.

9.1.3 Spent Fuel Cooling

The spent fuel pool cooling system is designed to remove the decay heat from the fuel assemblies. The cooling system consists of two 100 percent capacity spent fuel cooling pumps and heat exchangers. The makeup of the spent fuel pool is normally supplied from the non-seismic cycled condensate storage tank. In addition make up to the spent fuel pool can be provided by the seismic Category I portion of the service water system.

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

The spent fuel pool cooling system with either train operating, has the capacity to maintain the spent fuel pool water temperature below 120 degrees Fahrenheit with 25 percent of a core stored in the pool. This is within the 140 degree Fahrenheit limit stated in the Standard Review Plan 9.1.3, "Spent Fuel Pool Cooling and Cleanup System."

Based on our review, we conclude that the system design meets the requirements of General Design Criterion 61, including seismic design, provisions to prevent uncovering the fuel, and provisions for assured makeup and is, therefore, acceptable.

9.1.4 Fuel Handling System

The fuel handling system is designed to safely handle fuel assemblies from receipt of new fuel to shipping of spent fuel. The system is designed to conduct all spent fuel transfer and storage operations underwater to insure adequate shielding during refueling.

The arrangement of the fuel handling area includes a 110-ton overhead crane for the handling of the spent fuel shipping cask. Single failure proof protection is designed into the cask crane and lifting yoke to prevent a possible drop of the spent fuel cask. Travel of the spent fuel cask handling crane is limited by electrical interlocks to prevent the cask crane from carrying the spent fuel cask over the spent fuel storage pool; thus dropping or tipping of a spent fuel cask into the spent fuel pool is precluded.

The Final Safety Analysis Report and supplements providing information by the crane manufacturer have addressed all areas identified in our Branch Technical Position, Auxiliary and Power Conversion System Branch 9-1, "Overhead Handling Systems for Nuclear Power Plants." All areas of our Branch Technical Position Auxiliary and Power Conversion System Branch 9-1, "Overhead Handling Systems for Nuclear Power Plants," have been satisfactorily addressed. In addition the applicant committed to performing inspections in accordance with American National Standards Institute Standard B.30.2.

We reviewed the overhead crane handling system for Zimmer with respect to the guidance and criteria of Branch Technical Position, Auxiliary and Power Conversion System Branch 9-1 that was implemented in accordance with the proposed Regulatory Guide 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants," dated February 1976.

The Zimmer overhead crane did not meet the nondestructive weld joint inspection guidance of position C.1.d of the proposed Regulatory Guide 1.104 because the crane was built and delivered to the site well before the issuance of the guidance cited above. However, we conclude from our review, that the crane manufacturer has designed the joints and welds to prevent lamellar tearing and that the applicant's inspection program for the crane, following completion of construction, meets the American National Standards Institute B.30.2 cited in the Branch Technical Position, cited above, and provides a level of safety equivalent to Regulatory Guide 1.104 and, therefore, is acceptable.

We reviewed the adequacy of the applicant's design necessary for safe operation of the fuel handling system during normal, abnormal and accident conditions. We conclude that the design is in conformance with the positions of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," including the position regarding protection of the spent fuel storage facility from the impact of unacceptable heavy loads carried by overhead cranes and is, therefore, acceptable.

9.2 Water Systems

9.2.1 Service Water System

The service water system is designed to provide cooling water to remove heat from the residual heat removal and reactor building closed cooling water systems during normal plant cooldown, plant shutdown and accident conditions. The service water system also supplies coolant to the diesel generators and is used to flood the reactor vessel if required during the post loss-of-coolant accident period.

The service water systems consist of two independent full capacity trains. Each train has two full capacity pumps. The service water system is designed to seismic Category I requirements and protected to withstand adverse environmental occurrences. Each train of the service water system is powered from a separate emergency essential safeguards features bus.

Based on our review, we conclude that the design criteria and bases for the service water system meet the requirements of General Design Criterion 44 regarding their ability to transfer heat from safety related components to the ultimate heat sink, and General Design Criteria 45 and 46 regarding tests and inspections. We, therefore, conclude that the design of the service water system is acceptable.

9.2.2 Cooling System for Reactor Auxiliaries (Reactor Building Closed Cooling Water System)

During normal plant operations the reactor building closed cooling water system removes heat from the auxiliary equipment housed in the reactor building. The essential portions of the reactor building closed cooling water system are seismic Category I. The reactor building closed cooling water system consists of two independent full capacity trains. Each train of reactor building closed cooling water system is operable from a separate emergency bus.

The reactor building closed cooling water systems heat exchangers reject heat to the service water system. The reactor building closed cooling water system is designed to supply the spent fuel pool cooling heat exchangers from the non-essential portions of the system. At our request the applicant made provisions to supply water from the essential portion of the reactor building closed cooling water system through removable spool pieces to one heat exchanger of the spent fuel pool cooling system.

Based on our review we conclude that the design criteria and bases for the reactor building closed cooling water system meet the requirements of General Design Criterion 44 regarding its ability to transfer heat from safety related components to the ultimate heat sink and General Design Criteria 45 and 46 regarding tests and inspection. We, therefore, conclude that the design of the reactor building closed cooling water system is acceptable.

9.2.3 Ultimate Heat Sink

The ultimate heat sink will dissipate heat from the service water system and provide water for safe shutdown of the plant under accident conditions. The ultimate heat sink for Zimmer, Unit No. 1 is the Markland Pool located on the Ohio River.

The Markland Pool will dissipate the maximum heat load due to the total integrated decay heat, station auxiliary system heat release and sensible heat removed from the containment and primary system following a design basis loss-of-coolant accident.

The applicant demonstrated to our satisfaction 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 system to withstand the most severe natural phenomena expected and a single failure of man-made structural features.

Based on our review, we conclude that the ultimate heat sink is designed in accordance with the positions of Regulatory Guide 1.27 and is, therefore, acceptable.

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

The compressed air system includes a control air system which is designed seismic Category I and a station air system which is not safety related. The station and control air systems are not required to effect a safe reactor shutdown or for control during long term recovery since all air-operated valves or controls involved in safety related functions are either provided with individual accumulators which are seismically designed to properly position the valve, or are designed to fail in the safe position. The automatic depressurization system which requires a compressed gas has a nitrogen system with the piping and valves designed to seismic Category I requirements. This portion of the system is fully redundant.

We reviewed the applicant's design for the compressed air system necessary for the continued presence of supply air to safety related components during anticipated plant operation conditions, and conclude that the system design is acceptable. We also conclude that the operation of the non-safety related station air system will not adversely affect any safety related function.

9.3.2 Equipment and Floor Drainage System

The equipment and floor drainage system is designed to collect leakage or spillage from equipment in the reactor building, auxiliary building and turbine building. The waste from radioactive equipment drains and floor drains are discussed in section 11 of this report.

Leak detection is provided in the sumps for all emergency core cooling system equipment and rooms. Each component of the emergency core cooling system is in a separate room with its own sump and duplex pumps. Instrumentation provided in the control room to indicate excessive emergency core cooling system component leakage will consist of a sump high-high level alarm. High and low level switches are provided in each sump to start and stop the sump pump automatically.

The condensate and demineralized water storage tanks are physically located such that their contents will not enter the plant facility in the event of a failure. Failure of other tanks located outside or within the turbine, auxiliary and fuel areas will not cause a flood of sufficient depth to endanger any safety related equipment. Tanks in the control building are small enough such that any water loss is within the capacity of the floor drain system.

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

9.3.3 Main Steam Isolation Valve Leakage Control System

The main steam isolation valve leakage control system is designed to control and minimize the release of fission products which could leak through the closed main steam isolation valves after a loss-of-coolant accident. This is accomplished by directing the leakage through bleed lines into an area served by the standby gas treatment system for processing prior to release to the atmosphere.

Each system utilizes dilution air flow from the auxiliary building to the blower suction to mix with and decrease the steam temperature. The system is designed to seismic Category I requirements.

The main steam isolation valve leakage control system controls are provided with interlocks actuated from safety systems to prevent inadvertent main steam isolation valve leakage control system operation.

The applicant states that the design and operational objective of the main steam isolation valve leakage control system will be established to allow main steam isolation valve leakage rates up to approximately 100 standard cubic feet per hour for each main steam isolation in each line. (This design basis while acceptable for sizing the main steam isolation valve leakage control system, does not provide an acceptable basis to allow an increase in the main steam isolation valve leak rate.) In addition, an additional interlock is provided so that the operation of an inboard system is prevented should an inboard main steam isolation valve fail to fully close.

We reviewed the adequacy of the applicant's design for the main steam isolation valve leakage control system necessary to control the release of fission products after a loss-of-coolant accident. We conclude that the present design is acceptable.

9.4 Air Conditioning, Heating, and Ventilation Systems

9.4.1 Control Room Heating, Ventilating and Air Conditioning System

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

The control room system consists of two 100 percent air-conditioning systems and each containing a supply air filtration unit. During chlorine or ammonia accident conditions the control room air is automatically recirculated through the air filtration units. The entire control room heating, ventilating and air conditioning system is designed to seismic Category I requirements and all outside intakes are tornado missile protected.

The control room heating, ventilating and air conditioning systems is designed to maintain the control room under positive pressure. Redundant smoke and radiation detectors will monitor the minimum outside air supply with alarms in tne control room. Initiation of these alarms will automatically isolate the outside air supply to the control room and start the emergency makeup air filtration unit. We conclude that the control room heating ventilation and air condition system meet the acceptance requirements of Standard Review Plan 9.4.1, "Control Room Area Ventilation System."

9.4.2 Reactor Building Heating, Ventilating and Air Conditioning System

The reactor building ventilation system equipment is not required to function under any but normal station operating conditions however, the reactor building recirculation fans and secondary containment isolation valves are part of the standby gas treatment system and are designed to operate during abnormal conditions. The duct system is designated as seismic Category I.

Based on our review and evaluation of the design of the reactor building heating, ventilating and air conditioning system we conclude that the reactor building heating ventilating and air conditioning systems are acceptable.

Fuel Pool Area Ventilation System (Reactor Building Heating, Ventilating and Air Conditioning System)

The fuel pool portion of the reactor build ventilation system is designed to maintain the atmosphere above the refueling floor within acceptable temperature and humidity limits for personnel and equipment, and also to maintain the area at a negative pressure, and to mitigate the consequences of a fuel handling accident by filtration of the exhaust air. The system consists of a normal supply and exhaust system; for emergency conditions the normal supply and exhaust systems are isolated and airborne contaminates are exhausted through the reactor building standby gas treatment system.

The exhaust from the fuel handling area during normal operation is discharged through the station vent by the normal exhaust system without filtration. A slight negative pressure is maintained in the fuel building by this exhaust system. The reactor building standby gas treatment system, and the reactor building ventilation system are designed to operate in conjunction with one another to mitigate the consequences of the fuel handling accident. These systems are designed to seismic Category I requirements. In the event of a fuel handling accident, a high radiation signal from the radiation monitors in the exhaust air duct automatically actuates the standby gas treatment system. Pneumatic operated dampers in the normal supply and exhaust ventilation system fail closed to direct contaminated exhaust through the redundant charcoal filter banks in the standby gas treatment system prior to discharge to the atmosphere through the station vent.

Based on our review, we conclude that the design of the reactor building ventilation system in conjunction with the standby gas treatment system meets the recommendations of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and therefore, is acceptable.

9.4.3 Emergency Core Cooling System Equipment Areas Heating, Ventilating and Air Conditioning System

The emergency core cooling system equipment area cooling system consists of a fan coil unit for each emergency core cooling system equipment cubicle. Each system is seismic Category I and powered from the essential buses serving the cubicle from which the equipment is powered. Full redundancy exists throughout the entire emergency core cooling system equipment area. The seismic Category I reactor building closed cooling water system is used in the cooling coils. Ventilation air for the emergency core cooling system equipment cubicles is provided by the redundant reactor building heating, ventilating and air cnditioning system.

We reviewed the adequacy of the applicant's design necessary to maintain a suitable environment for the emergency core cooling system equipment during normal, abnormal and accident conditions. We conclude that the design is acceptable based on it meeting the guidance provided in Standard Review Plan 9.4.5, "Engineered Safety Feature Ventilation System."

9.4.4 Diesel Generator Building Heating, Ventilating and Air Conditioning System

The diesel generator building heating, ventilating and air conditioning system is designed to maintain a suitable environment for the operation of the diesel generators and their auxiliary components during all modes of plant operation, including accident conditions. Independent diesel generator heating and ventilation systems, and air supply and exhaust systems are provided for each of the three diesel generators to satisfy the required environmental conditions and combustion air requirements during diesel operation.

The diesel generator room ventilation system is designed to seismic Category I requirements and to maintain the diesel generator rooms below 122 degrees Fahrenheit whenever the diesel generators are in operation. This is within the recommended ambient temperature design rating of the diesel generators.

The combustion air supply is drawn from the room ventilation air supply. Any meteorological changes or accident conditions cannot affect all diesel air supplies. The outside air intakes and exhausts are tornado missile protected.

We reviewed the ventilation system design for the diesel generators and conclude that they are acceptable.

9.4.5 Emergency Switchgear Area Air Conditioning System

The air conditioning system for the emergency switchgear area provides cooled air to the emergency switchgear rooms and the battery rooms. The system consists of two 100 percent capacity, seismic Category I cooling trains, for each switchgear rooms. The battery rooms receive air from the switchgear rooms during battery charging operation. The battery rooms are provided with separate exhaust fans so that they can be maintained at a negative pressure with respect to the switchgear rooms. The battery rooms exhaust fans discharge to the turbine building. The reactor building component cooling water system removes heat from the water cooled condensing units to limit the maximum temperature inside the switchgear rooms to 104 degrees Fahrenheit and inside the battery rooms to 75 degrees Fahrenheit which is below the design temperature of the equipment.

Based on our evaluation, we determined that the design of the air conditioning systems for the emergency switchgear area contains sufficient component redundancy and physical separation to meet the single failure criterion so that air conditioning and ventilation is assured during accident conditions, and therefore, the systems are acceptable.

9.4.6 Service Water Pump Structure Heating, Ventilating and Air Conditioning System

The service water pump structure ventilation system consists of a ventilation system for the pump house accessible areas, a heat removal system for the cooling tower makeup water pump room, and a heat removal system for each service waterpump room. Two 50 percent capacity fan-coil units are provided for each service water pump room which functions at a maximum temperature of 104 degrees Fahrenheit during service water pump operation. The essential portions of the system are designed to seismic Category I requirements.

Operation of the service water pump structure accessible areas ventilation system is not required for safe shutdown but the service water pump room heat removal system is required for both normal and abnormal station conditions. We reviewed the adequacy of the applicant's design necessary to maintain a suitable environment for the essential equipment during normal, abnormal and accident conditions and conclude that the design is acceptable.

9.5 Fire Protection Systems

As a result of investigations conducted by us on the fire protection systems, further requirements were imposed to improve the capability of the fire protection system to prevent unacceptable damage that may result from a fire. We requested that the applicant conduct a re-evaluation of the proposed fire protection system for Zimmer and that the applicant compare these systems, in detail with the guidelines of Appendix A to Branch Technical Position Auxiliary and Power Conversion System Branch 9.5-1, "Guidelines for Fire Protection for Nuclear Plants."

The applicant has provided the requested reevaluation and responded to all our positions and concerns resulting from our review of the reevaluation. We will provide our evaluation of the fire protection system in a supplement to this report after we have completed our review of the applicant's responses to our concerns and positions.

9.6 Diesel Generator Systems

9.6.1 Diesel Generator Fuel Oil Storage and Transfer System

The fuel oil storage and transfer system is designed to provide fuel oil storage and transfer capability to allow operation of each standby diesel generator for at least seven days.

The fuel oil system consists of two separate and independent trains, one for each diesel generator. Each system includes a day tank which holds a 550 gallon supply of fuel oil for each standby diesel. The fuel oil system is designed to seismic Category I requirements. The fuel oil storage tanks are located in individual missile protected cubicles and the transfer pumps are located inside the diesel generator building.

The fuel oil transfer pumps are powered from separate emergency buses. Based on its independent evaluation, we have determined that the design of the fuel oil systems meet our single failure criteria.

Based on our review of the diesel generator fuel oil system, we conclude the system has adequate capacity and can perform its designated safety functions in accordance with the guidance and criteria in Standard Review Plan 9.5.4, "Emergency Diesel Engine Fuel Oil Storage and Transfer System," and is, therefore, acceptable.

9.6.2 Diesel Generator Auxiliary Systems

The diesel generator auxiliary systems include the diesel generator cooling water system, diesel generator starting system, and the diesel generator lubrication system.

The diesel generator cooling water system is an integral part of the diesel generator. It is designed to maintain the temperature of the diesel engine within a safe operation range. The diesel generator cooling water system is a closed cooling system and the heat is rejected to the station service water system which is seismic Category I. When the engine is idle, the engine water is heated by electric heaters to keep the engine warm and ready to accept loads within the prescribed time interval. The system is designed to seismic Category I requirements and meets the guidance and criteria described in the Standard Review Plan 9.5.5, "Emergency Diesel Engine Cooling Water System."

Each of the standby diesel generators is provided with independent compressed air starting systems consisting of two air compressors, each compressor supplying a separate storage tank. Each tank is capable of providing three starts without recharging from the air compressors. The starting air system is designed to seismic Category I requirements and meets the guidance and criteria described in the Standard Review Plan 9.5.6, "Emergency Diesel Engine Starting System."

Each diesel generator is provided with a lubrication system, which is an integral part of the diesel generator, designed to supply lubricating oil to the diesel generators. The system circulates lube oil through the engine for heating when the engine is idle and for cooling when the engine is operating. The lube oil is cooled by the diesel generator cooling water system and heated by an electric heater. The system is designed to seismic Category I requirements and meets the guidance and criteria described in the Standard Review Plan 9.5.7, "Emergency Diesel Engine Lubrication System."

Based on our review and meeting the guidance and criteria in the cited Standard Review Plans, we conclude that the diesel generator auxiliary systems design meet their designated safety functions, have the needed capacity and are, therefore, acceptable.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion system transfers heat energy from the nuclear reactor to the turbine generator and converts it by conventional means to electrical energy. The Zimmer Nuclear Power Station steam and power conversion system consists of the main steam supply system, turbine generator, turbine bypass system, feedwater system, circulating water system, main condenser and condenser evacuation system, turbine gland sealing system, and condensate storage and cleanup systems. We reviewed the design criteria and design bases for these systems which are necessary for safe operation. The review and conclusions are discussed in the following subsections of this section.

10.2 Turbine Generator

The turbine-generator consists of a tandem arrangement of a double flow high-pressure turbine and four-flow low pressure turbine driving a direct-coupled generator at 1800 revolutions per minute. The turbine is equipped with an electro-hydraulic control system. The speed of the turbine is controlled by modulating the turbine inlet steam control valves.

The following are some of the conditions under which the turbine control system is designed to trip the turbine: turbine overspeed, condenser low vacuum, excessive thrust bearing wear, generator electric trips, low hydraulic fluid pressure, and manual local or remote turbine trips.

Overspeed protection is accomplished by two independent systems, namely the electrohydraulic system and the mechanical overspeed system. The electro-hydraulic system will close the governor and interceptor valves at less than 103 percent of rated speed. If 111 percent of rated speed is reached, the mechanical overspeed sensor will trip all steam valves (throttle, governor, reheat stop and interceptor valves). An electrical overspeed trip device will also trip valves to drain the trip oil and thus causing closure of all the valves at 110 percent of rated speed.

As a result of our review, we conclude that the turbine-generator overspeed protection design can meet its designated functions and is, therefore, acceptable.

10.3 Main Steam Supply System

The steam generated in the reactor is routed to the high pressure turbine by means of four main steam lines. Each main steam line contains two main steam isolation valves. One main steam isolation valve is located immediately inside of the drywell and the other immediately outside containment. The main steam supply system is designed to seismic Category I requirements up to the turbine stop valves.

The main steam isolation valves are designed to provide positive isolation against steam flow associated with a main steam line break. They are pneumatic operated, fast-closing valves. Operating air is supplied to the valves from the plant air system and a seismic Category I air accumulator provides backup operating air for each valve. The main steam isolation valve is designed to withstand the dynamic forces under the postulated steam line break flow conditions.

We reviewed the adequacy of the applicant's design necessary for the safety-related function of the main steam supply system under normal, abnormal, and accident conditions. We conclude that the design of the main steam supply system conforms with the single failure criterion, the seismic position of Regulatory Guide 1.29, "Seismic Design Classification," and valve closure time requirements and is, therefore, acceptable.

10.3.1 Steam and Feedwater System Materials

The mechanical properties of materials selected for Class 2 and 3 components of the steam and feedwater system satisfy Appendix I of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, or Parts B and C of Section II of the Code. The fracture toughness properties of the ferritic materials satisfy the requirements of Articles NC-2300 and ND-2300 of Section III, of the American Society of Mechanical Engineers Code.

The controls imposed upon austenitic stainless steel comply with the Position Materials Engineering Branch 5-1 on Regulatory Guide 1.31, "Control of Stainless Steel Welding," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices performed in accordance with these requirements provide reasonable assurance that stress corrosion cracking will not occur during the design life of the plant. The controls placed upon concentrations of leachable impurities in nonmetallic thermal insulation used on austenitic stainless steel components of the steam and feedwater systems are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

The welding procedures used in limited access areas satisfy the recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility." The onsite cleaning and cleanliness controls during fabrication satisfy the positions given in Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

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

10.4 Other Features of the Steam and Power Conversion System

10.4.1 Circulating Water System

The circulating water system supplies the main steam condenser with cooling water from the cooling tower. The cooling tower receives makeup from the Ohio River. We reviewed the consequences (flooding) resulting from a failure of this system with respect to affecting safety-related equipment of the plant. The condenser is connected to the circulating water piping using expansion joints located between the condenser and the motor operated butterfly valves on both sides of the condenser. There will not be any safety-related equipment located in the turbine building or flow paths between the turbine building and the safety features building. The flood level in the plant that would result from the failure of an expansion joint will not endanger any safety related equipment.

We reviewed the adequacy of the applicant's design for safe operation of the circulating water system during normal, abnormal, and accident conditions. We conclude that the design of the circulating water system is acceptable.

10.4.2 Condensate Cleanup System

In boiling water reactors, the tubes in the main condenser and other heat exchangers are barriers between the reactor coolant and cooling water. It is, therefore, of some safety significance to maintain an appropriate chemical composition of the reactor coolant to minimize corrosion of these tubes. Inleakage in the main condenser can contaminate the water with undesirable elements or suspended solids. Outleakage through leaking tubes, when reactor coolant pressure is higher than cooling water pressure, can result in radioactive water contaminating the cooling water. A condensate cleanup system, used to maintain high quality condensate consists of several parallel-operating deep-bed demineralizer vessels designed for continuous treatment of condensate flow through deep-bed demineralizers. Vent gases, chemical wastes, and other waste water from the condensate cleanup system are sent to the radwaste system for treatment and disposal. We reviewed the design of the condensate cleanup system and conclude that it satisfies the positions set forth in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," and is, therefore, acceptable.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

The radioactive waste ma..agement systems are designed to provide for controlled handling and treatment of liquid, gaseous and solid wastes. The liquid radioactive waste system processes wastes from equipment and floor drains, phase separator decantation, decontamination and laboratory wastes, demineralizer regenerants, and laundry and shower wastes. The gaseous radioactive waste system includes a charcoal delay system to allow decay of short lived noble gases removed from the main condenser and treatment of ventilation exhausts through high efficiency particulate air filters and charcoal adsorbers as necessary to reduce releases of radioactive materials to "as low as is reasonably achievable" levels in accordance with 10 CFR, Part 20 and 10 CFR, Part 50.34a. The solid radioactive waste system provides for the solidification, packaging and storage of radioactive wastes generated during station operation prior to shipment offsite to a licensed facility for burial.

In our evaluation of the liquid and gaseous radioactive waste systems, we considered: (1) the capability of the systems for keeping the levels of radioactivity in effluents "as low as is reasonably achievable" based on expected radwaste inputs over the life of the plant, (2) the capability of the systems to maintain releases below the limits in 10 CFR, Part 20 during periods of fission product leakage at design levels from the fuel, (3) the capability of the systems to meet the processing demands of the station during anticipated operational occurrences, (4) the quality group and seismic design classification applied to the equipment and components and structures housing these systems, (5) the design features that are incorporated to control the releases of radioactive materials in accordance with General Design Criterion 60 and (6) the potential for gaseous release due to hydrogen explosion in the gaseous radwaste system.

In our evaluation of the solid radioactive treatment system, we considered: (1) system design objectives in terms of expected types, volumes and activities of waste processed for offsite shipment, (2) packaging and conformance to applicable Federal packaging regulations, and provisions for controlling potentially radioactive air-borne dusts during baling operation, and (3) provisions for onsite storage prior to shipping.

In our evaluation of the process and effluent radiological monitoring and sampling systems we considered the system's capability: (1) to monitor all normal and potential pathways for release of radioactive materials to the environment, (2) to control the release of radioactive materials to the environment, and (3) to monitor performance of process equipment and detect radioactive material leakage between systems.

The quantities of radioactive materials that will be released in liquid and gaseous effluents and the quantity of radioactive waste that will be shipped offsite are provided in the Final Environmental Statement for Zimmer Nuclear Power Station, Unit No. 1. In making these determinations we considered waste flows, activity levels and equipment performance consistent with expected normal plant operation, including anticipated operational occurrences, for an assumed 30 years of normal plant operation.

The estimated releases of radioactive materials in liquid and gaseous effluents were calculated using the BAW-GALE Code described in NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWRs)", dated April 1976. The principal parameters used in these calculations along with their bases, are given in NUREG-0016.

In conformance with Section V.B of Appendix I to 10 CFR, Part 50, the applicant submitted, on June 4, 1976, information necessary to evaluate the capability of the Zimmer Nuclear Power Station, Unit No. 1, for keeping levels of radioactivity in effluents to unrestricted areas, "as low as is reasonably achievable". In these submittals, the applicant chose to comply with the Commission's September 4, 1975 Annex to Appendix I, in lieu of performing a cost-benefit analysis as required by paragraph II.D of Appendix I to 10 CFR, Part 50.

Based on the following evaluation, we conclude that the liquid and gaseous radioactive waste treatment systems for Zimmer Nuclear Power Station, Unit No. 1, are capable of maintaining releases of radioactive materials in liquid and gaseous effluents to as low as is reasonably achievable levels in accordance with 10 CFR, Part 50.34a, Sections II.A, II.B, II.C of Appendix I to 10 CFR, Part 50, and the optional alternative to the cost-benefit analysis required by Section II.D. of Appendix I as provided in the Annex to Appendix I.

Based on our evaluation, as described below, we find the proposed liquid, gaseous and solid radioactive waste systems and associated process and effluent radiological monitoring and sampling systems to be acceptable.

11.2 System Description and Evaluation

11.2.1 Liquid Radioactive Waste Treatment System

The liquid radioactive waste treatment system consists of process equipment and instrumentation necessary to collect, process, monitor and recycle or dispose of radioactive liquid wastes. The liquid radwaste system is designed to collect and process wastes based on the origin of the waste in the plant and the expected levels of radioactivity. All liquid waste is processed on a batch basis to permit optimum control of releases. Prior to being released, samples will be analyzed to determine the types and amounts of radioactivity present.

Based on the results of the analyses, the waste will be recycled for eventual reuse in the plant, retained for further processing, or released under controlled conditions to the environment.

The liquid radioactive waste treatment system consists of the equipment drain subsystem, floor drain subsystem, chemical waste subsystem, filter sludges and concentrates subsystem, and laundry waste subsystem.

The equipment drain subsystem processed the low conductivity wastes consisting of reactor building, radwaste building, and turbine building equipment drains waste, drywell equipment and floor drains waste, condensate and radwaste demineralizer backwash, and auxiliary building drains waste. The equipment drain subsystem consists of a 25,000 gallon collector tank, two 25,000 gallon surge tanks, a filter, three (one spare) mixed bed demineralizers in a series configuration, and three 20,000 gallon sample tanks. We estimated the equipment drain subsystem waste input flow to be approximately 17,500 gallons per day and assumed that five percent of the treated process steam will be released to the environment via the service water discharge canal. The remainder will be recycled to the condensate storage tanks for eventual reuse within the plant. The design capacity of the equipment drain subsystem is 432,000 gallons per day. The difference between the expected flow and design flow provides adequate reserve for processing surge flows.

The floor drain subsystem processes the high conductivity wastes consisting of reactor building, turbine building, and radwaste building floor drains waste. The floor drains subsystem consists of a 20,000 gallon collector tank, a 25,000 gallon surge tank, a filter, and two 20,000 gallon sample tanks. We estimated the waste input flow to the floor drain subsystem to be approximately 5,000 gallons per day. The waste will be filtered, sampled, and either discharged to the environment via the discharge canal or directed to the equipment drain subsystem or chemical waste subsystem for further treatment. We assumed that all the floor drains waste will be polished in the equipment drains subsystem and released to the environment. The design capacity of the floor drain subsystem is 432,000 gallons per day. The difference between the expected flow and design flow provides adequate reserve for processing surge flows.

The chemical waste subsystem will process condensate and radwaste demineralizer regenerants, shop decontamination solutions, reactor building and turbine building decontamination drains and cask cleaning drains. The chemical waste subsystem consists of two 30,000 gallon collector tanks, two 30 gallon per minute evaporators, and two 30,000 gallon monitor tanks. The design capacity of one evaporator (43,200 gallons per day) is sufficient, relative to the expected chemical waste subsystem input flow of 4,500 gallons per day, to handle surge flows. Treated chemical waste will be sampled and released to the environment via the discharge canal or directed to equipment drains subsystem for polishing or to the condensate storage tanks for reuse in the plant. We assumed that all of the treated chemical waste will be processed in the equipment drains subsystem and that 90 percent of the processed waste will be recycled for reuse in the plant. The remaining 10 percent will be discharged to the Ohio River through the service water discharge canal.

The filter sludges and concentrates subsystem will process filter sludges from the floor drain subsystem and equipment drain subsystem and filter/demineralizer sludges from the reactor water cleanup system and fuel pool cleanup system and collect evaporator concentrates from the chemical waste subsystem. The filter sludges and concentrates subsystem consists of two 3,500 gallon concentrate tanks, a 10,000 gallon floor drain sludge tank, a 10,000 gallon waste sludge tank, an 8,000 gallon fuel pool phase separator, and two 8,000 gallon cleanup phase separators. The evaporator concentrates and concentrated sludges will be sent to the solid radwaste system for solidification and eventual offsite shipment for burial. The clear supernatant liquid which is separated from the concentrated sludge will be pumped to the equipment drains subsystem for processing.

The laundry waste subsystem will process detergent wastes and personnel decontamination wastes. The laundry waste subsystem consists of two 1,500 gallon collector tanks, a filter, a four gallon per minute reverse osmosis unit, and two 1,500 gallon sample tanks. We estimated the waste input flow to the laundry waste subsystem to be approximately 450 gallons per day and assumed that all of the permeate will be released to the environment via the discharge canal. The design capacity of the reverse osmosis unit (5,760 gallons per day) is sufficient, relative to the expected input flow, to handle surge flows.

The seismic and quality group classification of the liquid radwaste equipment are based on criteria which were acceptable during the construction permit licensing stage. The liquid radioactive waste treatment system is located in the radwaste building. The design parameters of the principal components considered in the liquid radwaste system evaluation are listed in Table 11-1. We determined that the provisions incorporated in the design of the liquid radwaste equipment and structure housing the equipment conforms to Branch Technical Position, Effluent Treatment Systems Branch 11-1 (Revision 1), "Design Guidance for Radioactive Waste Management Systems Installed in Light-Water Cooled Nuclear Power Reactor Plants." (Replaced by Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light-Water Cooled Nuclear Cooled Nuclear Power Plants." We, therefore, find the applicant's radwaste building and liquid waste treatment system design to be acceptable.

The liquid radioactive waste treatment system is designed to control the release of radioactive materials due to overflows from tanks outside containment by

TABLE 11-1

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS CONSIDERED IN RADWASTE SYSTEM EVALUATIONS

Components	No.	Capacity Each	Quality Group	Seismic Design Classification
Equipment Drain Subsystem				
Waste collector tank	1	25,000 gallons	a	a
Waste surge tank	2	25,000 gallons	a	a
Waste demineralizer	3	300 gallons per minute	a	a
Waste sample tank	3	20,000 gallons	a	a
Floor Drain Subsystem				
Floor drain collector tank	1	20,000 gallons	a	a
Floor drain surge tank	1	25,000 gallons	a	a
Floor drain sample tank	2	20,000 gallons	a	a
Chemical Waste Subsystem				
Chemical waste tank	2	30,000 gallons	a	a
Waste evaporator	2	30 gallons per minute	a	a
Evaporator monitor tank	2	30,000 gallons	a	a
Filter Sludges and Concentrates Subsystem				
Concentrates waste tank	2	3,500 gallons	a	a
Floor drain sludge tank	1	10,000 gallons	a	a
Waste sludge tank	1	10,000 gallons	a	a
Fuel pool phase separator	1	8,000 gallons	a	a
Cleanup phase separator	2	8,000 gallons	a	a
Laundry Waste Subsystem				
Laundry drain tank	2	1,500 gallons	a	a
Reverse osmosis unit	1	4 gallons per minute	a	a
Laundry sample tank	2	1,500 gallons	a	a
Liquid Radwaste System				
Discharge tank	2	30,000 gallons	a	a
Process Offgas System				
Offgas preheater	2	b	a	a
Recombiner	2	b	a	a
Offgas condenser	2	b	a	a
Cooler condenser	2	b	a	a
Guard bed	2	b	a	a
Prefilter	2	b	a	a
Cyclic dryer	4	b	a	a
Gas cooler	2	b	a	a

TABLE 11-1 (Continued)

Charcoal bed	5	b	a	a
After filter	2	b	a	a
Vacuum pump	2	b	a	a

[a]In accordance with Branch Technical Position, Effluent Treatment Systems Branch 11-1 (Revision 1). (Replaced by Regulatory Guide 1.143.)

[b]The design basis flow rate in the process offgas system is based on 12.5 standard cubic feet per minute of air inleakage to the main condenser.

providing level instrumentation with alarm annunciation and overflow lines to collector tanks or sumps from which the waste may be transferred for treatment. We consider these provisions to be capable of preventing the uncontrolled release of radioactive materials to the environment.

We determined that during normal operation, including anticipated operational occurrences, the liquid radioactive waste treatment systems are capable of reducing the release of radioactive materials in liquid effluents to approximately 1.9 curies per year excluding tritium and dissolved gases, and 30 curies per year of tritium. Based on our evaluation, the radioactive materials in liquid effluents will not result in total body doses to an individual in an unrestricted area greater than three millirem per year or any organ dose greater than 10 millirem per year, in accordance with Section II.A of Appendix I to 10 CFR, Part 50. Also, the calculated release of radioactive material in the liquid effluents, exclusive of tritium and dissolved gases, will be less than five curies per year and the total body and any organ dose will be less than five millirem per year from the site, in accordance with the option to Section II.D of Appendix I as provided in the Annex to Appendix I. 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, Appendix I to 10 CFR, Part 50, and the Annex to Appendix I to 10 CFR, Part 50.

We determined that the liquid radwaste treatment systems will be capable of reducing the release of radioactive materials in liquid effluents to concentrations below the limits in 10 CFR, Part 20, during periods of fission product leakage from the fuel at design levels.

11.2.2 Gaseous Radioactive Waste Treatment System

The gaseous radioactive waste treatment system is designed to process gaseous wastes based on the origin of the wastes in the plant and the expected levels of radioactivity.

The gaseous waste treatment system consists of the process offgas system, mechanical vacuum pump offgas system, drywell purge system, gland seal condenser offgas system, and building ventilation system.

The process offgas system is designed to collect and delay fission product noble gases removed from the condenser by the air ejectors. In the process offgas system, the gas flows through a hydrogen recombiner, a condenser, a guard bed, a prefilter, a dryer, a gas cooler, three charcoal beds in series, and an after-filter. Except for the fifth charcoal bed, which is shared, the process offgas system consists of two separate trains of equipment which provide 100 percent redundancy in the processing of the gaseous wastes. The five charcoal beds will be maintained at 40 degrees Fahrenheit and contain five tons of charcoal each. We consider the system capacity and design to be adequate for meeting the demands of the station during normal operation, including ancitipated operational occurrences. The system design includes hydrogen analyzers upstream and downstream of the recombiner which will activate an alarm upon exceeding a present hydrogen concentration and indicate that switchover to the standby recombiner is required. In addition to the protective instrumentation, the pressure boundary of the process offgas system is designed to withstand a hydrogen explosion.

We find the design provisions incorporated to reduce the potential of hydrogen explosion and to mitigate the effects of an explosion to be acceptable.

The seismic and quality group classification of the process offgas system are based on criteria which were acceptable during the construction permit licensing stage. The process offgas system is located in the auxiliary building which is a seismic Category I structure. The parameters of the principal components considered in the process offgas system evaluation are listed in Table 11-1. We determined that the provisions incorporated in the design of the process offgas system conform to Branch Technical Position, Effluent Treatment Systems Branch 11-1 (Revision 1). (Replaced by Regulatory Guide 1.143.) We, therefore, find the process offgas system and structure housing the system to be acceptable.

The mechanical vacuum pump offgas system will be used during unit startup to remove air from the main condenser. The mechanical vacuum pump exhaust will be discharged directly to the atmosphere via the reactor building vent.

The drywell purge system is designed to process the drywell atmosphere through high efficiency particulate air filters and charcoal adsorbers prior to discharge to the reactor building vent.

The gland seal condenser offgas system will vent the noncondensable gases in the steam used for gland sealing purposes. The gases will be vented directly to the atmosphere via the reactor building vent. The steam used for gland sealing will be clean steam from the gland seal steam evaporator.

The plant building ventilation systems are designed to induce air flows from potentially less radioactive contaminated areas to areas having a greater potential for radioactive contamination. Ventilation exhaust from the radwaste building and laundry room will be processed through high efficiency particulate air filters prior to release to the reactor building vent. Ventilation air from the reactor, auxiliary and turbine buildings will be released without treatment to the reactor building vent.

We determined that the proposed gaseous radwaste treatment and plant ventilation systems are capable of reducing the release of radioactive materials in gaseous effluents to approximately 11,000 curies per year of noble gases, 0.5 curies per year of iodine-131, 30 curies per year of tritium, 25 curies per year of argon-41, 9.5 curies per year of carbon-14, and 0.06 curies per year of particulates.

Based on our evaluation, the radioactive materials in gaseous effluents will not result in an air dose to an individual in an unrestricted area greater than 10 millirads per year for gamma radiation, 20 millrads per year for beta radiation, or 15 millirem per year for radioiodine and radioactive particulates in accordance with Sections II.B and II.C of Appendix I to 10 CFR, Part 50. Also, the effluents from the site will not result in an annual gamma air dose greater than 10 millirads per year a beta air dose greater than 20 millirads per year, a release of iodine-131 greater than one curie per reactor, or a dose from radioiodine and radioactive particulates released greater than 15 millirem, in accordance with the alternative to Section II.D. of Appendix I as provided in the Annex to Appendix I. We conclude that the gaseous radwaste treatment system will reduce gaseous radioactive effluents to as low as is reasonably achievable levels in accordance with 10 CFR, Part 50.34a, Appendix I to 10 CFR, Part 50, and the Annex to Appendix I to 10 CFR, Part 50.

We determined that the gaseous radwaste treatment systems and plant ventilation systems will be capable of reducing the release of radioactive materials in gaseous effluents to concentrations below the limits of 10 CFR, Part 20 during periods of fission product leakage from the fuel at design levels.

11.2.3 Solid Radwaste Treatment System

The solid radwaste treatment system is designed to collect and process wastes based on their physical form and need for solidification prior to packaging. "Wet" solid wastes, consisting of spent demineralizer bead resins, evaporator bottoms, filter sludges, filter/demineralizer sludge, and laundry reverse osmosis concentrates will be combined with cement to form a solid matrix and sealed in the shipping containers. Dry solid wastes, consisting of ventilation air filters, contaminated clothing and 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 annually will be approximately 31,000 cubic feet of "wet" solid waste containing approximately 1800 curies and 4700 cubic feet of "dry" solid waste containing less than five curies total.

During the baling operation of compressible dry wastes, the air flow in the vicinity of the baler will be exhausted by a fan through a high efficiency particulate air filter to reduce the potential for airborne radioactive dusts.

The seismic and quality group classifications of the solid radwaste system are based on criteria which were acceptable for the construction permit licensing stage. The solid radwaste system is located in the radwaste building. We find the applicants solid radwaste system design to be acceptable in accordance with the guidance and criteria described in Branch Technical Position, Effluent Treatment Systems Branch 11-1 (Revision 1). (Replaced by Regulatory Guide 1.143.) We find that the solid radwaste system design conforms to the guidance and criteria in Branch Technical Position Effluent Treatment Systems Branch 11-3 (Revision 1), "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light Water-Cooled Nuclear Power Reactor Plants," except for Section B II, "Assurance of Complete Solidification." However, the applicant has committed to provide a process control program to satisfy Section B II; and, therefore, we find the solid radwaste system design will be in conformance with Branch Technical Position, Effluent Treatment Systems Branch II-3.

Storage facilities are provided for 18 liners (65 cubic feet) and 125 drums (55 gallons). We find the storage capacity adequate for meeting the demands of the station.

Wastes will be packaged in accordance with the requirements of 10 CFR, Part 20, 10 CFR, Part 71 and 49 CFR, Parts 170-178, and shipped to a licensed burial site in accordance with Commission and Department of Transportation regulations.

11.3 Process and Effluent Radiological Monitoring and Sampling Systems

The process and effluent radiological monitoring and sampling systems are designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in plant discharges to the environment. Certain liquid and gaseous streams will be continuously monitored for radioactivity. Monitors on selected effluent release lines automatically terminate discharges should radiation levels exceed a predetermined value. Table 11-2 indicates the proposed location, number, type, and sensitivity of each continuous monitor. Systems which are not amenable to continuous monitoring or for which detailed isotopic analyses are required will be sampled and analyzed in the plant laboratory. The sampling system will provide representative liquid and gaseous samples to effectively monitor the operation of the plant and provide isotopic analyses for determining the radioactive materials in liquid and gaseous effluents. Sample points are located at each tank in the liquid radwaste treatment system for sampling tank contents both before and after each processing step. In the gaseous radwaste treatment system, sample points are located at the inlet and outlet of the process offgas system charcoal bed trains and in the reactor building vent.

We reviewed the locations and types of effluents and process monitoring and sampling provided. Based on the plant design and on the continuous monitoring locations and sampling locations, we conclude that all normal and potential release pathways will be monitored. We also determined that the sampling and

TABLE 11-2

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEM

Stream Monitor	Number and Type	Monitor Classification	Monitor Sensitivity
Liquid			
Service water from turbine building concentrator condensers	1-Inline	-Scintillator	10^{-7} microcurie per milliliter (CS-137)
Reactor building closed cooling water system	2-Inline	-Scintillator	10-7 microcurie per milliliter (Cs-137)
Liquid radwaste system discharge	1-Inline	-Scintillator	10-7 microcurie per milliliter (Cs-137)
Service water from residing heat removal heat exchanger	2-Inline	-Scintillator	10-7 microcurie per milliliter (Cs-137)
Service water discharge	1-Offline	-Scintillator	10^{-7} microcurie per milliliter (Cs-137)
Gaseous			
Main steamline	4-Inline	-Ion Chamber	1 millirad per hour
Air ejected offgas (pretreated)	1-Offline	-Geiger-Muller	1 millirad per hour
Air ejector offgas (post-treated)	2-Offline	-Geiger-Muller	10 counts per minute
Off-gas vent pipe	1-Offline	*	*
Plant vent stack plenum	4-Inline	-Geiger-Muller	0.01 millirad per hour
Plant vent stack	1-Offline	*	*
Fuel pool vent plenum	[illegible]-Inline	-Geiger-Muller	0.01 millirad per hour

*Monitors for gross gamma activity; particulates and iodine are continuously collected and periodically analyzed.

monitoring provisions will be adequate for detecting radioactive material leakage to normally uncontaminated systems and for monitoring plant processes which affect radioactivity releases. On this basis we consider the monitoring and sampling provisions to meet the requirements of General Design Criteria 13, 60 and 64 and the guidelines of 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."

11.4 Evaluation Findings

In our evaluation, we calculated releases of radioactive materials in liquid and gaseous effluents for normal operation including anticipated operational occurrences based on expected radwaste inputs over the life of the plant.

In our evaluation we determined that the applicant's design of the liquid and gaseous waste treatment systems satisfies the design objectives set forth in rule making 50-2 specified in the option provided by the Commission's September 4, 1975 amendment to Appendix I and, therefore, meets the requirements of Section II.D of Appendix I of 10 CFR, Part 50.

We conclude that the liquid and gaseous radwaste treatment systems will reduce radioactive materials in effluents to "as low as is reasonably achievable" levels in accordance with 10 CFR, Part 50.34a and, therefore, are acceptable.

We considered the potential consequences resulting from reactor operation with a fission product release rate consistent with a noble gas release rate to the reactor coolant of 100 microcurie per megawatt thermal per 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, Appendix B, Table II.

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

We 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 conforms to the acceptance criteria and guidance as set forth in Branch Technical Position, Effluent Treatment Systems Branch 11-1 (Revision 1). (Replaced by Regulatory Guide 1.143.)

We 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 that the measures proposed by the applicant are consistent with our acceptance criteria as set forth in Branch Technical Position, Effluent Treatment Systems Branch 11-1 (Revision 1).

Our review of the radiological process and effluent monitoring systems included the provisions for sampling and monitoring all normal and potential effluent discharge paths in the conformance with General Design Criterion 64, for providing automatic termination of effluent releases and assuring control over releases of radioactive material in effluents in conformance with General Design Criterion 60 and Regulatory Guide 1.21 mentioned above, for sampling and monitoring plant waste process streams for process control in conformance with General Design Criterion 63, for conducting sampling and analytical programs in conformance with the guidelines in Regulatory Guide 1.21, and for monitoring process and effluent streams during postulated accidents. The review included piping and instrument diagrams and process flow diagrams for the liquid, gaseous, and solid radwaste systems and ventilation systems; and the location of monitoring points relative to effluent release points. 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 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 systems to the applicable regulations and guides referenced above, as well as to our technical positions and industry standards.

12.0 RADIATION PROTECTION

The Zimmer Nuclear Power Station Final Safety Analysis Report provides information on the methods for radiation protection including the facility design and layout, equipment design, and a description of the health physics program. This information includes an estimate of occupational radiation exposure to plant personnel, the shielding provided to reduce radiation levels, the ventilation arrangement to control the flow of potentially contaminated air and the radiation monitoring employed to measure levels of radiation in potentially occupied areas and to measure airborne radioactivity throughout the plant. A health physics program is provided for plant personnel and visitors during reactor operation, maintenance, refueling, radwaste handling and inservice inspections. We reviewed and evaluated the description and analysis of the radiation protection program included in the Final Safety Analysis Report and the 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 design and program features are consistent with the guidelines of Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable." It is the applicant's written policy to maintain occupational radiation exposure as low as is reasonably achievable. In response to requests by us, the applicant added extensive material to Chapter 12 of the Final Safety Analysis Report related to maintaining occupational radiation exposures as low as is reasonably achievable. On the basis of our review, we conclude that the applicant's radiation protection program is acceptable and will provide reasonable assurance that doses to personnel will be less than the limits established by 10 CFR, Part 20 and will be maintained as low as is reasonably achievable, consistent with the guidelines of Regulatory Guide 8.8. Details are discussed in the following sections.

12.1 Shielding

The applicant considered means to keep external and internal radiation exposures to personnel as low as is reasonably achievable, including both individual and total man-rem doses. Shielding is designed to control radiation exposure during operation such that (1) doses to plant personnel, contractors, and authorized site visitors will be as far below the limits in 10 CFR, Part 20 as practicable, in conformance with the suggested design features in Regulatory Guide 8.8; (2) doses to the offsite general public from direct and air-scattered radiation will be a small fraction of the limits in 10 CFR, Part 20.

The applicant provided five radiation zones as a basis for classifying occupancy and access restrictions on various areas within the plant site boundary. Maximum design dose rates are established for each zone and used as input for shielding of the respective zones in each building. For example, design radiation levels in operating areas where personnel are expected to be working for a 40-hour week will be less than 1.0 millirem per hour.

For each process system, the mode of operation with the highest expected radiation dose rate to personnel provided the basis for shielding design. The ANISN code was used to design the inner, sacrificial shield for neutron and gamma attenuation; all other shields were designed for gamma attenuation by the standard point attenuation kernel, numerically integrated over the volume of the source, using the ISOSHLD-II and OAD codes. Labyrinths and penetrations were analyzed by the albedo or Monte Carlo method. We consider the assumptions used in the shielding calculations to be conservative, and the models and codes used acceptable.

The magnitudes of all calculated radiation levels are based on failed fuel operation with the following source term assumptions: (1) plant operation at maximum power; (2) noble gas release rate from the core equivalent to 0.1 curie per second after 30 minutes decay; (3) concentrations in the reactor water based on fission product equilibrium halogen concentrations; and (4) concentrations of other fission products and activation products based on operating experience with boiling water reactors.

Radiation protection concepts directed to keeping personnel exposures below regulatory limits were used throughout in the design and construction of the plant. Shielding design and radiation zoning were based on whichever mode (operating or shutdown) involved the higher projected doses. To the extent practicable, major sources are in individually labyrinthed, shielded cubicles, with instrumentation outside the shielding. Valve stations are shielded, remotely operated, or provided with extension stems as practicable. Pipes and ducts are routed through high-zoned low-access areas where practicable; shielding is provided for pipe chases and penetrations, which are offset to minimize streaming.

The applicant has a radiation protection design review committee chaired by a Certified Health Physicist, with shield design engineers on the committee. This committee made several changes to reduce radiation exposure to personnel.

During our review, we requested the applicant to modify the area outside the mixed bed demineralizer cubicle on the 510 foot 6-inches level to allow for control of radiation fields from pipes containing spent resin pumped to the spent resin tank. The applicant made shielding and radiation zone modifications in that area. The applicant also committed to have the radiation protection design review group assure that similar situations did not exist in other parts of the plant. As a result of that group's review, another similar piping layout was identified and corrected. No other similar piping arrangements have been identified.

The applicant's area radiation monitoring system is designed to provide (1) indication in the control room of gamma radiation levels at selected locations where radioactive materials may be present or inadvertently introduced; (2) information necessary for decisions on deployment of personnel in the event of an accidental inplant release of radioactive material; (3) information necessary for detection of unauthorized or inadvertent movement of material in the plant, or abnormal migrations of radioactive material from plant process streams; (4) alarming for abnormal radiation conditions; (5) local alarms or indicators at all points where a substantial increase in radiation levels could affect personnel in the area; and (6) indication that a channel is inoperable. Locations were selected to include those in which personnel perform regular duties in radiation areas, or infrequent duties in areas where there is a high probability of significant changes in radiation levels; or any other area where surveillance is desired. The above objectives and location criteria are in conformance with 10 CFR, Parts 50 and 70, and are acceptable.

The applicant based the estimate of annual man-rem exposure experience from design and operation of other boiling water reactors, such as Nine Mile Point, Oyster Creek, and Quad Cities and on design features in the Zimmer plant intended to assure that occupational radiation exposures are maintained as low as is reasonably achievable. At currently operating modern boiling water reactors, total occupational annual exposures have been of the order of 400-500 man-rem. The applicant expects that through intense precautions taken to maintain occupational radiation exposure as low as is reasonably achievable that the overall annual exposure at Zimmer will be approximately 300 man-rem. The bases for the applicant's exposure estimates are reasonable, and consistent with the acceptance criteria in our Standard Review Plan.

We conclude, based on information presented in the Final Safety Analysis Report, that the applicant designed a facility to keep radiation exposures within the applicable limits of 10 CFR, Part 20. In his design and arrangement he has considered the recommendations of Regulatory Guide 8.8 to reduce unnecessary exposure during operations. Based on our review, we find that the shielding and arrangement of the plant are acceptable.

12.2 Ventilation

The plant ventilation system is designed to maintain a suitable environment for personnel and equipment. Among the design objectives of this system are the protection of operating personnel from possible airborne radioactivity and the assurance that maximum expected airborne radioactivity concentrations will be maintained within the limits of 10 CFR, Part 20. These design objectives are acceptable. To meet these objectives, several design criteria are used including: (1) air-flow from areas of least radioactive contamination to areas of progressively greater radioactive contamination followed by exhaust to ventilation ducts; and (2) maintenance of slight negative pressures in selected areas. These design criteria are in accordance with the recommendations of Regulatory Guide 8.8 and, therefore, are acceptable. The drywell and suppression pool chamber purge system will provide a means of reducing the airborne contamination to allow personnel access into the drywell. The radwaste building ventilation system provides flow to various cubicles to maintain potential airborne radioactivity levels from flowing into noncontaminated areas.

The bases and methods of estimating sources of airborne radioactivity in the plant and expected levels of airborne concentrations in various areas of the plant are described in Section 12.2.3 of the Final Safety Analysis Report. These bases, including rates of leakage and partition factors, are conservatively determined and the calculations result in concentrations of airborne radioactivity below the requirements to 10 CFR, Part 20.

To assure compliance with the standards of 10 CFR, Part 20, the applicant has an airborne radioactivity monitoring system. This system consists of a fixed continuous air monitoring system, a fixed air sampling system, a semi-portable continuous air monitoring system and portable air sampling equipment. These monitors will provide sufficient sensitivity to detect one maximum permissible concentration in one hour in most cubicles. Access to rooms for which the one maximum permissible concentration in one hour can not be determined will be preceded by a measurement of airborne radioactivity. We consider the airborne radioactivity monitoring program acceptable.

Based on our review we determined that the ventilation system, as described in detail in Section 9.4 of the Final Safety Analysis Report, meets the radiation protection design objectives, considers the recommendations of Regulatory Guide 8.8, and will maintain doses from airborne radioactive materials below the limits of 10 CFR, Part 20. We conclude that the ventilation system is acceptable.

12.3 Health Physics Program

The Health Physics Program objective is to provide administrative control of onsite personnel to assure that occupational radiation exposures are within the limits of 10 CFR, Part 20, and are as low as is reasonably achievable, consistent with the intent of Regulatory Guide 8.8 and other applicable Regulatory Guides. The Rad-Chem Supervisor has the responsibility for administering this program. He influences decisions on plant operation that can affect the radiation safety of workers, in that he is a member of the Station Review Board and he reports directly to the Plant Superintendent.

The Health Physics Program is designed to ensure that: (1) operations, maintenance, and technical personnel, are trained to the extent required for their duties, consistent with 10 CFR, Parts 19 and 20, and Regulatory Guide 1.8, "Personnel Selection and Training"; (2) detailed procedures are prepared and approved for all aspects of the Radiation Protection Program; (3) appropriate access control procedures are followed to separate potentially contaminated areas from clean areas, and that positive control is provided for each entry in a high radiation area; (4) potential transfer of radioactive contamination is controlled by monitoring personnel, equipment, tools, and clothing; (5) radiation levels are measured and posted and personnel are monitored and provided with appropriate bioassay; (6) complete radiation exposure records are maintained; and (7) personnel access to high radiation areas, and maintenance work in radiation areas are controlled by use of a Radiation Work Permit, which must be approved by the Rad-Chem Group, and which specifies any special requirements for the job.

The radiation protection facilities include access control points, high and low level laboratories, counting room, instrument calibration room, offices, decontamination and laundry area, and change room.

Based on our review, we conclude that these facilities are sufficient to maintain occupational exposures as low as is reasonably achievable, and are consistent with Regulatory Guide 8.8.

The applicant will provide equipment to be used for radiation protection which includes: protective clothing, respiratory protective equipment, air sampling equipment, portable radiation measuring instruments, calibration sources, counting room instrumentation, area monitors, airborne activity monitors, laboratory equipment and special shielding materials.

Based on our review, we conclude that the numbers and types of this equipment will be adequate to provide reasonable assurance that exposures to personnel can be maintained as low as is reasonably achievable.

All persons entering a controlled area are provided with a TLD to monitor beta-gamma radiation, persons entering a radiation area are also provided with a self-reading dosimeter. A neutron badge is provided for personnel who enter areas where the rate exceeds five millirems per hour or could exceed 15 millirems during a given month. The personnel neutron dosimetry program will be conducted in accordance with Regulatory Guide 8.14, "Personnel Neutron Dosimeters." Finger rings, wrist badges, or other dosimeters, as well as alarming dosimeters, are provided as appropriate. Whole body counting will be performed periodically to assess intake of radioactive materials. Bioassay may also be used as necessary.

Based on the information provided in the application, and the responses to our requests, we conclude that the applicant is implementing a radiation protection program that is acceptable and will keep radiation exposures as low as is reasonably achievable.

13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Qualifications

Zimmer Nuclear Power Station, Unit 1, operational activities are conducted under the onsite supervision of the Station Superintendent. He reports to the Manager Electric Production who in turn reports to the Vice President of Engineering Services and Electric Production.

The Station Superintendent is responsible for the safe, efficient, and reliable operation of the station. Reporting to the Station Superintendent is an Assistant Superintendent and a staff of approximately 100 full-time employees. Reporting to the Assistant Superintendent is a radiation/chemistry group responsible for plant radiation protection and chemistry with a staff of approximately 18 persons; an instrument and controls group responsible for instrument and controls maintenance with a staff of approximately 13 persons; a maintenance group responsible for all electrical and mechanical maintenance at the station with a staff of approximately 20 persons; a technical engineering group responsible for reactor engineering and other general engineering with a staff of approximately eight persons; and an operations group responsible for the day-to-day operation of the plant with a staff of approximately 25 persons. In addition, a Training Coordinator reports to the Assistant Superintendent. A Station Quality Engineer reports directly to the Station Superintendent.

Reporting to the Operating Supervisor of the operations group are the plant operating shifts. The shift crew for plant operation will consist of five persons, one of whom will be a licensed senior operator and two of whom will be licensed operators. In addition, a radiation chemist will be onsite at all times.

We reviewed the qualification requirements for station personnel described in Section 13.1 of the Final Safety Analysis Report and finds they meet those qualifications described in American National Standards Institute N18.1-1971, "Standard for the Selection and Training of Personnel for Nuclear Power Plants." We reviewed the qualifications of key personnel assigned to the Zimmer Station and found them acceptable, except the position of Maintenance Supervisor which is vacant and the position of Reactor Engineer. Their qualifications with regard to educational backgrounds, experience, and technical specialties meet those described in American National Standards Institute N18.1. The applicant will augment the position of Reactor Engineer with an individual who meets the qualification requirements for that position for the initial test program. Upon completion of the initial test program, the individual currently assigned the position of

Reactor Engineer will then have the necessary experience requirements and qualifications. We find this approach acceptable. The applicant has committed to fill the position of Maintenance Supervisor by March 1, 1979, and the qualifications of this individual will be reviewed at that time.

Offsite technical support for the plant staff will be provided by personnel from the Cincinnati Gas and Electric Company Electric Production Department, Licensing and Environmental Affairs, and General Engineering Department. Health physics support will be provided from outside Cincinnati Gas and Electric Company by contract.

We conclude, subject to the satisfactory review of the qualifications of the individual assigned to the position of Maintenance Supervisor, that the organizational structure and qualifications of the plant personnel meet Regulatory Guide 1.8, Rev. 1, "Personnel Selection and Training," and are satisfactory to provide an acceptable operating staff. We further conclude that the applicant has the necessary resources or has made acceptable arrangements to secure offsite technical support for the operation of the facility.

13.2 Training Program

The Station Superintendent of the Zimmer Plant has the overall responsibility for the selection and training of plant personnel. At the plant, the day-to-day administration of the training program is carried out by the Training Coordinator.

The replacement and retraining ·nforms to the requirements of 10 CFR, Part 50, 10 CFR, Part 55, Appendix A, and follows the guidance given in American National Standards Institute Standard N18.1.

Complete records of all training administered will be maintained.

All station personnel not requiring Commission licenses receive General Retraining, as applicable to their normal duties, consisting of appropriate plans and procedures, radiological health and safety, station security procedures, and the emergency plan.

A training program is also conducted for refresher training of professional technical personnel and for technicians and maintenance personnel.

All new employees will receive training in radiation safety, emergency plan, security, quality assurance, industrial safety and job functions.

On the basis of our review, we conclude that the training programs meet the requirements for approval.

13.3 Emergency Planning

In response to our requests Q422.1 through 422.13, in Revisions 33, 36, 37 and 42 to the Final Safety Analysis Report dated September 30, 1977, November 30, 1977, December 16, 1977, and April 28, 1978 (the latter being a complete update) the applicant submitted revised emergency plans for the Zimmer station, designated as Appendix F to the Final Safety Analysis Report. We reviewed these plans against the requirements of Appendix E to 10 CFR, Part 50, Parts I, III and IV (the applicable regulations). We find that these plans exceed the minimum requirements of the regulations such that there is reasonable assurance that appropriate measures can and will be taken in the event of an emergency to protect public health and safety and prevent damage to property. The basis for these findings are summarized below following the outline of the minimum requirements specified in Part IV of Appendix E.

The organization for coping with emergencies has been developed around normal organizational structures, i.e., the shift operating staff under the shift supervisor and station superintendent, the Cincinnati Gas and Electric Company organization and local, State and Federal government agencies, and local support services normally responsible for the protection of the public. The shift operating staff has been assigned the authority and responsibility for immediate actions, including activating major Cincinnati Gas and Electric Company organizational response and off-shift station personnel, and initiating support requests of civil and private services as necessary or desirable.

The station organizational chart is shown in Figure 13.1-2 of the Final Safety Analysis Report. Relationships between the station organization and local, State and Federal agencies and local support services are illustrated in Final Safety Analysis Report, Figure F-1, reproduced here as Figure 13-1. As indicated in this figure, arrangements have been made for medical, fire, police, transportation and radiological support services in an emergency. Draft State and local agency emergency plans and letters documenting emergency planning and emergency response agreements reached by Cincinnati Gas and Electric Company with local, State and Federal agencies and local support services are reproduced in Appendix F of the Final Safety Analysis Report. We reviewed these documents for evidence of compatibility of the plans of the applicant and interfacing agencies and support services and find that our minimum acceptance criteria stated in the Standard Review Plan 13.3, "Emergency Planning," are met or exceeded.

Means for notification of various elements of the emergency response organization include the station telephones, public address system, sirens and alarms, "pagers" for contact with the designated Emergency Duty Officer, the utility communication system, and radio communication capabilities for contact with primary local offsite support agencies and station monitoring teams. A message verification scheme will be developed and tested for emergency contacts with primary offsite support agencies.

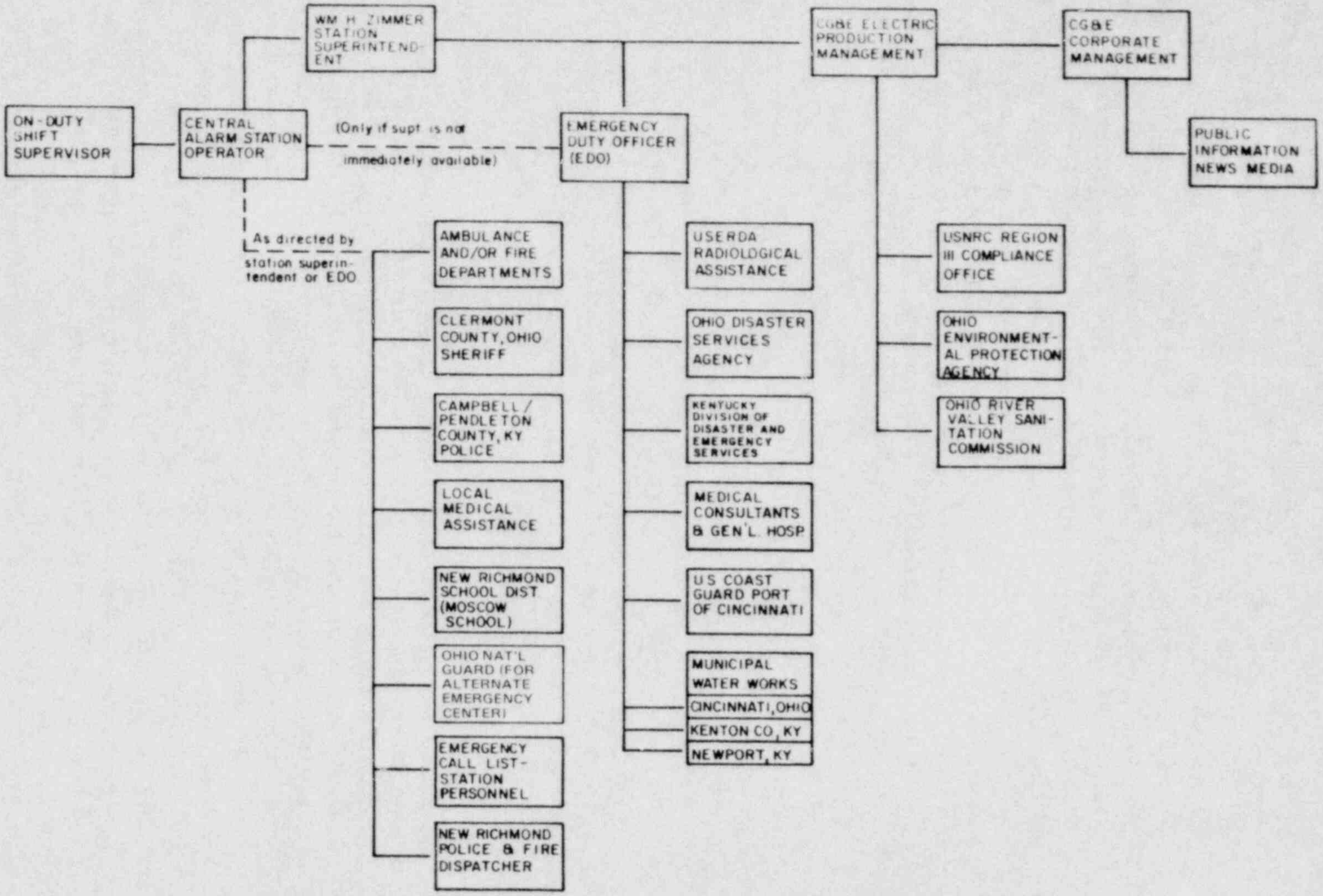

FIGURE 13.1 OFFSITE ORGANIZATION NOTIFICATION SEQUENCE

Telephone and radio communications facilities would be available at the offsite emergency control center at the Ohio National Guard Armory near Felicity, Ohio should it be activated.

Means for determining the magnitude of releases of radioactivity include routine analyses of inventories in plant systems, fixed area and portable radiation detectors, and fixed monitors in the local environs. Information from process and station-status monitors (e.g., temperature, pressure, water level, flow rate, valve status, and meteorological instruments) would also be available. A variety of possible emergency conditions, means of detection, immediate actions and required notifications have been considered in the emergency plans (cf. Table F-2 of the Final Safety Analysis Report). Incidents considered include natural phenomena (floods, earthquakes), fires, personnel injuries and abnormal releases of toxic chemicals and radioactivity. For each postulate the applicant described a probable detection means, an emergency classification and the specific initial action required, including notifications of various parts of the emergency organization as appropriate. For example, notification of offsite agencies would occur if a release rate in excess of a technical specification were detected, mobilization of offsite agencies would be recommended at a projected whole body dose level of 50 millirem offsite, and evacuation of persons in the environs would be recommended at a projected whole body dose level of five rem or a thyroid dose of 25 rem at the site boundary. Details of these criteria are presented in Final Safety Analysis Report Table F-3. The applicant developed action levels relating in-plant measurements to these projected dose criteria and meteorological parameters, by which specific levels of emergency would be declared and specific actions would be initiated. These pre-determined action levels are illustrated in Figures F-2 and F-3 of the Final Safety Analysis Report. For the particular case of the most serious accident conservatively analyzed for siting purposes, i.e., the design basis loss-of-coolant accident, the applicant projected that a whole body dose of five rem would not be exceeded offsite for two hours after the initiating event; for this same postulated incident the thyroid dose is projected to exceed twenty-five rem (via inhalation), offsite, after one and one-half hours. The United States Environmental Protection Agency recommends that protective actions be seriously considered given projected doses of these magnitudes; as noted above, the applicant would recommend protective actions at these levels, and mobilization of offsite agencies well below these levels. Appropriate protective actions offsite would be initiated by the cognizant local authorities under the general directions of the County Sheriffs. Because of the history of floods in the area, the local and State agencies have had substantial experience responding to emergencies, including the conduct of evacuations.

Provisions for maintaining the plans up to date include biennial renewal of agreements between the applicant and offsite agencies and local support services, annual review of the emergency plans by the station review board, quarterly inventory and maintenance of station emergency supplies and equipment, and training programs and drills for station and offsite agency and support services personnel. The

Office of Inspection and Enforcement of the Nuclear Regulatory Commission staff will conduct at least biennial audits of the emergency plans and procedures, including interviews with offsite agencies and support services. The applicant committed to cooperating with offsite agencies in the conduct of a coordinated drill before loading fuel.

The plans provide for first aid, ambulance, medical and decontamination supplies and services. First aid and decontamination rooms are located adjacent to the turbine room and machine shop in the store room (n.b. Final Safety Analysis Report Figure F-12). Typical supplies and equipment in these areas are listed in Final Safety Analysis Report Tables F-10 and F-11. The station will maintain its own ambulance onsite, but arrangements have been made by the applicant for backup ambulance support with the Moscow and New Richmond Life Squads. Medical and hospital services are coordinated by Cincinnati Gas and Electric Company's medical doctor.

Arrangements have been made by the applicant for receipt and treatment of injured personnel at Christ Hospital and Bethesda Hospital, and for injuries including radiation exposure or contamination, at the Radioisotope Laboratory of Cincinnati General Hospital. Radioisotope Laboratory has established a decontamination suite in the basement of Logan Hall for treatment of contaminated patients, including minor surgery. Final Safety Analysis Report Figures F-8, F-9 and F-10 show the locations of these hospitals with respect to the Zimmer station.

In addition to the Cincinnati Gas and Electric training discussed in Section 13.2 of the Final Safety Analysis Report, training specifically related to emergency planning is discussed in the Final Safety Analysis Report, Appendix F, Section F7.1. Such includes annual retraining of emergency duty supervisors and control room shift personnel, including presentations and walk-throughs of accident assessment and immediate and supplementary actions in an emergency. Emergency team members participate in training and retraining programs which consist of presentations and practical factors in first aid, firefighting, damage control and rescue. Coordinated training will be offered to selected offsite agencies and support services personnel, including annual site familiarity training for local fire department personnel.

At least annually, announced emergency drills will be conducted to test the adequacy of training and content of specific implementing procedures and to test emergency equipment. All drills will be critiqued. Coordinating drills will be held annually with participating offsite agencies, testing as a minimum the communication links and a warning authentication scheme.

General guidelines and criteria have been developed to determine when, following an accident, reentry of the facility may be appropriate, or operation may be continued. When emergency actions can be taken to alleviate a potentially hazardous situation, e.g., to prevent substantial and extreme loss of property, eliminate further escape of effluents, or to control fires, whole body exposures of 12.5 rem could be allowed for individuals participating in corrective actions. An exposure limit for voluntary entry of areas to remove injured persons is 75 rem. Limits for emergency personnel performing emergency first aid, decontamination, transportation or medical treatment services to injured persons are 25 rem whole body and 125 rem thyroid. General guidelines for recovery and reentry are as follows: all action would be preplanned, every reasonable effort would be made to limit radiation exposures, risk vs benefit factors of the particular situation would be considered, and reentry of a contaminated area would be supervised and authorized by Cincinnati Gas and Electric management. Continued or renewed operation of the facility during or after an emergency would be governed by the Nuclear Regulatory Commission regulations and the operating license conditions for the facility.

We reviewed the Zimmer Power Station Emergency Plan (Section 13.3 and Appendix F of the Final Safety Analysis Report) and the applicant's fire protection systems reviews submitted under cover letters dated February 4, 1977, September 30, 1977, November 31, 1977 and November 30, 1977 and find that these plans include measures for coping with fire emergencies that conform with the applicable provisions of Regulatory Guide 1.101, Revision 1, "Emergency Planning for Nuclear Power Plants." In particular a satisfactory written agreement is in effect with the Washington Township Fire Department, Moscow, Ohio, which assures the availability of additional trained personnel and equipment for fire fighting support when called upon. In addition, the applicant has provided for annual training of these personnel to assure their necessary familiarity with the plant, access procedures, and radiation protection precautions, and for their participation in an annual drill or test exercise. (We note that the Clermont County (Ohio) emergency plan calls for the New Richmond Fire District to provide backup support to the Washington Township Fire Department, via mutual aid agreements between area fire departments).

In summary, we conclude that the Zimmer Power Station Emergency Plan meets or exceeds our requirements contained in Regulatory Guide 1.101, Rev. 1 and Appendix E to 10 CFR, Part 50 and that the protective actions proposed therein are feasible including evacuation, if necessary.

13.4 Review and Audit

The applicant described proposed provisions for the review and audit of plant operations. They include the Station Review Board that will provide a continuing review of plant operations and the offsite Operations Review Committee that will provide an independent review and audit of plant operations. We reviewed the provisions for the review and audit of plant operations and finds that they meet those described in Section 4 of American National Standards Institute N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," and are acceptable. The details of these provisions will be included in the facility Technical Specifications.

13.5 Station Procedures

All safety-related operating, maintenance testing and modification activities are conducted in accordance with approved, written procedures meeting the requirements of Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operations)," and American National Standards Institute N18.7-1972. Areas covered include system operating procedures, plant operating procedures, special procedures, alarm response procedures, procedures performed by nonlicensed personnel including maintenance and testing activities and administrative control procedures. The applicant's provision meets the requirements of 10 CFR, Parts 50.54(i), (j), (k), (l) and (m). Procedures addressing activities associated with safety-related structures, systems and components are forwarded to the Station Review Board for review and comment. Upon final approval by the Station Superintendent, a procedure becomes available for use.

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

13.6 Plant Records

The applicant has stated that he will maintain plant records in accordance with Regulatory Guide 1.88, "Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records," and we find this acceptable.

13.7 Industrial Security

The applicant submitted a security plan on September 10, 1975. We reviewed the plan and three subsequent revisions to the plan submitted between July 30, 1976 and November 19, 1976 and conclude that the security plan as amended is in conformance with existing criteria (i.e., Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage").

In conformance with 10 CFR, Part 73.55, the applicant submitted an amended security plan dated May 25, 1977. This amended security plan has been reviewed and evaluated by us and a security plan review team will visit the plant site as part of this overall evaluation. The results of the our evaluation have been discussed with the applicant so that the amended security plan can be modified, as needed, to meet the performance requirements of Part 73.55. We will report on this matter in a supplement to this report.

In accordance with 10 CFR, Part 73.55, full implementation of the Zimmer security plan will be required prior to granting an operating license.

14.0 INITIAL TEST PROGRAMS

14.1 Initial Tests and Operations

We reviewed the information provided in the Final Safety Analysis Report (through Amendment 82, Revision 51) pertaining to the applicant's initial test program. This review included an evaluation of:

(1) The applicant's organization and staffing for the development, conduct, and evaluation of the test program.

(2) The qualifications and experience of the principal participants managing and supervising the test program.

(3) The administrative controls that will govern the development, conduct, and evaluation of the test program.

(4) The degree of participation of the plant operating and technical staff in the test program.

(5) The applicant's requirements pertaining to the trial-use of plant operating and emergency procedures during the test program.

(6) The schedule for conducting the test program.

(7) The sequence of testing to be followed.

(8) The methods for conducting individual tests and the acceptance criteria to be used in evaluating the test results for plant structures, systems and components.

(9) The test program's conformance with applicable Regulatory Guides including 1.20 (June 1975), "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing"; 1.41 (March 1973), "Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments"; 1.52 (June 1973), "Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants"; 1.68 (November 1973), "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors"; 1.68.1 (January 1977), "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power

Plants"; and, 1.80 (June 1974), "Preoperational Testing of Instrument Air Systems."

Our review also included an evaluation of the applicant's method for review of reactor plant operating experiences that is being conducted to determine where improvement or emphasis may be warranted in his initial test program. We conclude, with the exceptions noted below, that the information provided in the application shows that an acceptable initial test program will be conducted in accordance with the acceptance criteria in Section 14.2 of the Standard Review Plan.

14.1.1 Preoperational Test Program

The applicant's proposed tests of the essential direct current systems do not include demonstration of the capability of essential loads to operate at the direct current systems design bases minimum voltage level. The applicant submitted some justification for omitting this testing in Final Safety Analysis Report, Revision 46 (Amendment 76). We reviewed this information and concluded that the justification is not adequate. We will require either that the applicant include these demonstrations in his preoperational tests of the 125 volt and 250 volt direct current systems or provide further technical justification for their omission. We will report resolution of this matter in a supplement to this report.

14.1.2 The Startup Test Program

Because of changes to the test program recommended by the General Electric Company, the applicant had not submitted revised descriptions of several of the startup tests until recently. Therefore, we have not completed our review of this phase of the test program. We will complete our evaluation of the startup test program and report the results in a supplement to this report.

15.0 ACCIDENT ANALYSIS

Introduction

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 affect 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 will be 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 (cladding) damage will occur and that peak nuclear vessel pressure will not exceed 110 percent of the design pressure (American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Class I requirements are met if nuclear system pressure remains below 1375 pounds per square inch gauge, which is 110 percent of the 1250 pounds per square inch gauge design pressure). These two requirements will demonstrate, respectively, that the first radioactive material barrier (the cladding) and the second barrier (the pressure vessel) will be 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 reactor coolant pressure boundary occur. This is done by insuring that peak fuel enthalphy remains below 280 calories per gram, the limit used in Regulatory Guide 1.77, "Assumption Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," for the pressurized water reactor rod ejection accident analysis and accepted by us for use as a fuel safety limit for boiling water reactors. The 280 calories per gram energy density value will provide 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 cladding temperature must remain below 2200 degrees Fahrenheit, as stated in 10 CFR, Part 50.46 for the loss-of-coolant accident analysis. The Zimmer reactor 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 4.4 and Section 6.3 of the Final Safety Analysis Report.

15.1 Abnormal Operational Transients

The applicant analyzed several events expected to occur one or more times in the life of the plant. It is to be demonstrated that all of these events are terminated without exceeding specified acceptable fuel design limits (minimum critical power ratio remains greater than 1.07) and the reactor coolant pressure stays below 110 percent of design) as required by General Design Criteria 20 and 15.

The applicant used the following conservative assumptions:

(1) Steam flow rate corresponding to 105 percent of design;

(2) Fuel burnup conditions which provide conservative results;

(3) Scram reactivity based on 80 percent of total rod worth which accounts for a stuck control rod as required by General Design Criterion 26; and,

(4) Void feedback modified to produce conservative power effects.

The transients were analyzed with the methods described in NEDO-10802, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," which is still under consideration by us. In this regard, three turbine trip tests were performed at the Peach Bottom, Unit 2 plant. The purpose of the tests was to provide experimental data for code verification and to improve the understanding of integral plant behavior under transient conditions. The results from this program have raised some questions about the analytical methods since not all the test data were conservatively predicted by the current licensing methods (As discussed in subsection 4.4.1 of this report). We will report resolution of this matter in a supplement to this report.

The transients analyzed were protected by the following reactor scrams in accordance with General Design Criterion 20:

(1) Nuclear system high pressure;

(2) Reactor vessel low water level;

(3) Turbine stop valve closure;

(4) Turbine control valve fast closure;

(5) Main steam line isolation valve closure; and,

(6) Neutron monitoring system scram.

Time delays to trip for each scram signal were included in the analyses.

The recirculation flow control system was reviewed by us as discussed in subsection 7.7.3 of this report.

The series of postulated events which involve loss of steam flow to the turbine are: (1) generator load rejection; (2) turbine trip; (3) main steam isolation valve closure; and (4) loss of condenser vacuum. The main steam isolation valve closure results in the largest reactor coolant system overpressure transient (1199 pounds per square inch gauge--which is less than 110 percent design pressure).

The applicant's analysis of the generator load rejection without bypass results in the largest reduction in the minimum critical power ratio (reduction of 0.17). Based on these results, the minimum operation limit minimum critical power ratio is 1.24. For this minimum initial operating minimum critical power ratio (1.24), turbine trip without bypass results in a minimum transient value of 1.09. We conclude from our review that the applicant's analysis is acceptable.

There are two transients which could cause unplanned additions to coolant inventory. One, is the actuation of the high pressure core spray system the other is increase in feedwater flow rate. High pressure core spray system actuation has a negligible effect because high pressure core spray system flow is small compared to recirculation flow. When feedwater flow is increased because of maximum flow demand, reactor water level is increased which trips the turbine and feedwater pumps. Initiation of scram is subsequently caused by turbine stop valve closure. The reactor core isolation cooling system and high pressure core spray system are actuated subsequently when low water level is reached. The applicant stated that the minimum critical power ratio is not limiting for this event. We agree with the applicant's statement based on our review.

The transients which result in increased cooling are: (1) loss of feedwater header, (2) recirculation flow controller failure, increasing flow. In the first transient scram is initiated on high average power range monitor power; the minimum critical power ratio for this transient is calculated to be 1.11 which is acceptable.

The Zimmer Final Safety Analysis Report contains analysis of two similar events in the category of flow controller failure resulting in increased recirculation flow - opening of one or both of the recirculation system flow control valves. As a result of the increase in core flow, neutron flux attains a level of 164 to 237 percent of rated value, causing a high flux scram, but surface heat flux never exceeds 75 percent. The minimum critical power ratio remains above the safety limit and the increase in reactor coolant system pressure is slight. These results comply with applicable criteria and are acceptable.

Transients which involve reduction in coolant inventory include: (1) pressure regulator failure (Open), (2) inadvertent safety/relief valve opening, (3) loss of feedwater flow.

An open pressure regulator would cause excessive steam flow to the turbine, resulting in system depressurization, formation of voids with consequent swelling of the water level in the reactor vessel. The resultant level increase may cause turbine trip or the main steam isolation valves may close when turbine pressure becomes less than 825 pounds per square inch absolute. Minimum critical power ratio is reported to remain above 1.24 while the pressure rise is not significant.

Inadvertent safety/relief valve opening causes the plant to operate at a lower pressure and a reduced power level. Changes in minimum critical power ratio are calculated to be negligible. No immediate operator action is required; however, the operator must eventually decide on whether the valve can be closed or he must shut the plant down. We conclude that the applicable criteria have been met, thus the transient is found to be acceptable.

Loss of feedwater causes the coolant level in the reactor vessel to drop which actuates the high pressure core spray system and reactor core isolation cooling system; the main steam isolation valves close and recirculation pumps trip. There is no significant change in fuel thermal margins. Pressure in the reactor vessel remains below 110 percent design pressure because of safety valve operation. The results of the transient are found acceptable because they satisfy all appropriate design criteria.

Partial and complete loss of flow transients include: (1) single pump trip, (2) two pump trip, and (3) recirculation flow control failure - decreasing flow.

Two pump trip may occur as a result of loss of power. As the pumps coastdown, voids are formed causing level swell in the reactor vessel and the neutron flux to decrease. Turbine trip is assumed to occur because of high water level. The applicant reports that the minimum critical power ratio remains above the operating limit (1.24). The pressure rise calculated during the initial portion of the transient is limited by the assumed opening of the turbine bypass valve.

In order to confirm the transient flow coastdown behavior analyzed above, pump coastdown characteristics will be verified during preoperational testing. Any deviations which would suggest a nonconservative analyses of transients or accidents must be explained.

The transient initiated by recirculation flow control failure, as reported by the applicant, is not as severe as two pump trip and is therefore, acceptable.

There are three transients which could result in reactivity insertions: (1) continuous rod withdrawal at power, (2) continuous rod withdrawal at startup, and (3) startup of an idle recirculation loop.

The rod block monitor prevents rod withdrawal beyond the point at which the minimum critical power ratio is 1.08 during power operation. At startup, the average power range monitors and intermediate range monitors provide adequate protection against an out-of-sequence rod withdrawal. Both rod withdrawal incidents are, therefore, prevented from causing fuel damage by violating minimum critical power ratio limits. Since there is no pressure effect in these transients, the results are acceptable. Startup of an idle recirculation loop is not a limiting transient.

Loss of auxiliary power results in reduction of condenser vacuum, in turn, turbine trip. Subsequently, complete loss of condenser vacuum results in main steam isolation valve closure and scram. The results of the transient shows no significant change in fuel margin. System pressure, relieved by relief mode operation of the safety/relief valves results in pressure increases less than 100 pounds per square inch above normal operating pressure.

Loss of instrument air is reported to cause a normal plant shutdown. This transient could occur as a result of air line rupture, loss of station auxiliary power and offsite power. No design safety limits are violated and, thus, the transient is acceptable.

We are currently reviewing the use of certain nonsafety-grade equipment used to mitigate the consequences of some abnormal operational transients such as feedwater flow control failure on a generic basis. The conclusions reached from this review, which may affect the operating minimum critical power ratio, will be discussed in a supplement to this report. In order to assure an acceptable level of performance for Zimmer, our position is that the equipment relied upon to mitigate the most limiting transient. the excess feedwater event, namely the turbine bypass system and level 8 highwater level trip, be identified in the plant technical specifications with regard to availability, setpoints and surveillance testing. The applicant has been requested to submit his plan for implementing this requirement along with any system modifications that may be required to fulfill the requirements.

We find the results of the analyses of abnormal operating occurrence acceptable pending completion of the generic review of credit taken for nonsafety-grade equipment.

15.2 Accidents

In our review we noted that the inputs for recirculation pump trip and reactor scram following generator load rejection or turbine trip originated in the turbine building which is not seismically qualified. The applicant was required to analyze the consequences of a safe shutdown earthquake event without taking credit for nonseismically qualified equipment including reactor scram or recirculation pump trip from these trip inputs and, in the case of load rejection, including the effect of turbine overspeed on recirculation flow. For the worst case, the analysis showed that six percent of the fuel rods would experience boiling transition. We calculated the radiological consequences of this event assuming that six percent of the fuel rods perforated and found that the resulting doses to the public (subsection 15.3.7 of this report) are a small fraction of the dose guidelines of 10 CFR Part 100 and are, therefore, acceptable.

The applicant analyzed a pump shaft seizure accident. Reactor scram is sufficient to preclude violating the safety limits minimum critical power ratio (1.07) and therefore no fuel damage occurs. The reactor vessel pressure decreases throughout the event. The results of these analyses are acceptable since neither fuel or primary system behavior were predicted to violate our criteria.

The General Electric Company topical report NEDO-20626, "Studies of BWR Designs for Mitigation of Anticipated Transients Without Scram," is a generic study of methods to mitigate the consequences of anticipated transients without scram. The results of our review of this report are presented in WASH-1270, "Status Report on Anticipated Transients Without Scram," dated December 9, 1975.

The status report specified additional analyses and design changes needed to meet the safety objectives of WASH-1270. In September 1976 General Electric Company submitted additional generic analyses in conformance with the Status Report requirements.

A reevaluation of the potential risks from anticipated transients without scram has been published in NUREG-0460, Volumes 1 and 2, "Anticipated Transients Without Scram for Light Water Reactors." The status of NUREG-0460 is described below:

(1) In April 1978, we published the report NUREG-0460 on anticipated transients without scram. The recommendations included design criteria for plants such as Zimmer and recommended rulemaking to establish such criteria.

(2) The report, at present (September 1978), is under review by the Advisory Committee on Reactor Safeguards and the Office of Nuclear Reactor Regulation. After completion of the review, now estimated by January 1979, the Office Director, Nuclear Reactor Regulation, will forward his recommendations to the Commission.

(3) After deliberation, the Commission will act upon the matter. Whether it will agree to rulemaking is speculative at this time. If rulemaking is initiated by the Commission, we would expect that any rule adopted would include an implementation plan for all classes of plants.

The Zimmer unit would be required to provide plant modifications in conformance with anticipated transients without scram criteria and schedular requirements provided in the rule or as adopted by the Commission.

15.3 Design Basis Accidents

15.3.1 Radiological Consequences of Accidents

The applicant calculated the offsite doses resulting from the various postulated design basis accidents in order to demonstrate the effectiveness of the engineered safety features. In addition, we independently performed similar calculations for the loss-of-coolant, fuel handling, and control rod drop accidents and compared our results with those of the applicant. Our acceptance criteria are that the doses from these postulated accidents (as evaluated by us) 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 operating license review stage are 25 rem whole body and 300 rem thyroid.

On the basis of our experience with the evaluations of the steam line break accident for boiling water reactor plants of similar design, we conclude 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 acceptably 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 for this accident, will be appropriately small fractions of the guidelines values of 10 CFR, Part 100.

15.3.2 Loss-of-Coolant Accident (Radiological Considerations)

Direct Leakage Contributions

A design basis loss-of-coolant accident was postulated for the William H. Zimmer Nuclear Power Station. The station includes secondary containment systems to mitigate the offsite doses resulting from a loss-of-coolant accident.

In calculating the consequences of the postulated loss-of-coolant accident, we used the conservative assumptions presented in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors." The primary containment was assumed to leak at a constant rate of 0.635 percent of the containment volume per day for the duration of the accident. Our assumptions are shown in Table 15-2.

The pressure within the reactor building is at a negative pressure of -0.25 inches water gauge during normal operation and the air in this region is exhausted through the normal ventilation system. Upon receipt of a loss-of-coolant-accident signal, the normal ventilation system is switched off and the standby gas treatment system is switched on. The applicant's analysis indicates that during this change-over a pressure transient occurs within the reactor building such that the pressure increases to a maximum of -0.16 inches of water gauge and then returns to a value of -0.25 inches of water gauge. The total duration of this transient was conservatively estimated to be as long as five minutes. Thus the pressure within the reactor building is not expected to become positive, although it may exceed a value of -0.25 inches of water gauge during this period. Maintenance of a negative pressure is required as a criterion for precluding direct outleakage. A value of -0.25 inches of water is typically used to assure that no local positive pressure exists in portions of the secondary containment building volume due to various atmospheric effects such as winds and temperature differences.

We evaluated the specific features of the reactor building ventilation system and noted that the air volume in the reactor building annulus is about ten times larger than that of the primary containment (2.65×10^6 cubic feet vs. 2.73×10^5 cubic feet), and that the air currents and flow patterns established by the normal ventilation system (using high velocity jet diffusers) provided considerable coverage of air mixing throughout the reactor building air volume. Due to the presence of significant mixing, we would expect that the primary containment leakage would be mixed thoroughly with the reactor building air prior to treatment by the standby gas treatment system. We conclude, therefore, that it is conservative to assume only a portion of the reactor building air as being available for mixing during the first five minutes after a loss-of-coolant accident signal but that direct leakage is not likely to occur.

Nonetheless, in our analysis we assumed conservatively that the primary containment leakage was mixed with only 50 percent of the reactor building air during this period and that exfiltration occurred from 0-5 minutes. We found that the resulting loss-of-coolant accident doses were within the guideline values of 10 CFR, Part 100. We further examined the sensitivity of our assumption concerning the degree of mixing. We found that the loss-of-coolant accident doses were within the guideline values of 10 CFR, Part 100 even for mixing fractions well below 50 percent of the reactor building air assumed in the analysis.

The resultant calculated doses from this release path are shown in Table 15-1; at the exclusion radius these are 12 rem to the thyroid and 10 rem to the whole body and at the low population zone these are three rem to the thyroid and four rem to the whole body.

Main Steam Isolation Valve Leakage Contribution

The loss-of-coolant accident doses include the contribution of activity released by the main steam isolation valves. For this contribution we assumed that one of the four inboard main steam isolation valves failed to close, thus allowing contaminated steam to travel to the outboard main steam isolation valves and be released directly to the environment for 20 minutes until the reactor operator can activate the main steam isolation valves leakage control system. For the other steam lines, contaminated steam is isolated by the inboard valve. In the line with the failed valve the outboard main steam isolation valve is assumed to leak at its Technical Specification leak rate of 11.5 cubic feet per hour, all of which is collected by the leakage control system after 20 minutes. This leakage is processed directly by the reactor building recirculation and standby gas treatment system before it is released to the environment. The other three lines leak only clean steam during this time. After 2.9 hours plug flow calculations indicate that the clean steam will have been replaced with contaminated steam and we assume that the pressure from the trapped steam in the 20 foot length of steam line between the inboard and outboard valves of the three other steam lines reached 35 pounds per square inch or less and, by design, the inboard leakage control system would begin to function. This results in a total leakage of 46 cubic feet per hour of contaminated steam for the time periods beyond 2.9 hours. The assumptions for the dose model for the main steam line leakage control system are listed in Table 15-2. The radiological consequences of the loss-of-coolant accident are reported in Table 15-1.

Emergency Core Cooling Leakage Contribution

As part of the loss-of-coolant accident we also evaluated the consequences of leakage of containment sump water which is circulated by the emergency core cooling system after a postulated loss-of-coolant accident. We assumed the sump water contains a mixture of iodine fission products in agreement with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." After the loss-of-coolant accident this water is circulated into the reactor building to be cooled. If a source of leakage should develop a portion of the iodine could become gaseous and exit to the reactor building atmosphere. Our calculation of the dose resulting from operational leakage of engineered safety feature equipment is a small fraction of 10 CFR Part 100 guidelines. Areas housing equipment circulating containment sump water

TABLE 15-1

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

Postulated Accident	0-2 Hour Doses, Exclusion Area Boundary, rem		0-30 Day Doses, Low Population Zone, rem	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss-of-Coolant	12	10	3	4
Main Steam Isolation Valve	2	3	3	[illegible]
Total*	14	13	6	7
Fuel Handling	15	9	F 1	F 1
Control Rod Drop	57	11	6	F 1

*Includes contribution from Main Steam Isolation Valve leakage

TABLE 15-2

ASSUMPTIONS USED TO CALCULATE LOSS-OF-COOLANT ACCIDENT DOSES

Power Level	2533
Operating Time (years)	3
Core Fraction Released to Drywell (percent):	
Noble Gases	100
Iodine	25
Primary Containment Leak Rate (percent per day)	.635
Containment Free Volume (cubic feet)	2.730×10^{5}
Reactor Building Free Volume (cubic feet)	2.65×10^{6}
Reactor Building Mixing Fraction (percent) (after positive pressure period)	50
Bypass Leakage (percent)	0
Reactor Building Recirculation System Flow Rate (cubic feet per minute):	
Exhaust	2300
Recirculation	77,000
Standby Gas Treatment System Filter Efficiencies for Iodines (percent)	
Elemental	99
Organic	99
Particulate	99
Main Steam Isolation Valve Filtered Contaminated Leakage (cubic feet per hour):	
0-2.9 hours	11
> 2.9 hours	46
Minimum Exclusion Area Boundary, EB (meters)	250
Low Population Zone Distance, LPZ (meters)	4827
Atmospheric Diffusion (X/Q) values (seconds per cubic meter):	
0-2 hours, Exclusion Area Boundary	7.1×10^{-3}
0-8 hours, Low Population Zone Boundary	[illegible].2 $\times 10^{-5}$
8-24 hours, Low Population Zone Boundary	5.4×10^{-5}
1-4 days, Low Population Zone Boundary	2.8×10^{-5}
4-30 days, Low Population Zone Boundary	9.5×10^{-6}

are located in the reactor building and releases to the reactor building resulting from leakage of that sump water would be treated by the charcoal filters of the standby gas treatment system.

We conclude that doses resulting from the postulated leakage of post-loss-of-coolant accident recirculation water from pump seals, valve packings, etc., are low and, when added to the direct leakage loss-of-coolant accident doses, result in total doses that are within the guideline values of 10 CFR Part 100.

15.3.3 Fuel Handling Accident

For this accident, it is as d that a fuel assembly is dropped by the refueling crane into the reactor core or spent fuel pool. Should such an accident occur, the radiation detectors will automatically isolate the reactor building, shut off and isolate the normal ventilation system, and start the standby gas treatment system. The applicant stated that the transport time for any gases released from the pool to the isolation valves in the ventilation system is 11 seconds and that the value closure will not exceed 10 seconds. Our evaluation confirms the applicant's analysis. With the operation of this engineered safety feature, fission products released to the reactor building air will be filtered before release, even if all the fission products are released immediately.

In the evaluation of the fuel handling accident, the applicant assumed that the cladding on 125 fuel rods is damaged (equivalent to more than two fuel assemblies) 24 hours after reactor shutdown. With respect to the fuel handling accident, the applicant calculated that 63 fuel rods in the dropped assembly will fail, plus 62 additional rods in the struck assemblies. Based on lower fuel temperatures in the 8x8 design, we conclude generically that the consequence of the fuel handling accident with the 8x8 fuel assembly would not exceed that with the 7x7 assemblies. We independently evaluated the radiological consequences of this accident using these assumptions as well as the conservative assumptions given in Regulatory Guide 1.25 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

All of the radioactivity is assumed to be released after processing by the standby gas treatment system filters. Our assumptions are listed in Table 15-3. The resultant calculated doses are shown in Table 15-1 and are 15 rem to the thyroid and nine rem to the whole body at the exclusion boundary and at the low population zone are less than one rem to the thyroid and to the whole body. These doses are well within the 10 CFR, Part 100 guidelines.

TABLE 15-3

ASSUMPTIONS USED TO CALCULATE FUEL HANDLING ACCIDENT DOSES

Power Level (megawatts thermal)	2533
Total Number of Fuel Rods in Core	35,280
Number of Fuel Rods Damaged	125
Power Peaking Factor	1.5
Shutdown Time (hours)	24
Fraction of Fuel Rod Activity Released to Pool Iodines and Noble Gases (percent)	10
Pool Decontamination Factors:	
Iodines	100
Noble Gases	1
Reactor Building Standby Ventilation System Filter Efficiency (percent)	99
Atmospheric Diffusion Coefficient Value:	
@ Exclusion Area Boundary (seconds per cubic meter)	7.1×10^{-3}
@ Low Population Zone Boundary (0-8 hours)	9.2×10^{-5}

15.3.4 Control Rod Drop Accident

For the postulated control rod drop accident, it is assumed that a bottom entry control rod has been fully inserted and has stuck in this position, the drive becomes uncoupled and withdrawn from the rod. Subsequently, it is assumed that the rod falls out of the core inserting an amount of reactivity corresponding to the worth of the rod.

The radiological consequences of this accident were evaluated using the criteria described below. These criteria are the same as those used in the construction permit review and are more conservative than the criteria subsequently issued in the Standard Review Plan 15.4.9, "Spectrum of Rod Drop Accidents (BWR)." The most reactive control rod assembly is assumed to drop out of the core causing 770 fuel rods to exceed a calculated energy input of 170 calories per gram. These rods were assumed to perforate, releasing 100 percent of the contained noble gases and 50 percent of the contained halogens to the reactor coolant system. Of the halogens released from the affected rods, 90 percent are assumed to be retained in the primary system and one-half of the remaining halogens are assumed to be removed by plate-out. All of the noble gases and 2.5 percent of the halogens are assumed to be released from the primary system through the condenser vacuum pump system to the atmosphere. A conservative ground level release was assumed, using X/Q values determined by onsite measurements. Our assumptions are listed in Table 15-4. For this accident, the 24-hour time interval is the full course of the accident. The resulting doses shown in Table 15-1 are well within the 10 CFR, Part 100 guidelines.

The analysis of the rod drop accident was performed by General Electric Company on a generic basis and presented in NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," and Supplements 1 and 2 to this report. We have reviewed these reports. NEDO-10527 has been accepted by us in a letter to the General Electric Company dated 4/17/74. The results of our review of the supplements is presented in our report to the Advisory Committee on Reactor Safeguards (Rusche to Farley, dated June 7, 1976) detailing the resolution of Generic Item II A-2. These analyses are performed under the following conditions:

(1) Technical Specification values of scram time,

(2) Measured rod drop velocities plus 3 standard deviations, and

(3) Worst time in life.

On the bases of these analyses, a value for the inserted reactivity worth which produces a resultant peak fuel enthalpy greater than 280 calories per gram is obtained. This value has been determined to be greater than 0.013 Wk/k.

TABLE 15-4

ASSUMPTIONS USED TO CALCULATE CONTROL ROD DROP ACCIDENT DOSES

Number of Fuel Rods Involved	770
Fraction of Fission Product Inventory Released to Coolant (percent):	
Noble Gases	100
Iodines	50
Peaking Factor	1.5
Iodine Fraction Released to Condenser (percent)	10
Iodine Fraction Plate-Out in Condenser (percent)	90
Condenser Leak Rate (percent per day)	1.0
Atmospheric Diffusion (X/Q) Values (seconds per cubic meter)	
0-2 hours, Exclusion Area Boundary	7.1×10^{-3}
0-8 hours, Low Population Zone Boundary	9.2×10^{-3}

Zimmer is equipped with a rod worth minimizer and a rod sequence control system which restricts rod motion to the banked position withdrawal sequence (see subsection 4.3. of this report). This sequence limits the maximum potential rod worth to less than 0.01 $\Delta k/k$ for normal operation and to less than 0.013 $\Delta k/k$ for the maximum permitted number (8) of inoperable rods. We conclude that Zimmer is adequately protected against a rod drop accident having a peak fuel enthalphy of 280 calories per gram.

While the Zimmer plant meets all current requirements for the postulated rod drop accident, recent research results suggest that present fuel damage limits should be reevaluated. Although we are currently reviewing these fuel damage limits, no decision has been made to alter requirements of the rod drop accident analysis. Should any such licensing requirements be changed in the future, the effects of such changes would be evaluated for all plants including Zimmer.

15.3.5 Operation of Fuel Assembly in Improper Position

The consequences of operating a fuel assembly in an improper position were evaluated by the applicant. The assembly average enrichment is the same for each bundle in the core. However, burnable poison loadings in the peripheral bundles are smaller than those in the interior bundles. The operation of a peripheral bundle in an interior position would result in increased heat generation rates and reduced minimum critical power ratio in the misplaced bundle. Analysis shows that the bundle power would be increased by 8.1 percent, the nodal power increased by six percent and the minimum critical power ratio reduced by eight percent. These changes are well within the operating margins for Zimmer and we conclude that normal operation with a misloaded bundle does not violate any safety limits.

15.3.6 Postulated Radioactive Releases Due to Liquid Tank Failures

The consequences of component failures for components located outside the reactor containment which could result in releases of liquid containing radioactive materials to the environs were evaluated. Considered in our evaluation were (1) the radionuclide inventory in each component assuming a one percent operating power fission product source term, (2) a component liquid inventory equal to 80 percent of its design capacity, (3) the mitigating effects of plant design including overflow lines and the location of storage tanks in curbed areas designed to retain spillage, and (4) the effects of site geology and hydrology.

The applicant incorporated provisions in the design to retain releases from liquid overflows as discussed in subsection 11.2.1 of this report. In the event of a spill, we postulated liquid flow directly to the groundwater beneath the location of the tank. The flow was assumed to seep through the ground and flow along the shortest pathway to the Ohio River, adjacent to the site, and mix with the lowest average annual flow of record for the Ohio River. We calculated an overall dilution factor of 1.4×10^7

Based on our evaluation, the potential tank failure resulting in the greatest quantity of activity released to the environment is failure of the waste collector tank. The tank is assumed to contain radionuclides at approximately 25 percent of the primary coolant activity level for the design basis fission product inventory stated above. In our evaluation, we determined the liquid transit time for the leakage to the river to be 46.7 days. Considering the leakage dilution and transit time, the calculated radionuclide concentrations in the Ohio Rive result in values that are small fractions of the limits of 10 CFR, Part 20, Appendix B, Table II, Column 2, for unrestricted areas. Based on the foregoing evaluation, we conclude that the provisions ir porated in the applicant's design to mitigate the effects of component failu s involving contaminated liquids are acceptable.

15.3.7 Generator Load Rejection/Turbine Trip

We have performed an independent analysis of the event described in subsection 15.2 of this report. It is postulated that a seismic event occurs causing a turbine trip or load rejection and the multiple sensors located in the non-seismic turbine building fail to provide a reactor scram signal or a recirculation pump trip signal. It was assumed that the only pathway for relief of the pressurized primary coolant is the relief valve discharge to the suppression pool. Flow to the consenser would be stopped by the turbine stop, throttle and bypass valves.

At a peak clad temperature less than 1420 degrees Fahrenheit, fuel clad perforation is not expected to occur, at least not in the five-second period for closure of the main steam line isolation valves. If it is nevertheless assumed that 6 percent of the fuel rod cladding perforates as a result of reaching boiling transition the resulting radioactivity would be discharged via the relief valves to the suppression pool.

Our position involving release of fuel rod gap activity is that the fuel rod gaps contain 10 percent of the rod's iodine activity and 10 percent of the rod's noble gas activity. With 6 percent of the total rods release of gap activity, a total of 0.6 percent of the core activity would have been released to the primary containment (suppression pool) from whence it is conservatively postulated to leak at the design basis leak rate.

Since our analysis of the postulated loss of coolant accident for the Zimmer plant assumed that a 25 percent of the core iodines and 100 percent of the core noble gases were available for leakage from the primary containment, the calculated thyroid and whole body doses would be less than 1 percent of the loss-of-coolant accident doses.

Based on the calculated low consequences of this accident, we conclude that the design meets the seismic requirements of Appendix A to 10 CFR, Part 100 and that additional seismic design requirements are not required to mitigate the consequences of a turbine trip without bypass.

16.0 TECHNICAL SPECIFICATIONS

The technical specifications in a license define certain features, characteristics and conditions governing operation of a facility that cannot be changed without prior approval of the Commission. The finally approved technical specifications will be made a part of the operating license. Included will be sections covering safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

At the time of submittal of the Final Safety Analysis Report, the applicant had proposed technical specifications in Chapter 16. Shortly thereafter, we informed the applicant that we intended to use the Standard Technical Specifications for Boiling Water Reactors as the basis for development of the final technical specifications for Zimmer Unit 1.

The Standard Technical Specifications for BWR/5 plants to be used as the basis for the plant technical specifications have been updated as a result of their application to technical specifications for other plants and also of continued discussion with General Electric Company and applicants with boiling water reactors.

On the basis of our review we conclude that normal plant operation within the limits of the finally approved technical specifications will not result in potential offsite exposures in excess of the 10 CFR, Part 20 limits. Furthermore, the limiting conditions for operation and surveillance requirements will assure that necessary engineered safety features will be available in the event of malfunctions within the plant.

17.0 QUALITY ASSURANCE

General

The description of the quality assurance program for the operations phase of the William H. Zimmer Nuclear Power Station, Unit No. 1 is contained in Section 17.2 of the Final Safety Analysis Report through Revision 50. Our evaluation of this quality assurance program is based on a detailed review of this information and discussions with representatives of Cincinnati Gas & Electric Company to assess if the quality assurance program for the operations phase complies with the requirements of Appendix B to 10 CFR, Part 50 and supplemental guidance contained in Regulatory Guides 1.8, "Personnel Selection and Training"; 1.30, "Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment"; 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Component of Water-Cooled Nuclear Power Plants"; 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants"; 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants"; 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants"; 1.58, "Qualification of Nuclear Power Plant Inspection, Examination and Test Personnel"; 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants"; 1.74, "Quality Assurance Terms and Definitions"; 1.88, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records"; and 1.94, "Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants", as well as American National Standards Institute Standards N18.7-1976, N45.2.8-1975, N45.2.12 (Draft 4, Revision 2) and N45.2.13-1976.

17.1 Organization

The organizational structure responsible for the operation of Zimmer and for the establishment and execution of the operations phase quality assurance program is shown in Figure 17-1. The President of Cincinnati Gas and Electric Company has the overall responsibility for the engineering, design, procurement, construction, operation, and quality assurance activities of Zimmer. He has delegated the authority for quality assurance to the Principal Quality Assurance & Standards Engineer who manages the quality assurance organization.

The Principal Quality Assurance & Standards Engineer reports to the Manager-General Engineering Department but maintains open lines of communication to the Vice President-Engineering Services & Electric Production and the

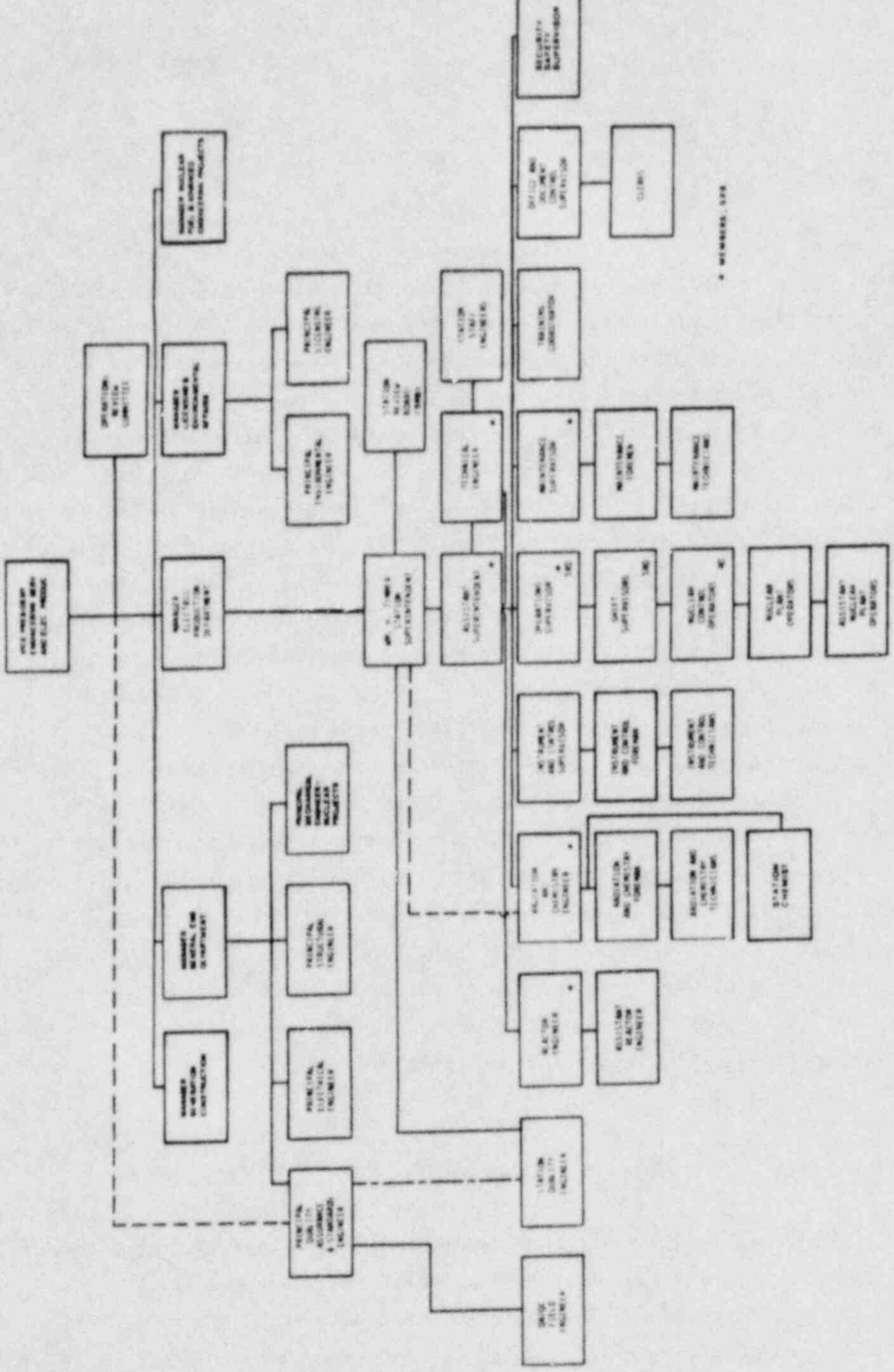

FIGURE 17-1 CG&E ORGANIZATION FOR QUALITY ASSURANCE

Station Quality Engineer. The Principal Quality Assurance & Standards Engineer has the responsibility for establishment, effective implementation, and control of the quality assurance program. The quality assurance organization has the authority to identify quality problems; to initiate, recommend, or provide solutions; to verify implementation of solutions; and to stop unsatisfactory work or stop further processing of unsatisfactory material.

The Manager-General Engineering Department and the Manager-Electric Production Department are responsible for the development, direction, and overall coordination of engineering support activities and for the operation and maintenance of Zimmer, respectively, during the operating phase of the station. The Manager of Electric Production has delegated the responsibility of the day-to-day operation and maintenance activities for Zimmer to the Superintendent of the Electric Production Department.

The Station Superintendent, who reports to the Manager of Electric Production Department, is primarily responsible for operating Zimmer in compliance with the requirements of the operating license and the Quality Assurance Manual. The resolution of disputes on the quality assurance program requirements arising between quality assurance/quality control personnel and other department personnel will follow the lines of responsibility designated in Figure 17-1 culminating at the highest level of management necessary for resolution.

The Station Quality Engineer, who reports to the Station Superintendent, communicates directly with the Principal Quality Assurance & Standards Engineer in matters relating to the policies and practices of the operational quality assurance program. He independently monitors maintenance and operation activities to assure the implementation of the operations quality assurance program at the plant. Both the quality assurance organization and the Station Quality Engineer have been given the responsibility for: reviewing and approving quality related documents (e.g., instructions, procedures, drawings, and specifications); performing vendor quality assurance prequalifications; assuring that procurement documents contain requirements which can be inspected and controlled to meet predetermined acceptance criteria; surveillance, inspection, and auditing of vendors; documenting and reporting to responsible management any nonconformance discovered in the course of surveillance or audit; assuring corrective actions are effective and accomplished in a timely manner; and auditing of maintenance and operation activities.

The Station Review Board, which is chaired by the Station Superintendent, is responsible for the technical aspects (e.g., reviews all procedures, and changes thereto, proposed tests and experiments, and proposed modifications to plant systems that affect nuclear safety) of the station but coordinates its activities with the Station Quality Engineer to ensure that applicable quality aspects of station operation are satisfied. Membership in the Station Review Board is shown in Figure 17-1.

The Operations Review Committee reports to the Vice President-Engineering Services & Electric Production and is responsible for independent reviews and audits to assure that the operation of the plant is performed in a safe manner and is consistent with license provisions, administrative and quality assurance procedures, and Cincinnati Gas and Electric Company policies. The Vice President-Engineering Services & Electrical Production may instruct the Operating Review Committee to conduct an audit of any activity at any time to determine compliance with or the effectiveness of the policies and practices established in the quality assurance program. The Operating Review Committee is composed of a Chairman and four members, who collectively have the experience and qualifications to review and audit the designated activities, and qualified technical consultant(s) as required.

17.2 Quality Assurance Program

The quality assurance program for the operation of Zimmer implements the requirements of Cincinnati Gas and Electric Company's quality assurance policies via the Quality Assurance Manual, administrative, and operating procedures. These three levels of documents control quality related activities involving safety-related items to comply with the requirements of Appendix B to 10 CFR Part 50. Cincinnati Gas and Electric Company's program requires that implementing documentation encompasses: detailed controls for translating codes, standards, regulatory requirements, technical specifications, engineering and process requirements into drawings, specifications, procedures and instructions; developing, reviewing, and approving procurement documents, including changes; prescribing all quality related activities by documented instructions, procedures, and drawings; issuing and distributing approved documents; purchasing items and services; identifying materials, parts and components; performing special processes; inspecting and/or testing materials, equipment, processes or services; calibrating and maintaining measuring and test equipment; handling, storing and shipping of items; identifying the inspection, test and operating status of items; identifying and dispositioning nonconforming items; correcting conditions adverse to quality; preparing and maintaining quality assurance records; and auditing of activities which affect quality.

An indoctrination and training program is established to assure that personnel performing activities affecting quality are knowledgeable in quality assurance/quality control requirements, implementing procedures and instructions; and demonstrate a high level of competence and skill in the performance of their quality related activities.

Quality is verified through checking, review, surveillance, inspection, testing and audit of quality related activities. The quality assurance program requires that quality verification be performed by individuals who are not directly responsible for performing the actual work activity. Inspections are performed in accordance with procedures, instructions and/or checklists approved by the Station Superintendent and the Principal Quality Assurance and Standards Engineer.

Inspections are performed by qualified personnel who are trained in accordance with Cincinnati Gas and Electric Company's training programs. Nondestructive examination personnel are certified in accordance with Society for Nondestructive Testing Standard-TC-1A.

External audits of vendors and service contractors and internal audits of all aspects of the quality assurance program are conducted by the quality assurance organization. In addition, the Station Quality Engineer provides a direct onsite audit function, and the Operating Review Committee provides an independent management evaluation of plant maintenance and operation activities. Audits are performed in accordance with preestablished written procedures by appropriately trained personnel not having direct responsibilities in the areas being audited. The audit function, which is conducted at scheduled intervals and/or on a random unscheduled basis, includes an objective evaluation of the effectiveness of implementation of the quality assurance program; the adequacy of and compliance with quality assurance policies, practices, procedures and instructions; the adequacy of work areas, activities, processes, items, and records, and product compliance with applicable engineering drawings and specifications.

The quality assurance program requires documentation of audit results and review by management having responsibility in the area audited to determine and take corrective action needed, if any. Followup audits are performed to d nine that nonconformances are effectively corrected and that the corrective actio precludes the effectiveness of the quality assurance program, are also reported to responsible management including the President for review and assessment.

17.3 Conclusion

Our review of the Zimmer program description for the operations phase has verified that the criteria of Appendix B to 10 CFR, Part 50 have been adequately addressed in the Zimmer quality assurance program.

Based on our detailed review and evaluation of the quality assurance program description contained in Section 17.2 of the Final Safety Analysis Report, through Revision 50, for the William H. Zimmer Nuclear Power Station, Unit No. 1, we conclude that:

(1) The quality assurance organization of the Cincinnati Gas and Electric Company is provided sufficient independence from cost and schedule (when opposed to safety considerations), sufficient authority to effectively carry out the operations quality assurance program, and sufficient access to management at a level necessary to perform their quality assurance functions.

(2) The quality assurance program description contains adequate quality assurance requirements and a comprehensive system of planned and systematic controls which address each of the criterion of Appendix B to 10 CFR, Part 50 in an acceptable manner.

18.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The William H. Zimmer Nuclear Power Station application for a one unit facility is being reviewed by the Advisory Committee on Reactor Safeguards. 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 us to resolve any issues raised as a result of the Committee's review.

19.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and that all of the directors and principal officers of the applicants* are United States citizens. The applicants are 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 Cincinnati Gas and Electric Company agreed to safeguard any such data which might become involved in accordance with the requirements of 10 CFR, Part 50. The applicants will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes will be involved. For these reasons and in the absence of any information to the contrary, we find that the activities to be performed will not be inimical to the common defense and security.

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*Cincinnati Gas and Electric Company
Columbus and Southern Ohio Electric Company
Dayton Power and Light Company

20.0 FINANCIAL QUALIFICATIONS

20.1 Introduction

The Nuclear Regulatory Commission's regulations relating to the determination of an applicant's financial qualifications for a facility operating license appear in Section 50.33(f) and Appendix C to 10 CFR, Part 50. At our request, Cincinnati Gas and Electric Company, Columbus and Southern Ohio Electric Company, and Dayton Power and Light Company submitted financial information regarding estimated operating and decommissioning costs for the Zimmer Nuclear Power Station, Unit No. 1, along with additional material covering the applicants' financial status. The following analysis summarizes our review of this submittal and addresses each applicant's financial qualifications to operate, and, if necessary, permanently shut down and safely maintain the subject facility.

20.2 Estimated Operating and Shutdown Costs

For the purpose of estimating the facility's operating costs, the applicants assumed that 1979 would be the first full year of commercial operation. Estimates of the total annual cost of operating the Zimmer plant for each of the first five years are presented in Table 20-1. The unit costs (mills per kilowatt-hour) are based on a net electrical capacity of 792 Megawatt-electric.

TABLE 20-1

ESTIMATE OF TOTAL ANNUAL COST

	Plant Capacity Factor	Operating Cost Estimate (thousands)	Mills/KWh
1979	65.4%	$140,705	31.01
1980	58.6%	$132,963	32.70
1981	75.4%	$135,395	25.88
1982	76.8%	$131,658	24.71
1983	76.2%	$130,827	24.75
5-year average	70.5%	$134,310	27.91

The estimates of operating costs cover operating and maintenance expenses (including fuel expense), depreciation, taxes, and a return on investment.

The applicants based their estimate of decommissioning costs on a report published in November 1976 by the Atomic Industrial Forum entitled, "An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives." This study concluded that the most economical type of decommissioning would be either temporary mothballing or temporary entombment for a cooling period of about 104 years, followed by dismantling and removal of the radioactive structures of the facility. Temporary entombment becomes the most economical choice if it is assumed that a security force will be required to guard a temporarily mothballed facility for the entire 104-year cooling period. For purposes of their cost estimates, the applicants assumed that such a security force would be required with temporary mothballing. Consequently, the applicants' cost estimates are based on temporarily entombing the facility at the end of its 33-year life, allowing the radiation levels to decay for 104 years, and then dismantling and removing only the contaminated structures. Interpolating from the Atomic Industrial Forum estimates and adjusting for inflation, the applicants arrived at estimated total decommissioning costs for the Zimmer plant of $17,864,851 in 1979 dollars. This can be broken down as follows:

Initial temporary entombment	$8,555,348
Total surveillance and maintenance for 104 years	$7,663,864
Dismantling and removal of contaminated structures after 104 years of cooling	$1,645,639
Total decommissioning cost	$17,864,851

The Nuclear Regulatory Commission had Battelle (Pacific Northwest Laboratories) do a technical review of the Atomic Industrial Forum's study on reactor decommissioning, which concluded that the total costs presented in that study appear to be realistic.

20.3 Source of Funds

The applicants expect to cover all operating costs through revenues generated from their system-wide sales of electricity and in proportion to their ownership interests: 40 percent for Cincinnati Gas and Electric Company, 28.5 percent for Columbus and Southern Ohio Electric Company, and 31.5 percent for Dayton Power and Light Company. All three applicants are investor-owned utilities providing electric and/or gas service to residential, commercial, and industrial customers in Ohio. For the 12 months ended September 30, 1977, the unit prices per kilowatt-hour from system-wide sales of electric power for Cincinnati Gas and Electric Company, Columbus and Southern Ohio Electric Company, and Dayton Power and Light Company were 3.24 cents, 3.62 cents, and 3.10 cents, respectively. These prices are in excess of the projected operating costs presented above and in addition, do not reflect possible rate increases during the first five years of Zimmer's commercial operation.

Furthermore, the applicants have consistently demonstrated the ability to achieve revenues sufficient to cover all operating costs and interest charges. Table 20-2 presents financial data on revenues and net income for each applicant during the five years ended 1976.

TABLE 20-2

REVENUES/NET INCOME (MILLIONS)

CINCINNATI GAS & ELECTRIC COMPANY

1976	1975	1974	1973	1972
$547.1/$54.4	$479.9/$49.7	$416.1/$45.5	$349.2/$50.2	$325.0/$47.1

COLUMBUS AND SOUTHERN OHIO ELECTRIC COMPANY

1976	1975	1974	1973	1972
$280.3/$54.3	$259.1/$42.4	$188.6/$21.1	$158.4/$25.9	$136.0/$21.1

DAYTON POWER AND LIGHT COMPANY

1976	1975	1974	1973	1972
$397.9/$43.4	$347.6/$41.2	$296.6/$29.7	$229.0/$27.0	$219.9/$27.1

The applicants intend to obtain the funds required for decommissioning the plant through annual depreciation charges over the service life of the facility which will be deposited with a trustee. Based on a 6 percent annual inflation rate from 1975 through the final dismantling/removal of contaminated structures in the year 2116 and a 5 percent tax-free interest rate on funds deposited with the trustee, the annual payments over the 33-year plant life required to provide the necessary funds for each of the three components of decommissioning, as well as the total amount payment, would be as follows:

Initial temporary entombent	$730,963
104 years of surveillance and maintenance	$1,110,584
Dismantling and removal of contaminated structures after 104 years of cooling	$376,803
Total annual decommissioning fund deposit over 33 years operating lifetime	$2,218,350

Such a funding plan for anticipated decommissioning costs would, of course, require the approval of the Ohio Public Service Commission. To the best of our knowledge, this approval has not yet been obtained. Nevertheless, even in the absence of a plan specifically setting aside funds for decommissioning purposes, we feel that there is reasonable assurance that the applicants could cover the estimated decommissioning costs insofar as they are relatively small in comparison to the applicants' financial resources.

20.4 Conclusion

In accordance with the regulations cited above, an applicant must demonstrate that it has reasonable assurance of obtaining the necessary funds to cover the estimated costs of the activities contemplated under the license. Based upon the preceding analysis, we conclude that Cincinnati Gas and Electric Company, Columbus and Southern Ohio Electric Company, and Dayton Power and Light Company have satisfied this reasonable assurance standard and are, therefore, financially qualified to operate and, if necessary, shut down and safely maintain the Zimmer Nuclear Power Station, Unit No. 1. Our conclusion is based upon the applicants' demonstrated ability to achieve revenues sufficient to cover all operating costs and interest charges, and the favorable comparison between their current unit prices for electricity and the projected unit costs of this facility. We have requested additional financial information which will update the financial evaluation based on 1980 as the first full year of plant operation. While we do not expect that this additional inormation will change our conclusion, we will report our reevaluation in a supplement to this report.

21.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

21.1 General

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR, Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licenses for facilities such as power reactors under 10 CFR, Part 50.

21.2 Preoperational Storage of Nuclear Fuel

The Commission's regulations in 10 CFR, Part 140 require that each holder of a construction permit under 10 CFR, Part 50, who is also the holder of a license under 10 CFR, Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR, Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR, Part 70 license. Payment of an annual indemnity fee is required.

The applicant will furnish the Commission proof of financial protection in the amount $1,000,000 in the form of a Nuclear Energy Liability Insurance Association Policy (Nuclear Energy Liability Policy, facility form No. NF-210). Further, the applicant will execute an indemnity agreement with the Commission effective as of the date of its preoperational fuel storage license. The applicant will pay the annual indemnity fee applicable to preoperational fuel storage.

21.3 Operating Licenses

Under the Commission's regulations, 10 CFR, Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for William H. Zimmer, Unit 1, (which has a rated capacity in excess of 100,000 electrical kilowatts), is the maximum amount available from private sources, which is currently $450 million.

Accordingly, licenses authorizing operation of William H. Zimmer, Unit 1, will not be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended so that the policy limits have been increased, to meet the requirements of the Commission's regulations for reactor operation. Similarly, operating licenses will not be issued until an appropriate amendment to the present indemnity agreement has been executed. The applicant will be required to pay an annual fee for operating license indemnity as provided in the Nuclear Regulatory Commission's regulations, at the rate of $12 per thousand kilowatts of thermal capacity authorized in his operating license. On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR, Part 140 have been satisfied and that, prior to issuance of the operating licenses, the applicant will be required to comply with the provisions of 10 CFR, Part 140 applicable to operating licenses, including those as to proof of financial protection in the requisite amount and as to the execution of an appropriate indemnity agreement with the Commission.

22.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, we have determined that, upon favorable resolution of the outstanding matters described herein, we will be able to conclude that:

(1) The application for a facility operating license filed by the Cincinnati Gas and Electric Company, dated September 10, 1976, as amended, complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR, Chapter 1; and

(2) Construction of the William H. Zimmer Nuclear Power Station, Unit 1 (the facility), has proceeded and there is reasonable assurance that it will be substantially completed, in conformity with Construction Permit No. CPR-88, the application as amended, the provisions of the Act, and the rules and regulations of the Commission, and

(3) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(4) There is reasonable assurance (a) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (b) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR, Chapter 1; and

(5) The applicant is technically and financially qualified to engage in the activities authorized by the license, in accordance with the regulations of the Commission set forth in 10 CFR, Chapter 1, and

(6) The issuance of the license will not be inimical to the common defense and security or to the health and safety of the public.

Before an operating license will be issued to the Cincinnati Gas and Electric Company for operation of Unit 1, the unit must be completed in conformity with the construction permit, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power levels must be verified by the Commission's Office of Inspection and Enforcement prior to issuance of the license.

Further, before the operating license is issued, the applicant will be required to satisfy the applicable provisions of 10 CFR, Part 140.

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2.0 SITE CHARACTERISTICS

2.3 Meteorology

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5. Zimmer - Final Safety Analysis Report, Environmental Report - Docket No. 50-358.

3.0 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

Codes & Standards

General

1. ASME Boiler and Pressure Vessel Code, 1968 Edition, Section III, "Nuclear Vessels," American Society of Mechanical Engineers.

2. ASME Boiler and Pressure Vessel Code, 1971 Edition, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

3. ANSI B31.7, 1969 Edition, "Nuclear Power Piping," American National Standards Institute.

4. Draft ASME Code for Pumps and Valves for Nuclear Power (Dated November 1968), American Society of Mechanical Engineers.

5. ASME Boiler and Pressure Vessel Code, 1968 and 1971 Editions, Section VIII, Division 1, "Pressure Vessels," American Society of Mechanical Engineers.

6. ANSI B31.1.0-1967, "Power Piping," American National Standards Institute.

7. API Standard 620, Fourth Edition, February 1970, "Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks," American Petroleum Institute.

8. API Standard 650, Third Edition, July 1966, "Welded Steel Tanks for Oil Storage," American Petroleum Institute.

9. ANSI B96.1-1971, "Specification for Welded Aluminum-Alloy Field-Erected Storage Tanks."

10. Standards of Turbine Exchanger Manufacturers Association, Fifth Edition, 1968.

Materials

1. ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, including Addenda through Summer 1972.

2. ASME Boiler and Pressure Vessel Code, Section III, 1972 Summer Addenda including Appendix G, "Protection Against Non-Ductile Failure."

3. ASTM Specification E 208-69, "Standard Method for Conducting Dropweight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," Annual Book of ASTM Standards, Part 31, 1973.

4. ASTM Specification E 185-73, "Surveillance Tests on Structural Materials in Nuclear Reactors," Annual Book of ASTM Standards, Part 30, July 1973.

5. ASME Specification, SA 370, "Methods and Definitions of Mechanical Testing of Steel Products," ASME Boiler and Pressure Code, Section II, Part A - Ferrous.

6. ASTM E 23, "Notched Bar Impact Testing of Metallic Materials," Book of ASTM Standards.

7. ASTM E 208, "Standard Method for Conducting Dropweight Test to Determine Nilductility Transition Temperature of Ferritic Steels," Book of ASTM Standards.

8. ASTM A 708, "Recommended Practice for Detection of Susceptibility of Intergranular Corrosion in Severely Sensitized Austenitic Stainless Steel," Book of ASTM Standards.

9. NUREG 75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975.

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Section 3.5 - Missile Protection

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3.8-1 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.

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4.0 REACTOR

1. General Electric Standard Safety Analysis Report (GESSAR), April 1973.

2. "General Electric BWR Generic Reload Application for 8x8 Fuel," General Electric Report, NEDO-20360, Revision 1 (Proprietary version NEDE-20360-1P), November 1974.

3. "GEGAP-III: A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods," General Electric Report, NEDO-20181 (Proprietary version NEDC-20181), November 1973.

4. V. A. Moore (USAEC) letter to I. S. Mitchell (GE), "Modified GE Model for Fuel Densification," March 22, 1974.

5. "Creep Collapse Analysis of BWR Fuel Using Safe-Colaps Model," General Electric Report, NEDO-20606-A (Proprietary version NEDE-20606), August 1976.

6. D. F. Ross (USNRC) letter to G. Sherwood (GE), "Fission Gas Release at High Burnup," November 23, 1976.

7. G. C. Potts, "Urania-Gadolinia Nuclear Fuel Physical and Irradiation Characteristics and Material Properties," General Electric Report, NEDO-20943 (Proprietary version NEDE-20943-P), January 1977.

8. "Supplemental Information for Plant Modifications to Eliminate Significant In-Core Vibration," General Electric Report, NEDO-21156 (Proprietary version NEDE-21156) January 1976.

9. "Safety Evaluation Report on the Reactor Modifications to Eliminate Significant In-Core Vibration in Operating Reactors with 1-Inch Bypass holes in the Core Support Plate," USNRC Report, March 1976.

10. K. W. Holtzclaw (GE), Fuel Rod Bowing Presentation to USNRC Staff, November 13, 1974.

11. "8x8 Fuel Surveillance P[illegible]am, Quad Cities 1 Fuel Assembly," General Electric Report, NEDM-21430, Propri[illegible], October 1976.

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11.0 RADIOACTIVE WASTE MANAGEMENT

1. U. S. Atomic Energy Commission Concluding Statement of Position of the Regulatory Staff (and its Attachment) - Public Rulemaking Hearing on: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criteria "As Low as Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactors, Docket Number RM 50-2, Washington, D.C., February 20, 1974.

2. Staff of the U. S. Nuclear Regulatory Commission, "Final Environmental Statement related to the operation of Zimmer Nuclear Power Station, Unit No. 1," Docket No. 50-358, Washington, D. C., June 1977.

APPENDIX A

CHRONOLOGY - RADIOLOGICAL HEALTH AND SAFETY REVIEW OF THE WM. H. ZIMMER NUCLEAR POWER STATION, UNIT 1 APPLICATION FOR OPERATING LICENSE

May 9, 1975	Cincinnati Gas & Electric Company, et al. (the Applicant) tendered its application for a facility operating license (General Information, Final Safety Analysis Report and the Environmental Report).
June 2, 1975	Applicant letter submitting a Mark II schedule and program related to suppression pool information.
June 16, 1975	NRC letter rejecting the FSAR portion of the tendered application.
June 25, 1975	NRC letter requesting ECCS information required by 10 CFR 50.46 and Appendix K to 10 CFR Part 50.
June 30, 1975	Applicant letter submitting drawings related to the suppression pool.
August 18, 1975	Meeting held at the site between NRC staff and the Applicant for the purpose of discussing the Applicant's responses to questions related to the FSAR rejection.
August 20, 1975	Applicant letter (retendered application) submitting information requested by NRC letter dated June 16, 1976.
August 27, 1975	Issued summary of meeting held on August 18, 1975.
August 28, 1975	NRC letter accepting the operating license application for docketing.
September 10, 1975	Docketed application (Amendment No. 22) for facility operating license.
September 10, 1975	Applicant submits Physical Security Plan.
September 11, 1975	Applicant submits information related to the Mark II containment supporting program.

September 17, 1975	NRC letter transmitting notice of receipt.
September 24, 1975	Notice of Receipt of Application ...and Notice of Opportunity for Hearing published in the Federal Register (40 F.R. 43959).
October 1, 1975	NRC letter advising that the suppression pool drawing submittal does not contain sufficient information.
October 6, 1975	NRC letter advising that the General Electric Topical Reports NEDO-10678 and NEDO-10698 are unacceptable for referencing in the FSAR.
October 22 and 23, 1975	Meetings held in Bethesda, Maryland between NRC and Applicant representatives to discuss EI&CS Round-one review matters.
October 24, 1975	NRC letter advising that GE Topical Reports NEDO-10739 and its supplemental information are unacceptable as information to be incorporated by reference in the Zimmer 1 FSAR.
October 24, 1975	Applicant letter providing a schedule for submitting information requested by NRC letter dated October 1, 1975.
October 28, 1975	Amendment No. 23 submitted. The amendment contains information requested by NRC letter dated June 16, 1975.
October 29, 1975	Summary of meetings held on October 22 and 23, 1975 issued.
November 3, 1975	NRC letter transmitting the review schedule.
November 6, 1975	Applicant letter incorporating BWR Mark II containment information in the Zimmer 1 docket.
November 11, 1975	Site visit by NRC representatives for the purpose of NRC staff reviewers to observe the plant and site in its present stage of construction prior to finalizing Round-one Requests.
November 18, 1975	Amendment No. 24 filed, which contains further responses to questions issued by NRC letter dated June 16, 1975.
November 18, 1975	Issued summary of the November 11, 1975 site meeting.
November 19, 1975	NRC letter requesting additional information.
November 24, 1975	Issued correction to meeting summary dated October 29, 1975.

December 4, 1975	Round-one Request issued.
December 18, 1975	Further Round-one Requests issued.
December 31, 1975	Further Round-one Requests issued.
January 8, 1976	Further Round-one Requests issued.
January 14, 1976	Applicant letter submitting proprietary information related to the nuclear design and ECCS.
January 23, 1976	Special prehearing conference held in Cincinnati, Ohio.
January 27, 1976	Further Round-one Requests issued.
January 28, 1976	Meeting held in Bethesda, Maryland for the purpose of discussing electrical drawings.
February 4, 1976	Issued summary of meeting held on January 28, 1976.
February 5, 1976	Applicant letter responding to NRC letters dated October 6 and October 24, 1975 relating to GE Topical Reports.
February 6, 1976	Amendment No. 25 received, which responds to Round-one Requests issued December 4, 1975.
February 11, 1976	Applicant letter submitting (Supplement 1 to the FSAR) cable routing drawings.
February 13, 1976	NRC letter requesting Appendix I information.
February 23, 1976	NRC letter transmitting draft Guides 1.AA, 1.BB, 1.CC, 1.DD, 1.EE and 1.FF.
March 3, 1976	Applicant letter (Supplement 2 to the FSAR) submitting earthwork operations and soil stress information.
March 3, 1976	Amendment No. 26 submitted. The amendment response to Round-one Requests issued by NRC letters (published in the FR March 25, 1976, 41 F.R. 12361) dated December 18 and 31, 1975 and January 8, 1976.
March 10, 1976	Issued further Round-one Requests, which supplement those issued on December 4, 1975.

March 15, 1976	Applicant letter transmitting the "Mark II Containment Design Assessment Report," which is in response to NRC letters dated April 18 and 21, 1975, and October 1, 1975.
March 19, 1976	Notice of Hearing on Applications for Operating License issued by the ASLB.
March 19, 1976	Amendment No. 27 filed, which submits responses to Round-one Requests transmitted by NRC letters dated December 18, 1975, December 31, 1975 and January 27, 1976.
March 22, 1976	Amendment No. 28 filed. The amendment responds to Requests issued by NRC letter dated November 19, 1975.
March 31, 1976	Amendment No. 29 filed, which responds to the Requests issued by NRC letter dated December 31, 1975.
April 1, 1976	Applicant letter providing further justification for requesting the withholding of proprietary information submitted by its letter dated January 14, 1976.
April 9, 1976	Amendment No. 30 submitted. The amendment responds to the Round-one Requests issued by NRC letters dated November 4, 18 and 31, 1975.
April 23, 1976	Amendment No. 31 filed. It responds to the Round-one Requests issued by NRC letters dated December 31, 1975 and March 10, 1976.
May 3, 1976	NRC letter making proprietary finding on information submitted by applicant letter dated January 14, 1976.
May 6, 1976	Meeting held between between NRC and applicant representatives to discuss the electrical review.
May 7, 1976	Amendment No. 32 submitted, which contains further responses to Requests issued by NRC letters dated December 31, 1975 and March 10, 1976.
May 10, 1976	NRC letter transmitting draft model technical specifications related to Appendix I requirements.
May 18, 1976	Issued summary of meeting held on May 6, 1976.
May 24, 1976	NRC letter advising of recent events and conclusions related to ATWS.

May 24, 1976	Applicant letter submitting electrical drawings (New Volume 11A).
May 24, 1976	Meeting between NRC and applicant representatives on the Zimmer review schedule.
May 25 and 26, 1976	Meetings held in Bethesda, Maryland between NRC and applicant representatives to discuss several Round-two positions.
May 28, 1976	Applicant letter responding to NRC letter dated May 3, 1976 relating to proprietary information.
June 4, 1976	Amendment No. 33 filed, which contains information required by Appendix I, and is in reference to NRC letter dated February 13, 1976.
June 9, 1976	Further Round-one requests issued.
June 10, 1976	Issued summary of meeting held on May 24, 1976.
June 11, 1976	Amendment No. 34 received. The amendment responds to Round-one Questions issued by NRC letters dated December 4, 1975 and March 10, 1976.
June 16, 1976	Issued summary of meetings held on May 25 and 26, 1976.
June 18, 1976	Round-Two positions issued.
June 30, 1976	Letter from applicant transmitting Amendment No. 35.
July 8, 1976	Letter to applicant requesting additional information.
July 12, 1976	Letter to applicant concerning Mark II owners and requesting additional information.
July 13, 1976	Letter to applicant requesting additional information.
July 16, 1976	Letter from applicant transmitting Amendment No. 36.
July 27, 1976	Letter from applicant transmitting proprietary information in response to staff requests for additional information.
July 27, 1976	Letter from applicant transmitting a document entitled, "T-49 Flammability Control System Qualification Test" in response to staff request.

July 28, 1976	Letter from applicant transmitting application for extension of construction permit completion dates.
July 30, 1976	Letter from applicant transmitting Amendment No. 37.
August 12, 1976	Letter from applicant transmitting Revision 1 to the Industrial Security Plan.
August 12, 1976	Letter from applicant transmitting a list of significant valves which have contributed to delays in construction at the Zimmer site.
August 25, 1976	Letter from applicant transmitting document entitled, "Flammability Control System Qualification Test Specification."
August 27, 1976	Letter from applicant transmitting Amendment No. 38.
September 7, 1976	Letter from applicant transmitting a document entitled, "NEDO-21071, Standard Product Report-Flammability Control System (Thermal Recombiner)."
September 8, 1976	Letter from applicant transmitting additional copies of report submitted by letter dated July 27, 1976.
September 13, 1976	Letter to applicant concerning procedural changes for filing applications for construction permits and facility operating licenses.
September 15, 1976	Letter to applicant requesting additional information.
September 16, 1976	Letter to applicant requesting additional information.
September 17, 1976	Letter from applicant transmitting Amendment No. 39.
September 20, 1976	Letter from applicant providing submittal date for revised Mark II Containment Design Assessment Report.
September 27-29, 1976	Site vist - electrical review.
September 30, 1976	Letter to applicant regarding fire protection reevaluation.
September 30, 1976	Letter from applicant regarding proposed resolution of ATWS by installation of alternate reactor scram system (ARS).

October 7, 1976	Letter to applicant transmitting milestone dates for areas affected by delay in completion of requests for additional information.
October 7, 1976	Letter to applicant requesting additional information.
October 14, 1976	Summary of electrical site review on September 27-29, 1976.
October 22, 1976	Letter from applicant transmitting Amendment No. 40 and Revision 2 (proprietary) to Industrial Security Plan.
October 26, 1976	Letter from applicant providing submittal date for information requested in staff letter dated September 30, 1976.
October 26, 1976	Letter to applicant requesting additional information.
October 29, 1976	Letter from applicant transmitting Amendment No. 1 to the Mark II Containment Design Assessment Report.
October 29, 1976	Letter from applicant transmitting Amendment No. 41.
November 10, 1976	Letter to applicant requesting additional information.
November 15, 1976	Letter from applicant transmitting errata sheets for Amendment No. 1 to Mark II Containment Design Assessment Report submitted by letter dated October 29, 1976.
November 16, 1976	Meeting with applicant to discuss Zimmer plant electrical system.
November 16, 1976	Summary of meeting held October 27-29, 1976.
November 17, 1976	Letter to applicant requesting additional justification for withholding information submitted by letter dated July 27, 1976 from public disclosure.
November 19, 1976	Letter from applicant transmitting Amendment No. 42 and Revision 3 (proprietary) to Industrial Security Plan.
November 19, 1976	Meeting with applicant to discuss review status of the Zimmer Mark II containment.
November 19, 1976	Letter from applicant requesting staff to incorporate report submitted by General Electric Company entitled, "Mark II Containment Dynamic Forcing Function Information Report" in the Zimmer docket.

November 24, 1976	Letter to applicant regarding security clearances for review of classified SANDIA safeguards reports.
November 24, 1976	Letter from applicant providing revised date for submittal of responses to staff requests for additional information.
November 30, 1976	Letter to applicant requesting additional information.
November 30, 1976	Letter to applicant requesting additional information regarding applicant's request to extend construction permit completion dates.
November 30, 1976	Summary of meeting held November 19, 1976.
December 3, 1976	Meeting with applicant to discuss staff position regarding containment bypass leakage and associated tests and surveillance.
December 8, 1976	Letter to applicant transmitting page inadvertently omitted from November 30, 1976 request for additional information.
December 10, 1976	Summary of meeting held December 3, 1976.
December 14, 1976	Letter to applicant requesting additional information.
December 17, 1976	Letter to applicant transmitting additional guidance for the Zimmer fire protection reevaluation.
December 20, 1976	Meeting with applicant to discuss the Zimmer plant electrical systems.
December 27, 1976	Summary of meeting held December 20, 1976.
December 27, 1976	Letter to applicant regarding staff letter dated November 17, 1976, granting requesting to withhold information submitted on July 27, 1976 from public disclosure until January 19, 1977.
January 10, 1977	Letter to applicant requesting additional information.
January 10, 1977	Letter from applicant transmitting supplemental affidavits in support of request to withhold information from public disclosure, which was discussed in staff letter dated November 17, 1976.
January 10, 1977	Letter to applicant requesting additional information.

January 14, 1977	Letter from applicant transmitting Amendment No. 43.
January 18, 1977	Letter to applicant requesting additional information regarding Mark II Pool Dynamics Load Program.
January 26, 1977	Letter to applicant requesting additional information regarding the Mark II Containment Design Assessment Report.
February 4, 1977	Letter from applicant transmitting Amendment No. 44 and Zimmer Fire Protection Evaluation Report.
February 8, 1977	Letter to applicant granting July 27, 1976 request to withhold proprietary information from public disclosure contained in NEDE-21071-P, "Flammability Control System (Thermal Recombiner)."
February 9, 1977	Site visit to discuss emergency planning and industrial security matters.
February 10, 1977	Meeting with applicant to discuss applicant's responses to staff requests for additional information in the area of initial tests and operations.
February 12, 1977	Letter to applicant requesting additional information.
February 14, 1977	Meeting with applicant to discuss applicant's responses to staff requests for additional information in the area of containment systems.
February 14, 1977	Summary of site visit on February 9, 1977.
February 18, 1977	Letter from applicant transmitting Amendment No. 45.
February 18, 1977	Summary of meeting held on February 10, 1977.
February 22, 1977	Letter from applicant transmitting proposed submittal dates for responses to staff requests for additional information.
February 23, 1977	Meeting with applicant to discuss status of Zimmer review and schedular problems requiring management attention.
February 25, 1977	Summary of meeting held on February 14, 1977.
February 25, 1977	Letter to applicant transmitting Amendments to 10 CFR Parts 50 and 73 concerning physical protection of licensed activities in nuclear power reactors against industrial sabotage.

February 28, 1977	Letter from applicant transmitting Amendment No. 46.
February 28, 1977	Letter from applicant transmitting Amendment No. 2 to the Mark II Containment Design Assessment Report.
March 10, 1977	Letter to applicant requesting additional information.
March 10, 1977	Letter to applicant regarding NRC evaluation of fuel handling accidents for all nuclear power plants.
March 11, 1977	Letter from applicant transmitting Amendment No. 47.
March 14, 1977	Letter from applicant regarding ATWS.
March 14, 1977	Meeting with applicant to discuss applicant's position concerning Mark II containment dynamic load back-up.
March 15, 1977	Meeting with applicant to discuss containment wetwell bypass leakage and containment isolation valves.
March 17, 1977	Summary of meeting held February 23, 1977.
March 21, 1977	Summary of meeting held March 14, 1977.
March 22, 1977	Summary of meeting held March 15, 1977.
March 29, 1977	Meeting with applicant to discuss accident analysis dose calculations for the Zimmer plant.
March 30, 1977	Meeting with applicant to discuss resolution of wetwell bypass allowable leakage and subcompartment analysis.
March 31, 1977	Meeting with applicant to discuss responses to staff requests for additional information regarding the electrical review.
April 1, 1977	Letter from applicant transmitting Amendment No. 3 to the Mark II Containment Design Assessment Report.
April 1, 1977	Letter from applicant transmitting Amendment No. 48.
April 14, 1977	Letter from applicant transmitting annual financial statements as required by 10 CFR 50.71b.
April 14, 1977	Letter to applicant requesting additional information regarding instrument trip setpoint valves.

April 14, 1977	Summary of meeting held March 30, 1977.
April 14, 1977	Summary of meeting held March 31, 1977.
April 15, 1977	Letter from applicant transmitting Amendment No. 50.
April 18, 1977	Letter to applicant requesting additional information.
April 18-22, 1977	Meeting with applicant at General Electric Company facilities in San Jose, California to discuss design review of recirculation system and reactor manual control system.
April 20, 1977	Letter from applicant transmitting supplemental information to Mark II Containment Design Assessment Report entitled, "Containment Structure Sensitivity to Pool Hydrodynamic Loads."
April 22, 1977	Letter to applicant transmitting standard format for meteorological data submitted on magnetic tape.
April 26, 1977	Letter from applicant providing additional information regarding request to extend construction permit completion dates.
April 27, 1977	Letter from applicant transmitting document entitled, "Information for Antitrust Review of Operating License Application."
May 2, 1977	Letter from applicant transmitting Amendment No. 51.
May 2, 1977	Summary of meeting held on April 18-22, 1977.
May 4, 1977	Letter to applicant transmitting Intrusion Detection Systems Handbook, SAND 76-0554, dated November 1976.
May 12, 1977	Meeting with applicant to discuss revisions to the Final Safety Analysis Report.
May 13, 1977	Letter from applicant providing submittal date for information requested in staff letter dated April 14, 1977 concerning instrument trip setpoint valves.
May 23, 1977	Summary of meeting held May 12, 1977.

May 24, 1977	Meeting with applicant to discuss Zimmer review schedule and its potential impact on planned fuel load date.
May 25, 1977	Letter from applicant transmitting Amendment No. 52 and Revision 4 (proprietary) to the Industrial Security Plan.
May 27, 1977	Letter from applicant transmitting Amendment No. 53.
June 1, 1977	Letter to applicant transmitting Order extending the latest construction completion date from January 1, 1977 to April 1, 1980.
June 6-8, 1977	Site visit to discuss fire protection review.
June 7, 1977	Summary of meeting held May 24, 1977.
June 8, 1977	Letter from applicant transmitting Amendment No. 4 to the Mark II Containment Design Assessment Report.
July 6, 1977	Meeting with applicant to discuss resolution of Appendix J matters in the Safety Evaluation Report.
July 12, 1977	Summary of meeting held July 6, 1977.
July 14, 1977	Meeting with applicant to discuss status of Safety Evaluation Report open issues and completion date.
July 20, 1977	Letter from applicant requesting withholding from public disclosure, proprietary report entitled, "Qualification Report Flammability Control System (Thermal Recombiner)."
July 22, 1977	Letter from applicant transmitting Amendment No. 54.
July 22, 1977	Letter from applicant transmitting Amendment No. 5 to the Mark II Containment Design Assessment Report.
July 29, 1977	Letter from applicant transmitting Amendment No. 55.
August 4, 1977	Summary of meeting held July 14, 1977.
August 5, 1977	Letter to applicant requesting additional information.
August 11, 1977	Letter to applicant requesting additional information.

August 12, 1977	Letter from applicant transmitting Amendment No. 6 to the Mark II Containment Design Assessment Report.
August 16, 1977	Letter to applicant requesting additional information.
August 16, 1977	Letter from applicant transmitting Amendment No. 56.
August 19, 1977	Letter to applicant transmitting information to be provided to staff for Upgraded STS Bases Program.
August 23-24, 1977	Meeting with applicant to discuss status of Safety Evaluation Report for electrical review; onsite power systems and drawing review of load sequencing system.
August 29, 1977	Letter to applicant transmitting "Nuclear Plant Fire Protection Functional Responsibilities" for use as supplemental guidance for the review of organizational and administrative aspects of fire protection evaluations.
August 30, 1977	Letter from applicant transmitting Amendment No. 57.
September 8, 1977	Letter to applicant requesting additional information concerning suppression pool temperature limit (enclosure withheld from public disclosure-proprietary).
September 14, 1977	Summary of meetings held August 23-24, 1977.
September 15, 1977	Letter from applicant transmitting Amendment No. 58.
September 15-16, 1977	Meeting with applicant to discuss vessel and core support loads, control rod drive return lines, equipment operability, feedwater nozzles and wetwell sprays.
September 19, 1977	Letter to applicant transmitting Federal Register notice of a petition for rulemaking regarding physical searches of individuals entering a protected area of a nuclear power plant and requesting applicant to post notice for information of all employees.
September 21, 1977	Letter to applicant requesting additional information concerning Mark II Containment Design Assessment Report.
September 21, 1977	Letter to applicant requesting additional information.

September 23, 1977	Letter from applicant providing response to staff letter dated August 19, 1977, which invited applicant's participation in a program to upgrade the bases section of the generic standard technical specifications for Zimmer.
September 28, 1977	Summary of meeting held September 15-16, 1977.
September 30, 1977	Letter from applicant transmitting Amendment No. 59.
October 4-5, 1977	Meeting with applicant to discuss electrical review Safety Evaluation Report input.
October 5, 1977	Letter from applicant providing submittal dates for staff requests for additional information.
October 6, 1977	Summary of meetings held October 4-5, 1977.
October 20, 1977	Letter to applicant requesting additional information.
October 21, 1977	Meeting with applicant to discuss 10 CFR Part 50, Appendix J and reverse pressurization of the drywell.
October 21, 1977	Letter from applicant transmitting Revision 2 to the Zimmer Fire Protection Evaluation Report.
October 21, 1977	Letter from applicant transmitting Amendment No. 60.
November 1, 1977	Letter to applicant concerning diesel generator operating status indication.
November 1, 1977	Letter to applicant granting requests to withhold proprietary document from public disclosure entitled, "Flammability Control System (Thermal Recombiner)."
November 2, 1977	Letter from applicant transmitting Amendment No. 61 and Revision 3 to the Zimmer Fire Protection Evaluation Report.
November 2, 1977	Letter from applicant transmitting Amendment No. 7 to the Mark II Containment Design Assessment Report.
November 3, 1977	Letter to applicant concerning BWR relief valve control system and associated containment loads.
November 10, 1977	Summary of meeting held October 21, 1977.

November 11, 1977	Letter from applicant transmitting Amendment No. 62 and Revision 4 to the Zimmer Fire Protection Evaluation Report.
November 14, 1977	Letter from applicant transmitting Amendment No. 63.
November 17, 1977	Letter from applicant providing submittal date for response to staff request for additonal information concerning instrumentation and control systems area.
November 18, 1977	Letter from applicant transmitting Amendment No. 64.
November 21-22, 1977	Site visit to review reactor systems.
November 21 1977	Letter from applicant concerning actuation of main steam safety/relief valves.
November 23, 1977	Letter to applicant requesting additional financial information.
November 23, 1977	Letter to applicant requesting additional information concerning pressure vessel fracture toughness properties.
November 28, 1977	Letter from applicant providing submittal date for financial information requested by staff letter dated November 23, 1977.
November 28, 1977	Letter to applicant transmitting amendment to 10 CFR Part 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors vs Industrial Sabotage," which delays implementation of physical search requirements of reactor personnel until August 24, 1978.
November 29, 1977	Meeting with applicant to discuss startup testing as part of facility operating license review for Safety Evaluation Report.
November 30, 1977	Letter from applicant transmitting Amendment No. 65 and Revision 5 to the Zimmer Fire Protection Evaluation Report.
December 12, 1977	Letter from applicant transmitting information regarding operating status indication of diesel generators at the Zimmer plant in response to staff letter dated November 1, 1977.
December 14, 1977	Letter from applicant transmitting financial information requested by staff letter dated November 12, 1977, revised pages to September 10, 1975 amendment and supplement to application, and update of survey of milk animals near Zimmer plant.

December 16, 1977	Letter from applicant transmitting Amendment No. 66.
December 30, 1977	Letter from applicant transmitting Amendment No. 67.
January 5, 1978	Meeting with applicant to discuss Zimmer plant specific resolution SRSS load combinations, downcomer lateral loads, and relief valve actuation.
January 6, 1978	Summary of December 16, 1977 Mark II owner's group meeting with staff.
January 10, 1978	Summary of May 18 and 19, 1977 Mark II owner's group meeting with staff.
January 11, 1978	Meeting notice for January 26, 1978 meeting with CG&E on Industrial Security.
January 12, 1978	Meeting notice for January 25, 1978 meeting with CG&E on drywell/wetwell allowable bypass.
January 12, 1978	Summary of January 5, 1978 meeting with CG&E on Zimmer review issues.
January 12, 1978	Letter to CG&E requesting additional information
January 13, 1978	Letter from CG&E (1-11-78) requesting NRC staff site visits.
January 13, 1978	Letter from CG&E (1-11-78) regarding Fire Protection Administrative Controls.
January 18, 1978	Letter to CG&E requesting additional Mark II information.
January 18, 1978	Summary of December 14, 1977 Mark II owner's group meeting with staff.
January 23, 1978	Letter to CG&E concerning review issues.
January 26, 1978	Letter from CG&E (1-23-78) concerning interim response to second actuation of main steam safety valves 50.55(e) issue.
January 31, 1978	Letter from CG&E (1-27-78) transmitting Amendment 68, Revision 39 to FSAR.
February 1, 1978	Letter from CG&E (1-31-78) transmitting Amendment 8 to Zimmer Design Assessment Report.

February 2, 1978	Letter to CG&E concerning inservice testing program for pumps and valves.
February 8, 1978	Meeting notice for February 14, 1978 meeting with CG&E on drywell/wetwell allowable bypass.
February 9, 1978	Letter to CG&E concerning second actuation of safety valves.
February 14, 1978	Letter from CG&E (2-13-78) regarding fracture toughness properties of reactor vessels.
February 14, 1978	Letter to CG&E requesting additional Round 2 information.
February 21, 1978	Summary of February 14, 1978 appeals meeting with CG&E on drywell/wetwell allowable bypass.
February 22, 1978	Summary of February 15, 1978 meeting with CG&E on Zimmer review issues.
February 23, 1978	Letter from CG&E (2-21-78) regarding assessment of conservatism in transient analysis methods.
February 28, 1978	Letter to CG&E transmitting staff position on the use of austenitic stainless steel in BWRs.
March 2, 1978	Letter from CG&E (2-20-78) transmitting Amendment 9 to Mark II Design Assessment Report.
March 2, 1978	Letter from CG&E (2-28-78) transmitting Amendment 69, Revision 40 to the FSAR.
March 3, 1978	Letter from CG&E (2-28-78) transmitting Revision 6 to Fire Protection Report.
March 15, 1978	Letter to CG&E requesting additional Mark II information.
March 16, 1978	Meeting notice for March 22-23, 1978 meeting with CG&E on Fire Protection.
March 16, 1978	Meeting notice for March 31, 1978 site visit to Zimmer site on preoperational testing.
March 17, 1978	Meeting notice for April 11-12, 1978 meeting with CG&E on review information schedules.

March 17, 1978	Summary of April 4, 1978 meeting with CG&E on BWR Equipment Seismic Qualification.
March 17, 1978	Letter to CG&E summarizing the staff's position of the February 14, 1978 appeals meeting.
March 27, 1978	Summary of March 22, 1978 meeting with CG&E on fire protection.
March 28, 1978	Summary of March 21, 1978 meeting with CG&E on Zimmer design assessment.
March 29, 1978	Summary of March 3, 1978 meeting with Mark II owners on the containment pool dynamic loads program.
April 3, 1978	Letter from CG&E (3-31-78) transmitting Amendment 70, Revision 41 to the FSAR.
April 3, 1978	Letter from CG&E (3-30-78) transmitting annual financial statements for Zimmer applicants.
April 14, 1978	Letter to CG&E concerning dewatering of compacted backfill material.
April 17, 1978	Letter from CG&E (4-14-78) transmitting Amendment 10 to the Zimmer Design Assessment Report.
April 17, 1978	Summary of April 13, 1978 meeting with CG&E on mass-energy release calculations.
April 18, 1978	Summary of April 12, 1978 meeting with CG&E on issues information schedules.
April 18, 1978	Summary of April 11, 1978 meeting with CG&E on SRV discharge devices.
April 18, 1978	Summary of April 6, 1978 meeting with GE in San Jose to discuss Pool Dynamic Loads Program.
April 21, 1978	Letter to CG&E concerning main steam isolation valve leakage control system for Zimmer.
April 25, 1978	Summary of April 5, 1978 meeting with Mark II owners at Livermore to discuss pool dynamic loads.

May 1, 1978	Letter from CG&E (4-28-78) transmitting Amendment 71, Revision 42 to the FSAR.
May 2, 1978	Meeting notice for May 24, 1978 Zimmer site visit by members of Containment Systems Branch.
May 3, 1978	Meeting notice for June 28, 1978 meeting with CG&E on Design Assessment Report and discharge devices.
May 3, 1978	Meeting notice for May 31, 1978 meeting with CG&E on SRV discharge devices and DAR.
May 8, 1978	Letter to CG&E concerning Mark II containment review request for additional information.
May 10, 1978	Letter to CG&E concerning short term mass and energy release rates.
May 10, 1978	Letter to CG&E transmitting request for additional information.
May 12, 1978	Letter from CG&E (5-12-78) concerning Closure Report.
May 30, 1978	Letter from CG&E (5-26-78) transmitting Amendment 72, Revision 43 to FSAR.
June 16, 1978	Meeting notice for July 13, 1978 meeting with CG&E concerning Zimmer plant staffing.
June 16, 1978	Meeting notice for July 14, 1978 meeting with CG&E concerning fire protection.
June 22, 1978	Summary of May 16-17, 1978 meetings with the Mark II owners on pool dynamics program.
June 22, 1978	Letter to CG&E concerning a request for additional information.
July 3, 1978	Letter from CG&E (6-26-78) transmitting Amendment 11 to the Design Assessment Report.
July 10, 1978	Meeting notice for July 7, 1978 Zimmer site visit concerning fire protection.
July 10, 1978	Letter from CG&E (7-5-78) concerning second actuation of relief valves.

July 10, 1978	Summary of June 28, 1978 meeting with CG&E on SRV discharge devices and Closure Report.
July 11, 1978	Letter from CG&E (7-10-78) transmitting Amendment 73, Revision 44 to the FSAR and the Closure Report.
July 19, 1978	Summary of July 13-14, 1978 meetings with CG&E concerning plant staffing and fire protection.
August 1, 1978	Letter to CG&E transmitting Appendix I model Tech Specs for BWRs.
August 1, 1978	Letter to CG&E concerning second actuation of safety relief valves.
August 2, 1978	Letter from CG&E (7-31-78) transmitting Amendment 75, Revision 45 to the FSAR.
August 2, 1978	Letter to CG&E (8-2-78) transmitting manpower requirements for operating reactors.
August 9, 1978	Meeting notice for August 14, 15, 16, 17, 1978 site visit of SQRT team.
August 9, 1978	Meeting notice for August 24, 1978 meeting on SER status.
August 11, 1978	Letter to CG&E (8-11-78) requesting additional information.
August 11, 1978	Letter to CG&E (8-11-78) concerning reactor protection system power supplies.
August 29, 1978	Summary of August 24, 1978 management meeting with CG&E on licensing review.
August 30, 1978	Letter from CG&E (8-29-78) transmitting Amendment 76, Revision 46 to the FSAR and Revision 7 to Fire Protection Evaluation Report.
September 11, 1978	Letter from CG&E (9-6-78) regarding dewatering of backfill position.
September 15, 1978	Summary of August 14, 15, 16, 1978 site visit.
September 29, 1978	Letter from CG&E (9-29-78) transmitting Amendment 77, Revision 47, to FSAR and Revision 8 to the Fire Protection Evaluation Report.

September 29, 1978	Letter to CG&E (9-22-78) requesting additional information.
September 29, 1978	Letter to CG&E (10-2-78) requesting additional information.
October 5, 1978	Letter from CG&E (10-2-78) regarding reactor protection system.
October 10, 1978	Letter to CG&E (10-10-78) requesting additional information.
October 11, 1978	Meeting notice for November 11, 1978 meeting NSSS & BOP piping systems.
October 30, 1978	Letter from CG&E (10-30-78) transmitting Amendment 78, Revision 48, to FSAR.
November 27, 1978	Letter to CG&E (11-27-78) requesting updated financial data.
November 28, 1978	Summary of November 9, 1978 meeting with CG&E on ABS evaluation.
November 30, 1978	Letter from CG&E (11-30-78) transmitting Amendment 79, Revision 49, to FSAR.
December 1, 1978	Letter from CG&E (12-1-78) on fire brigade manpower.
December 5, 1978	Letter to CG&E (12-5-78) requesting additional information.
December 8, 1978	Letter from CG&E (12-3-78) transmitting resumes of Reactor Engineer and Maintenance Supervisor.
December 13, 1978	Letter from CG&E (12-13-78) concerning NUREG-0313.
December 15, 1978	Letter from CG&E (12-15-78) transmitting Amendment 80, Revision 9, to Fire Protection Report.
December 18, 1978	Letter from CG&E (12-18-78) transmitting Amendment 81, Revision 50, to FSAR.
December 22, 1978	Letter from CG&E (12-22-78) transmitting response to staff positions.

APPENDIX B

ACRS GENERIC CONCERNS

The Advisory Committee on Reactor Safeguards periodically issues a report listing various matters of generic concern applicable to large light water reactors. These are items which the Committee and we, 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 on November 15, 1977.

The status of our efforts leading to resolution of all unresolved 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 dated December 4, 1978.

For several of the items we have provided in this report specific discussions particularizing for this facility the generic status in the status report. These items are listed below with the appropriate subsection numbers of this report where such discussions are included. The group numbering corresponds to that in the November 15, 1977 report of the Committee.

For those generic matters applicable to the William H. Zimmer Nuclear Power Station design, but for which specific conclusions .. actions have not been identified for consideration on this application, our status report of December 4, 1978 provides the appropriate information.

GROUP II - Resolution Pending

(1) Turbine Missiles - This item is resolved for the Zimmer Station by the protection provided by the applicant (subsection 3.5 of this report).

(2) Effective Operation of Containment Sprays in a Loss-of-Coolant Accident - Not applicable to the Zimmer Station.

(3) Possible Failure of Pressure Vessel Post-Loss-of-Coolant Accident by Thermal Shock - This item is under generic review as indicated in our status report to the ACRS dated December 4, 1978.

(4) Instruments to Detect (Severe) Fuel Failures - This item is partly resolved, as reported in the November 15, 1977 letter from the ACRS to the Commission. Instrumentation to detect fuel failures associated with normal operation and transients (limited fuel failures) has been shown to be adequate. The adequacy of instrumentation to detect failures associated with more rapid events during which substantial fuel failures could occur has not been demonstrated and this concern is considered unresolved. Further work is necessary to determine (a) the adequacy of current instrumentation for these rapid events, and (b) the need for additional instrumentation. Research administered by the Office of Reactor Safety Research and studies conducted under contracts administered by the Office of Nuclear Reactor Regulation should provide the information required to evaluate instrumentation limitations and needs. In the interim, we have not identified any credible event (transient or accident sequence) for which a rapid fuel failure detection system would prevent "substantial" fuel failure (including fuel melt) and loss of coolable geometry. (See our December 4, 1978 status report.)

(5A) Monitoring for Loose Parts Inside the Reactor Pressure Vessel - This item is resolved for the Zimmer Station (subsection 4.4.1).

(5B) Montioring for Excessive Vibration Inside the Reactor Pressure Vessel - This item is under generic review as indicated in our status report to the ACRS dated December 4, 1978.

(6) Common Mode Failures - This item is under generic review as indicated in our
(6A) status report to ACRS dated December 4, 1978.
(6B)
(6C)

(7) Behavior of Reactor Fuel Under Abnormal Conditions - This item is under generic review as indicated in our status report to ACRS dated December 4, 1978.

(8) Boiling Water Reactor Recirculation Pump Overspeed During a Loss-of-Coolant Accident - This item is under generic review as indicated in our status report to the ACRS dated December 4, 1978.

(9) The Advisability of Seismic Scram - A seismic scram is not proposed for the Zimmer Station and we will not require such a scram (See letter, dated May 19, 1977, from E. Case, Acting Director, Office of Nuclear Reactor Regulation, to ACRS Chairman Bender; Subject: "The Advisability of a Seismic Scram," and our December 4, 1978 status report.)

(10) Emergency Core Cooling System Capability for Future Plants - This item is under generic review as indicated in our status report to ACRS dated December 4, 1978.

GROUP II A - Resolution Pending - Items Since December 18, 1972

(1) Ice Condenser Containments - Not applicable to the Zimmer Station.

(2) Pressurized Water Reactor Pump Overspeed During a Loss-of-Coolant Accident - Not applicable to the Zimmer Station.

(3) Steam Generator Tube Leakage - Not applicable to the Zimmer Station.

(4) ACRS/NRC Periodic (10-Year) Review of All Power Reactors - This item is under generic review as indicated in our status report to the ACRS dated December 4, 1978.

GROUP II B - Resolution Pending - Items Added Since February 13, 1974.

(1) Computer Reactor Protection Systems - Not applicable to the Zimmer Station.

(2) Qualification of New Fuel Geometries - This item is resolved for the Zimmer Station by the analysis and confirmatory programs conducted (subsection 4.2 of this report).

(3) Behavior of Boiling Water Reactor Mark III Containment - Not applicable to the Zimmer Station.

(4) Stress Corrosion Cracking in Boiling Water Reactor Piping - This item is resolved for the Zimmer Station by position taken by us and corrective actions by the applicant (subsection 5.2 of this report).

GROUP II C - Resolution Pending - Items Added Since March 12, 1975

(1) Locking Out of Emergency Core Cooling System Power-Operated Valves - Not applicable to the Zimmer Station.

(2) Design Features to Control Sabotage - On February 24, 1977 the Commission published new requirements for the physical protection of nuclear power plants against acts of sabotage (10 CFR 73.55). The new rule requires compliance at the operating license stage. As a result of our review of the applicant's preliminary plans for physical security, we conclude 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. (See our December 4, 1978 status report.)

(3A& 3B) Decontamination and Decommissioning of Reactors - These items are under generic review as indicated in our status report to ACRS dated December 4, 1978.

(4) Vessel Support Structures - This item is resolved for the Zimmer Station by position taken by us (subsection 5.2.1 of this report).

(5) Water Hammer - This item is resolved for the Zimmer Station by the specific design (subsection 6.3.2 of this report).

(6) Maintenance and Inspection of Plants - This item is resolved for the Zimmer Station (subsection 12.3 of this report).

(7) Behavior of Boiling Water Reactor Mark I Containments - Not applicable to the Zimmer Station.

GROUP II D - Resolution Pending - Items Added Since April 16, 1976

(1A) Safety-Related Interfaces Between Reactor Island and Balance-of-Plant - Not applicable to the Zimmer Station. (See our December 4, 1978 status report.)

(1B) Systems Interaction in Nuclear Power Plants - This item is under generic review as indicated in our status report to the ACRS dated December 4, 1978.

(2) Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment - This item is under generic review as indicated in our status report to the ACRS dated December 4, 1978.

GROUP II E - Resolution Pending - Items Added Since February 24, 1977

(1) Soil-Structure Interactions - This item is being evaluated as part of Task Action Plan A-40, "Seismic Design Criteria," currently under development. The soil-structure interaction evaluation is currently underway and its objective is to determine limits and conditions of applicability as well as estimates of conservatism in the definition of seismic input and soil-structure interaction currently used in the seismic analysis of nuclear power plants. (See our December 4, 1978 status report.)

NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER (Assigned by DDC) NUREG-0528
4. TITLE AND SUBTITLE (Add Volume No., if appropriate) Safety Evaluation Report related to operation of the Wm. H. Zimmer Nuclear Power Station, Unit 1	2. (Leave blank)
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S)	5. DATE REPORT COMPLETED MONTH / YEAR
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) U. S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation Washington, D. C. 20555	DATE REPORT ISSUED MONTH January YEAR 1979
	6. (Leave blank)
	8. (Leave blank)
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Same as 9 above	10. PROJECT/TASK/WORK UNIT NO.
	11. CONTRACT NO.
13. TYPE OF REPORT Safety Evaluation Report	PERIOD COVERED (Inclusive dates)
15. SUPPLEMENTARY NOTES	14. (Leave blank)

16. ABSTRACT (200 words or less)

A safety evaluation of the Cincinnati Gas and Electric Company's application for a license to operate its Wm. H. Zimmer Nuclear Power Station, Unit 1 (Docket No. 50-358), located on the Ohio River in Clermont County, Ohio, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. It consists of a technical review and staff evaluation of applicant information on: (1) population density, land use, and physical characteristics of the site area; (2) design, fabrication, construction, testing criteria, quality assurance program, and performance characteristics of plant structures, systems, and components important to safety; (3) expected response of the facility to anticipated operating transients, and to postulated design basis accidents; (4) applicant engineering and construction organization, and plans for the conduct of plant operations; and (5) design criteria for a system to control the plant's radiological effluents. The staff has concluded that the plant can be operated by the Cincinnati Gas and Electric Company without endangering the health and safety of the public provided that the outstanding matters discussed in the report are favorably resolved.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
17b. IDENTIFIERS/OPEN-ENDED TERMS	

18. AVAILABILITY STATEMENT	19. SECURITY CLASS (This report)	21. NO. OF PAGES
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